Appendix 5A. Tables

Table 5-1 System Design and Operating Parameters

	Unit 1	Unit 2
Plant Design Life, years	40	40
Nominal Operating Pressure, psig	2,235	2,235
Total System Volume Including Pressurizer and Surge Line, ft^3	13,084	11,861
System Liquid Volume, Including Pressurizer Water Level, ft^3 (55% full – Unit 1) (55 % full – Unit 2)	12,429	11,206
Pressurizer Spray Rate, maximum gpm	900	900
Pressurizer Heater Capacity, kw	1,662	1,662
Thermal Design Parameters		
NSSS Power, %	100	100
NSSS Power, MWt	3,488	3,430
NSSS Power, 10^6 BTU/hr	11,956	11,714
Analyzed Power, MWt	3,479	3,479
Analyzed Power, 10^6BTU/hr	11,881	11,881
Licensed Reactor Power, MWt	3,469	3,411
Licensed Reactor Power, 10^6 BTU/hr	11,837	11,639
Thermal Design Flow, Loop gpm	95,500	94,250
Reactor 10^6 lb/hr	145.3	143.4
Reactor Coolant Pressure, psia	2,250	2,250
Reactor Coolant Temperature, °F	Unit 1	Unit 2
Core outlet	617.4	620.8
Vessel outlet	614.9	616.7
Core average	585.3	589.6
Vessel average	585.1	587.5
Vessel/core inlet	555.3	558.3
Steam Generator outlet	555.3	558
Steam Generator		
Steam Temperature °F	549	541
Steam Pressure, psia	1021	970
Steam Flow, 10^6 lb/hr total	15.5	15.12

	Unit 1	**Unit 2**
Feed Temperature, °F	442	440
Moisture, % max	0.25	
Zero Load Temperature °F	557	
Hydraulic Design Parameters		
Pump Design Point, Flow (gpm)	101,000	101,000
Head (ft)	279	286
Mechanical Design Flow, gpm	105,000	105,000
System Pressure Drops @ Best Estimate Flow		
	Unit 1	Unit 2
Reactor Vessel ΔP, psi	45.1	45.1
Steam Generator ΔP, psi	33.0	38.3
Hot Let Piping ΔP, psi	1.3	1.3
Pump Suction Piping ΔP, psi	3.3	3.3
Cold Leg Piping ΔP, psi	3.3[(1)]	3.3[(1)]
Pump Head, feet	279	286

Note:

1. Includes pump weir ΔP of 2.o psi

Table 5-2. Applicable Code Addenda for RCS Components

Reactor Vessel	ASME III, 1971 Edition through Winter '71 - Unit 1 ASME III, 1971 Edition through Winter '72 - Unit 2
Steam Generator	ASME III, 1986 Edition no addenda - Unit 1 ASME III, 1971 Edition through Winter '72 - Unit 2
Pressurizer	ASME III, 1971 Edition through Winter '72
CRDM Housing	ASME III, 1974 Edition through Summer '74
CRDM Head Adapter	ASME III, 1971 Edition through Winter '72
Reactor Coolant Pump	ASME III, 1971 Edition through Summer '73
Reactor Coolant Pipe	ASME III, 1974 Edition
Surge Lines	ASME III, 1974 Edition
Valves	
Pressurizer Safety Dresser	ASME III, 1974 Edition through Summer '74
Gate, Globe and Check	
Westinghouse Valve Division	ASME III, 1974 Edition through Summer '74
Borg-Warner, NVD	ASME III, 1971 Edition through Summer '73
Packless Globe and Check	
Kerotest	ASME III, 1974 Edition through Summer '73
CC1	ASME III, 1974 Edition through Winter '74
Rockwell	ASME III, 1977 Edition through Summer '78
Walworth	ASME III, 1971 Edition through Summer '73
Fisher	ASME III, 1974 Edition through Summer '74 Addendum

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

Table 5-3. ***Code Cases Applicable for Operation, Maintenance, and Testing Activities***

Number	***Description***
N-92(1698)	*Waiver of Ultrasonic Transfer Method*
N-98(1705-1)	*Calibration Block Tolerance*
N-112(1730)	*Acceptance Standards for Components*
N-113(1731)	*Basic Calibration Blocks for Ultrasonic Examination of Welds 10 to 14 In. Thick*
N-118(1738)	*Acceptance Standards - Surface Indications Cladding*
N-198-1	*Exemption from Examination for ASME Class 2 Piping Located at Containment Penetrations*
N-209	*Conditional acceptance of Identifiable, Isolated, or Random Rounded Indications*
N-210	*Exemptions to Hydrostatic Test Repairs*
N-211	*Recalibration of Ultrasonic Equipment Upon Change of Personnel*
N-242	*Steam Generator Power Operated Relief Valves*
	1/4" Plugs
	1/4" 3000# THR'D Half Couplings
	1/4" 3000# S. W. 90° Elbows
	1/4" 3000# S. W. Full Couplings
	3/4" 3000# THR'D Caps
	1" 6000# S. W. Half Couplings
	1" 9000# S. W. Full Couplings
	2" 6000# S. W. Special Weld Boss
	2"x3/4" 6000# S. W. Special Reducer
N-416-1	*Alternatative pressure test requirements for welded repairs*

Table 5-4. ASME Code Cases Used For Catawba Units 1 & 2 Class 1 Components

Equipment	Unit 1	Unit 2
Steam Generators	N-411-1	1355
	N-20-3	1493
	N-71-15	1484
	N-474-1 2142 2143	1528
		1498
Pressurizer	1528	----
RC Pipe/Fitting Fab.	1423-2	1423-2

Table 5-5. Typical Plant Thermal-Hydraulic Parameters. HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

	Units	**2-Loop**	**3-Loop**	**4-Loop**	**Catawba**
Heat Output, Core	MWt	1,780	2,652	3,411	3,411
System Pressure	psia	2,250	2,250	2,250	2,250
Coolant Flow	gpm	178,000	265,500	354,000	397,200 (Unit 1) 387,600 (Unit 2)
Average Core Mass Velocity	10^6 lb/hr-ft^2	2.42	2.33	2.50	2.62
Inlet Temperature	°F	545	544	552.5	556.3 (558.3, Unit 2)
Core Average T_{mod}	°F	581	580	588	586.8 (589.6, Unit 2)
Core Length	Ft	12	12	12	12
Average Power Density	kw/l	102	100	104	104
Maximum Fuel Temperature	°F	<4100	<4200	<4200	<4200
Fuel Loading	kg/l	2.7	2.6	2.6	2.6
Pressurizer Volume	Ft^3	1000	1400	1800	1800
Pressurizer Volume Ratioed to Primary System Volume		0.157	0.148	0.148	0.144
Peak Surge Rate for Pressurizer Safety Valve Sizing Transient	Ft^3/sec	21.8	33.2	41.0	34.74
Pressurizer Safety Valve Flow at 2500 psia - +3% Accumulation	Ft^3/sec	26.1	36.1	43.3	43.225
Ratio of Safety Valve Flow to Peak Surge Rate		1.197	1.087	1.056	1.244
Full Power Steam Flow per Loop	lb/sec	1078	1076	1038	1056.5
Nominal Shell-side Steam Generator Water Mass per Loop	lb	100,300	106,000	106,000	122,600 (Unit 1) 103,370 (Unit 2)

Table 5-6. Class 1 Primary Components Material Specifications

Reactor Vessel Components (Unit 1)	
Head Plates	SA-533, Gr. B, Class 1 (vacuum treated)
Shell, Flange and Nozzle Forgings, Nozzle Safe Ends	SA-508, Class 2, SA-182, Type 308L (Weld Metal Buildup)
CRDM and/or ECCS Appurtenances, Upper Head	SB-166 or 167 and SA-182, Type F304
Instrumentation Tube Appurtenances, Lower Head	SB-166
Closure Studs, Nuts, Washers Inserts and Adaptors	SA-540, Gr. B-24
Core Support Pads	SB-166 with Carbon less than 0.10%
Monitor Tubes and Vent Pipe	SA-312, Type 316 or SB-166
Vessel Supports, Seal Ledge and Head Lifting Lugs	SA-516, Gr. 70 Quenched and Tempered or SA-533, Gr. B, Class 1 or SA-509 Class 2 (vessel supports may be of weld metal build up of equivalent strength)
Cladding and Buttering	Stainless steel weld metal analysis A-7 and Ni-Cr-Fe weld metal F-Number
Reactor Vessel Components (Unit 2)	
Shell and Head Plates (other than core region)	SA-533, Gr. A, B or C, Class 1 or 2 (vacuum treated)
Shell Plates (core region)	SA-533, Gr. A or B, Class 1 (vacuum treated)
Shell, Flange and Nozzle Forgings, Nozzle Safe Ends	SA-508, Class 2 or 3, SA-182, Type F304 or F316
CRDM and/or ECCS Appurtenances, Upper Head	SB-167 and SA-182, Type F304
Instrumentation Tube Appurtenances, Lower Head	SB-166 or 167 and SA-182, Type F304, F304L or F316
Closure Studs	SA-540, Class 3, Gr. B-24
Nuts and Washers	SA-540, Class 3, Gr. B-23
Core Support Pads	SB-166 with carbon less than 0.15% (with an aim of less than 0.10%)
Monitor Tubes and Vent Pipe	SA-312 or 376, Type 316 or SB-167
Vessel Supports, Seal Ledge and Head Lifting Lugs	SA-516, Gr. 70 quenched and tempered or SA-533, Gr. A, B or C, Class 1 or 2 (vessel supports may be of weld metal build up of equivalent strength)

Cladding and Buttering		Stainless steel weld metal with a corrosion resistance equal to or better than Type 304 Acceptable Values: Cr - 18% minimum Ni - 8% minimum C -.08% maximum[1] Ferrite Content - 5-15% Ni-Cr-Fe alloy Fe content shall be 15%maximum.

Steam Generator Components (Units 1 & 2)		
Pressure Plates		SA533 GR A, B or C, Class 1 or 2
Pressure Forgings (including nozzles and tubesheet)		SA508 Class 2, 2a or 3
Nozzle Safe Ends	Unit 1	SA 336-F316N/316LN
	Unit 2	N/A
Channel Heads	Unit 1	SA 508 Class 3
	Unit 2	SA216 Grade WCC
Tubes	Unit 1	SB163 Alloy 690, Code Case N-20-3
	Unit 2	SB163 Ni-Cr-Fe, Annealed
Cladding and Buttering	Unit 1	SFA 5.9 ER 309L/ER 308L
	Unit 2	SFA 5.9 ER 309L - SS
Closure Bolting		SA193 Gr B-7
Pressurizer Components (Units 1 & 2)		
Pressure Plates		SA533 Gr A, Class 2
Pressure Forgings		SA508 Class 2
Nozzle Safe Ends		SA182 Type 316L
Cladding and Buttering		Stainless Steel Weld Metal Analysis A-8 and Ni-Cr-Fe Weld Metal F-Number 43
Closure Bolting		SA193 Gr B-7/SA 194 Gr 7
Reactor Coolant Pump (Units 1 & 2)		

Pressure Forgings	SA182 F 304, F 316, F-347 or F 348
Pressure Casting	SA351 Gr CF8, CF8A or CF8M
Tube & Pipe	SA213, SA376 or SA312 - Seamless Type 304 or 316
Pressure Plates	SA240 Type 304 or 316
Bar Material	SA479 Type 304 or 316
Closure Bolting	SA193, SA320, SA540, SA453, Gr 660
Flywheel	SA533 Gr B, Class 1
Reactor Coolant Piping (Units 1 & 2)	
Reactor Coolant Pipe	SA351 Gr CF8A centrifugal casting
Reactor Coolant Fittings	SA351 Gr CF8A
Branch Nozzles	SA182 Code Case 1423-2 Gr 304N
Surge Line	SA 376 Gr 304
Auxiliary Piping 1/2" through 12" and wall schedules 40S through 80S (ahead of second isolation valve)	ANSI B36.19
All other Auxiliary piping (ahead of second isolation valve)	ANSI B36.10
Socket weld Fittings	ANSI B16.11
Piping Flanges	ANSI B16.5
Control Rod Drive Mechanism (Units 1 & 2)	
Latch Housing	SA182 Gr F304 or SA351 Gr CF8
Rod Travel Housing	SA182 Gr F304 or SA336 Gr F8
Cap	SA479 Type 304
Welding Materials	Stainless Steel Weld Metal Analysis A-8

Note:

1. For multilayer cladding where the first layer is Type 309 material, the carbon content of the first layer shall be 0.1% maximum.

Table 5-7. Class 1 and 2 Auxiliary Components Material Specifications

Valves		
Bodies	SA182 Type F316 or SA351 Gr CF8 or CF8M	
Bonnets	SA182 Type F316 or SA351 Gr CF8 or CF8M	
Discs	SA182 Type F316 or SA564 Gr 630	
Pressure Retaining Bolting	SA453 Gr 660	
Pressure Retaining Nuts	SA453 Gr 660 or SA194 Gr 6	
Auxiliary Heat Exchangers		
Tube Sheets	SA240 Type 304, SA 182 Gr F304, SA516 Gr 70 with Stainless Steel Cladding (Analysis A-8)	
Tubes	SA213 TP 304, SA 249 TP 304	
	Tube Side	Shell Side
Heads	SA240, Type 304, 304L, SA182 Gr F304 SA403 Type 304	SA285 Gr C, SA516 Gr 70
Nozzle Necks	SA240 Type 304 SA312 Type 304, SA479 Type 304	SA106 Gr B
Shells	SA240 Type 304	SA106 Gr B, SA285 Gr C, SA315, Gr CF8A SA516 Gr 70
Flanges	SA182 Gr F304	SA105
Auxiliary Pressure Vessels, Tanks, Filters, etc.		
Shells & Heads	SA-351 Gr CF8A, SA-312 TP304, SA-182 Gr F304, SA240 Type 304 or SA264 consisting of SA537 Gr B with Stainless Steel Weld Metal Analysis A-8 Cladding	
Flanges & Nozzles	SA182 Gr F304 and SA105 or SA350 Gr LF2, LF3 with Stainless Steel Weld Metal Analysis A-8 Cladding	
Piping	SA312 and SA240 TP304 or TP316 Seamless	
Pipe Fittings	SA403 WP394 Seamless	
Closure Bolting & Nuts	SA193 Gr B7 and SA194 Gr 2H	
Auxiliary Pumps		
Pump Casing & Heads	SA351 Gr CF8 or CF8M, SA182 Gr F304 or F315	
Flanges & Nozzles	SA182 Gr F304 or F316, SA403 Gr WP316L Seamless	
Piping	SA312 TP304 or TP316 Seamless	
Stuffing or Packing Box Cover	SA351 Gr CF8 or CF8M, SA240 TP304 or TP316, SA182 Gr F304	
Pipe Fittings	SA403 Gr WP316L Seamless, SA213 TP304	

Closure Bolting & Nuts	SA193 Gr B6, B7 or B8M and SA194 Gr2H or Gr 8M, SA193 Gr B6, B7 or B8M; SA453 Gr 660; and Nuts, SA194 Gr 2H, Gr 8M, Gr 6, and Gr 7

Table 5-8. Reactor Vessels Internals for Emergency Core Cooling

Forgings	SA182 Type F304
Plates	SA240 Type 304
Pipes	SA312 Type 304 Seamless or SA376 Type 304
Tubes	SA213 Type 304
Bars	SA479 Type 304 & 410
Castings	SA351 Gr CF8 or CF8A
Bolting	SA193 GrB8M (65 MYS/90MTS) Code Case 1613 Inconel 750 SA637 Gr688 Type 2
Nuts	SA193 Gr B-8
Locking Devices	SA479 Type 304

Table 5-9. Deleted Per 1998 Update

Table 5-10. Leakage Detection Sensitivity

Detection Device	Parameter Monitored	Readout Location	Leak Rate	Sensitivity	Response Time
Containment Atmosphere Particulate Radioactivity Monitor	Deleted Per 2007 Update	Deleted Per 2007 Update	Deleted Per 2007 Update	Deleted Per 2007 Update	Deleted Per 2007 Update
	Radioactivity accumulated on filters from samples of containment air.	Control Room	1 gpm	The monitor sensitivities are given in Table 11-20	Assume leakage activity containing only current realistic coolant activity, then leakage will be detected in 10 hours or less during Mode 1.
Containment Floor and Equipment Sump Level Indicator	Water level in sump	Control Room	1 gpm	1 gpm within one hour of leakage reaching the sump.	Leak detection within one hour.
Ventilation Unit Condensate Drain Tank Level Indicator	Water level in tank	Control Room	1 gpm	1 gpm within one hour of leakage reaching the tank.	Leak detection within one hour.
Incore Instrumentation Sump	Water level in sump	Control Room	1 gpm	1 gpm within four hours of leakage reaching the sump.	Leak detection within four hours.

Table 5-11. Reactor Vessel Quality Assurance Program

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

	RT[1]	UT[1]	PT[1]	MT[1]
Forgings				
1. Flanges		*yes*	*yes*	*yes*
2. Studs, Nuts		*yes*		*yes*
3. Head Adaptors		*yes*	*yes*	
4. Head Adaptor Tube		*yes*	*yes*	
5. Instrumentation Tube		*yes*	*yes*	
6. Main Nozzles		*yes*		*yes*
7. Nozzle Safe Ends (Unit 2)		*yes*	*yes*	
8. Shells (Unit 1)		*yes*		*yes*
Plates		*yes*		*yes*
Weldments				
1. Main Seam	*yes*	*yes*		*yes*
2. CRD Head Adaptor Connection			*yes*	
3. Instrumentation Tube Connection			*yes*	
4. Main Nozzle	*yes*	*yes*		*yes*
5. Cladding		*yes*	*yes*	
6. Nozzle Safe Ends				
(Forging - Unit 2)	*yes*	*yes*	*yes*	
7. Nozzle Safe Ends	*8.*	*9.*	*10.*	*11.*
(Weld deposit - Unit 1)	*yes*	*yes*	*yes*	
8. Head Adaptor Forging to Head	*9.*	*10.*	*11.*	*12.*
Adaptor Tube	*yes*		*yes*	
9. All Full Penetration Ferritic				
Pressure Boundary Welds				
Accessible After Hydrotest	*yes*		*yes*	
10. All Full Penetration Nonferritic				
Pressure Boundary Welds Accessible				
After Hydrotest		*yes*		
11. Seal Ledge			*yes*	
12. Head Lift Lugs			*yes*	

	RT[1]	***UT[1]***	***PT[1]***	***MT[1]***
13. Core Pad Welds		*yes*		

Note:

1. RT – Radiographic
UT – Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle

Table 5-12. Initial (Unirradiated) Toughness Properties for the Catawba Unit 1 Reactor Vessel[3]

Component	Heat No.	Mat'l Spec. No.	Cu (%)	P (%)	Ni (%)	T_{NDT} °F	T_{cv}[1] (50 Ft-Lb) (35 Mil) °F	RT_{NDT} °F	Shelf Energy NPWD[1] Ft-Lb
Closure Head Dome	55888-1	A533B,CL.1	-	.011	0.61	-4	70	10	86
Closure Head Ring	007055	A508,CL.2	-	.006	0.86	16	23	16	101
Closure Head Flange	527038	”	.05	.013	0.83	-4	-2	-4	104
Vessel Flange	411212	”	-	.004	0.86	-31	-47	-31	153
Inlet Nozzle	526827	”	.05	.010	0.75	-13	7	-13	87
“ ”	526829	”	.07	.010	0.74	-4	43	-4	86
“ ”	526859	”	.04	.013	0.77	-4	23	-4	81
“ ”	526857	”	.05	.012	0.75	-13	23	-13	77
Outlet Nozzle	526827	”	.05	.011	0.75	-22	27	-22	84
“ ”	526829	”	.07	.010	0.75	-4	2	-4	87
“ ”	526859	”	.04	.011	0.77	-13	38	-13	81
“ ”	526857	”	.05	.013	0.80	-4	38	-4	60
Nozzle Shell Forging 06	411077	”	-	.007	0.85	-40	34	-26	101
Inter. Shell Forging 05	411343	”	.086[6]	.004	0.858[6]	-40	52	-8	134[4]
Lower Shell Forging 04	527708	”	.04	.008	0.83	-13	16	-13	134[4]
Bottom Head Ring	527428	”	.06	.013	0.77	-4	74	14	68
Bottom Head Segment	55292-1	A533B,CL.1	-	.006	0.59	-22	5	-22	79
“ ” ”	”	”	-	.006	0.59	-13	34	-13	79
“ ” ”	55163-2	”	-	.011	0.61	-4	38	-4	80
“ ” ”	”	”	-	.011	0.61	-13	74	14	70

Component	Heat No.	Mat'l Spec. No.	Cu (%)	P (%)	Ni (%)	T_{NDT} °F	T_{cv}[1] (50 Ft-Lb) (35 Mil) °F	RT_{NDT} °F	Shelf Energy NPWD[1] Ft-Lb
" " Dome	55178-1	"	-	.010	0.64	-31	84	24	64
Nozzle Shell to Inter. Shell Weld (P710)			.03	-	0.75	-	-	10[2]	92[8]
Inter. Shell to Lower Shell Weld Root (P710)			.03	.009	0.75[7]	-0[2]	-	0[2]	
Lower Shell to Bot. Head Ring Weld (P710)			.03	-	0.75	-	-	10[2]	92[8]
Inter. Shell to Lower Shell Weld (R747)			.039[6]	.010	.724[6]	-76	-9	-51	130[4]

Weld Control No.	Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot. No.
P710	NIMO	899680	Grau Lo	P23
R747[5]	"	895075	Grau Lo	P46

Component	Heat No.	Mat'l Spec. No.	Cu (%)	P (%)	Ni (%)	T_{NDT} °F	T_{cv}[1] (50 Ft-Lb) (35 Mil) °F	RT_{NDT} °F	Shelf Energy NPWD[1] Ft-Lb

Notes:

1. Estimated per NRC Standard Review Plan Section 5.3.2 from data obtained in the principle working direction
2. Estimated per NRC Standard Review Plan Section 5.3.2 from charpy tests performed at 10°F
3. Source (except where noted otherwise): S. E. Yanichko, "Catawba Unit No. 1 Reactor Material Toughness Properties, Westinghouse Internal Calc-Note dated June 29, 1978 (located in reactor vessel materials archives with MCTRs).
4. Source: WCAP-15609, Rev. 1, "MOX Fuel Effects on Reactor Vessel Integrity at Catawba Units 1 and 2 and McGuire Units 1 and 2", dated March 2003, and Westinghouse Owners Group Calcnote 92-016, "WOG USE Program – Onset of Upper Shelf Energy Calculations, J. M. Chicots, 1/19/93 [MUHP-5080].
5. Used for Surveillance Program Weldment.
6. ATI-94-012-T003, Rev. 2, "A Review of Materials Data for the Catawba 1 Reactor Pressure Vessel, dated March 1999.
7. Source: Check analysis reported weld deposit analysis reported in De Rotterdame Drodgdak Mattschappu N.V. (The Rotterdam Dockyard Company or RDM) Welding Material Test Report.
8. Source: Catawba License Renewal Application

Table 5-13. Initial (Unirradiated) Toughness Properties for the Catawba Unit 2 Reactor Vessel[3]

Component	Heat No.	Mat'l Spec No.	Cu (%)	P (%)	Ni (%)	T_{NDT} °F	T_{cv}[1] Ft-Lb[1] (35 Mil) °F	RT_{NDT} °F	Shelf Energy NPWD[1] Ft-Lb
Closure Head Dome	B8607-1	A533B,CL.1	.13	.007	0.64	- 40	50	- 10	106
Closure Head Torus	B8608- 1	”	.07	.007	0.62	- 20	57	- 3	118
Closure Head Flange	B8601- 1	A508 CL.2	-	.010	0.70	10	<10	10	152
Vessel Flange	B8602- 1	”	-	.010	0.71	10	<10	10	175
Inlet Nozzle	B8609- 1	”	-	.010	0.81	- 20	<10	- 20	119
“ ”	B8609- 2	”	-	.010	0.78	- 20	<10	- 20	124
“ ”	B8609- 3	”	-	.008	0.85	- 20	<40	- 20	109
“ ”	B8609- 4	”	-	.006	0.85	- 20	97	37	97
Outlet Nozzle	B8610- 1	”	-	.008	0.73	- 10	<50	- 10	141
“ ”	B8610- 2	”	-	.006	0.78	- 10	<50	- 10	144
“ ”	B8610- 3	”	-	.004	0.80	- 20	<40	- 20	140
“ ”	B8610- 4	”	-	.006	0.80	- 10	<50	- 10	150
Nozzle Shell	B8604- 1	A533B,CL.1	.11	.007	0.61	- 10	84	24	96
“ ”	B8604- 2	”	.11	.007	0.61	- 10	86	26	89
“ ”	B8604- 3	”	.07	.009	0.53	- 20	110	50	70
Intermediate Shell	B8605- 1	”	.082[1]	.011	0.618[4]	- 10	75	15	89
“ ”	B8605- 2	”	.08[4]	.012[5]	0.613[4]	- 20	93	33	82
“ ”	B8616- 1	”	.045[4]	.010	0.595[4]	0	72	12	92
Lower Shell	B8806- 1	”	.057[4]	.011[5]	0.56[4]	- 60	66	6	83
“ ”	B8806- 2	”	.057[4]	.011[5]	0.593[4]	- 40	50	- 10	102

Component	Heat No.	Mat'l Spec No.	Cu (%)	P (%)	Ni (%)	T_{NDT} °F	T_{cv}[1] Ft-Lb[1] (35 Mil) °F	RT_{NDT} °F	Shelf Energy NPWD[1] Ft-Lb
" "	B8806- 3	"	.057[4]	.011[5]	0.593[4]	- 40	68	8	105
Bottom Head Torus	B8613- 1	"	.14	.010	0.48	- 40	52	- 8	113
Bottom Head Dome	B8612- 1	"	.14	.010	0.48	- 40	65	5	124
Nozzle Shell Vert. Weld Seams (G1.36)			.156[6]	.012[6]	0.059[6]	- 50	<10	-50	>112
Nozzle Shell to Inter. Shell Weld Seam (G1.50)			.153[6]	.016[6]	0.077[6]	- 40	<20	-40	>102
Inter. & Lower Vert. Weld Seams (G1.45)			.04	.005	.12	- 80	<-20	-80	>130
Inter. & Lower Shell Weld Seam (G1.45)									

Weld Control No.	Weld Wire		Flux	
	Type	Heat No.	Type	Lot No.
G1.36	B- 4	51912	Linde 0091	3490
G1.50	B- 4	5P5622	"	1122
G1.45[2]	B- 4	83648	"	3536

Notes:

1. Estimated per NRC Standard Review Plan Section 5.3.2.
2. Used for Surveillance Program Weldment.
3. Source (except where noted otherwise): WCAP-11941, Analysis of Capsule Z from the Duke Power Company Catawba Unit 2 Reactor Vessel Radiation Surveillance Program, S.E. Yanichko et al, table A-3, September 1988.
4. Source: ATI-94-012-T002, Rev. 2, "A Review of Materials Data for the Catawba 2 Reactor Pressure Vessel, dated December 1999.
5. Source: Combustion Engineering's (C-E's)
6. Source: Combustion Engineering Report NPSD-1039, Rev. 2.

Table 5-14. Comparison of Initial (Unirradiated) and Projected EOLE (54 EFPY) Fracture Toughness Properties Of The Catawba Unit 1 Reactor Vessel Beltline Region Material

		Initial Values			End-of-Life Extension at 54 EFPY					
Component	**Code No.**	**T_{NDT} (°F)**	**RT_{NDT} (°F)**	**USE[2] (ft-lb)**	**Inner Wall Fluence[2] 10^{19}n/cm^2**	**ΔRT_{PTS}[2] (°F)**	**Margin[2]**	**RT_{PTS}[2] (°F)**	**USE Drop[2] (%)**	**EOLE USE[2] (ft-lb)**
Inter. Shell Forging 05	411343[1]	-40	-8	134	2.60	35.8	17	45	10	121
Lower Shell Forging 04	527708	-13	-13	134	2.60	32.7	32.7	52	21	106
Weld (W05)	R747[1]	-76	-51	130	2.60	35.8	28	13	8	120

Notes:

1. Surveillance program materials.
2. Fluence, ΔRT_{PTS} and RT_{PTS}, and USE values taken from Westinghouse Report WCAP-17669-NP, "Catawba Unit 1 Measurement Uncertainty Recapture (MUR) Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations, "dated June 2013. Also see UFSAR Tables 5-42 and 5-44 for additional information.

Table 5-15. Comparison of Initial (Unirradiated) and Projected EOLE (54 EFPY) Toughness Properties for the Catawba Unit 2 Reactor Vessel

		Initial Values			End-of-Life Extension at 54 EFPY					
Component	**Code No.**	**T_{NDT} (°F)**	**RT_{NDT} (°F)**	**USE[3] (ft-lb)**	**Peak Inner Wall Fluence[2] 10^{19}n/cm^2**	**ΔRT_{PTS}[2] (°F)**	**Margin**	**RT_{PTS}[2] (°F)**	**USE (Drop[3] (%)**	**EOLE USE[3] (ft-lb)**
Inter. Shell	B8605-1[1]	-10	15	89	3.16	57.2	17	89	6.6	90
Inter. Shell	B8605-2	-20	33	82	3.16	66.3	34	133[4]	22	64
Inter. Shell	B8616-1	0	12	92	3.16	40.3	34	86	22	72
Lower Shell	B8806-1	-60	6	83	3.16	48.1	34	88	22	65
Lower Shell	B8806-2	-40	-10	102	3.16	48.1	34	72	22	80
Lower Shell	B8806-3	-40	8	105	3.16	48.1	34	90	22	82
Inter. to Lower Shell Circ. Weld Seam (101-171); Axial Weld Seams (101-142A, 101-124B), and (101-142B/C, 101-124A/C)	G1.45[1]	-80	-80	146	3.16	43.4	28	-9	11	130

Table 5-16. Catawba Unit 1 Closure Head Bolting Material Properties

Heat No.	Mat'l Spec. No.	Bar No.[1]	0.2 YS KSi	UNITS KSi	ELONG %	RA %	BHN	Energy At 10°F Ft-Lbs	Lateral Expansion Mils
Closure Head Studs									
35674	A540,B24	25K	142.5	160.4	18.8	57	341	42,40,40	12,16,28
35674	A540,B24	25T	136.9	157.1	18.0	57	331	38,38,39	12,8,12
35674	A540,B24	26K	143.9	161.8	19.0	57	331	46,46,46	32,28,28
35674	A540,B24	26T	138.2	162.7	19.4	56	341	44,44,43	32,24,28
35674	A540,B24	27K	141.5	161.8	18.2	56	331	40,39,39	12,28,12
35674	A540,B24	27T	141.5	160.5	18.4	56	331	44,43,44	16,24,24
35666	A540,B24	28K	143.6	162.8	18.4	52	321	48,48,46.5	28,24,24
35666	A540,B24	28T	145.0	164.9	18.0	52	331	45.5,46.5,49.5	28,20,24
35666	A540,B24	29K	145.7	163.8	18.2	55	341	42,40.5,40.5	24,16,20
35666	A540,B24	29T	145.7	163.8	18.4	55	331	38,38,40.5	20,16,24
35847	A540,B24	297K	143.6	160.4	18.0	59	321	49,52,52.5	32,24,24
35847	A540,B24	297T	145.7	161.8	18.0	59	341	49.5,53.5,53.5	32,32,35
Closure Head Nuts (Original supply. See Section 5.3.1.7.1 for discussion of alternative nuts)									
36627	A540,B24	328K	133.5	153.7	20.0	61	331	60,62,58	35,45,47
36627	A540,B24	328T	135.9	154.7	20.0	60	331	60.5,67,62	39,47,43
Closure Head Washers (Original supply. See Section 5.3.1.7.1 for discussion of alternative washers)									
36512	A540,B24	306K	131.2	153.7	19.6	60	331	48.5,49.5,49	35,35,32
36512	A540,B24	306T	132.5	153.7	20.2	60	321	52,52.5,50.5	43,43,43

Closure Head Studs									
Heat No.	**Mat'l Spec. No.**	**Bar No.**[1]	**0.2 YS KSi**	**UNITS KSi**	**ELONG %**	**RA %**	**BHN**	**Energy At 10°F Ft-Lbs**	**Lateral Expansion Mils**

Note:

1. K & T denote top and bottom of bar respectively.

Table 5-17. Catawba Unit 2 Closure Head Bolting Material Properties

Closure Head Studs									
Heat No.	**Mat'l Spec. No.**	**Bar No.[1]**	**0.2%YS KSi**	**UTS KSi**	**ELONG %**	**RA %**	**BHN**	**Energy At 10°F Ft-Lbs**	**Lateral Expansion Mils**
81874	A540,B24	144	149.5	163.0	16	51.4	331	47,47,47	27,27,26
”	”	144-1	146.5	160.0	17	54.7	341	50,49,49	31,28,29
”	”	146	153.5	167.0	16	52.7	341	44,44,45	25,26,26
”	”	146-1	145.0	160.0	17	54.4	331	51,49,49	30,28,28
”	”	149	146.5	160.0	17	54.8	352	50,49,49	27,29,26
”	”	149-1	147.0	161.0	17	52.7	341	49,49,49	28,26,27
”	”	153	152.0	164.0	16	53.8	341	53,52,52	31,30,32
”	”	153-1	140.5	155.0	18	54.4	331	46,47,47	26,26,26
”	”	157	142.5	158.0	17.5	54.4	341	51,51,51	30,28,31
”	”	157-1	148.0	161.5	17	54.0	341	47,48,47	27,30,27
”	”	163	149.8	163.0	16.5	55.1	331	49,49,49	28,26,29
”	”	163-1	145.0	159.0	16.5	54.3	341	51,51,53	30,31,33
82552	”	197	142.5	156.0	18	53.8	341	51,51,52	28,29,30
”	”	197-1	141.5	155.0	17	53.8	341	50,51,50	28,31,28
”	”	201	146.0	158.5	15.5	51.7	341	48,48,49	27,27,27
”	”	201-1	145.5	159.5	15.5	50.6	341	47,49,47	27,30,27
”	”	207	138.0	153.0	17.0	52.5	341	51,52,51	31,32,31
”	”	207-1	138.5	153.0	16.5	51.4	341	51,50,49	28,28,27
”	”	212	141.0	155.0	17.0	53.0	341	52,52,51	32,31,30

Heat No.	Mat'l Spec. No.	Bar No.[1]	0.2%YS KSi	UTS KSi	ELONG %	RA %	BHN	Energy At 10°F Ft-Lbs	Lateral Expansion Mils
Closure Head Studs									
”	”	212-1	144.0	157.0	16.0	49.8	352	51,50,49	31,29,29
Closure Head Nuts and Washers (Original supply. See Section 5.3.1.7.1 for discussion of alternative nuts)									
19632	A540,B23	69	141.5	155.0	17	56.4	321	55,55,55	33,30,29
”	”	69-1	145.5	158.0	17	55.0	321	49,47,47	30,28,25
”	”	73	145.5	158.0	16	51.9	-	55,56,55	34,33,31
”	”	73-1	142.5	156.0	16.5	54.5	-	48,47,45	28,26,25

Table 5-18. Reactor Vessel Design Parameters

Design Pressure, psig	2485
Design Temperature, °F	650
Overall Height of Vessel and Closure Head, ft (Bottom Head Outside Diameter to top of Control Rod Mechanism Adaptor)	43.833
Thickness of Insulation, minimum, in	3
Number of Reactor Closure Head Studs	54
Diameter of Reactor Closure Head/Studs, in (minimum shank)	6.75 (Unit 1) 6.8125 (Unit 2)
Inside Diameter of Flange, in	l67
Outside Diameter of Flange, in	205
Inside Diameter at Shell, in	l73
Inlet Nozzle Inside Diameter, in	27.5
Outlet Nozzle Inside Diameter, in	29
Clad Thickness, minimum, in	0.125
Lower Head Thickness, minimum, in	5.236 (Unit 1) 5.375 (Unit 2)
Vessel Belt-Line Thickness, minimum, in	8.464 (Unit 1) 8.625 (Unit 2)
Closure Head Thickness, in	6.496 (Unit 1) 7.0 (Unit 2)

Table 5-19. Chemical Composition Of The Catawba Unit 1 Reactor Vessel Beltline Region Material[3]

Element	Inter. Shell Forging 411343	Lower Shell Forging 527708	Weld Control No. R747[1]	Weld Control No. P710[2]
C	.21	.21	.069	.052
Mn	.76	.72	1.97	1.97
P	.004	.008	.010	.009
S	.006	.007	.010	.015
Si	.28	.33	.22	.25
Ni	.858[4]	.83	.724[4]	.75
Cr	.38	.33	.05	.04
Mo	.60	.55	.56	.46
Cu	.086[4]	.04	.039[4]	.03
V	<.01	<.01	-	.01
Co	.013	.01	-	-
Al	.04	.01	-	.014

Notes:

1. Submerged arc weld - NiMo wire (Heat No. 895075) and Grau Lo Flux (Lot No. P46) used to fabricate beltline region girth seam WO5 and surveillance weldment.
2. Submerged arc weld - NiMo wire (Heat No. 899650) and Grau Lo Flux (Lot No. P23) used to fabricate root of beltline region girth seam WO5.
3. Unless noted otherwise, the source of all composition data is the check analysis reported on DeRotterdame Drodgdak Mattschappu N.V. (The Rotterdam Dockyard Company or RDM) MCTRs or weld deposit analysis reported in RDM’s Welding Material Test Reports.
4. ATI-94-012-T003, Rev. 2, “A Review of Materials Data for the Catawba 1 Reactor Pressure Vessel, dated March 1999.

Table 5-20. Chemical Composition of the Catawba Unit 2 Reactor Vessel Beltline Region Material

	Intermediate Shell Plate Code. No. (Heat No.)			Lower Shell Plate Code No. (Heat No.)		Weld Control No. (Heat No.)	
Element	**B8605-1 (C-0543-1)**	**B8605-2 C-0543-2)**	**B8616-1 (A-0617-1)**	**B8806-1 (C-2288-1)**	**B8806-2 (C-2272-1)**	**B8806-3 (C-2272-2)**	**G1.45[2] (83648[)]**
C	.25	.25	.24	.24	.22	.22	.13
Mn	1.40	1.40	1.39	1.44	1.36	1.36	1.23
P	.011	.012	.010	.011	.011	.011	.005
S	.013	.012	.021	.014	.016	.016	.009
Si	.28	.28	.27	.23	.25	.25	.13
Ni[4]	.618	.613	.595	.56	.593	.593	.136
Cr	-	-	-	-	-	-	-
Mo	.57	.56	.54	.55	.54	.54	.59
Cu[4]	.082	.08	.045	.057	.057	.057	.042
V	-	-	ND[1]	ND[1]	ND[1]	ND[1]	.006

Notes:

1. Not detected.
2. Submerged arc weld -. type B4 wire (Heat No. 83648) and Linde 0091 flux (Lot No. 3536) used to fabricate all beltline region weld seams including the surveillance weldment: chemistry data taken rom WCAP-11941, “Analysis of Capsule Z from the Duke Power Company Catawba Unit 2 Reactor Vessel Radiation Surveillance Program”, dated September 1988
3. Unless noted otherwise, the source of all composition data is the check analysis reported on Combustion Engineering’s (C-E’s) MCTRs or weld deposit analysis reported in C-E’s Welding Material Test Reports.
4. All Cu and Ni values were taken from ATI-94-012-T002, Rev. 2, “A Review of Materials Data for the Catawba 2 Reactor Pressure Vessel, dated December 1999.

Table 5-21. Catawba Unit 1 Reactor Vessel Beltline Region Toughness Properties

Inter. To Lower Shell Weld Weld Code No. R747				Weld Metal For Pressure Vessel Core Region				Inter. To Lower Shell Weld Root Weld Core No. P710			
Temp. (°F)	Energy (ft/lb)	Lat. Exp. (Mils)	Shear (%)	Temp. (°F)	Energy (ft/lb)	Lat. Exp. (Mils)	Shear (%)	Temp (°F)	Energy (ft/lb)	Lat. Exp. (Mils)	Shear (%)
-148	3.5	4	0	-100	12	6	13	10	57.5	43	47
-112	8.5	12	16	-60	13	9.5	28	10	43.0	39	47
-76	21.5	23	33	-60	15	11.5	33	10	39.0	55	55
-40	45.0	39	43	-40	37	26	33				
-22	57.5	47	55	-40	26	18	28				
- 4	54.5	43	47	-16	40	31	42				
32	92.5	71	76	-16	60	45.5	37				
68	104.5	81	92	-16	54	39	50				
86	113.0	91	89	5	44	38	54				
122	123.5	83	93	25	91	66	72				
140	144.0	99	100	25	98.5	74	87				
158	129.0	94	98	75	119	86	87				
176	130.0	87	98	75	110	77	95				
212	126.5	87	100	120	132	90.5	100				
				120	119	87	100				
				210	133	90	100				
				210	130	90	100				
				210	124	89	100				
T_{NDT} = -76°F											
RT_{NDT} = -51°F											

Lower Shell Forging (04) Heat No. 527708 (Tang)				Lower Shell Forging (04) Heat No. 527708 (Axial)			
Temp. (°F)	Energy (ft/lb)	Lat. Exp (Mils)	Shear (%)	Temp. (°F)	Energy (ft/lb)	Lat. Exp. (Mils)	Shear (%)
-148	2.5	2	0	-148	5	4	0
-148	3.5	4	0	-148	3	4	0
-148	4.5	4	0	-148	3	4	0
-76	11.5	12	4	-76	10	8	13
-76	11.5	8	3	-76	31	16	10
-76	11.5	8	3	-76	20	24	11
- 4	57.5	47	29	-4	67	59	23
- 4	57.5	51	23	-4	82	67	35
- 4	53.5	47	23	-4	69	59	29
40	101.5	79	67	60	94	71	60
40	126.5	94	80	60	104	75	60
40	109.5	83	68	60	98	79	61
40	149.0	94	80	104	130	91	66
40	156.5	91	75	104	123	83	60
40	113.0	83	65	104	112	83	72
60	125.5	83	70	176	133	91	90
60	117.0	71	62	176	135	94	89
60	106.5	71	55	176	135	87	85
113	149.5	94	100				
113	135.5	91	85				
113	149.0	94	100				

Lower Shell Forging (04) Heat No. 527708 (Tang)				Lower Shell Forging (04) Heat No. 527708 (Axial)			
Temp. (°F)	Energy (ft/lb)	Lat. Exp (Mils)	Shear (%)	Temp. (°F)	Energy (ft/lb)	Lat. Exp. (Mils)	Shear (%)
176	155.0	99	98				
176	158.5	91	100				
176	151.0	91	100				
T_{NDT} = -13°F							
RT_{NDT} = -13°F							

Inter. Shell forging (05) Heat No. 411343				Inter. Shell forging (05) Heat No. 411343 (Axial)				Inter. Shell Equip. (05) Heat No. 411343 (Tang)			
Temp. (°F)	Energy (ft/lb)	Lat. Exp. (Mils)	Shear (%)	Temp. (°F)	Energy (ft/lb)	Lat. Exp. (Mils)	Shear (%)	Temp. (°F)	Energy (ft/lb)	Lat. Exp. (Mils)	Shear (%)
-148	5	4	0	-100	2	0	0	-148	2.0	6	0
-148	5	4	0	-40	8	5.5	0	-148	3.0	2	0
-148	5	4	0	-40	16	11	0	-148	4.5	4	0
-76	10	4	8	-15	48	34	25	-76	8.5	8	3
-76	17	16	10	0	27	19	20	-76	3.5	4	0
-76	8	4	4	0	35	25	30	-76	8.5	4	0
-4	43	59	29	0	51	38	30	-4	45.5	39	27
-4	58	28	11	20	60	44	34	-4	16.5	20	16
-4	54	39	16	20	56	44	40	-4	42.0	35	17
60	86	67	50	20	69	52	45	40	101.5	79	52
60	83	63	44	75	81	60	52	40	102.5	79	49
60	80	63	49	75	89	67	61	40	100.0	71	49

Inter. Shell forging (05) Heat No. 411343				Inter. Shell forging (05) Heat No. 411343 (Axial)				Inter. Shell Equip. (05) Heat No. 411343 (Tang)			
Temp. (°F)	Energy (ft/lb)	Lat. Exp. (Mils)	Shear (%)	Temp. (°F)	Energy (ft/lb)	Lat. Exp. (Mils)	Shear (%)	Temp. (°F)	Energy (ft/lb)	Lat. Exp. (Mils)	Shear (%)
104	115	83	76	120	111	77	82	40	87.0	71	47
104	117	75	80	120	119	79.5	81	40	86.5	71	38
104	112	83	74	150	135	90	100	40	57.5	51	38
176	138	99	80	210	137	88	100	60	118.0	75	65
176	137	91	80	210	132	91	100	60	122.5	79	57
176	136	91	85	210	133	87	100	60	92.0	63	52
								113	140.0	87	85
								113	129.5	91	76
								113	155.0	99	90
								176	152.5	87	100
								176	156.5	87	100
								176	152.5	94	100
										T_{NDT} = -40°F	
										RT_{NDT} = -8°F	

Table 5-22. Catawba Unit 2 Reactor Vessel Beltline Region Toughness Properties

Intermediate Shell Course (Transverse Data)											
Plate B8605 - 1				Plate B8605 - 2				Plate B8616-1			
Temp. (°F)	Energy (ft/lb)	Lat. Exp. Mils	Shear %	Temp. °F	Energy (ft/lb)	Lat. Exp. Mils	Shear (%)	Temp (°F)	Ener (ft/l	La Exp (Mils	Shear (%)
- 80	5	2	3	- 40	5	4	0	- 40	7	6	0
- 80	6	5	3	- 40	4	4	0	- 40	11	8	0
- 40	9	6	13	- 40	6	5	0	- 40	10	7	0
- 40	16	10	18	10	15	12	5	10	26	18	20
- 40	18	13	9	10	22	18	10	10	30	19	20
0	34	21	29	10	14	12	5	10	29	21	20
0	35	27	25	40	25	22	15	40	35	30	20
0	41	35	29	40	37	27	20	40	42	32	25
40	35	27	38	40	37	26	20	40	50	36	30
40	52	41	43	75	42	31	25	74	53	41	40
40	54	40	34	75	50	37	30	74	52	39	40
80	59	47	41	75	37	30	25	74	56	43	40
80	64	48	44	100	70	54	50	100	65	49	60
80	71	50	45	100	58	42	40	100	64	47	60
100	53	43	47	100	63	46	40	100	69	51	60
100	66	50	55	160	78	60	95	160	89	68	100
100	77	54	59	160	83	63	95	160	93	69	100
120	66	47	62	160	86	64	100	160	95	71	100

Intermediate Shell Course (Transverse Data)											
Plate B8605 - 1				Plate B8605 - 2				Plate B8616-1			
Temp. (°F)	Energy (ft/lb)	Lat. Exp. Mils	Shear %	Temp. °F	Energy (ft/lb)	Lat. Exp. Mils	Shear (%)	Temp (°F)	Ener (ft/l	Lat Exp (Mils	Shear (%)
120	84	61	68								
120	94	70	100								
180	92	64	100								
180	96	72	100								
180	99	71	100								
240	100	71	100								
240	102	78	100								
320	84	65	100								
320	100	75	100								
T_{NDT} = -10°F				T_{NDT} = -20°F				T_{NDT} = 0°F			
RT_{NDT} = 15°F				RT_{NDT} = 33°F				RT_{NDT} = 12°F			

Lower Shell Course (Transverse Data)											
Plate B8806 - 1				Plate B8806 - 2				Plate B8806 - 3			
Temp. (°F)	Energy (ft/lb)	Lat. Exp. Mils	Shear %	Temp. °F	Energy (ft/lb)	Lat. Exp. Mils	Shear (%)	Temp. (°F)	Ener (ft/l	Lat Exp (Mils	Shear (%)
- 40	16	11	0	- 40	16	12	0	- 40	14	9	0
- 40	15	9	0	- 40	16	12	0	- 40	9	6	0
- 40	13	10	0	- 40	15	10	0	- 40	18	13	5

Intermediate Shell Course (Transverse Data)											
Plate B8806 - 1				Plate B8806 - 2				Plate B8806-1			
Temp. (°F)	Energy (ft/lb)	Lat. Exp. Mils	Shear %	Temp. °F	Energy (ft/lb)	Lat. Exp. Mils	Shear (%)	Temp (°F)	Energy (ft/lb)	Lat. Exp. (Mils)	Shear (%)
10	36	26	15	10	38	25	15	10	35	23	15
10	40	28	20	10	35	23	15	10	26	19	10
10	38	28	20	10	34	21	15	10	44	33	20
40	48	37	30	40	53	34	25	40	39	24	15
40	49	36	30	40	61	42	40	40	43	31	20
40	46	34	30	40	49	33	20	40	44	31	20
74	60	45	50	74	61	40	40	74	53	39	30
74	53	41	50	74	59	40	40	74	58	43	35
74	70	52	70	74	64	44	40	74	72	50	60
100	77	62	90	100	86	61	70	100	75	50	70
100	72	59	80	100	75	52	60	100	81	52	70
100	84	64	100	100	69	46	60	100	88	59	80
160	82	61	100	160	108	74	100	160	102	69	100
160	85	65	100	160	98	69	100	160	103	71	100
160	82	64	100	160	100	72	100	160	110	73	100
T_{NDT} = - 60°.F				T_{NDT} = -40°F				T_{NDT} = - 40°F			
RT_{NDT} = - 6°F				RT_{NDT} = -10°F				RT_{NDT} = 8°F			

Intermediate And Lower Shell Vertical Weld Seams And Girth Weld Seam Weld Code No G1.45			Weld Metal for Pressure Vessel Core Region			
Temp (°F)	**Energy (ft/lb)**	**Lat. Exp. (mils)**	**Temp (°F)**	**Energy (ft-lb)**	**Lat. Exp. (mils)**	**Shear (%)**
- 20	108	74	-140	4	1	9
- 20	84	56	-140	5	2	13
- 20	79	54	-80	6	1	13
10	130	84	-80	8	3	18
10	129	76	-80	26	11	18
10	132	85	-60	11	6	28
			-60	15	10	33
T_{NDT} = -80°F			-60	15	6	28
RT_{NDT} = - 80°F			-40	46	31	47
			-40	58	41	40
			-40	73	51	52
			0	58	45	65
			0	96	60	73
			0	101	72	71
			40	121	79	96
			40	125	78	93
			40	135	80	84
			80	131	79	93

Intermediate And Lower Shell Vertical Weld Seams And Girth Weld Seam Weld Code No G1.45			Weld Metal for Pressure Vessel Core Region			
Temp (°F)	Energy (ft/lb)	Lat. Exp. (mils)	Temp (°F)	Energy (ft-lb)	Lat. Exp. (mils)	Shear (%)
			80	138	88	96
			80	147	86	94
			120	142	88	100
			120	146	88	100
			120	151	87	100
			220	139	86	100
			220	148	91	100
			320	152	90	100
			320	164	86	100

Table 5-23. Reactor Coolant Pump Design Parameters

Unit Design Pressure, psig	2485
Unit Design Temperature, °F	650[1]
Unit Overall Height, ft	27.6
Seal Water Injection, gpm	8
Seal Water Return, gpm	3
Cooling Water Flow, gpm	436
Maximum Continuous Cooling Water Inlet Temperature, °F	105
Pump	
Capacity, gpm	101,000 ± 2000
Developed Head, ft	289 ± 12
NPSH Required, ft	Figure 5-13
Suction Temperature, °F	557.8
Pump Discharge Nozzle, Inside Diameter, in	27-1/2
Pump Suction Nozzle, Inside Diameter, in	31
Speed, rpm	1185
Water Volume, ft^3	78.6 [2]
Total Pump/Motor Weight (dry), lbs	201, 200
Motor	
Type	Drip proof, squirrel cage induction, water/air cooled
Power, Hp	7000
Voltage, volts	6600
Phase	3
Frequency, Hz	60
Insulation Class	Class F
Starting Current, Amps	3000 Amp @ 6600 volts
Input, Hot Reactor Coolant	492 ± 17 amps
Input, Cold Reactor Coolant	654 ± 23 amps
Pump Moment of Inertia, 1b–ft^2 minimum	95,000

Notes:

1. Design temperature of pressure retaining parts of the pump assembly exposed to the reactor coolant and injection water on the high pressure side of the controlled leakage seal shall be that temperature determined for the parts for a primary loop temperature of 650°F.
2. Composed of reactor coolant in the casing and of injection and cooling water in the thermal barrier.

Table 5-24. Reactor Coolant Pump Quality Assurance Program

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

	RT[1]	***UT[1]***	***PT[1]***	***MT[1]***
Castings	*yes*		*yes*	
Forgings				
1. Main Shaft		*yes*	*yes*	
2. Main Studs		*yes*	*yes*	
3. Flywheel (Rolled Plate)		*yes*	*yes*	*yes*
Weldments				
1. *Circumferential*	*yes*		*yes*	
2. *Instrument Connections*			*yes*	

Note:

1. RT – Radiographic
UT – Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle

Table 5-25. Steam Generator Design Data

	Unit 1	**Unit 2**
Design Pressure, reactor coolant side, psig	2485	2485
Design Pressure, steam side, psig	1185	1185
Design Temperature, reactor coolant side, °F	650	650
Design Temperature, steam side, °F	600	600
Total Heat Transfer Surface Area, ft^2	79,800	48,165
Maximum Moisture Carryover, wt percent	0.25	0.25
Overall Height, ft-in	67-8	67-8
Number of U-Tubes	6633	4570
U-Tube Nominal Diameter, in.	.6875	.750
Tube Wall Nominal Thickness, in.	.040	.043
Number of Manways	3	4
Inside Diameter of Manways, in.	21	16
Number of Inspection Ports		
2.0" Dia.	12	4(2)
2.5" Dia.	0	2
2.7" Dia.	N/A	1 on S/G 2C only
6.0" Dia.	10	5
Design Fouling Factor(Btu/hr °F $ft^2)^{-1}$	0.00002(1)	0.00005
Preheat Section	NA	0.00010

Notes:

1. There is no specified design fouling factor for McGuire or Catawba Unit 1. Fouling factor of 0.00002 is used for start-up, and various fouling factors are used thereafter for performance analysis under variable fouling.
2. 6 for S/G 2A

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

Table 5-26. ***Steam Generator Quality Assurance Program***

	RT[1]	UT[1]	PT[1]	MT[1]	ET[2]
Tubesheet					
1. Forging		yes		yes	
2. Cladding					
Unit 1		yes	yes[3]		
Unit 2		yes[2]	yes		
Channel Head					
1. Casting (Unit 2)	yes			yes	
2. Forging (Unit 1)		yes			
3. Cladding			yes		
Secondary Shell & Head					
1. Plates		yes			
Tubes		yes			yes
Nozzles (Forgings)		yes		yes	
Weldments					
1. Shell, longitudinal					
Unit 1	yes	yes		yes	
Unit 2	yes			yes	
2. Shell, circumferential					
Unit 1	yes	yes		yes	
Unit 2	yes			yes	
3. Cladding (channel head- tubesheet joint cladding restoration)			yes		
4. Steam and feedwater nozzle to shell					
Unit 1	yes	yes		yes	
Unit 2	yes			yes	
5. Support brackets				yes	
6. Tube to tubesheet			yes		
7. Instrument connections (primary and secondary)					
Unit 1		yes		yes	
Unit 2				yes	

	RT[1]	***UT***[1]	***PT***[1]	***MT***[1]	***ET***[2]
8. Temporary attachments after removal				*yes*	
9. After hydrostatic test (all welds and complete cast channel head - where accessible)				*yes*	
10. Nozzle safe ends (if forgings)	*yes*		*yes*		
11. Nozzle safe ends (if weld deposit)					
Unit 1	*yes*				
Unit 2			*yes*		

Notes:

1. *RT – Radiographic*
   UT – Ultrasonic
   PT - Dye Penetrant
   MT - Magnetic Particle
   ET - Eddy Current
2. *Flat Surfaces Only*
3. *Weld Deposit Areas Only*

Table 5-27. Reactor Coolant Piping Design Parameters

Reactor Inlet Piping, inside diameter, in	27-1/2
Reactor Inlet Piping, nominal wall thickness, in	2.32
Reactor Outlet Piping, inside diameter, in	29
Reactor Outlet Piping, nominal wall thickness, in	2.45
Coolant Pump Suction Piping, inside diameter, in	31
Coolant Pump Suction Piping, nominal wall thickness, in	2.60
Pressurizer Surge Line Piping, nominal pipe size, in	14
Pressurizer Surge Line Piping, nominal wall thickness, in	1.405
Reactor Coolant Loop Piping	
Design Pressure, psig	2485
Design Temperature, °F	650
Pressurizer Surge Line	
Design Pressure, psig	2485
Design Temperature, °F	680
Pressurizer Safety Valve Inlet Line	
Design Pressure, psig	2485
Design Temperature, °F	680
Pressurizer (Power-Operated) Relief Valve Inlet Line	
Design Pressure, psig	2485
Design Temperature, °F	680

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

Table 5-28. ***Reactor Coolant Piping Quality Assurance Program***

	RT[1]	***UT[1]***	***PT[1]***
Fittings and Pipe (Castings)	*yes*		*yes*
Fittings and Pipe (Forging)		*yes*	*yes*
Weldments			
1. Circumferential	*yes*		*yes*
2. Nozzle to runpipe (Except no RT for nozzles less than 6 inches)	*yes*		*yes*
3. Instrument connections			*yes*
Castings	*yes*		*yes (after finishing)*
Forgings		*yes*	*yes (after finishing)*

***Note**:*

1. RT – Radiographic
UT – Ultrasonic
PT – Dye Penetrant

Table 5-29. Design Bases for Residual Heat Removal System Operation

Residual Heat Removal System Start Up.	~4 hours after Reactor Shutdown
Reactor Coolant System Initial Pressure, psig	~385
Reactor Coolant System Initial Temperature, °F	≤350
Component Cooling Water Design Temperature, °F	105
Cooldown Time, Hours After Initiation Of Residual Heat Removal System Operation	~16
Reactor Coolant System Temperature At End Of Cooldown, °F	200
Decay Heat Generation At 20 Hours After Reactor Shutdown, BTU/hr	78.2×10^6

Table 5-30. Residual Heat Removal System Component Data

Residual Heat Removal Pump		
Number	2	
Design Pressure, psig	600	
Design Temperature, °F	400	
Design Flow, gpm	3000	
Design Head, ft	375	
Maximum Calculated Runout Flow (ECCS), gpm	3800	
NPSH Required at test 3980 gpm, ft	16	
NPSH Available at 3980 gpm, ft. NPSH available does not include the losses associated with the ECCS Sump Strainer. NPSH available will increase throughout the event as the containment sump pool temperature decreases.	33	
Assumed "Runout Flow" per SER Supplement 2	5300	
Resolution of Confirmatory Issue 22, gpm		
NPSH Required at 5300 gpm, ft.	22.75	
NPSH Available at 5300 gpm, ft. (Second case which includes 2 ft. of water above floor level inside containment as reviewed and approved by NRC in SER Supplement 2).	24.0	
Power, HP	400	
Residual Heat Exchanger		
Number	2	
Design Heat Removal Capacity, BTU/hr	30.96×10^6	
Estimated UA, BTU/hr °F	2.07×10^6	
	Tube-Side	Shell-Side
Design Pressure, psig	600	150
Design Temperature, °F	400	200
Design Flow, 1b/hr	1.48×10^6	2.48×10^6
Inlet Temperature, °F	140	105
Outlet Temperature, °F	119.0	117.5
Material	Austenitic Stainless Steel	Carbon Steel
Fluid	Reactor Coolant	Component Cooling Water

Table 5-31. Failure Mode and Effects Analysis-Residual Heat Removal System Active Components-Plant Cooldown Operation

Component	Failure Mode	Effect on System Operation[1]	Failure Detection Method[2]	Remarks
1. Motor operated gate valve ND2A (ND37A analogous)	a. Fails to open on demand ("open" manual mode CB switch selection).	a. Failure blocks reactor coolant, flow from hot leg of RC loop B through train A of RHRS. Fault reduces the redundancy of RHR coolant train provided. No effect on safety for system operation. Plant cooldown requirements will be met by reactor coolant flow from hot leg of RC loop C flowing through train B of RHRS, however, time required to reduce RCS temperature will be extended.	a. Valve position indication (closed to open position change) at CB; RC loop B hot leg pressure indication (NCP5120) at CB; RHR train A discharge flow indication (NDP5190) and low flow alarm at CB; and RHR pump discharge pressure indication (NDP5090) at CB	1. Valve is electrically interlocked with the containment sump isolation valve (NI185A), the RWST isolation valve (FW27A), the RHR to charging pump suction line isolation valve (ND28A), the residual spray valve (NS43A) and with a prevent open pressure interlock (PB-405A) of RC loop B hot leg. The valve can not be opened remotely from the CB if any of the indicated isolation valves is open or if RC loop pressure exceeds 385.5 psig.
2. Motor operated gate valve. ND1B (ND36B analogous).	a. Same failure modes as those stated for item #1.	a. Same effect on system operation as that stated for item #1.	a. Same methods of detection as those stated for item #1	1. Same remarks as those stated for item #1 except for pressure interlock (PB-403A) control.

Component	Failure Mode	Effect on System Operation[1]	Failure Detection Method[2]	Remarks
3. Residual heat removal pump A, (pump B analogous)	a. Fails to deliver working fluid.	a. Failure results in loss of reactor coolant flow from hot leg of RCS loop B through the train A of RHRS. Fault reduces the redundancy of RHR coolant trains provided. No effect on safety for system operation. Plant cooldown requirements will be met by reactor coolant flow from hot leg of RC loop C flowing through train B of RHRS. However, time required to reduce RCS temperature will be extended	a. Open pump switchgear circuit indication at CB; circuit breaker position monitor light for group monitoring of components at CB; common breaker trouble alarm at CB; RCS loop B hot leg pressure indication (NCP5120) at CB; RHR train A discharge flow indication at CB (NDP5190); and pump discharge pressure indication (NDP5090) at CB.	1. The RHRS shares components with the ECCS. Pumps are tested as part of the ECCS testing program (see Section 6.3.4) Pumps failure may also be detected during ECCS testing.

Component	Failure Mode	Effect on System Operation[1]	Failure Detection Method[2]	Remarks
4. Motor operated globe valve ND25A (ND59B analogous)	a. Fails to open on demand (open manual mode CB switch selection).	a. Failure blocks miniflow line to suction of RHR pump A during cooldown operation or checking boron concentration level of coolant in train A of RHRS. No effect on safety for system operation. Operator may establish miniflow for RHR pump A operation by opening CVCS letdown control valve (NV135) to allow flow to CVCS.	a. Valve position indication (closed to open position change) at CB.	1. Valve is automatically controlled to open when pump discharge is less than 533 gpm and to close when the discharge exceeds 1400 gpm. The valve protects the pump from dead heading during ECCS operation. No auto position for this valve. While the pump is ON, the valve may be manually positioned whenever the flowrate is between 533 gpm and 1400 gpm. While the pump is OFF, the valve automatically closes.

Component	Failure Mode	Effect on System Operation[1]	Failure Detection Method[2]	Remarks
	b. Fails to close on demand	b. Failure allows a portion of RHR heat exchanger A discharge flow to be bypassed to suction of RHR pump A. RHRS train A is degraded. No effect on safety for system operation. Cooldown of RCS within the specified cooldown rate may be accomplished through operator action of throttling flow control valve ND26 and with redundant RHRS train B.	b. Valve position indication (open to closed position change) and RHRS train A discharge flow indication (NDP5190) at CB.	

Component	Failure Mode	Effect on System Operation[1]	Failure Detection Method[2]	Remarks
5. Air diaphragm operated butterfly valve ND27 (ND61 analogous)	a. Fails to open on demand ("auto" mode CB switch selection on manual/auto station)	a. Failure prevents coolant discharged from RHR pump A from bypassing RHR heat exchanger A resulting in mixed mean temperature of coolant flow to RCS being low. RHRS train A is degraded for the regulation of controlling temperature of coolant. No effect on safety for system operation. Cooldown of RCS within established specification rate may be accomplished through operator action of throttling flow control valve ND26 and controlling cooldown with redundant RHRS train B.	a. Valve position indication closed to open position change) at CB; pump A discharge flow temperature recording (NDCR5060) at CB; and RHRS train A discharge to RCS cold leg flow indication (NDP5190) at CB.	1. Valve is designed to fail "closed" and is electrically wired so that solenoid of the air diaphragm operator is energized to open the valve. Solenoid valve receives "S" signal to close valve for ECCS operation. Valve is normally "closed" during power operations.

Component	Failure Mode	Effect on System Operation[1]	Failure Detection Method[2]	Remarks
	b. Fails to close on demand ("auto" mode CB switch selection on manual/auto station).	b. Failure allows coolant discharged from RHR pump A to bypass RHR heat exchanger A resulting in mixed mean temperature of coolant flow to RCS being high. RHRS train A is degraded for the regulation of controlling temperature of coolant. No effect on safety for system operation. Cooldown of RCS with in established specification rate may be accomplished through operator action of throttling flow control valve ND26 and controlling cooldown with redundant RHRS train B, however, cooldown time will be extended.	b. Same methods of detection as those stated above except open to closed valve position change indication at CB.	

Component	Failure Mode	Effect on System Operation[1]	Failure Detection Method[2]	Remarks
6. Air diaphragm operated butterfly valve ND26 (ND60 analogous)	a. Fails to close on demand for flow reduction through heat exchanger.	a. Failure prevents control of coolant discharge flow from RHR heat exchanger A resulting in loss in being able to adjust mixed mean temperature of coolant flow to RCS. No effect on safety for system operation. Cooldown of RCS within established specification rate may be accomplished by operator action of controlling cooldown with redundant RHRS train B.	a. Same methods of detection as those stated for item #5, except degree of valve being open position indication at CB.	1. Valve is designed to fail "open" and is electrically wired so that the solenoid of the air diaphragm operator is energized to close the valve. The solenoid receives "S" signal to open the valve. The valve is normally "open" during power operations.
	b. Fails to open on demand for increased flow through heat exchanger.	b. Same effect on system operation as that stated above.	b. Same methods as those stated above.	

Component	Failure Mode	Effect on System Operation[1]	Failure Detection Method[2]	Remarks
7. Motor operated globe valve (ND24A(ND58B analogous)	a. Fails closed	a. Failure blocks flow from train A of RHRS to CVCS letdown heat exchanger. Fault prevents (during the initial phase of plant cooldown) equalizing boron concentration of coolant in RHRS train A and in the RCS using the RHR cleanup line to CVCS. No effect on safety for system operation. Operator can balance boron concentration levels by cracking open flow control valve ND26 to permit flow to cold leg of loop B of RCS in order to balance levels using normal CVCS letdown flow. Later during cooldown, letdown flow comes from train B of RHRS.	a. CVCS letdown flow indication (NVP5530) at CB.	1. Valve is normally “closed” to align the RHRS for ECCS operation during plant power operation.

Component	Failure Mode	Effect on System Operation[1]	Failure Detection Method[2]	Remarks
8. Air diaphragm operated globe valve NV135.	a. Fails to open on demand.	a. Failure blocks flow from train A and B of RHRS to CVCS letdown heat exchanger. Fault prevents use of RHR cleanup line to CVCS for balancing boron concentration levels of RHRS trains A and B with RCS during initial cooldown operation. Later in plant cooldown RHRS letdown flow is blocked. No effect on safety for system operation. Operator can balance boron concentration as stated above for item 7. Water clarity can alternately be maintained utilizing the FW pump through KF system purification loop.	a. Valve position indication (degree of opening) at CB and CVCS letdown flow indication (NVP5530) at CB.	1. The valve is normally closed and designed to Fail closed.
				2. Valve is a component of the CVCS that interfaces with the RHRS during plant cooldown.

Component	Failure Mode	Effect on System Operation[1]	Failure Detection Method[2]	Remarks
9. Motor operated gate valve FW27A (FW55B analogous)	a. Fails to close	a. Failure to close results in loss of ability to open the associated trains RCS loop suction valve. In this case the alternate RHR loop is used for RHR cooling.	a. Valve position indication (open to closed position change) at CB and valve (closed) monitor light and alarm at CB.	1. Valve is a component of the ECCS that performs an RHR function during plant cooldown. Valve is normally "open" to align the RHRS for ECCS operation during plant power operation.

List of acronyms and abbreviations

Auto--Automatic	RC --Reactor Coolant
CB --Control Board	RCS --Reactor Coolant System
CVCS--Chemical and Volume Control System	RHR --Residual Heat Removal
ECCS--Emergency Core Cooling System	RHRS--Residual Heat Removal System
MO --Motor Operated	RWST--Refueling Water Storage Tank

Notes:

1. See list at end of table for definition of acronyms and abbreviations used.
2. As part of plant operation, periodic tests, surveillance inspections and instrument calibrations are made to monitor equipment and performance. Failures may be detected during such monitoring of equipment in addition to detection methods noted.

Table 5-32. Pressurizer Design Data

Design Pressure, psig	2485
Design Temperature, ^{0}F	680
Surge Line Nozzle Diameter, in	14
Heatup Rate of Pressurizer Using Heaters Only, $^{0}F/hr$	55
Internal Volume ft^3	1800

Table 5-33. Pressurizer Quality Assurance Program

[HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

	RT[1]	UT[1]	PT[1]	MT[1]
Heads				
1. Plates		yes		
2. Cladding			yes	
Shell				
1. Plates		yes		
2. Cladding			yes	
Heaters				
1. Tubing[2]		yes	yes	
2. Centering of element	yes			
Nozzle (Forgings)		yes	yes[3]	yes[3]
Weldments				
1. Shell, longitudinal	yes			yes
2. Shell, circumferential	yes			yes
3. Cladding			yes	
4. Nozzle Safe End	yes	yes[4]	yes[4]	
5. Instrument Connection			yes	
6. Support Skirt, Longitudinal Seam	yes			yes
7. Support Skirt to Lower Head		yes		yes
8. Temporary Attachments (after removal)				yes
9. All external pressure boundary welds after shop hydrostatic test				yes

Notes:

1. *RT – Radiographic*

   UT – Ultrasonic

   PT – Dye Penetrant

   MT – Magnetic Particle

2. *Or a UT and ET*
3. *MT or PT*
4. *Weld Overlay Installation, UT and PT*

Table 5-34. Pressurizer Relief Tank Design Data

Design Pressure, psig		100
Normal Operating Pressure, psig		3
Final Operating Pressure, psig		50
Rupture Disc Release Pressure, psig	Normal	91
	Range	86-100
Normal Water Volume, ft^3		1350
Normal Gas Volume, ft^3		450
Design Temperature, ^0F		340
Maximum Initial Operating Water Temperature, ^0F		120
Maximum Final Operating Water Temperature, ^0F		200
Total Rupture Disc Relief Capacity at 100 psig, 1b/hr		1.6×10^6
Cooling Time Required Following Maximum Discharge		
(Approximate), hr		1

Table 5-35. Relief Valve Discharge To The Pressurizer Relief Tank

Reactor Coolant System	
3 Pressurizer Safety Valves	Figure 5-3
3 Pressurizer Power-Operated Relief Valves	Figure 5-3
Residual Heat Removal System	
2 Residual Heat Removal Pump	Figure 5-17
Suction Line from the Reactor	
Coolant System Hot Legs	
Chemical and Volume Control System	
1 Seal Water Return Line	Figure 9-89
1 Letdown Line	Figure 9-89

Table 5-36. Reactor Coolant System Boundary Valve Design Parameters

Reactor Coolant Boundary Valves	Parameter
Design Pressure, psig	2485
Pre-Operational Hydrotest, psig	3107
a. Design Temperature, ^{0}F Other than pressurizer safety and power operated relief valves	650
b. Pressurizer safety and power operated relief valves	680

Table 5-37. Pressurizer Valves Design Parameters

Pressurizer Spray Control Valves	Parameters
Number	2
Design pressure, psig	2485
Design temperature, ^{0}F	650
Design flow for valves full open, each, gpm	450
Pressurizer Safety Valves	
Number	3
Maximum relieving capacity, ASME rated flow, 1b/hr (per valve)	420,000
Set pressure, psig	2485
Fluid	Saturated steam
Backpressure:	
Normal, psig	3 to 5
Design, psig	500
Pressurizer Power Relief Valves	
Number	3
Design pressure, psig	2485
Design temperature, ^{0}F	680
High pressure setpoint, psig	2335
Relieving capacity, 1b/hr (per valve)	210,000
Fluid	Saturated Steam
Low pressure setpoint, psig (NC-32B and NC-34A only)	400 psig
Relieving capacity, gpm (per valve)	1060
Fluid	Water (@60^{0}F)

Table 5-38. Component Supports. Loading Combinations and Code Requirements

Loading Combination	**Code or Stress Requirements**	
Normal and Upset Conditions	**Preliminary Design**	**Final Design**
1. DL + OL + LL	AISC with Allowable Stresses of F_s	ASME[2]
2. DL + OL + OBE		
Faulted Conditions		
3. DL + OL + SSE	Note 1.	ASME[2]
4. DL + OL + SSE + LOCA		

DL	=	Dead Load, including own weight of the support.
OL	=	Normal Operating Load: These loads are associated with plant operations in addition to weight of permanent equipment.
LL	=	Live Load, including construction loads.
OBE	=	Operating Basis Earthquake load.
SSE	=	Safe Shutdown Earthquake load.
LOCA	=	Accident loads including reactions due to pipe rupture and thermal loads.
AISC	=	"Specifications for Design, Fabrication and Erection of Structural Steel Buildings," Seventh Edition, 1973.
F_s	=	Steel allowable stresses as specified in AISC Part 1.
F_y	=	Yield stress of structural steel.
ASME	=	1974 ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, through Summer of 1974 addenda, including Appendix F and Appendix XVII.

Notes:

1. For loading combinations (3) and (4) which are ultimate loading conditions, the allowable stresses for the structural steel are as follows:

Type of Stress	**Allowable Stress**
Tension, Compression and Bending	$0.9\ F_y$
Shear	$0.55\ F_y$
Compression with Buckling	$1.7\ F_s$

2. Allowable stresses are not specified for the Reactor Coolant Pump bolts as material is not defined in Section III, Appendix I of the ASME Code. Allowable stresses shown for the preliminary design are used in the final design check of these bolts.

Table 5-39. Materials

Material used in these supports include:		
Plate	- SA-516 Grade 70 - SA-533, Class 1 - A-588 - SA-36	- SA-516 Grade 60 - SA-516 Grade 55 - SA-106, Grade B - SA-240, Type 304
Rod	- SA-306 Grade 70 - SA-306 Grade 60	
Bolts	- SA-637 Grade 688, Type 2 - 4340 (Modified)	- SA-325 - SA-193, GR B7
Forging	- A-471, Class 9 - SA-540 Grade B24	SA 540 Grade B22

Table 5-40. Reactor Vessel Material Surveillance Program - Withdrawal Schedule

Unit 1 Capsules	Vessel Location	Withdrawal Time (EOC)	EOC Date	EFPY at Withdrawal	Lead Factor	Fluence (n/cm^2 x 10^{19})	Reference
Z	301.5°	1	8/8/86	0.79[e]	3.85	0.292	WCAP-11527
Y	241°	6	7/10/92	4.98[e]	3.73	1.31	WCAP-13720
W	121.5°	14	11/18/03	14.69[e]	4.00	3.51	Note 6
X	238.5°	10	11/28/97	9.29[e]	3.88	2.41	WCAP-15117 Note 3
U	58.5°	10	11/28/97	9.29[e]	3.88	2.41	WCAP-15117 Note 3
V	61°	10	11/28/97	9.29[e]	3.72	2.31	WCAP-15117
Ex-vessel Dosimetry	NA	16	11/11/2006	17.35[e]	NA	--	WCAP-16869-NP, Rev 1
Unit 2 Capsules							
Z	301.5°	1	12/23/87	0.86	4.13	0.323	WCAP-11941
X	241°	5	1/23/93	4.52	4.14	1.23[a]	WCAP-13875
W	121.5°	14	3/17/06	13	4.28	3.0[d]	Note 3
U	58.5°	Note 4	Note 4	Note 4	---	---	--
Y	238.5°	9	9/13/98	9.24	4.33	2.49	WCAP-15243
V	61°	9	9/13/98	9.24	4.13	2.38[b][c]	WCAP-15243
Ex-vessel Dosimetry	NA	14	4/5/2006	13	NA	--	--

a. Approximate fluence at vessel ¼ thickness location, at 32 EFPY

b. Approximate fluence at vessel inner wall location, at 32 EFPY

c. Approximate fluence at vessel ¼ thickness location, at 54 EFPY

d. Approximate fluence at vessel inner wall location at 54 EFPY

e. The fluence evaluation supporting this effort did not consider 0.11 Effective Full Power Years (EFPY) of neutron exposure resulting from pre-commercial operation of Catawba Unit 1, between January 1985 and June 1985. The impact of this omission on the fluence evaluation results has been assessed to be negligible, and the results remain valid within the 20% uncertainty criterion for fluence calculations. Future fluence evaluations will consider this pre-commercial operation phase of Catawba Unit 1.

Notes:

1. EFPY= Effective Full Power Year
2. EOC=End of Cycle
3. Capsule specimens have been removed and stored at Westinghouse after reading dosimetry. These specimens are available for testing or additional irradiation if ever deemed necessary.

4. Capsule U is not available for irradiation and testing.
5. For CNS-2 Capsule X was discovered in the Y location, Capsule Y was in the X location. Values listed are actual corrected as-found locations.
6. CNS-1 Capsule W was placed in the spent fuel pool following removal.

Table 5-41. Reactor Coolant System Pressure Isolation Valves

Valve Number	Function
NI59	Accumulator Discharge
NI60	Accumulator Discharge
NI70	Accumulator Discharge
NI71	Accumulator Discharge
NI81	Accumulator Discharge
NI82	Accumulator Discharge
NI93	Accumulator Discharge
NI94	Accumulator Discharge
NI124	Safety Injection (Hot Leg)
NI125	Residual Heat Removal (Hot Leg)
NI126	Safety Injection (Hot Leg)
NI128	Safety Injection (Hot Leg)
NI129	Residual Heat Removal (Hot Leg)
NI134	Safety Injection (Hot Leg)
NI156	Safety Injection (Hot Leg)
NI157	Safety Injection (Hot Leg)
NI159	Safety Injection (Hot Leg)
NI160	Safety Injection (Hot Leg)
NI165	Safety Injection/Residual Heat Removal (Cold Leg)
NI167	Safety Injection/Residual Heat Removal (Cold Leg)
NI169	Safety Injection/Residual Heat Removal (Cold Leg)
NI171	Safety Injection/Residual Heat Removal (Cold Leg)
NI175	Safety Injection/Residual Heat Removal (Cold Leg)
NI176	Safety Injection/Residual Heat Removal (Cold Leg)
NI180	Safety Injection/Residual Heat Removal (Cold Leg)
NI181	Safety Injection/Residual Heat Removal (Cold Leg)
ND1B	Residual Heat Removal
ND2A	Residual Heat Removal
ND36B	Residual Heat Removal
ND37A	Residual Heat Removal

Table 5-42. RT PTS Calculations for Catawba Unit 1 Beltline Region Materials at 54 EFPY

Material	CF	Fluence @ 54 EFPY (10^{19} n/cm^2)	FF	RT_{NDT} (U)	Δ RT $_{PTS}$	M	RT $_{PTS}$ °F
Upper Shell Forging 06	123.5	0.116	0.4472	-26	55.2	34.0	63
Intermediate Shell Forging 05	58	2.60	1.2559	-8	72.8	34.0	99
→ Using Surveillance Capsule Data	28.5	2.60	1.2559	-8	35.8	17.0	45
Lower Shell Forging 04	26	2.60	1.2559	-13	32.7	32.7	52
Bottom Head Ring 03	37	0.195	0.5634	14	20.8	20.8	56
Upper to Intermediate Shell Circumferential Weld W06	41	0.116	0.4472	10	18.3	18.3	47
Intermediate to Lower Shell Circumferential Weld W05	54	2.60	1.2559	-51	67.8	56.0	73
→ Using Surveillance Capsule Data	28.5	2.60	1.2559	-51	35.8	28.0	13
Lower Shell to Bottom Head Ring Weld W04	41	0.195	0.5634	10	23.1	23.1	56

Data Sources:

All Materials: Westinghouse Report WCAP-17669-NP, "Catawba Unit 1 Measurement Uncertainty Recapture (MUR) Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations," dated June 2013.

Table 5-43. RT PTS Calculations for Catawba Unit 2 Beltline Region Materials at 54 EFPY

Material	CF	Fluence @ 54 EFPY (10^{19} n/cm^2)	FF	$RT_{NDT(U)}$	ΔRT_{PTS}	M	RT_{PTS} °F
Bounding Nozzle Shell Material	77	0.11	0.43	50	32.9	32.9	115.8
Bounding Nozzle Weld Material	81	0.11	0.43	-40	34.6	34.6	29
Intermediate Shell Plate B8605-1	51	3.16	1.3	15	66.3	34	115
→ Using Surveillance Capsule Data	44	3.16	1.3	15	57.2	17	89
Intermediate Shell Plate B8605-2	51	3.16	1.3	33	66.3	34	133
Intermediate Shell Plate B8616-1	31	3.16	1.3	12	40.3	34	86
Lower Shell Plate B8806-1	37	3.16	1.3	6	48.1	34	88
Lower Shell Plate B8806-2	37	3.16	1.3	-10	48.1	34	72
Lower Shell Plate B8806-3	37	3.16	1.3	8	48.1	34	90
Intermediate, Lower and Intermediate to Lower Shell Weld Seams	37.3	3.16	1.3	-80	48.5	48.5	17
→ Using Surveillance Capsule Data	33.4	3.16	1.3	-80	43.4	28	-9

Data Sources:

Bounding Nozzle Materials: Internal calculation DPC-1201.01-00-0006, CNC-1201.01-00-0020, “USE and RTPTS Values for Reactor Vessel Nozzle Region Locations”, Rev. 0, dated July 2002.

All Other Beltline Materials: WCAP-15449, Rev. 1, “Evaluation of Pressurized Thermal Shock for Catawba and McGuire Units 1 & 2 @ 54 EFPY”, dated October 2002.

Table 5-44. Evaluation of Upper Shelf Energy for Catawba Unit 1 Beltline Region Materials at 54 EFPY

Material	Weight % of Cu	¼ T EOL Fluence (10^{19} n/cm^2)	Unirradiated USE (ft-lb)	Projected USE Decrease (%)	Projected USE @ 54 EFPY (ft-lb)
Upper Shell Forging 06	0.16	0.070	101	14	87
Intermediate Shell Forging 05	0.09	1.565	134	21	106
→ Using Surveillance Capsule Data	0.09	1.565	134	10	121
Lower Shell Forging 04	0.04	1.565	134	21	106
Bottom Head Ring 03	0.06	0.117	68	12	60
Upper to Intermediate Shell Circumferential Weld W06	0.03	0.070	92	10	83
Intermediate to Lower Shell Circumferential Weld W05	0.04	1.565	130	21	103
→ Using Surveillance Capsule Data	0.04	1.565	130	8	120
Lower Shell to Bottom Head Ring Weld W04	0.03	0.117	92	12	81

Data Sources:

All Materials: Westinghouse Report WCAP-17669-NP, "Catawba Unit 1 Measurement Uncertainty Recapture (MUR) Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations," dated June 2013.

Table 5-45. Evaluation of Upper Shelf Energy for Catawba Unit 2 Beltline Region Materials at 54 EFPY

Material	Weight % of Cu	¼ T EOL Fluence (10^{19} n/cm^2)	Unirradiated USE (ft-lb)	Projected USE Decrease (%)	Projected USE @ 54 EFPY (ft-lb)
Bounding Nozzle Shell Material	0.11	0.063	65	10.0	58.5
Bounding Nozzle Weld Material	0.16	0.063	102	16.0	85.7
Intermediate Shell Plate B8605-1	0.08	1.88	89	6.6	90
Intermediate Shell Plate B8605-2	0.08	1.88	82	22	64
Intermediate Shell Plate B8616-1	0.05	1.88	92	22	72
Lower Shell Plate B8806-1	0.06	1.88	83	22	65
Lower Shell Plate B8806-2	0.06	1.88	102	22	80
Lower Shell Plate B8806-3	0.06	1.88	105	22	82
Intermediate Shell Longitudinal Weld Seams 101-142A, B, C	0.04	1.13 1.88 1.88	146	10 11 11	131 130 130
Intermediate Shell to Lower Shell Circumferential Weld Seams	0.04	1.88	146	11	130
Lower Shell Longitudinal Weld Seams 101-124 A, B, C	0.04	1.88 1.13 1.88	146	11 10 11	130 131 130

Data Soruces:

Bounding Nozzle Materials: Internal calculation DPC-1201.01-00-0006, CNC-1201.01-00-0020, "USE and RTPTS Values for Reactor Vessel Nozzle Region Locations", Rev. 0, dated July 2002.

All Other Beltline Materials: Westinghouse Letter DPC 00 069, dated October 22, 2000

Table 5-46. Summary of Reactor Coolant System Leakage Detection Instrumentation Exceptions and Comments to Regulatory Guide (RG) 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", (Rev. 0)

RG 1.45 Regulatory Position	Exception/Comment
C.2 Leakage to the primary reactor containment from unidentified sources should be collected and the flow rate monitored with an accuracy of one gallon per minute or better.	Incore sump alarm will detect a 1 gpm input within 4 hours of leakage reaching the sump.
C.5 The sensitivity and response time of each leakage detection system in regulatory position 3 above employed for unidentified leakage should be adequate to detect a leakage rate, or its equivalent, of one gpm in less than one hour.	Exception taken for containment particulate radiation monitor and incore sump level alarm. The particulate radiation monitor sensitivity will be 10-9 uCi/cc. The particulate monitor alarm setting wil be as low as practicable based on background and sufficiently high enough to prevent spurious alarms. Operability will be based on the sensitivity and surveillance testing. The incore sump alarm will actuate within 4 hours of leakage reaching the sump. Clarified Containment Floor and Equipment sump and Containment Ventilation Unit Condensate Drain Tank Level sensitivity of 1 gpm after leakage has reached the sump/tank.
C.6 The leakage detection systems should be capable of performing their functions following seismic events that do not require plant shutdown. The airborne particulate radioactivity monitoring system should remain function when subjected to the SSE.	Exception taken for the radioactivity monitoring system design for a seismic event.
C.7 Indicators and alarms for each leakage detection system should be provided in the main control room. Procedures for converting various indications to a common leakage equivalent should be available to the operators. The calibration of the indicators should account for needed independent variables.	Exception taken for incore sump indication in the control room – alarm only. The particulate radiation monitor and incore sump will alarm during the presence of a leak but are not converted to a leakage equivalent (e.g. gpm).
C.8 The leakage detection systems should be equipped with provisions to readily permit testing for operability and calibration during plant operation.	Exception taken for incore sump level alarm for testing and calibration during plant operation.