Appendix 5A. Tables

Table 5-1. System Design and Operating Parameters[1]

Unit Design Life, Years	40
Nominal Operating Pressure, psig	2235
Total System Volume, including pressurizer and surge line, ft^3	13,050
System Liquid Volume, including pressurizer water level (60% full), ft^3	11,155
Deleted Per 2014 Update	
System Thermal and Hydraulic Data Temperatures (Based on Thermal Design Flow)	
NSSS Power, MWt	3,483
NSSS Power, 10^6 BTU/hr	11,892
Licensed Reactor Power, MWt	3,469
Licensed Reactor Power, 10^6 BTU/hr	11,837
Thermal Design Flow, GPM/Loop	100,200
Total Reactor Coolant Flow, lb/hr	148×10^6
Reactor Vessel Inlet Temperature, °F	555.6
Reactor Vessel Outlet Temperature, °F	614.6
Steam Generator Primary Outlet Temperature, °F	555.1
Steam Generator at Full Power, psia	1020.7
Steam Generator Steam Temperature, °F	548.7
Steam Flow at Full Power, lb/hr (total)	15.5×10^6
Feedwater Inlet Temperature, °F	442.0
Pressurizer Spray Rate, max., gpm	900
Pressurizer Heat Capacity, kw	1800
Pressurizer Relief Tank Volume, ft^3	1800
Flows and Pressure Drops (Based on Best Estimate Flow)	
Best Estimate Flow, GPM/Loop	99,295
Pump Head at B.E. Flow, ft	276
Reactor Vessel ΔP, psi	46.7
Steam Generator ΔP, psi	33
Piping ΔP, psi	8.0

Note:

1. These values are approximate. Exact design values may be obtained from the NSSS Vendor (Westinghouse) or to the steam generator vendor (BWI)

Table 5-2. has been incorporated into Table 5-49

Table 5-3. Load Combinations and Operating Conditions

Load Combination (Except for RSGs - See Below)	Operating Condition
1. Normal (deadweight, thermal and pressure)	Normal Condition
2. Normal and Operating Basis Earthquake	Upset Condition
3. Normal and Safe Shutdown Earthquake	Faulted Condition
4. Normal and Design Basis Accident	Faulted Condition
5. Normal and Safe Shutdown Earthquake and Design Basis Accident	Faulted Condition

The following load combinations shall be used for the structural evaluation of the RSG (pressure boundary)

Loading Conditions	Service Loads/Combinations	ASME SECTION III Service Stress SUBSECTION NB Limit Level
Design	Deadweight Operating Basis Earthquake (OBE) Design Pressure Design Temperature Design Flow Thermal Internal Design Mechanical Loads	Design
Normal	Deadweight Thermal Internal Normal Mechanical Loads Normal Condition Transients	A
Upset	Deadweight Thermal OBE Internal Upset Mechanical Loads Upset Condition Transients	B
Emergency (See Note 1.)	Deadweight Thermal Internal Emergency Mechanical Loads Emergency Condition Transients	C
Faulted	Deadweight Thermal Safe Shutdown Earthquake (SSE) Internal Faulted Mechanical Loads Pipe Rupture Loads Faulted Condition Transients	D

Load Combination (Except for RSGs - See Below)	Operating Condition
Note:	
1. This loading condition is not part of McGuire Nuclear Station Design Bases, but is included here for new RSG only.	

Table 5-4. Faulted Condition Stress Limits for Class 1 Components([4,5])

System (or Subsystem) Analysis		Components Analysis	Stress Limits for Components	Test
		P_m	$P_m + P_b$	
ELASTIC	ELASTIC	Smaller of 2.4 S_m and 0.70 S_u	Smaller of 3.6 S_m and 1.05 S_u Note 2	
	Plastic	Larger of 0.70 S_u or S_y + 1/3(S_u - S_y)	Larger of 0.70 S_{ut} or S_y + 1/3 (S_{ut} - S_y)	
PLASTIC	Limit Analysis	0.9 L_1	Note 1	
	Plastic	Larger of 0.70 S_u or	Larger of 0.70 S_{ut} or	0.8 L_T
	Elastic	S_y + 1/3 (S_u - S_y)	S_y + 1/3 (S_{ut} - S_y)	Note 3

Note:

1. L_1 = Lower bound limit load with an assumed yield point equal to 2.3 S_m.
2. These limits are based on a bending shape of 1.5 for simple bending cases with different shape factors, the limits will be changed proportionally.
3. L_T = The limits established for the analysis need not be satisfied if it can be shown from the test of a prototype or model that the specified loads (dynamic or static equivalent) do not exceed 80 percent of L_T, where L_T is the ultimate load or load combination used in the test. In using this method, account shall be taken of the size effect and dimensional tolerances (similitude relationships) which may exist between the actual component and the tested models to assure that the loads obtained from the test are a conservative representation of the load carrying capability of the actual component under postulated loading for faulted conditions.
4. The use of these stress limits on Westinghouse-supplied components will be shown to be no less conservative than the limits of Appendix F to the ASME Code.

   If plastic component analysis is used with elastic system analysis or with plastic system analysis, the deformations and displacement of the individual system members will be shown to be no larger than those which can be properly calculated by the analytical methods used for the system analysis.

   S_y = Yield stress at temperature
   S_u = Ultimate stress from engineering stress-strain curve at temperature
   S_{ut} = Ultimate stress from the true stress-strain curve at temperature
   S_m = Stress intensity from ASME Section III at temperatue

System (or Subsystem) Analysis	Components Analysis	Stress Limits for Components	Test

5. Babcock and Wilcox replacement steam generators are designed to the requirements of Section III of the ASME Code for faulted conditions.

Table 5-5. Active and Inactive Valves in the Reactor Coolant System Boundary

Line	Valve	Line Size	Type A-Active [2] I - Inactive	Normal Position	Post-LOCA Position
RHR Suction	1)Motor gate	14"	I	Closed (interlocked) with RCS pressure)	Closed
	2)Motor gate	14"	I	Closed (interlocked with second RCS pressure instrument)	Closed
Loop Drains (each loop) Charging	1)Manual globe	2"	I	Closed	Closed
	2)Manual globe	2"	I	Closed	Closed
	1)Check	3"	A	Open	Closed
	2)Check	3	A	Open	Closed
RHR Return or Cold Leg Connections (each loop)	1)Check	10"	A	Closed	Open - for lowhead injection and accumulator injection
	2)Check	6"	A	Closed	Open - for lowhead injection
Accumulator	1)Check	10"	A	Closed	Open - for accumulator and low head injection
	2)Check	10"	A	Closed	Open - for accumulator injection
SIS - Boron Injection Tank (each loop)	1)Check	1½"	A	Closed	Open - for boron injection
	2)Manual Throttle	1½"	I	Open	Open

Line	Valve	Line Size	Type A-Active [2] I - Inactive	Normal Position	Post-LOCA Position
	3)Check	1½"	A	Closed	Open - for boron injection
High Head Hot Leg Connections (each loop)	1)Check	6"	A	Closed	Open - for highhead recirculation
	2)Check	2"	A	Closed	Open - for highhead recirculation
Low Head Hot Leg Connections (each loop)	1)Check	6"	A	Closed	Open - for lowhead recirculation
	2)Check	8"	A	Closed	Open - for lowhead recirculation
High Head Cold Leg Connections	1)Check	10"	A	Closed	Open - for high head injection and recirculation
	2)Check	2"	A	Closed	Open - For high head injection and recirculation
Excess letdown	1)Manual globe	1"	I	Open (Locked)	Open
	2)Air-op globe	1"	I(1)(5)	Closed (fail close)	Closed – on safety injection signal
	3)Air-op globe	1"	I(1)(5)	Closed (fail close)	Closed – on safety injection signal
Letdown	1)Manual globe	3"	I	Open	Open
	2)Air-op gate	3"	I(3)	Open (fail close)	Closed - on low pressurizer level signal
	3)Air-op gate	3"	I(3)	Open (fail close)	Closed - on low pressurizer level signal
Alternate Charging	1)Check	3"	A[1]	Closed	Closed

Line	Valve	Line Size	Type A-Active [2] I - Inactive	Normal Position	Post-LOCA Position
	2)Check	3"	A[1]	Closed	Closed
Pressurizer relief Valves (to PRT)	1)Motor gate	3"	I	Open	Open
	2)Air-op globe	3"	A[1]	Closed (fail close)	Closed - low pressurizer pressure signal
Pressurizer Safety Valves (to PRT)	1)Safety Valve	6"	A	Closed	Closed
Auxiliary Spray (from CVCS)	1)Check	2"	A	Closed	Closed
	2)Air-op globe	2"	I [(4)]	Closed (fail close)	Closed - operator action

Note:

1. There is a possibility for these valves to be open when the accident occurs.
2. Active here means the valve must change position to perform a safety or shutdown function and is not intended to designate that the valve is postulated to fail.
3. LOCA isolation assured via ‘S_t’ closure of the downstream letdown line containment isolation valves.
4. LOCA isolation assured via ‘S_s’ closure of the upstream changing line outboard series isolation valves, and inboard containment isolation check valve.
5. LOCA isolation assured by ‘S_t’ closure of downstream containment isolation valves.

Table 5-6. Deleted Per 1998 Update

Table 5-7. Applicable Code Addenda For Class 1 Equipment Within RCPB

1. Reactor Vessel	(Unit 1)	ASME III, 1971 Edition thru Summer 1971
	(Unit 2)	ASME III, 1971 Edition thru Winter 1971
Steam Generator		ASME III, 1986 Edition No Addendum
Pressurizer		ASME III, 1971 Edition
CRDM Housing		
Full Length		ASME III, 1971 Edition thru Summer 1971
Part Length		ASME III, 1971 Edition
CRDM Head Adapter		ASME III, 1971 Edition
Reactor Coolant Pump		ASME III, 1971 Edition thru Summer 1972
Reactor Coolant Pipe		ASME III, 1971 Edition thru Winter 1971
Surge Lines		ASME III, 1971 Edition thru Winter 1971
2. Many of these plants are purchased via multi-plant contracts. Therefore, the purchase order dates are not necessarily indicative of the applicable code addenda and it is not meaningful to supply them. As indicated in one (1) above, the code addenda required by the 10CFR 50.55a are met based on a CP date of 2/28/73.		

Table 5-8. Reactor Coolant Loop Piping Stress Analysis Summary[1]

	Hot Leg			Crossover Leg			Cold Leg		
Evaluation	**Max.**	**Location**	**Allowable**	**Max.**	**Location**	**Allowable**	**Max.**	**Location**	**Allowable**
Eq. 9 design stress intensity (ksi)	21.80	Hot Leg End of 50° Elbow	26.7	22.72	Crossover Leg Middle of 40° Elbow	26.7	25.41	Cold Leg Middle of 22° Elbow	26.7
Eq. 9 faulted stress intensity (ksi)	48.33	Hot Leg End of 50° Elbow	53.4	40.56	Crossover Leg Middle of 40° Elbow	53.4	51.44	Cold Leg End of 22° Elbow	53.4
Eq. 12 stress (ksi)	37.28	Hot Leg Middle of 50° Elbow	53.4	10.45	Middle of Crossover Leg 90° Elbow RCP Side	53.4	10.78	Cold Leg Middle of 22° Elbow	53.4
Eq. 13 stress intensity range (ksi)	39.8	RPV outlet nozzle	53.4	37.8	SG Outlet Nozzle	53.4	38.1	RPV Inlet Nozzle	53.4
Fatigue Usage Factor	0.6576	RPV outlet Nozzle	1.0	0.0511	SG Outlet Nozzle	1.0	0.106 8	RPV Inlet Nozzle	1.0

Note :

1. The stress intensities tabulated above from the Westinghouse analyses remain the bounding design basis for the reactor coolant loop piping. Stress intensities calculated for steam generator replacement are less than those shown.

Table 5-9. Relief Valve Discharge to the Pressurizer Relief Tank

Reactor Coolant System	
3 Pressurizer Safety Valves 3 Pressurizer Power Operated Relief Valves	Figure 5-1 Figure 5-1
Safety Injection System	
2 SIS Discharge to Hot Leg 1 SIS Discharge to Cold Legs 1 SIS Pump Suction	Figure 6-177 Figure 6-177 Figure 6-177
Residual Heat Removal System	
1 RHR Pump Suction from RCS Hot Leg 2 RHR Pump Discharge to Cold Legs 1 RHR Pump Discharge to Hot Legs	Figure 5-28 Figure 5-28 Figure 5-28
Chemical and Volume Control System	
1 Charging Pump Suction 1 Seal Water Return Line 1 Letdown Line	Figure 9-96 Figure 9-96 Figure 9-98
Containment Spray System	
2 CS Pump Suction from Containment Sump	Figure 6-194

Table 5-10. Reactor Coolant System Pressure Settings

Hydrostatic Test Pressure (Cold)	3107 psig
Design Pressure	2485
Nominal Operating Pressure	2235
High Pressure Reactor Trip	2385
Low Pressure Reactor Trip	1945
Safety Valves Lift Setpoint	2485
Power Operated Relief Valves NC-32B & NC-36B Lift Setpoint[3]	2335
Power Operated Relief Valve NC-34A Lift Setpoint[1,2,3]	+100 psi
Pressurizer Spray Valve Begin to Open[1]	+25
Proportional Spray Full On[1]	+75
Proportional Heaters Off[1]	+15
Proportional Heaters Full On[1]	-15
Backup Heaters On[1]	-25
High Pressure Alarm	
uncompensated pressure signal	2310 psig
compensated pressure signal	+75 psi
Low Pressure Alarm[1]	-25
PORV NC-32B and NC-34A Open (RCS Cold Leg ≤300°F)[3]	380

Note:

1. Input to this function is a compensated pressure signal
2. NC-34A is subjected to a 1 second lag in the initiating lift signal.
3. Each of the PORVs have a 2 second valve stroke time.

Table 5-11. Reactor Coolant Pressure Boundary Materials Class 1 Primary Components

Reactor Vessel Components	
Shell & Head Plates (other than core region)	SA533 Gr A, B or C, Class 1 or 2 (Vacuum treated)
Shell, Flange & Nozzle Forgings	SA508 Class 2 or 3
Nozzle Safe Ends	SA182 Type F304 or F316
CRDM & ECCS Appurtenances - Upper Head	SB166 or 167 and SA182 Type F304
Instrumentation tube Appurtenances-	SB166 or 167 and SA182 type 304,
Lower Head	F304L or F316
Closure Studs	SA540 Class 3 Gr B23 or B24
Closure Nuts	SA540 Class 3 or Gr B23 or B24
Closure Washers	SA540 Class 3 Gr B23 or B24
Core Support Pads	SB166 with Carbon less than 0.10%
Monitor Tubes & Vent Pipe	SA312 or 376 Type 304 or 316 or SB167
Vessel Supports, Seal Ledge & Head Lifting Lugs	SA516 Gr 70 Quenched & Tempered or SA533 Gr A, B, or C, Class 1 or 2. (Vessel Supports may be of weld metal buildup of equivalent strength)
Cladding	Stainless steel weld Metal Analysis A-7 and Ni-Cr-Fe Weld Metal F-Number 43
Steam Generator Components	
Pressure Plates	SA533 Gr B, Class 1
Pressure Forgings	SA508 Class 3
Nozzle Safe Ends	SA336-F316N/316LN
Channel Heads	SA-508 C13
Primary Divider Plate	SB-168 NO6690
Tubes	SB-163 Code Case N-20-3 Alloy 690
Cladding	SFA 5.9 ER 309L/ER 308L (equivalent to 304L) SFA 5.14 ER Ni-Cr3 (equivalent to Alloy 600, Inco82)
Closure Bolting	SA 193 Gr B-7
Closure Nuts	SA 194 Gr 7
Pressurizer Components	
Pressure Plates	SA533 Gr A, B, or C, Class 1 or 2
Pressure Forgings	SA508 Class 2 or 3

Nozzle Safe Ends	SA182 or 376 Type 316 or 316L and Ni-Cr-Fe Weld Metal F-Number 43
Cladding	Stainless Steel Weld Metal Analysis A-7 and Ni-Cr-Fe Weld Metal F-Number 43
Closure Bolting	SA540
Pressurizer Safety Valve Forgings	SA182 Type F316
Reactor Coolant Pump	
Pressure Forgings	SA182 Type 304, 316, or 348
Pressure Castings	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 Gr B7 or B8 SA540 Gr B23 or B24, SA453 Gr 660
Flywheel	SA533 Gr B, Class 1
Part Length Mechanism	
Pressure Housing	SA182 or SA312 Seamless Gr 304 and Code Case 1337-3
Bar Material	SA479 Type 304
Welding Materials	SFA 5.4 and 5.9 Type 308 and 308L and Ni-Cr-Fe Weld Metal F-Number 43
Reactor Coolant Piping	
Reactor Coolant Pipe	Code Case 1423-a Gr F302N or 316N, or SA351 Gr CF8A or CF8M centrifugal castings
Reactor Coolant Fittings	SA351 Gr CF8A or CF8M
Branch Nozzles	SA182 Gr F304 or 316 or Code Case 1423-1 Gr F304N or 316N
Surge Line & Loop Bypass	SA376 Type 304 or 316 or Code Case 1423-1 Gr F304N or 316N
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

Auxiliary Piping Valves (Class 1)	SA182 Type 304 or 316 or SA351 GR CF8, CF8A or CF8M
Welding Materials	SFA 5.4 and 5.9 Type 304 or 308L
Control Rod Drive Mechanism	
Pressure Housing	SA182 Gr F304 or SA351 Gr CF8
Pressure Forgings	SA182 Gr F304 or SA336 Gr F8
Bar Material	SA479 Type 304
Welding Materials	SFA 5.4 and 5.9 Type 308 or 308L

Note:

1. UPDATING INFORMATION CONTAINED IN THIS TABLE IS NOT REQUIRED. The material information contained in this Table is "Historical" in nature and is intended to be used for reference only. Actual materials are procured in accordance with the applicable equipment specification and ASME code requirements and are documented on the approved manufacturers drawings.

Table 5-12. Reactor Coolant Pressure Boundary Materials Class 1 and 2 Auxiliary Components

Valves	
Motor and Manual Operated Gate and Check Valves	
Bodys	SA182 Gr F316
Bonnets	SA182 Gr F316 or SA351 Gr CF8 or CF8M
Discs	SA182 Gr 316 or SA479 Type XM-19 Cond A
Stems	SA564 Type 630 Cond 1100°F Heat Treatment and SB637 Type 07718
Closure Bolting & Nuts	SA453 Gr 660 and SA194 Gr B6
Bonnet Retainer	SA479 Type S21800
Air Operated Valves	
Bodys	SA182 Gr F316 or SA351 Gr CF8 or CF8M
Bonnets	SA182 Gr F316 or SA351 Gr CF8 or CF8M
Discs	SA182 Gr F316 or SA546 Gr 630 Cond 1100°F Heat Treatment
Stems	SA182 Gr F318 or SA564 Gr 630 Cond 1100°F Heat Treatment or SA637 Gr 688 Type 2
Closure Bolting & Nuts	SA453 Gr 660 and SA194 Gr B6
Bonnet Retainer	SA182 Gr F316
Auxiliary Relief Valves	
Forgings	SA182 Gr F316
Disc	SA479 Gr 316
Miscellaneous Valves (2 inches and less)	
Bodys	SA479 Type 316 or SA351 Gr CF8 SA182 Type F316
Bonnets	SA479 Type 316 or SA351 Gr CF8
Disc	SA479 Type 316
Stems	SA479 Type 410 or Type 304 A276 Type 410 Cond. H
Closure Bolting & Nuts	SA453 Gr 660 and SA193 Gr B6
Auxiliary Heat Exchangers	
Heads	SA182 Gr F304 or SA240 Type 304 or 316
Flanges	SA182 Gr F304 or F316

Flange Necks	SA182 Gr F304 or SA240 Type 316 or SA312 Type 304 Seamless
Tubes	SA213 Type 304
Tube Sheets	SA240 Type 304 or 316 or SA182 Gr F304 or SA515 Gr 70 with Stainless Steel Weld Metal Analysis A-7 Cladding
Shells	SA351 Gr CF9
Pipe	SA312 Type 304 Seamless
Auxiliary Pressure Vessels Tanks, Filters, etc.	
Shells & Heads	SA240 Type 304 or SA264 Type 304 Clad to SA516 Gr 70 or SA516 with Stainless Steel Weld Metal Analysis A-7 Cladding
Flanges & Nozzles	SA182 Gr F304 and SA105 or SA350 Gr LF2 with Stainless Steel Weld Metal Analysis A-7 Cladding
Piping	SA312 Type 304 or Type 316 Seamless
Pipe Fittings	SA403 WP304 Seamless
Closure Bolting & Nuts	SA193 Gr B7 or B8 and SA194 Gr 2D
Auxiliary Pumps	
Pump Casings & Heads	SA351 Gr CF8 or CF8M, SA182 Gr F304 or F316
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
Pipe Fittings	SA403 Gr WP316L Seamless
Closure Bolting & Nuts	SA193 Gr B6, B7 or B8M and SA194 Gr 2H or Gr 8M

Note:

1. UPDATING INFORMATION CONTAINED IN THIS TABLE IS NOT REQUIRED. The material information contained in this Table is "Historical" in nature and is intended to be used for reference only. Actual materials are procured in accordance with the applicable equipment specification and ASME code requirements and are documented on the approved manufacturers drawings.

Table 5-13. Reactor Vessel Internals Materials for Emergency Core Cooling

Forgings	SA182 Type F304; SA182 304L (Unit 2 only)
Plates	SA240 Type 304
Pipes	SA312 Type 304 Seamless or SA376 Type 304
Tubes	SA213 Type 304; SA213 304L (Unit 2 only)
Bars	SA479 Type 304 & 410; SA276 Type XM-19 Nitronic 50, SA276 Type 304L (Unit 2 only)
Castings	SA351 Gr CF8 or CF8A
Bolting	SA (Pending) Westinghouse PD Spec. 70041EA
Nuts	SA193 Gr B-8
Locking Devices	SA479 Type 304
Weld Buttering	Stainless Steel Weld Metal Analysis A-7

Table 5-14. Reactor Coolant Chemistry Specification

Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration) at 250°C
Oxygen, ppm, maximum	Oxygen concentration of the reactor coolant is maintained below 0.1 ppm for operation above 250°F. Hydrazine may be used to chemically scavenge oxygen during the heatup.
Chloride, ppm, maximum	0.15
Fluoride, ppm, maximum	0.15
Hydrogen, cc(STP/kg H_2O)	
Normal Power Operation	25 - 50
Unit Refueling Shutdown	Less than 5
Total Suspended Solids, ppm, maximum	1.0
pH Control Agent (Li^7OH)	Variable, as necessary for pH control program
Boric Acid, ppm B	Variable from 0 to approximately 3000

Table 5-15. McGuire Unit 1 Reactor Vessel Toughness Table

Component	Material Specification/ Weld Number	Code Number	Heat Number/ Flux Type (Lot)	Cu (%)	Ni (%)	P(%)	T_{NDT} (°F)	RT_{NDT} (°F)	USE (ft-lb)
Closure head dome	A533BCL.1	B5086-1	A-8354-1	0.11	0.48	0.010	20	37[2]	65[2]
Closure head segments	A533BCL.1	B5087	A-8149-2	0.11	0.62	0.008	10	10[2]	89[2]
Closure head flange	A508CL.2	B5002	123X369VAI	--	0.75	0.010	40[2]	40[2]	101[2]
Vessel flange	A508CL.2	B4701	122W201VAI	--	0.73	0.010	29[2]	29[2]	101[2]
Inlet nozzle	A508CL.2	B5003-1 (113°)	ZV-3862-1	0.12	0.68	0.010	60[2]	60[2]	89[2]
Inlet nozzle	A508CL.2	B5003-2 (293°)	ZV-3862-2	0.10	0.71	0.012	60[2]	60[2]	88[2]
Inlet nozzle	A508CL.2	B5003-3 (247°)	ZV-3862-S1	0.10	0.69	0.009	60[2]	60[2]	79[2]
Inlet nozzle	A508CL.2	B5003-4 (67°)	ZV-3862-S2	0.10	0.69	0.010	60[2]	60[2]	77[2]
Outlet nozzle	A508CL.2	B5004-1 (202°)	AV2101-N8C6136	--	0.74	0.005	60[2]	60[2]	82[2]
Outlet nozzle	A508CL.2	B5004-2 (22°)	AV2120-N8C6137	--	0.74	0.007	60[2]	60[2]	75[2]
Outlet nozzle	A508CL.2	B5004-3 (338°)	AV2081-N8C6138	--	0.71	0.005	60[2]	60[2]	90[2]
Outlet nozzle	A508CL.2	B5004-4 (158°)	AV2234-N8C6139	--	0.79	0.006	60[2]	60[2]	81[2]
Upper shell	A533BCL.1	B5453-2	C-5168-1	0.14	0.58	0.011	10	15[3]	72.4[2]
Upper shell	A533BCL.1	B5011-2	C-4371-1	0.10	0.54	0.011	10	27[3]	68.3[2]
Upper shell	A533BCL.1	B5011-3	C-4387-1	0.13	0.56	0.010	0	0[3]	94.7[2]
Intermediate shell	A533BCL.1	B5012-1	C-4387-2	0.11[1]	0.61[1]	0.010	-30	34	101
Intermediate shell	A533BCL.1	B5012-2	C-4417-3	0.14	0.61	0.011	0	0	105

Component	Material Specification/ Weld Number	Code Number	Heat Number/ Flux Type (Lot)	Cu (%)	Ni (%)	P(%)	T_{NDT} (°F)	RT_{NDT} (°F)	USE (ft-lb)
Intermediate shell	A533BCL.1	B5012-3	C-4377-2	0.11	0.66	0.013	-20	-13	112
Lower shell	A533BCL.1	B5013-1	C-4315-1	0.14	0.58	0.009	-10	0	95
Lower shell	A533BCL.1	B5013-2	C-4374-2	0.10	0.51	0.010	-10	30	115
Lower shell	A533BCL.1	B5013-3	C-4371-2	0.10	0.55	0.010	0	15	103
Bottom head segment	A533BCL.1	B5458-1	C-5168-3	0.14	0.60	0.011	-70	-26[3]	90[3]
Bottom head segment	A533BCL.1	B5458-2	C-5175-3	0.15	0.54	0.014	-30	-15[3]	96[3]
Bottom head segment	A533BCL.1	B5458-3	C-5342-4	0.13	0.56	0.012	-20	2[3]	82[3]
Bottom head dome	A533BCL.1	B5085-1	C-9120-2	0.13	0.53	0.010	0	10[3]	79[3]
Intermediate shell longitudinal weld seams	2-442 A,B,C (Tandem)	M1.22[4]	20291/12008 Linde 1092 (3833/3854)	0.199	0.846	0.011	-60	-50	112
Intermediate shell to lower shell girth weld	9-442 (Single Wire)	G1.39	83640 Linde 0091 (3490)	0.051	0.096	0.006	-70	-70	143
Lower shell longitudinal weld seams	3-442 A,B,C (Tandem)	M1.32	21935/12008 Linde 1092 (3889)	0.213	0.867	0.015	0[2]	-50	124

Note:

All data was obtained or derived from manufacturers original Material Certification and Test Report (MCTR) data, except where noted below.

1. Calculated from a combination of manufacturers MCTRs and surveillance program data (see ATI-93-012-T001, Rev. 2).
2. Estimated per U.S. NRC Standard Review Plan, NUREG-0800, Branch Technical Position MTEB 5-2 (see also WCAP-10786, Table A-1).
3. Also estimated per U.S. NRC Standard Review Plan, but re-evaluation of charpy data in 2002 yielded new RT_{NDT} values.
4. Surveillance weldment.

Table 5-16. Deleted Per 1993 Update

Table 5-17. McGuire Unit 2 Reactor Vessel Toughness Table

Component	Material Specification / Weld Number	Code/Item/ Weld Number	Heat Number/ Flux Type (Lot)	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)[1]	Upper Shelf Energy MWD (ft-lb)	Upper Shelf Energy NMWD (ft-lb)
Closure head dome	A533B, Cl. 1	10	55154-1	--	0.63	0.011	-31	12	132	--
Closure head ring	A508 Cl. 2	09	007055	--	0.86	0.006	16	16	156	--
Closure head flange	A508 Cl. 2	08	526916	--	0.82	0.012	-13	1	155	--
Vessel flange	A508 Cl. 2	07	218572	--	0.82	0.016	- 4	- 4	179	--
Inlet nozzle	A508 Cl. 2	11 (67°)	526341-1	0.04	0.76	0.006	-13	-13	142	--
Inlet nozzle	A508 Cl. 2	12 (113°)	526395-1	0.05	0.73	0.009	-31	-31	108	--
Inlet nozzle	A508 Cl. 2	13 (247°)	526537	0.06	0.76	0.009	-22	-22	129	--
Inlet nozzle	A508 Cl. 2	14 (293°)	526537	0.06	0.78	0.009	-40	-40	132	--
Outlet nozzle	A508 Cl. 2	15 (22°)	526341	0.04	0.77	0.007	-13	- 7	122	--
Outlet nozzle	A508 Cl. 2	16 (158°)	525789	0.05	0.83	0.011	-40	-24	103	--
Outlet nozzle	A508 Cl. 2	17 (202°)	525789	0.05	0.84	0.010	-49	-16	116	--
Outlet nozzle	A508 Cl. 2	18 (338°)	526395-2	0.05	0.74	0.010	-40	-30	121[3]	--
Upper shell	A508 Cl. 2	06	411085	--	0.89	0.006	- 4	25	151	98[2]
Intermediate shell	A508 Cl. 2	05	526840	0.153[1]	0.793[1]	0.012	- 4	- 4[2]	147	94
Lower shell	A508 Cl. 2	04	411337-11	0.15	0.88	0.004	-30	-30[2]	152	141

Component	Material Specification / Weld Number	Code/Item/ Weld Number	Heat Number/ Flux Type (Lot)	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)[1]	Upper Shelf Energy MWD (ft-lb)	Upper Shelf Energy NMWD (ft-lb)
Bottom head ring	A508 Cl. 2	03	527428	0.06	0.77	0.013	- 4	15	109[3]	71[2]
Bottom head segment	A533B, C1.1	02-01 (240°)	55126-2-3	--	0.59	0.007	-49	--	--	--
Bottom head segment	A533B, Cl. 1	02-02 (180°)	55126-2-2	--	0.59	0.007	-49	- 2	136	--
Bottom head segment	A533B, Cl. 1	02-03 (120°)	55126-2-1	--	0.59	0.007	-40	-40	131	--
Bottom head segment	A533B, Cl. 1	02-04 (60°)	55292-2-3	--	0.58	0.006	-13	-13	141	--
Bottom head segment	A533B, Cl. 1	02-05 (0°)	55292-2-2	--	0.58	0.006	-13	-13	132	--
Bottom head segment	A533B, C1.1	02-06 (300°)	55292-2-1	--	0.58	0.006	-13	--	--	--
Bottom head dome	A533B, Cl. 1	01	55292-3	--	0.58	0.006	-40	-40	127	--
Intermediate to lower shell girth weld	Submerged arc weld	W05[4]	895075/ Grau Lo (P46)	0.039	0.724	0.010	-76	-68	--	132

Component	Material Specification / Weld Number	Code/Item/ Weld Number	Heat Number/ Flux Type (Lot)	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)[1]	Upper Shelf Energy MWD (ft-lb)	Upper Shelf Energy NMWD (ft-lb)

Note:

All data was obtained or derived from manufacturers original Material Certification and Test Report (MCTR) data, except where noted below.

1. Calculated from a combination of manufacturers MCTRs and surveillance program data (see ATI-94-012-T004, Rev. 2).
2. Estimated per U.S. NRC Standard Review Plan, NUREG-0800, Branch Technical Position MTEB 5-2 (see also WCAP-11029, Table A-1).
3. 100% shear not reached, upper shelf energy is greater than listed.
4. Surveillance weldment.

Table 5-18. Deleted Per 1993 Update

Table 5-19. Unit 1 Surveillance Weld Metal (Ht. 20291 & 12008, Linde 1092, Lot 3854) Charpy V-Notch Impacts

Temp.	Ft/Lbs.	% Shear	Mils Lat. Exp.	Temp. °F	Energy Ft/Lbs.	Shear %	Lat. Exp. Mils
-80	15	0	12	-100	4.5	6	1
-80	16	0	12	-100	4.5	6	1.5
-40	49	30	36	-35	11.5	23	11
-40	38	30	32	-35	30	27	26
-40	45	30	35	0	15	35	17
+10	71	40	52	0	31	40	29
+10	60	35	44	0	65	47	53
+10	73	40	49	25	75	60	61
+40	81	45	58	25	60	47	49
+40	90	50	63	50	74.5	60	63
+40	88	50	64	50	58	55	47
+110	100	90	81	100	105	93	84
+110	93	95	83	100	96	85	80
+160	111	100	82	150	105	92	86
+160	110	100	88	150	113	100	89
				210	113	100	87
				210	112	100	84
				210	110	100	86
					NDTT (°F): -60		

Table 5-20. Unit 1 Heat Affected Zone Charpy Data [3]

Temp.	Ft/Lbs.	% Shear	Mils Lat. Exp.	Temp. (°F)	Energy Ft-lb	Shear %	Lat. Exp. Mils
-80	21	1	14	-100	13	20	7
-80	19	1	12	-100	21	18	11
-40	25	20	22	-50	24	32	14
-40	37	25	28	-50	39.5	37	27
-40	43	30	32	-25	58	50	36
+10	43	35	34	-25	35	37	26
+10	40	30	33	10	90	65	61
+10	95	60	61	10	65	48	45
+40	58	40	42	50	122	80	76
+40	61	45	45	50	74	65	54
+40	62	40	46	50	63.5	70	46
+110	94	90	67	100	126	100	79
+110	98	90	71	100	83	90	69
+160	82	100	70	110	104	94	73
+160	80	100	70	150	132	100	86
				150	106	100	76
				210	131	100	81
				210	102	100	79
					NDTT (°F): - 50		

Table 5-21. Unit 1 Intermediate Shell Plate C-4387-2

Longitudinal Data				Transverse Data			
Temp. °F	**Energy Ft-Lb**	**Shear %**	**Lat. Exp. Mils**	**Temp. °F**	**Energy Ft-Lb**	**Shear %**	**Lat. Exp. Mils**
-40	17	0	12	-40	23	14	13
-40	19	0	16	-40	10	9	4.5
-40	12	0	9	-40	20	14	12
10	32	20	25	0	33	25	27
10	52	30	38	0	33	20	23
10	42	25	31	0	35	25	28
40	92	40	68	30	49	34	36
40	74	35	52	30	41	30	28
40	68	35	50	30	35	29	34
110	108	80	75	80	57	52	45
110	122	85	77	80	46	43	40.5
160	143	100	86	80	33	25	23
160	145	100	87	110	79	92	62
212	139	100	89	110	68	59	53
212	136	100	85	110	69	65	52.5
				210	103	100	80.5
				210	98	100	77
				210	103	100	80
NDTT (°F) = -30				NDTT (°F) = -30			

Table 5-22. Unit 1 Intermediate Shell Plate C-4417-3

Longitudinal Data				Transverse Data			
Temp. °F	Energy Ft-Lb	Shear %	Lat. Exp. Mils	Temp. °F	Energy Ft-Lb	Shear %	Lat. Exp. Mils
-40	12	0	13	-40	19	14	11.5
-40	17	0	15	-40	9	14	3.5
-40	11	0	10	-40	7	9	3.5
10	38	25	31	10	32	29	25
10	80	40	59	10	20	25	15
10	70	35	54	10	44	25	30.5
40	76	35	59	60	53	43	37
40	55	30	44	60	51	34	36
40	86	40	66	60	50	34	39
110	99	75	71	110	61	60	53
110	110	80	79	110	68	60	57
110	125	85	85	110	78	64	63
160	132	99	87	160	88	90	70
160	137	100	89	160	100	93	73
160	142	100	88	160	98	98	77
				210	102	100	80
				210	106	100	75
				210	106	100	78
	NDTT (°F) = 0						

Table 5-23. Unit 1 Intermediate Shell Plate C-4377-2

Longitudinal Data				Transverse Data			
Temp. °F	Energy Ft-Lb	Shear %	Lat. Exp. Mils	Temp. °F	Energy Ft-Lb	Shear %	Lat. Exp. Mils
-40	15	0	12	-20	43	14	31
-40	35	20	26	-20	10	18	5
-40	16	0	14	-20	21	14	14.5
10	78	35	60	40	53	34	36
10	60	30	46	40	60	34	48
10	61	30	47	40	47	29	34
40	71	45	55	75	70	50	50
40	101	50	72	75	73	55	56
40	104	50	73	75	64	52	46
110	127	85	84	110	80	59	57
110	123	85	75	110	83	79	62.5
110	129	85	86	110	89	77	66
160	158	100	88	160	101	100	80
160	153	100	91	160	112	100	82
160	156	100	90	160	104	100	76
				210	103	100	76
				210	113	100	82
				210	119	100	84
NDTT (°F) = -20							

Table 5-24. Unit 1 Lower Shell Plate C-4315-1

Longitudinal Data				Transverse Data			
Temp. °F	**Energy Ft-Lb**	**Shear %**	**Lat. Exp. Mils**	**Temp. °F**	**Energy Ft-Lb**	**Shear %**	**Lat. Exp. Mils**
-40	9	0	6	-40	16	18	11
-40	28	10	22	-40	26	14	15
-40	12	0	12	-40	18	14	15
10	46	15	35	10	17	23	16
10	45	15	36	10	43	30	30
10	46	15	35	10	37	34	26
40	69	25	50	50	50	43	39
40	65	25	47	50	48	40	40
40	53	20	40	50	49	43	36
110	95	70	70	90	65	59	54
110	105	80	77	90	58	56	52
110	97	70	69	90	61	61	51
160	132	100	90	150	86	90	67
160	130	100	87	150	90	100	72
160	125	100	84	150	93	100	68
				210	97	100	75
				210	93	100	72
				210	96	100	70

Note:

1. NDTT = -10°F

Table 5-25. Unit 1 Lower Shell Plate C-4374-2

Longitudinal Data				Transverse Data			
Temp. °F	Energy Ft-Lb	Shear %	Lat. Exp. Mils	Temp. °F	Energy Ft-Lb	Shear %	Lat. Exp. Mils
-40	15	0	12	-40	12	9	9
-40	22	0	19	-40	15	14	7
-40	14	0	12	-40	14	18	9
10	53	20	41	10	42	30	33
10	49	20	37	10	29	25	20
10	56	20	43	10	33	25	23.5
40	56	20	43	50	66	48	40
40	67	25	49	50	52	40	40
40	75	30	54	50	41	32	34
110	118	70	79	90	79	59	59
110	126	85	83	90	50	50	47
110	119	80	81	90	77	59	60
160	143	100	87	150	108	93	80
160	148	100	92	150	109	94	83
160	136	100	85	150	100	92	76
				210	113	100	83
				210	117	100	82.5
				210	116	100	81

Note:

1. NDTT = -10°F

Table 5-26. Unit 1 Lower Shell Plate C-4371-2

Longitudinal Data				Transverse Data			
Temp. °F	Energy Ft-Lb	Shear %	Lat. Exp. Mils	Temp. °F	Energy Ft-Lb	Shear %	Lat. Exp. Mils
-40	24	5	17	-40	16	9	10
-40	21	5	15	-40	19	14	12
-40	14	0	12	-40	12	9	8
10	40	10	29	10	36	29	24
10	51	20	36	10	32	29	22
10	38	10	25	10	30	29	20.5
40	48	20	32	60	43	38	30
40	51	20	36	60	45	42	38
40	50	20	37	60	48	38	33
110	97	70	68	100	63	59	50
110	100	70	64	100	70	59	54
110	85	70	60	100	68	54	50
160	128	100	86	160	86	92	70
160	134	100	87	160	93	96	71
160	130	100	85	160	90	96	71
				210	109	100	74
				210	100	100	75
				210	102	100	79
NDTT (°F) = 0							

Table 5-27. Unit 2 Heat Affected Zone Charpy Data

Temp. (°F)	Ft-Lbs.	% Shear	Mils Lat. Exp.	Temp. (°F)	Energy ft-lb	Shear (%)	Lat. Exp. Mils
-184	8.7	5	8	-150	3	0	1.5
-166	23.1	11	12	-125	6.5	0	0
-148	12.2	11	12	-125	3.5	0	0
-130	20.3	20	16	-100	29	18	16
-112	23.7	17	20	-75	73	56	48
-94	14.5	16	24	-75	34	47	26
-76	50.9	35	39	-50	43	20	28
-58	50.9	26	39	-16	66	69	43
-40	82.7	67	59	-16	67	64	48.5
-22	104.7	70	67	-16	78	77	51
+10.4	115.7	79	75	32	82	91	56
40	121.5	85	67	32	67	87	58
68	127.9	98	94	100	107	100	69
86	119.2	100	97	100	81	100	56
104	118.0	85	75	150	101	100	70
122	129.6	90	87	210	116	100	66
140	115.7	92	79	210	121	100	67
158	133.7	96	79	250	99	100	69
					NDTT (°F) = -76		

Table 5-28. Unit 2 All Weld Metal Charpy V-Notch Impacts

Temp. (°F)	ft-Lbs.	% Shear	Mils Lat. Exp.	Temp. (°F)	Energy ft-lb	Shear (%)	Lat. Exp. Mils
-184	3.5	0	4	-150	1	0	0
-166	5.2	6	8	-150	1	0	0
-148	9.3	5	12	-75	20	30	15.5
-130	13.9	16	16	-75	16.5	29	9.5
-112	11.0	11	12	-35	34	46	31
-94	26.0	16	16	-35	52	54	40
-76	28.9	30	28	-16	50	65	40
-58	28.9	34	32	-16	57	65	46
-40	33.6	47	28	-16	59.5	65	47
-22	54.4	56	47	25	93	81	67
+10.4	70.0	66	63	25	83.5	77	64
40	92.0	76	71	71	112	75	82
68	117.5	92	87	71	110	79	80
86	106.5	98	91	125	125	98	91
104	112.3	90	94	125	124	99	92
122	123.8	95	94	210	140	100	96
140	122.1	100	91	210	132.5	99	96.5
158	134.8	100	94	275	144.5	100	98
					NDTT (°F) = -76		

Table 5-29. Unit 2 Core Region

Lower Shell Forging, 04 NDT = -30°F				Intermediate Shell Forging, 05 NDT = -4°F			
Tangential Data				Tangential Data			
Temp. (°F)	Energy (Ft-Lb)	Shear (%)	Lat. Exp. (Mils)	Temp. (°F)	Energy (Ft-Lb)	Shear (%)	Lat. Exp. (Mils)
-148	5.8	0	4	-148	6.4	0	4
-148	8.1	0	8	-148	7.5	0	4
-148	7.5	0	8	-148	6.4	0	6
- 76	33.6	6	20	- 76	40.5	12	35
- 76	45.1	12	35	- 76	42.8	12	32
- 76	19.7	6	16	- 76	43.4	12	35
- 4	107.0	52	83	- 4	39.9	17	35
- 4	68.9	29	55	- 4	86.8	46	71
- 4	70.0	29	55	- 4	86.2	35	67
60	124.4	78	83	60	137.7	90	91
60	136.6	73	83	60	146.4	95	91
60	133.1	75	79	60	122.1	75	79
113	147.6	98	94	113	160.9	98	79
113	152.2	100	91	113	152.2	90	94
113	153.3	100	91	113	148.1	100	87
176	153.3	100	91	176	144.7	95	83
176	146.4	96	91	176	141.2	95	91
176	147.6	94	91	176	137.7	95	87

Axial Data				Axial Data			
Temp. (°F)	**Energy (Ft-Lb)**	**Shear (%)**	**Lat. Exp. (Mils)**	**Temp. (°F)**	**Energy (Ft-Lb)**	**Shear (%)**	**Lat. Exp. (Mils)**
-100	9	4	2	-100	5.5	2	1
-100	13	5	2	-100	7.5	2	1
-50	39	20	23	-50	21	15	12
-50	16	15	6	-25	29	15	20
-25	11	23	7	-25	28.5	15	18
0	74	45	47	0	34	30	26
0	71	42	48	25	49	45	40
0	38	34	21	25	53	45	39
30	85.5	55	52	56	68	54	47
30	90.5	63	56	56	63	54	46
30	98.5	65	62	56	67	54	
74	98	73	66	100	71.5	90	54
74	102	72	70	100	74	92	61
75	101	68	67	100	72	94	55
125	140	100	83	140	93	100	71
125	143	100	83	140	92.5	100	64
210	140	100	85	210	95	100	67
210	140	100	77	210	97	100	97
	NDTT (°F) = -4						

Table 5-30. Reactor Coolant Leakage Detection

Sensing Device	Parameter Monitored	Readout Location	Sensitivity
Scintillation Detector	Radioactivity accumulation on filter paper from sample of containment air	Control Room	The instrument sensitivity for air particulate is 10^{-9} μCi/cc. The response times are as follows: 1. For a 1.0 gal/min leak into Containment in mode 1, with activity from water activation products, leakage will be detected in less than 60 minutes.[1] 2. For a 1 gal/min leak into Containment, with corrosion products only, leakage will be detected in approximately 10 hours[1]
Containment Sump Floor and Equipment Level Monitor	Containment Floor and Equipment Sump Level	Control Room	A 1 gal/min leak is detectable in approximately 1 hour[2].
Deleted Per 2006 Update			
Scintillation Detector	Noble gas activity in condenser air ejector effluent	Control Room	A leak of < 30 gal/day is detectable[3].
Scintillation Detector	Steam Generator N-16 leakage detection monitor	Control Room	10^0 to 10^5 gallons per day leakage to individual steam generator indication for power levels from 40% to 100% rated power.
Volume Control Water Level Detector	Volume Control Tank water level	Control Room	A 1 gal/min leak is detectable[4].
Incore Instrument Sump Level Alarm	Incore Instrument Sump Level	Control Room	A leak of 1 gal/min is detectable[5].

Sensing Device	Parameter Monitored	Readout Location	Sensitivity

Note:

1. The sensitivities indicated assume instantaneous mixing, reference MCC-1223.03-00-0044.
2. Sensitivity is based on Reg. Guide 1.45 criteria to detect a 1 gpm leak in 1 hour, using OAC rate of change alarm for each sump level transmitter. The following are assumed: 1 gpm leak is detectable in 1 hour after leakage has reached the sump, embedded piping is considered in the alarm setpoint, leakage is cumulative between sumps A and B.
3. The sensitivity indicated is based upon the detector sensitivity in Table 11-28. The count rate of the detector is proportional to leak rate and coolant activity. With a typical background of 2.5 x 10^2 CPM, any combination of defects and leaks resulting in a count rate approximately 3 sigma above detector background is detectable.
4. The sensitivity indicated is based upon the change in water level in the volume control tank, pressurizer level, charging rate, letdown rate, reactor coolant pump seal water injection rate, and after condensate has reached the tank.
5. Leakage of 1 gal/min is detectable within four hours after reaching the sump.

Table 5-31. Deleted Per 1996 Update

Table 5-32. Reactor Vessel Design Parameters

Design/Operating Pressure, psig	2485/2235
Design Temperature, °F	650
Overall Height of Vessel and Closure	
Head, ft-in, (Bottom Head O.D. to top of control Rod Mechanism Adaptor)	43-9 5/8
Thickness of Insulation, min., in.	3
Deleted per 2017 Update	
Number of Reactor Closure Head Studs	54
Diameter of Reactor Closure Head Studs, in.	7
ID of Flange, in.	167.0
OD of Flange, in.	205
ID at Shell, in.	173
Inlet Nozzle ID, in.	27-1/2
Outlet Nozzle ID, in.	29
Clad Thickness, min., in.	5/32
Vessel Belt-Line Thickness, min., in.	8-1/2
Lower Head Thickness, min., in.	5-1/2
Closure Head Thickness, min., in.	6-1/2

Table 5-33. Surveillance Capsule Removal Schedule [Note 1]

Unit	Capsule	Withdrawal End of Cycle (EOC)	Lead Factor [Note 4]	Withdrawal EFPY (from plant startup	Fluence ($x10^{19}$ n/cm^2) [Note 4]	Reference
Deleted Per 2014 Update						
Unit 1	U	1	4.96	1.09	0.382	WCAP-10786
Unit 1	X	5	4.83	4.30	1.40	WCAP-12354
Unit 1	V	8	4.15	7.24	1.93	WCAP-13949
Unit 1 [Note 2]	Z	8	4.75	7.24	2.21	WCAP-13949
Unit 1	Y	11	4.20	10.21	2.65	WCAP-14993
Unit 1 [Note 3] [Note 5]	W	18	4.92	19.22	5.08	WCAP-17014-NP
Deleted Row per 2017 Update						
Deleted per 2014 Update						
Unit 2	V	1	4.16	1.03	0.302	WCAP-11029
Unit 2	X	5	4.81	4.16	1.38	WCAP-12556
Unit 2	U	7	4.66	6.05	1.90	WCAP-13516
Unit 2 [Note 2]	Y	8	4.03	7.18	1.94	WCAP-14231
Unit 2 [Note 2]	Z	8	4.60	7.18	2.21	WCAP-14231
Unit 2 [Note 6]	W	10	4.64	9.44	2.82	WCAP-14799
Deleted Row per 2017 Update						
Deleted Per 2014 Update						

Note 1:	All in-vessel surveillance capsules have been withdrawn. Thus, this table is a summary of all past surveillance capsule withdrawals.
Note 2:	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.
Note 3:	The management of Capsule W is controlled by Calvert Cliffs Unit 1. It was withdrawn at EOC 18 (19.22 EFPY) for evaluation. The dosimetry and weld specimens were analyzed. The plate specimens were stored at Westinghouse and are available for testing or additional irradiation if ever deemed necessary.
Note 4:	Capsule fluence and lead factors were updated in WCAP-17455.
Note 5:	15 specimens from Unit 1 Capsule W have been installed in the Shearon Harris Reactor Vessel as part of the EPRI PWR Supplemental Program (PSSP). PSSP Capsule CQL-P is scheduled for removal in 2028.

Unit	Capsule	Withdrawal End of Cycle (EOC)	Lead Factor [Note 4]	Withdrawal EFPY (from plant startup	Fluence ($x10^{19}$ n/cm^2) [Note 4]	Reference
Note 6:	12 specimens from Unit 2 Capsule W have been installed in the Shearon Harris Reactor Vessel as part of the EPRI PWR Supplemental Surveillance Program (PSSP). PSSP Capsule CQL-P is scheduled for removal in 2028.					

Table 5-34. Reactor Vessel Quality Assurance Program

	RT[1]	UT[1]	PT[1]	MT[1]
Forgings & Tubes				
1. Flanges		yes		yes
2. Studs		yes		yes
3. CRDM and UHI Adaptors		yes	yes	
4. CRDM and UHI Adaptor Tubes		yes	yes	
5. Instrumentation Tube		yes	yes	
6. Main Nozzles		yes		yes
Plates		yes		yes
Weldments				
1. Main Seam	yes	Yes[3]		yes
2. CRDM Head Adaptor to Head Connection			yes	
3. UHI Adaptor to Head Attachments	yes	yes	yes	
4. Instrumentation Tube Connection			yes	
5. Main Nozzles	yes	Yes[3]		yes
6. Cladding		Yes[2]	yes	
7. Nozzle-safe ends (weld deposit)	yes	Yes[3]	yes	
8. CRDM Head Adaptor Forging to Head Adaptor Tube	yes		yes	
9. UHI Adaptor Forging to Adaptor Tube	yes	Yes[3]	yes	
10. All Ferritic Welds Accessible After Hydrotest				yes
11. All Non-ferritic Welds Accessible After Hydrotest			yes	
12. Seal Ledge				yes
13. Head Lift Lugs				yes
14. Core Pad Welds		yes	yes	Yes

Note:

1. RT – Radiographic
   UT – Ultrasonic
   PT - Dye Penetrant
   MT - Magnetic Particle
2. UT of Clad Bond-to-base Metal
3. UT map for Section IX

Table 5-35. Reactor Coolant Pump Design Parameters

Design pressure, psig	2485
Design temperature, °F	650
Capacity per pump, gpm	99,000
Developed head, ft.	288
NPSH required, ft.	245
Suction temperature, °F	557.8
RPM	1186
Discharge nozzle, ID, inches	27-1/2
Suction nozzle, ID, inches	31
Overall unit height, ft.-in.	27'-7.2"
Water volume, ft^3	80
Total rotating inertia, ft-lb.	95,000
Weight, dry, lb.	201,900
Motor	
Type	AC induction, single speed totally enclosed water cooled
Power, H.P.	7000
Voltage, volts	6600
Insulation class	minimum B
Frequency, Hz	60
Phase	3
Starting Current, amps	3000
Input, hot reactor coolant, kw	5250 ± 180
Input, cold reactor coolant, kw	6850 ± 240
Seal water injection, gpm	8
Seal water return, gpm	3

Table 5-36. Reactor Coolant Pump Quality Assurance Program

	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 (For Bore)	yes
Weldments				
1. Circumferential	yes		yes	
2. Instrument Connections			yes	

Note:

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

Table 5-37. Steam Generator Design Data

Design pressure, reactor coolant side, psig	2485
Design pressure, steam side, psig	1185
Design temperature, reactor coolant side, °F	650
Design temperature, steam side °F	600
Total heat transfer surface area ft^2	79,800
Maximum moisture, carryover, wt percent	0.25
Overall height, ft-in.	68' - 1 3/8"
Number of U-tubes	6633
Tube wall thickness, nominal, in.	0.040
U-tube outer diameter, in.	0.6875
Number of manways	3
Number of handholes	10
ID of handholes, in.	6
ID of manways, in.	21

Table 5-38. Steam Generator Quality Assurance Program

	RT[1]	UT[1]	PT[1]	MT[1]	ET[1]
Tube Sheet					
1. Forging		yes		yes	
2. Cladding		yes	Yes[2]		
Channel Head					
1. Forging		yes		yes	
2. Cladding			yes		
Secondary Shell & Head					
1. Plates		yes			
Tubes		yes			yes
Nozzles (Forgings)		yes		yes	
Weldments					
1. Shell, longitudinal	yes	yes		yes	
2. Shell, circumferential	yes	yes		yes	
3. Cladding (Channel Head-Tube Sheet joint cladding restoration)			yes		
4. Steam and Feedwater Nozzle to shell	yes	yes		yes	
5. Support brackets				yes	
6. Tube to tube sheet			yes		
7. Instrument connections (primary and secondary)		yes		yes	
8. Temporary attachments after removal				yes	
9. After hydrostatic test (all welds and complete channel head - where accessible)				yes	
10. Nozzle safe ends (if forgings)	yes		yes		
11. Nozzle safe ends (if weld deposit)	yes		yes		

Note:

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

Table 5-39. Reactor Coolant Piping Design Parameters

Reactor inlet piping, ID, in.	27-1/2
Reactor inlet piping, nominal wall thickness, in.	2.30
Reactor outlet piping, ID, in.	29
Reactor outlet piping, nominal wall thickness, in.	2.42
Coolant pump suction piping, ID, in.	31
Coolant pump suction piping, nominal wall thickness, in.	2.58
Pressurizer surge line piping, ID, in.	11.50
Pressurizer surge line piping, nominal wall thickness, in.	1.25
Water volume, all loops and surge line, $ft.^3$	1030
Design/operating pressure, psig	2485/2235
Design Temperature, °F	650
Design Temperature, pressurizer surge line, °F	680
Design pressure, pressurizer relief line, psig	From pressurizer to safety valve 2485 psig, 680°F
Design Temperature, pressurizer relief lines, °F	From safety valve to pressurizer relief tank 500 psig, 500°F

Table 5-40. Reactor Coolant Piping Quality Assurance Program

	RT[1]	UT[1]	PT[1]
Fittings and Pipe (Castings)	yes		yes
Fittings and Pipe (Forgings)		yes	yes
Weldments			
1. Circumferential	yes		yes
2. Nozzle to runpipe (except no RT for nozzles less than 4 inches)	yes	yes	yes
3. Instrument connections		yes	yes

Note:

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

Table 5-41. 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	<450
Reactor Coolant System initial temperature, °F	<350
Component cooling water design, temperature, °F	95
Cooldown time, hours after initiation of RHRS operation	~16
Reactor Coolant System temperature at end of cooldown, °F	140
Decay heat generation at 20 hours after Reactor shutdown, BTU/hr	77.21×10^6
Reactor Coolant System flow rate during refueling mode, gpm	≥ 1000, and as required to maintain RCS temperature ≤ 140°F

Table 5-42. 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
Residual Heat Exchanger		
Number		2
Design Heat Removal Capacity, BTU/hr.		34.15×10^6
	Tube-Side	Shell-Side
Design Pressure, psig	600	150
Design Temperature, °F	400	200
Design Flow, lb/hr.	1.48×10^6	2.48×10^6
Inlet Temperature, °F	137	95
Outlet Temperature, °F	114	108.8
Material	Austenitic Stainless Steel	Carbon Steel
Fluid	Reactor Coolant	Component Cooling Water

Table 5-43. Pressurizer Design Data

Design Pressure	2485
Design Temperature, °F	680
Surge Line Nozzle Diameter, in	14
Heatup Rate of Pressurizer Using Heaters Only, °F/hr.	55
Internal Volume, cu. ft.	1800

Table 5-44. Pressurizer Quality Assurance Program

	RT[1]	UT[1]	PT[1]	MT[1]
Heads				
1. Plates		yes		
2. Cladding			yes	
Shell				
1. Plates		yes		
Cladding			yes	
Heaters				
1. Tubing [3]		yes	yes	
2. Centering of element	yes			
Nozzle (Forgings)		yes	yes[2]	yes[2]
Weldments				
1. Shell, longitudinal	yes			yes
2. Shell, circumferential	yes			yes
3. Cladding			yes	
4. Nozzle Safe End (if forging)	yes	yes[4]	yes[4]	
5. Instrument Connections			yes	
6. Support Skirt		yes		yes
7. Temporary Attachments (after removal)				yes
8. All welds and plate heads after hydrostatic tests				Yes

Note:

1. RT – Radiographic
   Deleted Per 2006 Update
   UT – Ultrasonic
   PT - Dye Penetrant
   MT - Magnetic Particle
2. MT or PT
3. or UT and ET
4. Weld Overlay Installation, UT and PT.

Table 5-45. Pressurizer Relief Tank Design Data

Design Pressure, psig	100
Rupture Disc Release Pressure, psig	100±5%
Design Temperature, °F	340
Total Rupture Disc Relief Capacity, lb/hr at 100 psig	1.6×10^6

Table 5-46. Reactor Coolant System Boundary Valve Design Parameters

Design/Normal Operating Pressure, psig	2485/2235
Pre-Operational Hydrotest, psig	3107
Design Temperature, °F	650

Table 5-47. Pressurizer Valves Design Parameters

	Parameters
Pressurizer Spray Control Valves	
Number	2
Design pressure, psig	2485
Design temperature, °F	650
Design flow for valves full open, each, gpm	450
Pressurizer Safety Valves	
Number	3
Maximum relieving capacity, ASME rated flow, lb/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, °F	680
High pressure setpoint, psig	2335
Relieving capacity, lb/hr (per valve)	210,000
Fluid	Saturated Steam
Low pressure setpoint, psig (NC-32B and NC-34A only)	380
Relieving capacity, gpm (per valve)	1060
Fluid	Water @60°F

Table 5-48. Component Supports Loading Combinations and Code Requirements

Loading Combination		Code Or Stress Requirements
1. DL + OL + LL 2. DL + OL + OBE		AISC with Allowable Stresses of F_s. Allowable stresses in concrete and reinforcement are in accordance to Chapter 10 of the AC1-318, 1962 Code, WSD.
3. DL + OL – SSE 4. DL + OL + SSE + LOCA		ACI-318, 1963 USD with no load factors applied to loadings. For allowable stresses in structural steel see Note 1.
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, 1969.
WSD	=	Working Stress Design
USD	=	Ultimate Strength Design.
F_s	=	Steel allowable stresses as specified in AISC Part 1.
F_y	=	Yield stress of structural steel.

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

Note:

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

Table 5-49. Design Transients for the Reactor Coolant System Including the BWI Replacement Steam Generators (RSGs)

Design Transients	**Allowable Occurrences**(1)	
	Unit 1	Unit 2
Normal (Level A) Transients		
(**) Plant Heatup	200	200
(**) Plant Cooldown	200	200
RHR Suction	200	200
RHR Injection	200	200
Refueling	80	80
(**) Plant Loading at 5%/min (15% to 100%)	13,200	13,200
(**) Plant Unloading at 5%/min (100% to 15%)	13,200	13,200
Small Step Load Increase		
15 - 25%	300	300
(**) 90 - 100%	2000	2000
Small Step Load Decrease		
25 - 15%	300	300
(**) 100 - 90%	2000	2000
(**) Large Step Load Decrease (100%-5%) with Steam Dump	200(2)	200(2)
Feedwater Cycling at No Load	2000	2000
(**) Steady State Fluctuations	Infinite(3)	Infinite(3)
Plant Loading and Unloading between 0% and 15% power	750 Load(2)	750 Load(2)
	750 Unload(2)	750 Unload(2)
Loop Out of Service		
Normal Pump Shutdown	80	80
Normal Pump Startup	70	70
Boron Concentration Equalization	26,400	26,400
Reactor Coolant Pump Startup/Shutdown		
Cold Conditions	750	750
Hot Conditions	3750	3750
RCS Venting		
Affected Loops	0(7)	0(7)
Unaffected Loops	0(7)	0(7)

Design Transients	Allowable Occurrences[1]	
	Unit 1	Unit 2
Vacuum Refill	480	480
Normal/Charging/Letdown Shutoff and Return to Service	60	60
Letdown Trip with Prompt Return to Service	200	200
Letdown Trip with Delayed Return to Service	20	20
Charging Trip with Prompt Return to Service	20	20
Charging Trip with Delayed Return to Service	20	20
Charging Flow 50% Decrease	24,000	24,000
Charging Flow 50% Increase	24,000	24,000
Letdown Flow 40% Decrease and Return to Normal	2000	2000
Letdown Flow 60% Increase	24,000	24,000
Letdown Shutoff and Momentary Excess Letdown	100	100
Switch of Charging Pump Suction to FWST and Back	180	180
Auxiliary Spray Actuation during Heatup	200	200
Auxiliary Spray Actuation during Cooldown	200	200
Pressurizer Relief Valve Operation	100	100
LTOP Pressurizer PORV Operation	200	200
Pzr Blk/Drn Vlv Operation	400	400
Pressurizer Safety Valve Operation	40	40
Upset (Level B) Transients		
(**) Loss of Load without Immediate Turbine or Reactor Trip	80	80
(**) Loss of Power (Blackout with Natural Circulation)	40	40
(**) Loss of Flow in One Loop	80	80
Reactor Trip from Full Power		
(**) Nominal	230	230
(*) Inadvertent Cooldown	0[7]	10
(*) Inadvertent RCS Depressurization	0[7]	20
Inadvertent Startup of an Inactive Loop	15	15
Control Rod Drop	0[7]	0[7]
Operating Basis Earthquake (OBE)		

Design Transients		**Allowable Occurrences**[1]	
		Unit 1	Unit 2
RSG		30 (20 cycles/occurrence)	
Reactor Coolant Pump, Pressurizer		20 (20 cycles/occurrence)	
Reactor Vessel		50 cycles	
Piping		See Section 3.7.3.1	
Excessive Feedwater Flow		45	45
Excessive Bypass Feedwater		0[7]	0[7]
Cold Feedwater to Dry, Pressurized RSG		2	2
Complete Loss of Flow		0[7]	0[7]
(**) Inadvertent Auxiliary Spray		10	10
Inadvertent SI Accumulator Blowdown during Plant Cooldown		4	4
High Head Safety Injection		17	17
Boron Injection		25	32
Faulted (Level D) Transients[4]			
(**) Reactor Coolant Pipe Break (Large LOCA)		1[5]	1[5]
(**) Large Steam Line Break		1	1
Safe Shutdown Earthquake (SSE)		1 (10 cycles)	
Steam Generator Tube Rupture		8	8
Cold Feedwater to Dry, Depressurized RSG		1	1
High Head Safety Injection		2	2
Boron Injection		2	2
Test Conditions			
(**) Turbine Roll Test		10[6]	10[6]
(**) Primary Side Hydrostatic Test		5	5
Secondary Side Hydrostatic Test		10	10
(**) Primary Side Leakage Test		50	50
Secondary Side Leakage Test		80	80
Tube Leak Test			
Secondary Side Pressure	200	600	600
	400	300	300
	600	180	180

Design Transients		Allowable Occurrences[1]	
		Unit 1	Unit 2
	840	80	80

Notes:

1. Allowed occurrences is the minimum controlling analyzed or postulated number of occurrences for the plant design life, including period of extended operation, considering ASME Section III and XI limits.
2. The RSGs are designed for several cycles of swapping of main feedwater supply between the auxiliary and the main feedwater nozzles with and without tempering flow during plant loading and unloading and large step load decrease. The design is for 100 swaps without tempering flow and 750 with tempering flow.
3. Pressurizer surge line is analyzed for 150,000 initial fluctuations and 3,000,000 random fluctuations. Various NSSS components are analyzed for 150,000 to 30,000,000 fluctuations of various minor amplitudes.
4. In accordance with the ASME Code, faulted conditions are not included in fatigue evaluations. Some components analyzed/piping segments are individually qualified for faulted events which are not listed because the unit as a whole is therefore not qualified for such.
5. This condition has been excused for RCS pressure boundary qualification by Leak Before Break. Some components analyzed after LBB was approved were not analyzed for this condition.
6. Since Turbine Roll Test is performed only prior to fuel load, the RSGs are not designed for this event.
7. Some components are not designed for this event, thus it is not considered to have been part of the analysis for any component. If it occurs, it will be dispositioned, by, for example, considering it as an another, actually analyzed event.

(**,*) Transient specified for Reactor Vessel design for (Units 1&2, Unit 2 only) [Section 5.4.4.4]

Table 5-50. Reactor Coolant System Pressure Isolation Valves

Valve Number	Function
NI60	Accumulator Discharge
NI71	Accumulator Discharge
NI59	Accumulator Discharge
NI70	Accumulator Discharge
NI82	Accumulator Discharge
NI94	Accumulator Discharge
NI81	Accumulator Discharge
NI93	Accumulator Discharge
NI134	Safety Injection (Hot Leg)
NI159	Safety Injection (Hot Leg)
NI156	Safety Injection (Hot Leg)
NI128	Safety Injection (Hot Leg)
NI124	Safety Injection (Hot Leg)
NI160	Safety Injection (Hot Leg)
NI157	Safety Injection (Hot Leg)
NI126	Safety Injection (Hot Leg)
NI129	Safety Injection (Hot Leg)
NI125	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

Valve Number	Function
ND2A	Residual Heat Removal

Table 5-51. RT $_{PTS}$ Calculations for McGuire Unit 1 Reactor Vessel Materials at 54 EFPY

Material	CF (ºF)	Fluence @ 54 EFPY ($x10^{19}$ n/cm^2)	FF	$RT_{NDT(U)}$ (ºF)	Δ RT $_{PTS}$ (ºF)	M (ºF)	RT $_{PTS}$ (ºF)
Upper Shell Plate B5453-2	99.1	0.0547	0.3072	15	30.4	30.4	76
Upper Shell Plate B5011-2	65	0.0547	0.3072	27	20.0	20.0	67
Upper Shell Plate B5011-3	89.8	0.0547	0.3072	0	27.6	27.6	55
Intermediate Shell Plate B5012-1	74.2	2.56	1.2521	34	92.9	34	161
→ Using Surveillance Capsule Data	63.5	2.56	1.2521	34	79.5	17	131
Intermediate Shell Plate B5012-2	100.3	2.56	1.2521	0	125.6	34	160
Intermediate Shell Plate B5012-3	74.9	2.56	1.2521	-13	93.8	34	115
Lower Shell Plate B5013-1	99.1	2.57	1.2531	0	124.2	34	158
Lower Shell Plate B5013-2	65	2.57	1.2531	30	81.4	34	145
Lower Shell Plate B5013-3	65	2.57	1.2531	15	81.4	34	130
Upper Shell Longitudinal Weld Seams 1-442A, B, C	201.3	0.0451	0.2767	-50	55.7	55.7	61
→ Using Surveillance Capsule Data	155.2	0.0451	0.2767	-50	43.0	28	21
Upper Shell to Intermediate Shell Circumferential Weld Seam 8-442	170.5	0.0547	0.3072	-56	52.4	62.4	59

Material	CF (ºF)	Fluence @ 54 EFPY ($x10^{19}$ n/cm^2)	FF	$RT_{NDT(U)}$ (ºF)	ΔRT_{PTS} (ºF)	M (ºF)	RT_{PTS} (ºF)
Intermediate Shell Longitudinal Weld Seams 2-442A, B, C	201.3	2.13	1.2055	-50	242.7	56	249
→ Using Surveillance Capsule Data	155.2	2.13	1.2055	-50	187.1	28	165
Intermediate Shell to Lower Shell Circumferential Weld Seam 9-442	37.5	2.47	1.2432	-70	46.6	46.6	23
Lower Shell Longitudinal Weld Seams 3-442 A, B,C	208.2	2.13	1.2055	-50	251.40	56	257
→ Using S/C Data from Diablo Canyon 2	186.4	2.13	1.2055	-50	224.7	28	203
Deleted Per 2014 Update							

NOTE: The information above was taken from WCAP-17455.

Table 5-52. RT $_{PTS}$ Calculations for McGuire Unit 2 Reactor Vessel Materials at 54 EFPY

Material	CF (°F)	Fluence @ 54 EFPY ($x10^{19}$ n/cm^2)	FF	$RT_{NDT(U)}$ (°F)	Δ RT $_{PTS}$ (°F)	M (°F)	RT $_{PTS}$ (°F)
Upper Shell Forging 06	123.9	0.0711	0.3521	25	43.6	34	103
Deleted Per 2014 Update							
Intermediate Shell Forging 05	117.2	2.41	1.2370	-4	145.0	34	175
→ Using Surveillance Capsule Data	85.5	2.41	1.2370	-4	105.7	17	119
Lower Shell Forging 04	115.8	2.48	1.2442	-30	144.1	34	148
Bottom Head Ring 03	37	0.336	0.6997	15	25.9	25.9	67
Upper Shell to Intermediate Shell Circumferential Weld Seam W06	82.9	0.0711	0.3521	10	29.2	29.2	68
Intermediate Shell to Lower Shell Circumferential Weld Seam W05	52.7	2.34	1.2296	-68	64.8	56	53
→ Using Surveillance Capsule Data	27.1	2.34	1.2296	-68	33.3	28	-7
Lower Shell to Bottom Head Ring Weld W04	41	0.336	0.6997	10	28.7	28.7	67

NOTE: The information above was taken from WCAP-17455.

Table 5-53. Evaluation of Upper Shelf Energy for McGuire Unit 1 Reactor Vessel Materials at 54 EFPY

Material	Weight % of Cu	¼ T EOLE Fluence ($x10^{19}$ n/cm^2)	Unirradiated USE (ft-lb)	Projected USE Decrease (%)	Projected USE @ 54 EFPY (ft-lb)
Upper Shell Plate B5453-2	0.14	0.033	72.4	11	64.4
Upper Shell Plate B5011-2	0.10	0.033	68.3	8.6	62.4
Upper Shell Plate B5011-3	0.13	0.033`	94.7	9.8	85.4
Intermediate Shell Plate B5012-1	0.11	1.525	101	23	77.8
→ Using Surveillance Capsule Data	0.11	1.525	101	11	89.9
Intermediate Shell Plate B5012-2	0.14	1.525	105	26	77.7
Intermediate Shell Plate B5012-3	0.11	1.525	112	23	86.2
Lower Shell Plate B5013-1	0.14	1.531	95	26	70.3
Lower Shell Plate B5013-2	0.10	1.531	115	21	90.9
Lower Shell Plate B5013-3	0.10	1.531	103	21	81.4
Upper Shell Longitudinal Weld Seams 1-442A, B, C	0.199	0.027	112	15	95.2
→ Using Surveillance Capsule Data	0.199	0.027	112	19	90.7
Upper Shell to Intermediate Shell Circumferential Weld Seam 8-442	0.183	0.033	109	15	92.7
Intermediate Shell Longitudinal Weld Seams 2-442 A, B, C	0.199	1.269	112	36	71.7
→ Using Surveillance Capsule Data	0.199	1.269	112	46	60.5
Intermediate Shell to Lower Shell Circumferential Weld Seam 9-442	0.051	1.472	143	21	113.0
Lower Shell Longitudinal Weld Seams 3-442 A, B, C	0.213	1.269	124	38	76.9

NOTE: The information above was taken from WCAP-17455.

Table 5-54. Evaluation of Upper Shelf Energy for McGuire Unit 2 Reactor Vessel Materials at 54 EFPY

Material	Weight % of Cu	¼ T EOLE Fluence ($x10^{19}$ n/cm^2)	Unirradiated USE (ft-lb)	Projected USE Decrease (%)	Projected USE @ 54 EFPY (ft-lb)
Upper Shell Forging 06	0.16	0.043	98	12	86.2
Intermediate Shell Forging 05	0.153	1.450	94	27	68.6
→ Using Surveillance Capsule Data	0.153	1.450	94	23	72.4
Lower Shell Forging 04	0.15	1.492	141	27	102.9
Bottom Head Ring 03	0.06	0.202	>71	13	>61.8
Upper Shell to Intermediate Shell Circumferential Weld W06	0.11	0.043	>71	12	>62.5
Intermediate Shell to Lower Shell Circumferential Weld W05	0.039	1.408	132	21	104.3
→ Using Surveillance Capsule Data	0.039	1.408	132	3.4	127.5
Lower Shell to Bottom Head Ring Weld W04	0.03	0.202	99	13	86.1

NOTE: The information above was taken from WCAP-17455.

Table 5-55. Summary of Reactor Coolant System Leakage Detection Instrumentation Exceptions to Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems" (Rev. 0)

Regulatory Position		Exception
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} μCi/cc. The particulate monitor alarm setting will 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 CFAE and CVUCDT sensitivity of 1 gpm after leakage has reached the sump/tank.
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.

Table 5-56. Ex-Vessel Dosimetry Capsule Summary Table

Unit	Capsule	Ex-Vessel Azimuthal Location [Note 1]	Axial Location	Installed Cycle	Removed Cycle	Best Estimate Flux (n/cm^2-s)	Reference
Unit 1	A	0°	Core Midplane	12	12	3.68E+08	WCAP -17799-NP
Unit 1	B	15°	Core Midplane	12	12	5.64E+08	WCAP -17799-NP
Unit 1	C	30°	Core Midplane	12	12	6.00E+08	WCAP -17799-NP
Unit 1	E	45°	Core Midplane	12	12	5.77E+08	WCAP -17799-NP
Unit 1	D	45°	Core Top	12	12	2.36E+08	WCAP -17799-NP
Unit 1	F	45°	Core Bottom	12	12	2.48E+08	WCAP -17799-NP
Unit 1	M	0°	Core Midplane	13	22	3.68E+08	WCAP -17799-NP
Unit 1	N	15°	Core Midplane	13	22	5.31E+08	WCAP -17799-NP
Unit 1	O	30°	Core Midplane	13	22	5.63E+08	WCAP -17799-NP
Unit 1	Q	45°	Core Midplane	13	22	5.53E+08	WCAP -17799-NP
Unit 1	P	45°	Core Top	13	22	1.96E+08	WCAP -17799-NP
Unit 1	R	45°	Core Bottom	13	22	2.40E+08	WCAP -17799-NP
Unit 2	G	0°	Core Midplane	12	12	4.11E+08	WCAP-17767-NP
Unit 2	H	15°	Core Midplane	12	12	6.35E+08	WCAP-17767-NP
Unit 2	I	30°	Core Midplane	12	12	6.80E+08	WCAP-17767-NP
Unit 2	K	45°	Core Midplane	12	12	5.46E+08	WCAP-17767-NP
Unit 2	J	45°	Core Top	12	12	5.04E+08	WCAP-17767-NP
Unit 2	L	45°	Core Bottom	12	12	8.84E+07	WCAP-17767-NP
Unit 2	S	0°	Core Midplane	13	21	4.02E+08	WCAP-17767-NP
Unit 2	T	15°	Core Midplane	13	21	5.52E+08	WCAP-17767-NP
Unit 2	U	30°	Core Midplane	13	21	5.86E+08	WCAP-17767-NP
Unit 2	W	45°	Core Midplane	13	21	5.93E+08	WCAP-17767-NP
Unit 2	X	45°	Core Top	13	21	4.97E+08	WCAP-17767-NP
Unit 2	V	45°	Core Bottom	13	21	6.22E+07	WCAP-17767-NP