Appendix 4A. Tables

Table 4-1. Reactor Design Comparison Table

	Robust Fuel Assembly
Thermal And Hydraulic Design Parameters	
1. Reactor Core Heat Output, (100%), MWt	3469
2. Reactor Core Heat Output, 10^6 Btu/hr	11836.7
3. Heat Generated in Fuel, %	97.4
4. System Pressure, Nominal, psia(1)	2280
5. System Pressure, Minimum Steady State, psia(1)	2250
6. Minimum DNBR at Nominal Conditions Limiting Channel	2.85 (WRB-2M)
7. Minimum DNBR at Design Transients Limiting Channel	1.45 (WRB-2M)
8. DNB Correlation	WRB-2M
COOLANT FLOW(3)	
9. Total Thermal Flow Rate, 10^6 lbm/hr	145.2
10. Effective Flow Rate for Heat Transfer, 10^6 lbm/hr	136.5
11. Effective Flow Area for Heat Transfer, ft^2	51.1
12. Average Velocity Along Fuel Rods, ft/sec	15.9
13. Average Mass Velocity, 10^6 lbm/hr-ft^2	2.67
COOLANT TEMPERATURE, °F(2)	
14. Nominal Inlet	553.1
15. Average Rise in Vessel	61.2
16. Average Rise in Core	65.0
17. Average in Core	587.3
18. Average in Vessel	585.1
HEAT TRANSFER	
19. Active Heat Transfer, Surface Area, ft^2	59,866
20. Average Heat Flux, Btu/hr-ft^2	192,579
21. Maximum Heat Flux for Normal Operation, Btu/hr-ft^2	481,447

Thermal And Hydraulic Design Parameters	Robust Fuel Assembly
22. Average Linear Power, kW/ft	5.53
23. Peak Linear Power for Normal Operation, kW/ft[(a)]	13.8
24. Peak Linear Power Resulting from Overpower Transients/Operator Errors (assuming a maximum overpower of 118%), kW/ft[(b)]	18.0
25. Peak Linear Power for Prevention of Centerline Melt, kW/ft	>18.0
26. Power Density, kW prr Liter of Core	104.5
27. Specific Power, kW per kg Uranium[(4)]	38.8
FUEL CENTRAL TEMPERATURE	
28. Peak at Peak Linear Power for Prevention of Centerline Melt, °F	Burnup Dependent
29. Pressure Drop (++)	
Across Core, psi	28.8 +/- 2.6
Across Vessel, Including Nozzle psi	51.2 +/- 4.6
Items 30-64	Deleted duplicate and historical 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.
   a. This limit is associated with the value of Fq = 2.50 and includes 2.6% gamma heating.
   b. See Section 4.3.2.2.6

   (+) Based on cold dimensions.

   (++) Based on best estimate reactor flow as discussed in Section 5.1. RFA pressure drops are based on Reference 98 of Section 4.4.7.
2. These values are typical values based on RCS flow of 400,000 gpm and a bypass flow of 6.0%.
3. These values are typical values based on RCS flow of 388,000 gpm and nominal inlet temperature of 553.1°F .
4. Typical values. May vary based on reload specific data.

Table 4-2. Analytic Techniques in Core Design

Analysis	Technique	Computer Code
Mechanical Design of Core Internals		
Loads, Deflections, and Stress Analysis	Static and Dynamic Modeling	Blowdown code, FORCE Finite element structural
Fuel Rod Design		
Fuel Performance Characteristics (temperature, internal pressure, clad strain, etc.)	Semi-empirical thermal model of fuel rod with consideration of fuel density changes, heat transfer, fission gas release, etc.	PAD
Nuclear Design		
1. Cross Sections and Group Constants	Microscopic and	Modified ENDF/B library
	Macroscopic constants for homogenized core regions	CASMO-3 or CASMO-4
	Group constants for control rods with self-shielding	CASMO-3 or CASMO-4
2. X-Y Power Distributions, Fuel Depletion, Critical Boron Concentrations, X-Y Xenon Distributions, Reactivity Coefficients	Collapsed 3-D, 2-Group NEM Based Nodal Code	SIMULATE-3P or SIMULATE-3 MOX
3. Axial Power Distributions, Control Rod Worths, and Axial Xenon Distribution	2-D and 3-D 2-Group Model Analysis Code	SIMULATE-3P or SIMULATE-3 MOX
4. Fuel Rod Power	Reconstructed Integral Rod Power	SIMULATE-3P or SIMULATE-3 MOX
5. Criticality of Reactor and Fuel Assemblies	1-D, Multi-Group Transport Theory	AMPX System of Codes
	3-D Monte Carlo	KENO-IV
Thermal-Hydraulic Design		
1. Steady-State	Subchannel analysis of local fluid conditions in rod bundles, including inertial and crossflow resistance terms, solution progresses from core-wide to hot assembly to hot channel	VIPRE-01

Analysis	Technique	Computer Code
2. Transient DNB Analysis	Subchannel analysis of local fluid conditions in rod bundles during transients by including accumulation terms in conservation equations; solution progresses from core-wide to hot assembly to hot channel	VIPRE-01

Table 4-3. Deleted Per 1992 Update

Table 4-4. Reactor Core Description (Units 1 and 2)

	Robust Fuel Assembly
Active Core	
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_2O/U Molecular Ratio, Lattice (68°F, 2250 psia)	~2.50
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 Barrel, 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. (Typical)(2)	219,819(1)
Zirconium Weight, lbs. (Cladding Surrounding Active Fuel)	41,966(1)
Number of Grids per Assembly	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™ -2280
Number of Guide Thimbles per Assembly	24
Composition of Guide Thimbles	ZIRLO™

	Robust Fuel Assembly
Inner Diameter of Guide Thimbles (upper part), in.	0.442
Outer Diameter of Guide Thimbles (upper part), in.	0.482
Inner Diameter of Guide Thimbles (lower part), in.	0.397
Outer Diameter of Guide Thimbles (lower part), in.	0.439
Inner Diameter of Instrument Guide Thimbles, in.	0.442
Outer Diameter of Instrument Guide Thimbles, in.	0.482
Fuel Rods	
Number	50,592
Outside Diameter, in.	0.374
Diameter Gap, in.	0.0065
Clad Thickness, in.	0.0225
Clad Material	ZIRLO™
Fuel Pellets	
Material	UO_2 Sintered
Density (percent of Theoretical)	95.5
Fuel Enrichments w/o[5]	
Reload Regions	0.711-5.00
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]
Rod Cluster Control Assemblies(Unit 1)	
Westinghouse Enhanced Performance (EP) RCCAs	
Neutron Absorber	80%, 15%, 5%
Composition	(Ag,In,Cd)

Rod Cluster Control Assemblies(Unit 1)	
Diameter, in.	
Upper	0.341
Lower	0.336
Density, lbs/in.^{3}	0.367
Cladding Material	Type 304 Cold Worked Stainless Steel, Chrome Plated
Number of Full Length Clusters	53
Number of Absorber Rods per Cluster	24
Full Length Assembly Weight, (dry), lb.	149
AREVA AIC HARMONI RCCAs	
Neutron Absorber	80%, 15%, 5%
Composition	(Ag,In,Cd)
Diameter, in.	
Upper	0.341
Lower	0.336
Density, lbs/in.^{3}	0.367
Cladding Material	Type 304 Cold Worked Stainless Steel, Ion-nitrated
Number of Full Length Clusters	53
Number of Absorber Rods per Cluster	24
Full Length Assembly Weight, (dry), lb.	149

Hybrid Ionitrided Rod Cluster Control Assemblies (Unit 2)	
Neutron Absorber	B_4C
Diameter, in.	0.294
Length, in.	102
Density, lbs/in^3	0.064
Tip Material	(Ag-In-Cd)
Composition	80%, 15%, 5% (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
Cladding Material	Type 136, Cold Worked Stainless Steel, Ionitrided
Cladding Thickness	.0385
Number of Full Length Clusters	53
Full Assembly Weight (dry), lb.	94
Burnable Poison Rod Loading & Initial Reactivity Worth	
Weight of Boron – 10 per foot of rod, lb/ft	Variable
Initial Reactivity Worth, %Δρ (hot)	0.0-~3.0 (typical)
Initial Reactivity Worth, %Δp (cold)	0.0-~2.2 (typical)
Excess Reactivity	
Maximum Fuel Assembly K∞ (Cold, Clean, Unborated Water)	Variable[3]
Maximum Core K∞ (Cold, Zero Power, Beginning of Cycle)	1.30[4]
WABAs	
Material	$A1_2O_3$-B_4
Inside Diameter, in.	0.225
Outside Diameter, in.	0.381
Clad Material	Zircaloy-4
Boron Loading	Proprietary

Note:

1. The values indicated are typical, for 17 x 17 Robust Fuel Assemblies, or Mk-BW fuel assemblies.
2. Not exact for every core. Total weight will vary as region UO_2 varies. See region specific data for the most current values.
3. Maximum Fuel Assembly k-infinities for cold clean unborated water are dependent upon the fuel assembly enrichment.
4. Variable, depending on cycle length and BA loading.
5. The fuel enrichments for the first core are 2.10w/o (Region 1), 2.60w/o (Region 2), 3.10w/o (Region 3) per Ref. 19 in Section 4.2.4.

Table 4-5. Nuclear Design Parameters [HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

Core Average Linear Power, kW/ft, including densification effects and gamma heating effects	5.44
Total Heat Flux Hot Channel Factor, F_Q	2.50
Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta N}^{H}$	Variable limit based on the magnitude and location of the axial peak, F_z.

Reactivity Coefficients (Reload Cycles)	**Tech Spec/Safety Analysis Design Limits**	**Best estimate**
Least-negative Doppler-only power coefficient, pcm/% Power	-9.5 to -6.0	-17.5 to -8.3
Distributed Doppler Temperature Coefficient, pcm/°F	-3.50 to -0.9	-2.0 to -1.2
Moderator Temperature Coefficient, pcm/°F	$<+7$ at $0 \le P \le 7$ [1] <0 at $P = 1.0$	+5 to -38
Rodded Moderator Density, pcm/gm/cc	$<0.43 \times 10^5$	0.38×10^5

Delayed Neutron Fraction and Lifetime	***First Cycle***	**Reload Cycle**
β_{eff} BOL	*0.0075*	0.0062
β_{eff} EOL	*0.0044*	0.0052
l BOL, μ sec	*19.4*	17
l EOL, μ sec	*18.1*	21
Control Rods		
Rod Requirements	*See Table 4-6*	See Table 4-6
Maximum Bank Worth, pcm[2]	*<2000*	~1250
Maximum Ejected Rod Worth	*See 15.0*	See 15.0
Boron Concentrations	***First Cycle***	**Reload Cycle**
Zero Power, K_{eff} = 1.00, Cold, ARO, 1 percent uncertainty included	*1504*	2000
Zero Power, K_{eff} = 1.00, Hot, ARO, 1 percent uncertainty included	*1406*	2100
Design Basis Refueling Boron Concentration	*2000*	2875
Zero Power, K_{eff} = 1.00, Hot, ARO	*1292*	2000
Full Power, No Xenon, K_{eff} = 1.0, Hot, ARO	*1177*	1800
Full Power, Equilibrium Xenon, K_{eff} = 1.0 Hot, ARO	*879*	1330
Reduction with Fuel Burnup, ppm/GWD/MTU [3]		See Figure 4-33

Notes:

1. See Figure 4-72

2. Note: 1 pcm = (percent mille rho) = $10^{-5}\ \Delta\lambda$ where $\Delta\lambda$ is calculated from two statepoint values of K_{eff} by 1n (k_2/K_1).
3. Gigawatt Day (GWD) = 1000 Megawatt Day (1000 MWD).

Table 4-6. 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 (Typical Reload Cycle)**
1. Control requirements					
a. Power Defect, %Δρ	*2.07*[2]		*3.08*[2]		2.85[4]
b. Rod Insertion Allowance, %Δρ	*0.50*		*0.50*		0.35[4]
2. Total Control, %Δρ	*2.57*		*3.58*		3.20
3. Estimated Rod Cluster Control Assembly Worth (53 Rods)	***Unit 1***	***Unit 2***	***Unit 1***	***Unit 2***	**Typical**
a. All full length assemblies inserted, %Δρ	*7.67*	*8.51*	*7.49*	*8.31*	6.77
b. All but one (highest worth) assemblies inserted, %Δρ	*6.48*	*7.19*	*6.33*	*7.02*	5.89
4. Estimated Rod Cluster Control Assembly credit with 10 percent adjustment to accommodate uncertainties (3.b.-10 percent), %Δρ	*5.83*	*6.47*	*5.70*	*6.32*	5.30
5. Shutdown margin available (4-2), %Δρ	*3.26*[3]	*3.90*[3]	*2.12*[3]	*2.74*[3]	2.10[1]

Note:

1. The design basis minimum shutdown is 1.3%.
2. *Includes Void Effects*
3. *The design basis minimum shutdown for Cycle 1 was 1.6%*
4. Includes allowances for transient xenon effects

Table 4-7. UO_2 Benchmark Critical Experiments

UO_2 Benchmark Critical Experiments for CASMO-3, TABLES-3 and SIMULATE-3 Methodology

No.	Ref.	General Description	Enrichment (w/o U^{235})	Reflector	Separating Material	Characterizing Separation (cm)	k_{eff}
2	37	UO_2 Rod Lattice	2.46	1037 ppm Water	-	-	1.0001±0.0005
3	37	UO_2 Rod Lattice	2.46	764 ppm Water	-	1.64	1.0000±0.0006
9	37	UO_2 Rod Lattice	2.46	Water	-	6.54	1.0030±0.0009
10	37	UO_2 Rod Lattice	2.46	143 ppm Water	-	4.91	1.0001±0.0009
11	37	UO_2 Rod Lattice	2.46	514 ppm Water	SS	1.64	1.0000±0.0006
13	37	UO_2 Rod Lattice	2.46	15 ppm Water	1.614% B/A1(1)	1.64	1.0000±0.0010
14	37	UO_2 Rod Lattice	2.46	92 ppm Water	1.257% B/A1(1)	1.64	1.0001±0.0010
15	37	UO_2 Rod Lattice	2.46	395 ppm Water	0.401% B/A1(1)	1.64	0.9998±0.0016
17	37	UO_2 Rod Lattice	2.46	487 ppm Water	0.242% B/A1(1)	1.64	1.0000±0.0010
19	37	UO_2 Rod Lattice	2.46	634 ppm Water	0.100% B/A1(1)	1.64	1.0002±0.0010

UO_2 Benchmark Critical Experiments for SCALE 4.4 Methodology

No.	Ref.	General Description	Enrichment (w/o U^{235})	Poison Material	Poison Thickness (cm)	Critical Separation (cm)		Critical No. of Rods
						X	Y	
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	Mutliple Fuel Clusters	4.31	None	-	2.83	19.81	432
57	60	Multiple Fuel Clusters	4.31	None	-	2.83	13.64	360

UO₂ Benchmark Critical Experiments for SCALE 4.4 Methodology								
No.	**Ref.**	**General Description**	**Enrichment (w/o U^{235})**	**Poison Material**	**Poison Thickness (cm)**	**Critical Separation (cm)**		**Critical No. of Rods**
						X	**Y**	
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
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
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	Multiple Fuel Clusters	4.31	SS-304	.302	2.83	8.10	396
71	60	Mulriple 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

UO_2 Benchmark Critical Experiments for SCALE 4.4 Methodology								
No.	Ref.	General Description	Enrichment (w/o U^{235})	Poison Material	Poison Thickness (cm)	Critical Separation (cm)		Critical No. of Rods
						X	Y	
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/o 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/o U^{235})	Poison Material	Distance from SS plate to Fuel Cluster (cm)	Length by widths 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/o U^{235})	Poison Material	Poison Thickness	Distance from SS plate to Fuel Cluster (cm)	Length by Width of Array	Critcal 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/o U^{235})	Boral Poison Loading (g B/cm^2)	Flux Trap Width (cm)	Flux Trap to Fuel Separation (cms)		Critical No. of Rods
						X	Y	
214	62	Neutron Flux Traps	4.31	0.36	3.73	0.295	0.295	952

No.	Ref.	General Description	Enrichment (w/o U^{235})	Boral Poison Loading (g B/cm^2)	Flux Trap Width (cm)	Flux Trap to Fuel Separation (cms)		Critical No. of Rods
						X	Y	
223	62	Neutron Flux Traps	4.31	0.36	3.73	4.077	4.077	858
224	62	Nuetron 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 4-8. Deleted Per 1996 Update

Table 4-9. Deleted Per 1996 Update

Table 4-10. Deleted Per 1996 Update

Table 4-11. Deleted Per 1996 Update

Table 4-12. 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 4-13. Deleted Per 1998 Update

Table 4-14. Deleted Per 1998 Update

Table 4-15. Deleted Per 1992 Update

Table 4-16. Deleted Per 1992 Update

Table 4-17. Deleted Per 1992 Update

Table 4-18. Deleted Per 1992 Update

Table 4-19. Deleted Per 2000 Update

Table 4-20. Deleted Per 1993 Update

Table 4-21. Void Fractions at Nominal Reactor Conditions With Design Hot Channel Factors

	Average	Maximum
Core	0.0	-
Hot Subchannel	0.3	1.0

Table 4-22. Statistically Combined Uncertainty Factors for Fq, FDeltaH, 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.