REACTOR DESIGN COMPARISON TABLE [1]

Thermal and Hydraulic Design Parameters		Initial Cycle	Typical Cycle Before MUR Power Uprate	Typical Cycle After MUR Power Uprate
1.	Reactor Core Heat Output, MWt	3,391	3,411	3468
2.	Reactor Core Heat Output, 10^6 Btu/hr	11,573.5	11,639	11,833
3.	Heat Generated in Fuel, %	97.4	97.4	97.4
4.	System Pressure, Nominal, psia	2,280	2,280	2,280
5.	System Pressure, Minimum Steady-State, psia	2,250	2,250	2,250
6.	Minimum Departure from Nucleate Boiling Ratio for Design Transients			
	Typical Flow Channel	1.80 (2)	1.69 (3)	1.69 (3)
	Thimble Flow Channel	1.77 (2)	1.61 (3)	1.61 (3)

[1] The fresh fuel assemblies for Cycle 21 and beyond will have **Optimized ZIRLO™** clad fuel rods and ZIRLO® guide thimbles, instrumentation tubes, mid-grids and IFM grids with balanced vanes. The option to remove thimble plugs will exist for Cycle 13 and beyond. This will increase the bypass flow and cause small changes in the core flow rates and temperatures.

[2] These numbers are based on Improved Thermal design Procedure in Reference 2.

[3] These numbers are based on Revised Thermal Design Procedure in Reference 3.

REACTOR DESIGN COMPARISON TABLE [1]

Thermal and Hydraulic Design Parameters		Initial Cycle	Typical Cycle Before MUR Power Uprate	Typical Cycle After MUR Power Uprate
COOLANT FLOW				
7.	Total Thermal Design Flow Rate, 10^6 lb_m/hr	142.7	134.4	134.7
8.	Effective Flow Rate for Heat Transfer, 10^6 lb/hr	136.3	127.5	125.15
9.	Effective Flow Area for Heat Transfer, ft^2	51.1	54.1	54.1
10.	Average Velocity Along Fuel Rods, ft/sec	16.7	14.6	13.5
11.	Average Mass Velocity, 10^6 lb_m/hr-ft^2	2.72	2.36	2.31
COOLANT TEMPERATURE, °F				
12.	Nominal Inlet	541.3	543.4 (4)	540.8 (4)
13.	Average Rise in Vessel	61.8	65.3 (4)	66.4 (4)
14.	Average Rise in Core	63.4	68.4 (4)	71.0 (4)
15.	Average in Core	574.3	579.3 (4)	578.05 (4)

4 Based on thermal design flow

REACTOR DESIGN COMPARISON TABLE [1]

	Thermal and Hydraulic Design Parameters	Initial Cycle	Typical Cycle Before MUR Power Uprate	Typical Cycle After MUR Power Uprate
16.	Average in Vessel	572.2	576.0 (4)	574.0 (4)
	HEAT TRANSFER			
17.	Active Heat Transfer, Surface Area, ft^2	59,700	57,505	57,505
18.	Average Heat Flux, Btu/hr-ft^2	188,700	197,180	200,477
19.	Maximum Heat Flux for Normal Operation, Btu/hr-ft^2	437,800(5)	460,420	468,114
20.	Average Thermal Output, kW/ft	5.41	5.45	5.54
21.	Maximum Thermal Output for Normal Operation, kW/ft	12.6 (6)	12.7	12.9
22.	Maximum Thermal Output at Maximum Overpower Trip Point (118% power), kW/ft	18.0 (7)	22.5	22.5
23.	Heat Flux Hot Channel Factor, F_Q	2.32 (8)	2.335	2.335

5 The value of 437,800 Btu/hr-ft^2 is associated with a Cycle 1 value of F_Q of 2.32.

6 This value of 12.6 kW/ft is associated with a Cycle 1 value of F_Q of 2.32.

7 See Section 3.3.2.2.6.

8 The value of F_Q = 2.32 was the value of F_Q for normal operation reported in the original FSAR.

REACTOR DESIGN COMPARISON TABLE [1]

Thermal and Hydraulic Design Parameters		Initial Cycle	Typical Cycle Before MUR Power Uprate	Typical Cycle After MUR Power Uprate
24.	Peak Fuel Central Temperature at 100% Power, °F	< 4700	< 4700	< 4700
25.	Peak Fuel Central Temperature at Maximum Thermal Output for Maximum Overpower Trip Point, °F	< 4700	< 4700	< 4700
FUEL ASSEMBLIES				
26.	Design	RCC Canless	RCC Canless	RCC Canless
27.	Number of Fuel Assemblies	193	193	193
28.	UO_2 Rods per Assembly	264	264	264
29.	Rod Pitch, in	0.496	0.496	0.496
30.	Overall Dimensions, in	8.426 x 8.426	8.426 x 8.426	8.426 x 8.426
31.	Fuel Weight (as UO_2), lb	222,739	204,200	204,200
32.	Zircaloy Weight, lb	50,913	45, 914	45, 914

REACTOR DESIGN COMPARISON TABLE [1]

	Thermal and Hydraulic Design Parameters	Initial Cycle	Typical Cycle Before MUR Power Uprate	Typical Cycle After MUR Power Uprate
33.	Number of Grids per Assembly	8 – Type R	6 – Flow mixer grids 2 – Non-flow mixer grids 3 – IFM grids 1 – Protective Grid	6 – Flow mixer grids 2 – Non-flow mixer grids 3 – IFM grids 1 – Protective Grid
34.	Loading Techniques	Out – In Checkerboard	3 – Region Low Leakage	3 – Region Low Leakage
	FUEL RODS			
35.	Number	50,952	50,952	50,952
36.	Outside Diameter, in	0.374	0.360	0.360
37.	Diametral Gap, in	0.0065	0.0062	0.0062
38.	Clad Thickness, in	0.0225	0.0225	0.0225
39.	Clad Material	Zircaloy-4	Zircaloy-4	**Optimized ZIRLO™** starting with Cycle 21

REACTOR DESIGN COMPARISON TABLE [1]

Thermal and Hydraulic Design Parameters		Initial Cycle	Typical Cycle Before MUR Power Uprate	Typical Cycle After MUR Power Uprate
FUEL PELLETS				
40.	Material	UO_2 Sintered	UO_2 Sintered 0.370 Enriched	UO_2 Sintered 0.370 Enriched
41.	Density (% of Theoretical)	95	95.5	95.5
42.	Diameter, in	0.3225	0.3088	0.3088
43.	Length, in	0.530 (4)	0.462 Axial Blankets	0.462 Axial Blankets
ROD CLUSTER CONTROL ASSEMBLIES				
44.	Neutron Absorber, Full/Part Length (9)	Ag-In-Cd	Ag-In-Cd	Ag-In-Cd
45.	Cladding Material	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked
46.	Clad Thickness, in	0.0185	0.0185	0.0185
47.	Number of Clusters, Full and Part Length (9)	53/0	53/0	53/0

[9] Part Length CRDMs were eliminated.

REACTOR DESIGN COMPARISON TABLE [1]

	Thermal and Hydraulic Design Parameters	Initial Cycle	Typical Cycle Before MUR Power Uprate	Typical Cycle After MUR Power Uprate
48.	Number of Absorber Rods per Cluster	24	24	24
	CORE STRUCTURE			
49.	Core Barrel, I.D./O.D., in	148.0/152.5	148.0/152.5	148.0/152.5
50.	Thermal Shield, I.D./O.D., in	158.5/164.0	158.5/164.0	158.5/164.0
	STRUCTURE CHARACTERISTICS			
51.	Core Diameter, in (Equivalent)	132.7	132.7	132.7
52.	Core Height, in (Active Fuel)	144.0	144.0	144.0
	REFLECTOR THICKNESS AND COMPOSITION			
53.	Top - Water plus Steel, in	10	10	10
54.	Bottom - Water plus Steel, in	10	10	10
55.	Side - Water plus Steel, in	15	15	15
56.	H_2O/U Molecular Ratio Core, Lattice (Cold)	2.41	2.73	2.73

REACTOR DESIGN COMPARISON TABLE [1]

Thermal and Hydraulic Design Parameters		Initial Cycle	Typical Cycle Before MUR Power Uprate	Typical Cycle After MUR Power Uprate
FEED ENRICHMENT, W/O				
57.	Region 1	2.10	4.0/2.6 [10]	4.0/2.6 [10]
58.	Region 2	2.60	4.0/2.6 [10]	4.0/2.6 [10]
59.	Region 3	3.10	4.0/2.6 [10]	4.0/2.6 [10]

[10] Reload enrichments are cycle-specific, 2.6 w/o value corresponds to the axial blanket.

ANALYTIC TECHNIQUES IN CORE DESIGN

Analysis	**Technique**	**Computer Code**	**Section Referenced**
Mechanical Design of Core Internals			
Loads, Deflections, and Stress Analysis	Static and Dynamic Modeling	Blowdown code, FORCE, Finite element structural analysis code, and others	14.3.3
Fuel Rod Design			
Fuel Performance Characteristics (temperature, internal pressure, clad stress, etc.)	Semi-empirical thermal model of fuel rod with consideration of fuel density changes, heat transfer, fission gas release, etc.	Westinghouse fuel rod design model	3.2.1.3.1 3.3.3.1 3.4.2.2 3.4.3.4.2
Nuclear Design			
1. Cross Sections and Group Constants	Microscopic data Macroscopic constants for homogenized core regions	Modified ENDF/B-V or ENDF/B-VI library PHOENIX-P	3.3.3.2 3.3.3.2
	Group constants for control rods with self-shielding	PHOENIX-P	3.3.3.2
Nuclear Design (Continued)			
2. X-Y Power Distributions, Fuel Depletion, Critical Boron Concentrations, X-Y Xenon Distributions,	3D, 2-Group Nodal Expansion Method	ANC	3.3.3.3

ANALYTIC TECHNIQUES IN CORE DESIGN

Analysis	Technique	Computer Code	Section Referenced
Reactivity Coefficients			
3. Axial Power Distributions, Control Rod Worths, and Axial Xenon Distribution	1-D, 2-Group Diffusion Theory	APOLLO	3.3.3.3
4. Fuel Rod Power	Integral Transport Theory	LASER	3.3.3.1
Effective Resonance Temperature	Monte Carlo Weighting Function	REPAD	
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	THINC-IV	3.4.3.4.1
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 solution progresses from core-wide to hot assembly to hot channel	THINC-I (THINC-III)	3.4.3.4.1

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.	Temperature 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 (operating basis earthquake and design basis earthquake)
16.	Blowdown Forces (due to cold and hot leg break)

Maximum Deflections Allowed For Reactor Internal Support Structure

Component	Allowable Deflections (in)	No Loss-of-Function Deflections (in)
Upper Barrel Radial inward Radial outward	 4.1 1.0	 8.2 1.0
Upper Package	0.10	0.15
Rod Cluster Guide Tubes	1.00	1.75

Reactor Core Description

Active Core	
Equivalent Diameter, in	132.7
Active Fuel Height, First Core, in	144.0
Height-to-Diameter Ratio	1.09
Total Cross Section Area, ft^2	96.06
H_2O/U Molecular Ratio, lattice (Cold)	2.73
Reflector Thickness And Composition	
Top - Water plus Steel, in	10
Bottom - Water plus Steel, in	10
Side - Water plus Steel, in	15
Fuel Assemblies	
Number	193
Rod Array	17 x 17
Rods per Assembly	264
Rod Pitch, in	0.496
Overall Transverse Dimensions, in	8.426 x 8.426
Fuel Weight (as UO_2), lb - per assembly	1058
Zircaloy Weight, lb - per assembly	238
Number of Grids per Assembly	2-R
	6-Z
	3-IFM
	1-P

Reactor Core Description

Composition of Grids	R-Inconel 718
	Zircaloy 4/ZIRLO™
	IFM - Zircaloy 4/ZIRLO™
	P-Debris Resistant- Inconel 718
Weight of Grids (Effective in Core), lb - per assembly	20.10
Number of Guide Thimbles per Assembly	24
Composition of Guide Thimbles	Zircaloy 4/ZIRLO™
Diameter of Guide Thimbles (upper part), in	0.442 I.D. x 0.474 O.D.
Diameter of Guide Thimbles (lower part), in	0.397 I.D. x 0.429 O.D.
Diameter of Instrument Guide Thimbles, in	0.440 I.D. x 0.474 O.D.
Fuel Rods	
Number	50,952
Outside Diameter, in	0.360
Diameter Gap, in	0.0062
Clad Thickness, in	0.0225
Clad Material	Zircaloy-4 / ZIRLO / **Optimized ZIRLO™**
Fuel Pellets	
Material	UO_2 Sintered
Density (percent of Theoretical)	Approx. 95.5
Maximum Fuel Enrichments w/o	4.95
Diameter, in	0.3088
Length, in	0.370 Enriched

Reactor Core Description

	0.462 Axial Blanket
Mass of UO_2 per Foot of Fuel Rod, lb/ft	0.336 [1]
Rod Cluster Control Assemblies	
Neutron Absorber	Ag-In-Cd
Composition	80%, 15%, 5%
Diameter, in	0.341
Density, lb/in^3	0.367
Cladding Material	Type 304, Cold Worked Stainless Steel
Clad Thickness, in	0.0185
Number of Clusters	
Full Length	53
Number of Absorber Rods per cluster	24
Full Length Assembly Weight (dry), lb	149
Excess Reactivity	
Maximum Fuel Assembly k (Cold, Clean, Unborated Water)	1.476 [2]
Maximum Core Reactivity (Cold, Zero Power Beginning of Cycle)	1.224 [2]

[1] Based on fuel at 95.5% theoretical density

[2] Typical values

Reactor Core Description

Integral Fuel Burnable Absorber	
Number	~8640 [2]
Material	ZrB_2
Coating Thickness, mil	~0.2
Boron 10 Loading, mg/in	2.25

Nuclear Design Parameters
(Best-Estimate Values Are Representative Of A Typical Cycle)

Core Average Linear Power, kW/ft, including densification effects[1]			5.45 / 5.54
Total Heat Flux Hot Channel Factor, F_Q			2.335
Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$			1.61 [1+0.3(1-P)]
Reactivity Coefficients[2]		**Design Limits**	**Best Estimate**
Doppler-only Power, Coefficients, pcm/°F			
	Upper Curve	-19.4 to -12.224	-12.4 to -7.9
	Lower Curve	-9.55 to -5.818	-10.9 to -7.5
Doppler Temperature Coefficient, pcm/°F		-3.20 to -0.91	-1.9 to -1.3
Moderator Temperature Coefficient, pcm/°F		+5 to -38[3]	< + 5.0[4]to -29.54[3]
Boron Coefficient, pcm/ppm			-10.9 to -7.6
Rodded Moderator Density, pcm/gm/cc		0.54 x 10 E + 05	0.40 x 10 E + 05
Radial Assembly Peaking Factor		**Design Limits**	**Best Estimate**
Radial Assembly Peaking Factor[5]			
	Unrodded		1.36 to 1.49
	D bank		1.51 to 1.58
	D + C		1.61 to 1.70
Boron Concentrations (ppm)		**Design Limits**	**Best Estimate**
Zero Power, K_{eff} = 0.99, Cold, Rod Cluster Control Assemblies Out			1804
Zero Power, K_{eff} = 0.99, Hot, Rod Cluster Control Assemblies Out			1930
Design Basis Refueling Boron Concentration		2400	1855

1 Before and After MUR power uprate values listed.
2 Uncertainties are referenced in Section 3.3.3.3.
3 Design limit dependent on vessel average moderator temperature. Value reported is for Cycle 14 temperature of 574.0 °F.
4 Administrative rod withdrawal limits are required if an MTC violation is observed during startup physics testing, as specified by an action statement in Technical Specification 3.1.3.A.1.
5 Typical values.

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Nuclear Design Parameters
(Best-Estimate Values Are Representative Of A Typical Cycle)

Zero Power, K_{eff} = 0.95, Cold, Rod Cluster Control Assemblies In			1754
Zero Power, K_{eff} = 1.00, Hot, Rod Cluster Control Assemblies Out			1795
Full Power, No Xenon, K_{eff} = 1.0, Hot, Rod Cluster Control Assemblies Out			1648
Full Power, Equilibrium Xenon, K_{eff} Hot, Rod Cluster Control Assemblies Out			1316
Reduction with Fuel Burnup Reload Cycle, ppm/GWD/MTU[6]			~84
Delayed Neutron Fraction and Lifetime		**Design Limits**	**Best Estimate**
β	eff BOL, (EOL	0.0075, (0.0040)	0.0062, (0.0050)
ℓ	BOL, (EOL) μsec[5]		20.1, (22.3)
Control Rods		**Best Estimate**	**Best Estimate**
Rod Requirements			See Table 3.3-3
Maximum Bank Worth, pcm			1380
Maximum Ejected Rod Worth			See Chapter 14
Bank Worth, pcm[7]		BOL, Xe free HZP	EOL, Xe free HZP
	Bank D	1135	1380
	Bank C	966	1222
	Bank B	851	1259
	Bank A	572	617

6 Gigawatt Day (GWD) = 1000 Megawatt Day (1000 MWD).
7 Note: For two statepoint values of k_{eff}, k_1 and k_2, the reactivity change in pcm (percent milli) is given by In (k_2/k_1) x10^5.

SHUTDOWN REQUIREMENTS AND MARGINS

	Typical Values	
	BOC	EOC
Control Rod Worth (pcm)		
Available Rod Worth Less Worst Stuck Rod	4856	5879
(A) less 10%	4371	5291
Control Rod Requirements (pcm)		
Reactivity Defects (Doppler, Tavg, RIA, Redistribution)	1431	2764
Void Allowance	50	50
(B) Total Requirements	1481	2814
(C) Available Shutdown Margin [(A) – (B)] (pcm)	2890	2477
(D) Required Shutdown Margin (pcm)	1300	1300
Excess Shutdown Margin [(C) – (D)] (pcm)	1590	1177

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AXIAL STABILITY INDEX PRESSURIZED WATER REACTOR CORE WITH A 12 FOOT HEIGHT

			Stability Index (hr^{-1})		
Burnup (MWD/MTU)	**F_Z**	**C_B (ppm)**	**Exp**	**Calc**	**Difference (Exp-Calc)**
1550	1.34	1065	-0.041	-0.032	-0.009
7700	1.27	700	-0.014	-0.006	-0.008
		Difference:	+0.027	+0.026	

COMPARISON OF MEASURED AND CALCULATED DOPPLER DEFECTS

Plant	Fuel Type	Core Burnup (MWD/MTU)	Measured (pcm)[1]	Calculated (pcm)
1	Air-filled	1800	1700	1710
2	Air-filled	7700	1300	1440
3	Air and helium-filled	8460	1200	1210

[1] 1 pcm = 10^{-5} $\Delta\rho$

Cook Nuclear Plant Unit 2 Reactor Design Comparison Table [1]

Thermal and Hydraulic Design Parameters	Initial Cycle	Typical Cycle Before MUR Power Uprate	Typical Cycle After MUR Power Uprate
Reactor Core Heat Output, MWt	3391	3411	3468
Reactor Core Heat Out, 10^6 BTU/hr	11,573.5	11,639	11,833
Heat Generated in Fuel, %	97.4	97.4	97.4
System Pressure, Nominal, psia [2]	2280	2280	2280
System Pressure, Minimum			
Steady-State, psia	2250	2250	2250
Minimum DNBR at Nominal Conditions			
Typical Flow Channel	3.03 [3]	2.42	2.66
Thimble (Cold Wall) Flow Channel	2.70 [3]	2.28	2.49
Design DNBR for Design Transients			
Typical Flow Channel	1.80 [4]	1.69 [5]	1.69 [5]
Thimble Flow Channel	1.77 [4]	1.61 [5]	1.61 [5]

1 The fresh fuel assemblies for Cycle 21 and beyond will have **Optimized ZIRLO™** clad fuel rods and ZIRLO® guide thimbles, instrumentation tubes, mid-grids and IFM grids with balanced vanes. The option to remove thimble plugs will exist for Cycle 13 and beyond. This will increase the bypass flow and cause small changes in the core flow rates, temperatures and pressure drops.

2 Pressure in the core. See Reference (1).

3 Based on Improved Thermal Design Procedure, Reference (84).

4 Including 31.1 percent rod bow penalty.

5 Value used in DNB analyses (RTDP Transients).

Cook Nuclear Plant Unit 2 Reactor Design Comparison Table [1]

Thermal and Hydraulic Design Parameters	Initial Cycle	Typical Cycle Before MUR Power Uprate	Typical Cycle After MUR Power Uprate
DNB Correlation	WRB-1	WRB-2	WRB-2
Coolant Flow [6]			
Total Thermal Design Flow Rate, 10^6 lb_m/hr	142.7	134.3	134.7
Best Estimate Flow, 10^6 lb_m/hr	148.4	145.2	145.2
Mechanical Design Flow, 10^6 lb_m/hr	154.3	154.5	154.5
Minimum Effective Flow Rate for Heat Transfer, 10^6 lb_m/hr	136.3	127.4	125.15
Effective Flow Area for Heat Transfer, ft^2	51.1	54.1	54.1
Average Velocity Along Fuel Rods, ft/sec	16.7	14.6	13.5
Average Mass Velocity, 10^6 lb_m/hr	2.72	2.35	2.31
Coolant Temperature [6]			
Nominal Inlet, °F	541.3	543.4	540.8
Average Rise in Vessel, °F	61.8	65.3	66.4
Average Rise in Core, °F	63.4	68.4	71.0
Average in Core, °F	574.3	579.3	578.05

6 Based on Thermal Design Flow.

Cook Nuclear Plant Unit 2 Reactor Design Comparison Table [1]

Thermal and Hydraulic Design Parameters	Initial Cycle	Typical Cycle Before MUR Power Uprate	Typical Cycle After MUR Power Uprate
Average in Vessel, °F	572.2 [7]	576.0	574.0
Heat Transfer			
Active Heat Transfer, Surface Area, ft^2	59,700	57,505	57,505
Average Heat Flux, BTU/hr-ft^2	188,700	197,180	200,477
Maximum Heat Flux for Normal Operation, BTU/hr-ft^2	437,800 [8]	460,420 [9]	468,114 [9]
Average Linear Power, kW/ft	5.41	5.45	5.54
Peak Linear Power for Normal Operation, kW/ft	12.6 [8]	12.7 [9]	12.9 [9]
Peak Linear Power Resulting from Overpower Transients/Operator Errors, (assuming a maximum overpower of 118%), kW/ft	18.0 [10]	22.5	22.0
Peak Linear Power for Prevention of Centerline Melt, kW/ft [11]	18.0	>22.5	>22.5
Fuel Central Temperature			
Peak at Peak Linear Power for Prevention of Centerline Melt, °F	4700	4700	4700

7 The vessel average temperature was increased to 573.8°F as per amendment 19 of May 13, 1980.
8 This limit is associated with the value of FQ = 2.32.
9 This limit is associated with the value of FQ = 2.335.
10 See Section 3.3.2.2.6.
11 See Section 3.4.2.2.6.

Cook Nuclear Plant Unit 2 Reactor Design Comparison Table [1]

Thermal and Hydraulic Design Parameters	Initial Cycle	Typical Cycle Before MUR Power Uprate	Typical Cycle After MUR Power Uprate
Pressure Drop [12]			
Across Core, psi	23.3 ± 2.3	27.0 ± 2.7	27.0 ± 2.7
Across Vessel, including nozzles, psi	43.2 ± 4.3	50.1 ± 5.0	50.1 ± 5.0

12 Based on Best Estimate Flow as discussed in 3.4.2.6.

VOID FRACTIONS AT NOMINAL REACTOR CONDITIONS [1]
WITH DESIGN HOT CHANNEL FACTORS

	Average	Maximum
Core	0.2%	-------
Hot Subchannel	0.9%	2.1%

[1] Based upon Minimum Measured Flow.

SYSTEM DESIGN AND OPERATING PARAMETERS
(TYPICAL CYCLES BEFORE AND AFTER MUR POWER UPRATE)[1]

	At 70°	At Hot[2]
Approximate total RCS volume (including pressurizer and surge line), with 0% steam generator tube plugging. (ft.^{3})	12,470	12,845
Approximate system liquid volume, (including pressurizer water) at maximum guaranteed power with 0% steam generator tube plugging. (ft.^{3})		12,019 [3]

SYSTEM THERMAL AND HYDRAULIC DATA
(BASED ON THERMAL DESIGN FLOW)

	Typical Cycle Before MUR Power Uprate	Typical Cycle After MUR Power Uprate[4]
NSSS Power, MWt	3423	3480
Reactor Power, MWt	3411	3468
Thermal Design Flows, gpm		
Active Loop	88,500	88,500
Reactor	354,000	354,000
Total Reactor Flow, 10^6lb/hr	134.4	134.4
Temperatures, °F		
Reactor Vessel Outlet	606.4	611.1
Reactor Vessel Inlet	541.2	545.1
Steam Generator Outlet	541.0	544.8
Steam Generator Steam	521.1	524.0
Feedwater	431.0	444.1
Steam Pressure, psia	820.0	840.9
Total Steam Flow, 10^6 lb/hr	14.78	15.37

1 The option to remove thimble plugs will exist for Cycle 13 and beyond. This will increase bypass flow and cause small changes in the core flow rates and temperatures.
2 This includes a 3% volume increase (1.3% for thermal expansion and 1.7% for pipe connections to the reactor coolant loops, volume in the rod drive mechanisms and calculation inaccuracies). Refer to Westinghouse letters AEP-97-151, AEP-98-078, AEP-98-082, AEP-98-161, and the Westinghouse IMP database SEC-SAI-4824-CO.
3 Total RCS Volume (12,845 ft.^{3}) - Pressurizer steam volume at full power (826 ft. ^{3}).
4 Based upon reactor loop average temperature of 578.1°F.

SYSTEM FLOW SUMMARY

Flows, gpm	Thermal Design [5]	Minimum Measured[6]	Best Estimate	Mechanical Design
4 Pumps Running, each loop	88,500	91,600	95,500	101,600

[5] Fixed value analyses (non-RTDP transients).
[6] DNB analyses values (RTDP transients).

COMPARISON OF THINC-IV AND THINC-I PREDICTIONS WITH DATA FROM REPRESENTATIVE WESTINGHOUSE TWO AND THREE LOOP REACTORS

Reactor	Power (MWt)	% Full Power	Measured Inlet Temp (ºF)	rms(ºF) THINC-I	(ºF) THINC-IV	Improvement (ºF) for THINC-IV over THINC-I
Ginna	847	65.1	543.7	1.97	1.83	0.14
	854	65.7	544.9	1.56	1.46	0.10
	857	65.9	543.9	1.97	1.82	0.15
	947	72.9	543.8	1.92	1.74	0.18
	961	74.0	543.7	1.97	1.79	0.18
	1091	83.9	542.5	1.73	1.54	0.19
	1268	97.5	542.0	2.35	2.11	0.24
	1284	98.8	540.2	2.69	2.47	0.22
	1284	98.9	541.0	2.42	2.17	0.25
	1287	99.0	544.4	2.26	1.97	0.29
	1294	99.5	540.8	2.20	1.91	0.29
	1295	99.6	542.0	2.10	1.83	0.27
Robinson	1427.0	65.1	548.0	1.85	1.88	0.03
	1422.6	64.9	549.4	1.39	1.39	0.00

COMPARISON OF THINC-IV AND THINC-I PREDICTIONS WITH DATA FROM REPRESENTATIVE WESTINGHOUSE TWO AND THREE LOOP REACTORS

Reactor	Power (MWt)	% Full Power	Measured Inlet Temp (ºF)	rms(ºF) THINC-I	(ºF) THINC-IV	Improvement (ºF) for THINC-IV over THINC-I
	1529.0	88.0	550.0	2.35	2.34	0.01
	2207.3	100.7	534.0	2.41	2.41	0.00
	2213.9	101.0	533.8	2.52	2.44	0.08