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SECTION 5

5.0 REACTOR COOLANT SYSTEM

5.1 SUMMARY DESCRIPTION

The Reactor Coolant (RC) System consists of the reactor vessel, two vertical once-through steam generators, four shaft-sealed reactor coolant pumps, an electrically heated pressurizer, and interconnecting piping. The system, located entirely within the Containment Vessel (CV), is arranged in two heat transport loops, each with two RC pumps and one steam generator. Reactor coolant is transported through piping connecting the reactor vessel to the steam generators and flows downward through the steam generator tubes transferring heat to the steam and water on the shell side of the steam generator. In each loop the reactor coolant is returned to the reactor through two lines, each containing an RC pump. In addition to serving as a heat transport medium, the coolant also serves as a neutron moderator and reflector and as a solvent for the soluble poison (boron in the form of boric acid) utilized in chemical shim reactivity control. System design conditions for the RC Pressure Boundary (RCPB) are listed in Table 5.1-1a. Major components of the RC pressure boundary are listed in Table 5.1-1b, and described in Section 5.5.

5.1.1 Pressure

The RC system's design, operating, and control setpoint pressures are listed in Table 5.1-2. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal leg, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics. The design pressures and data for the respective system components are listed in Tables 5.1-3 through 5.1-7.

5.1.2 Temperature

The design temperature selected for each component is above the maximum coolant temperature expected for that component under all normal and transient load conditions. The design and operating temperatures of the respective system components are listed in Tables 5.1-3 through 5.1-7. At the end of a cycle, the average reactor coolant temperature, T_{AVE}, may be reduced by 12 °F (less instrument error). This maneuver may result in a reactor coolant inlet temperature of approximately 547 °F. Reference 27 confirms that there should not be any significant effects on reactor coolant system materials and structural components and plant performance during normal and off-normal conditions associated with the T_{AVE} reduction.

5.1.3 Reaction Loads

All components in the RC system are supported and interconnected so that piping reaction forces result in combined mechanical and thermal stresses in equipment nozzles and structural walls which are within established code limits. Equipment and pipe rupture restraints are designed to absorb piping rupture reactor loads for the elimination of secondary accident effects and equipment foundation shifting.

The RC system is surrounded by concrete shield walls. These walls provide shielding to permit access into the containment vessel for inspection and maintenance of miscellaneous rotating equipment during rated power operation.

These shielding walls provide missile protection for the containment vessel.

Components of the RC system are designated as ASME Code Class A equipment, and are designed to maintain their functional integrity during an earthquake. The basic design guide for the seismic analysis is AEC publication TID-7024, "Nuclear Reactor and Earthquake."

In addition, each component of the system is designed to withstand the effects of cyclic loads due to temperature and pressure changes in the reactor system. These cyclic loads are introduced by normal unit load transients, reactor trip, and startup and shutdown operations. Design cycles are shown in Table 5.1-8.

5.1.4 Service

The service life of the RC system pressure components depends on the end-of-life material radiation damage, operational thermal cycles, quality manufacturing standards, environmental protection, and adherence to established operating procedures. The design service life is 40 years.

To establish the service life of the components of the RC system, the nuclear unit operating conditions involving the cyclic application of loads and thermal conditions have been established. The number of thermal and loading cycles to be used for design purposes is listed in Table 5.1-8. The transient cycles listed in this table are conservative and complete in that they include all significant modes of normal and emergency operation. A larger number of cycles of smaller magnitude than those described can be tolerated. The effect of individual transients and the sum of these transients are evaluated to determine the fatigue usage factor during the detail design and stress analysis. As specified in ASME Section III, the cumulative fatigue usage factor is less than 1.0 for the design cycles listed in Table 5.1-8.

NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification" required the re-evaluation of the cyclic fatigue of the Pressurizer Surge Line. Topical Report BAW 2127 (Reference 14), Final Submittal for Nuclear Regulatory Commission Bulletin 88-11 'Pressurizer Surge Line Thermal Stratification' with Supplements 2 (Reference 15) and 3 (Reference 16) describe the results of the revised evaluation. As part of this evaluation the design basis plant heatup and cooldown transients were completely redefined. Other transients were modified to include thermal stratification and striping. In addition to these changes, a number of transients were added and other modifications were made to the existing transients based on a review of the operating history and operating procedures of the plant. The number of design cycles for Ramp Loading and Unloading (8-100-8%) was significantly reduced because the plant is operated as a base loaded plant. Ramp loading and unloading have been separated in Table 5.1-8 and represent the lowered limit.

5.1.5 Material Radiation Damage

The beltline region of the reactor vessel, as defined in Appendix G of 10CFR50 (July, 1973), is the only region of the RC system that is exposed to significant neutron fluence levels. Therefore, a material radiation damage evaluation was performed on the beltline region. The reactor vessel material surveillance program will meet the requirements of the Appendix H of 10CFR50 (July, 1973).

The predicted irradiation damage adjustment of the reference temperature of the beltline region materials is described in B&W Topical Report BAW-10056A.

Figure 5.1-7 and Table 5.1-9 show the embrittling effect of neutron flux on steel plate and weldment specimens.

5.1.6 Volume Control

The only coolant removed from the RC system is the continuous bleed off through the Reactor Coolant Pump seals and the letdown to the makeup and purification system. The letdown flow rate is set at the desired rate by the operator positioning the letdown control valve and/or opening the stop valve for the letdown orifice. The Reactor Coolant Pump continuous bleed off is established by the design of the Reactor Coolant Pump seals.

To maintain a constant pressurizer water level, total makeup to the RC system must equal the letdown from the system. Total makeup consists of the seal injection water through the RC pump seals and reactor coolant makeup to the system through the RC makeup control valve. The pressurizer level controller provides automatic control of the valve to maintain the desired pressurizer water level. RC volume changes during station load changes exceed the capability of the RC makeup control valve and thus result in variations in pressurizer level. The level is returned to normal when the system returns to steady-state conditions.

5.1.7 Normal Operation

Normal operation of the RC system includes both power generating and hot shutdown operating phases. Power generation includes steady-state operation and normal unit load transients. During all phases of normal operation, the pressure of the RC system at the core outlet is maintained by the pressure controller, while the liquid level of the pressurizer is maintained by the makeup flow controller in the makeup and purification system.

When reactor power is less than the value corresponding to the steam generator low level limit, the reactor is controlled manually and RC average temperature varies with reactor power level. At power levels above the value corresponding to steam generator low level limit, the Integrated Control System (ICS) controls power based on the power demand and maintains a constant average RC average temperature.

During hot shutdown operations, when the reactor is subcritical, the temperature of the RC system is maintained by steam bypass to the main condenser. This is accomplished by the ICS, operating in the pressure control mode, which is set to maintain the pressure of the steam generator steam. Residual and decay heat from the core or operation of an RC pump(s) provides heat to overcome heat losses in the RC system.

During normal plant operation, the temperature of the water in the pressurizer quench tank remains at or below 120°F. If a pilot-operated relief valve (PORV) discharge occurs, the quench tank water temperature and level both increase. The quench tank is designed to be automatically restored to its nominal conditions if high temperature or high level setpoints are exceeded. Manual operator actions are required to stop the quench tank circulation pump when quench tank water temperature has been reduced to its normal operating temperature and to unisolate demineralized water to restore from a low level in the quench tank.

A SFAS signal to isolate containment will automatically close the quench tank inlet and outlet isolation valves. This would prevent automatic operation of the quench tank to maintain its normal water level and water temperature. If the quench tank pressure exceeds its normal operating range, the quench tank must be manually vented to the gaseous radwaste system.

5.1.8 Heatup

Heatup operations bring the reactor system from cold shutdown to no-load operating status. Before plant heatup, the RC loops are filled and vented, and the pressurizer is filled to the minimum level.

After filling and venting are completed, the pressurizer heaters are energized to pressurize the system. The pressure of the RC system is increased to obtain the required Net Positive Suction Head (NPSH) of the RC pumps. The RC pumps are then started one at a time to provide heating for the systems. During initial operation of the pumps, the pressure of the RC system is held above the minimum NPSH for the pumps and below the nil-ductility temperature limits for heatup.

The minimum liquid level in the pressurizer is maintained until the RC system reaches the hot no-load temperature. The pressurizer spray system is controlled manually and may be used in conjunction with the heaters to maintain system pressure in the proper range.

As the RC temperature increases, the pressurizer heaters maintain adequate suction pressure for the RC pumps. When the normal operating pressure of the pressurizer is reached, the pressurizer's heater and spray system controls are transferred from manual to automatic control.

5.1.9 Cooldown

Cooldown operations bring the reactor system from no-load operating status to cold shutdown. Before cooldown, the boron concentration of the reactor coolant is increased to that required for cold shutdown. Boric acid solution is added to the RC system from the chemical addition and boron recovery system via the makeup and purification system. If the cooldown is for refueling, the reactor coolant is degassed before cooldown to reduce the volume of hydrogen and fission gas.

Cooldown is a two-phase operation. During the first phase, RC system heat is removed by the steam generators. The Decay Heat Removal System is used during the second phase. When cooldown is initiated, fission product decay generates significant heat in the reactor core. The steam generators remove the decay heat, plus the heat input from operating RC pumps, and the sensible heat of the system components by steam formation. If an RC pump is operated, the pressurizer heaters are de-energized, and spray flow is used to cool the pressurizer to maintain the suction pressure required for the RC pump. If cooldown is required without the use of RC pumps, the RC system has a natural-circulation capability to remove decay heat during the first phase of cooldown. The analytical methods which are used to determine natural circulation performance of the RCS are described in Reference 9. The results of the analysis of the RCS operating under natural circulation conditions shows that this mode of operation provides an acceptable method of energy removal from the core transfer of energy to the secondary system through the steam generators.

The second phase of cooldown begins when the Decay Heat Removal System is actuated. The RC pumps are stopped when the decay heat system is actuated. Cooling of the pressurizer is continued with spray flow from the Decay Heat Removal System. When the pressurizer has been cooled, the spray is stopped and the RC system is depressurized by venting the pressurizer. The steam bubble is replaced with a nitrogen bubble.

5.1.10 Reliance on Process Systems

The principal heat removal systems that are interconnected with the RC system are the steam and feedwater systems and the Decay Heat Removal system. The steam generators and the steam, feedwater, and condensate systems are required to remove decay heat from the RC system over the range of temperatures from normal operating temperature to the decay heat system actuation temperature. All active components necessary for emergency core cooling in these systems are duplicated for reliability. Either steam generator is capable of removing decay heat and sensible heat.

Functional drawings of the steam and feedwater systems are shown in Figures 10.3-1 and 10.4-12. If the condensers are not available to receive the steam generated by decay heat, the resultant steam may be vented to the atmosphere. Auxiliary feed pumps supply water to the steam generators if the main feedwater pumps are inoperative. The auxiliary feedwater system is described in Subsection 9.2.7.

The Decay Heat Removal System is described in Chapter 6. The heat received by the Decay Heat Removal System is rejected to the Component Cooling Water system, which also possesses sufficient redundancy to ensure adequate operation. A functional drawing of the Decay Heat Removal System is presented in Figure 6.3-2A.

5.1.11 System Flow Diagram

A schematic flow diagram of the RC system at 110% design flow is presented in Figure 5.1-1. It denotes all major components, principal pressures, temperatures, flow rates, and coolant volume for the system.

5.1.12 System Functional Drawing

A functional drawing of the RC system and the primary sides of the auxiliary fluid systems and engineered safety feature systems connected to the RC system is presented in Figure 5.1-2, which shows the following:

a. The extent of the system located within the containment vessel.

b. The points of separation between the primary and secondary systems.

c. The isolation valves between the nuclear and non-nuclear sections of the systems.

5.1.13 System Elevation Drawings

A plan drawing and an elevation drawing showing principal dimensions of the RC system in relation to supporting or surrounding concrete structure and other equipment are shown in Figures 5.1-3 and 5.1-4, and the arrangement of the RC system is presented in Figures 5.1-5 and 5.1-6.

TABLE 5.1-1a

Design Conditions for RCPB

Component	Design pressure, psig	Design temperature, °F	Maximum test pressure [1]
Reactor vessel	2500	650	3125
Steam generator	2500	650	3125
Pressurizer	2500	670	3125
Reactor coolant piping	2500	650	3125
RC pumps	2500	650	3125

[1] When hydrostatically testing a system, the test pressure shall not exceed the maximum test pressure of any component in the system.

TABLE 5.1-1b

Components Within RCPB

Item	Design Pressure, psig	Design Temperature, °F	Shop hydrostatic test pressure, psig	Seismic	Active	Inactive
Valves-seal	3050	200	5400	I	X	
Injection flow	3050	200	5400	I	X	
Isolation	3050	200	5400	I	X	
	3050	200	5400	I	X	
Valves-pump seal	2500	200	5400	I	X	
Return isolation	2500	200	5400	I	X	
	2500	200	5400	I	X	
	2500	200	5400	I	X	
Valves-letdown	2500	650	5400	I		X
Coolers Inlet	2500	650	5400	I		X
Valves-letdown	2500	650	5400	I	X	
Coolers outlet	2500	650	5400	I	X	
Valves-HP	3050	200	5400	I	X	
Injection	3050	200	5400	I	X	
	3050	200	5400	I	X	
	3050	200	5400	I	X	
Valve-seal Return isolation	2500	300	5400	I	X	
Valve-makeup Isolation	3050	200	5400	I	X	
Valve-letdown Cooler isolation	2500	300	5400	I	X	
Valve-Pressurizer Spray control	2500	670	5400	I	X	
Valve-Pressurizer Spray isolation	2500	670	5775	I		X
Valve-Pressurizer Relief isolation	2500	670	5500	I		X

TABLE 5.1-1b (Continued)

Components Within RCPB

Item	Design Pressure, psig	Design Temperature, °F	Shop hydrostatic test pressure, psig	Seismic	Active	Inactive
Valve Pressurizer Pilot Operated Relief (PORV)	2500	670	3800	I		X
Valve Pressurizer	2500	670	3750	I		X
ASME Code Safety	2500	670	3750	I		X
Valves-LP	2500	350	5400	I	X	
Injection	2500	350	5400	I	X	
Valve-DH Removal Outlet	2500	350	5400	I		X
Valve-DH Removal Outlet	2500	650	5400	I		X
Coolers-Letdown	Tube side 2500	600	Tube side 4450			
	Shell side 200	350	Shell side 300	I		

TABLE 5.1-2

Tabulation of Reactor Coolant System Pressure Settings

Item	Pressure, psig
Design pressure	2500
Operating pressure	2155
Pressurizer ASME Code Safety Valves	2500
Pilot Operated Relief Valve (PORV)	2450
High-pressure trip	See Tech. Specs.
High-pressure alarm	2255
Low-pressure alarm	2055
Low-pressure trip	See Tech. Specs.

TABLE 5.1-3

Reactor Vessel Design Data

Item	Data
Design pressure, psig	2500
Hydrotest pressure, (cold), psig	3125
Design/operating temperature, °F	650/608**
Overall Height of Vessel and Closure Heat, ft-in.	39-0
Straight shell thickness, in.	8-7/16
Water volume, cu ft	4058
Thickness of insulation, in.	4
Number of reactor closure head studs	60
Flange ID, in.	165
Shell ID, in.	171
Inlet nozzle ID, in.	28
Outlet nozzle ID, in.	36
Core flooding water nozzle ID, in.	11-1/2
Diameter of reactor closure head studs, in.	6-1/2
Coolant operating temperature inlet/outlet, °F	*557/608
Reactor coolant flow, lb/hr	137.90×10^6
Shell cladding minimum thickness, in.	1/8
Shell cladding nominal thickness, in.	3/16
Closure head minimum thickness, in.	6-5/8
Lower head minimum thickness, in.	5
Control rod drive nozzles ID, in.	2.76
RV head continuous vent nozzle ID, in.	2.74
Axial power-shaping rod drive nozzle, ID, in.	2.76
Incore instrumentation nozzle dia, in.	3/4 Sch 160
Dry weight (est.) lb	
Vessel	664,100
Closure head	162,700
Studs, nuts, and washers	38,400

*At the end of a cycle, the average reactor coolant temperature, T_{AVG}, may be reduced by 12°F (see Section 5.1.2).
**The reactor vessel closure head along with associated appurtenances and the continuous vent line, have been analyzed for a revised head fluid operating temperature of 630F

TABLE 5.1-4

Pressurizer Design Data

Item	Data
Design/operating pressure, psig	2500/2185
Hydrotest pressure (cold), psig	3125
Design/operating temperature, °°F	670/650
Normal water volume, ft^3	800
Normal steam volume, ft^3	700
Design insurge, ft^3	250
Design outsurge, ft^3	250
Design spray flow, gpm	190
Design continuous spray recirculation flow rate, gpm	0.75-3.0
Design heatup rate, °°F/hr	100
Design cooldown rate, °°F/hr	100
Design heat loss through vessel insulation, Btu/hr-ft^2, max.	80
Electric heater capacity, kw	1638*
Overall height, ft-in.	44 ft. 11 ¾ in.
Shell OD, in.	96 3/8
Shell minimum thickness, in.	6.188
Dry weight, lb (estimated)	304,000
Surge line nozzle dia, in.	10
Spray line nozzle dia, in.	4
Pressurizer ASME code safety valve, in.	4 x 6
Vent nozzle dia, in.	1
Sample line nozzle, dia, in.	1
Thermowell dia, in.	3/8
Level sensing nozzle dia, in.	3/4
Heater bundle dia, in.	20 ¼
Manway opening ID, in	16
Pilot-operated relief valve, in.	2 ½ x 4

* Original design capacity was 1638 kw. Heater capacity is less than original due to loss of some heater function. An analyses was conducted with an upper bound of pressurizer heat loss of 210 kw assuming no leakage, with the results showing 85 kw of available heater capacity prevents loss of RCS sub-cooling margin for 15 hours following loss of offsite power. Controlled drawings indicate current heater capacity.

TABLE 5.1-5

Steam Generator Design Data

Item	Data per unit
Steam conditions at full load, outlet nozzles	
Steam flow, lb/hr	6.12×10^6
Steam temperature, °F	570 (35F superheat)
Steam pressure, psia	925.5
Feedwater temperature, °F	470
Reactor coolant flow, lb/hr	68.95×10^6
Reactor coolant side	
Design pressure, psig	2500
Design/operating temperature, °F	
Inlet	650/608
Outlet	650/557 (2)
Hydrotest pressure, psig	3125
Coolant volume (Hot), cu ft	2023.8
Secondary side	
Design pressure, psig	1150
Design temperature, °F	630
Hydrotest pressure, psig	1500 (limit 10 hydrostatic tests) (1)
Net volume, cu ft	3485
Dimensions	
Tubes, OD/min wall. in.	0.625/0.034
Overall height (including skirt), ft-in.	74-10 5/8
Shell OD. in.	148 1/8
Shell minimum thickness, in.	5 (Lower Tubesheet & Feedwater Connection) 5 1/4 (Upper Tubesheet)

(1) Hydrostatic test limit of 10 was specified in Replacement Once Through Steam Generator Certified Design Specification, TS-3985 Rev. 3.

(2) At the end of a cycle, the average reactor coolant temperature, T_{AVE}, may be reduced by 12°F (see Section 5.1.2).

TABLE 5.1-5 (Continued)

Steam Generator Design Data

Item	Data per unit
Shell minimum thickness, in.	3
Tube sheet thickness, in.	22.1875 (Unclad)
Dry weight, lb	975,287
Exposed tube length, ft-in.	52-5
Nozzles – reactor coolant side	
Inlet nozzle ID, in.	36
Outlet nozzle ID, in.	28
Continuous Vent Nozzle ID, in.	2 1/8
Manway ID, in.	16
Handholes, in.	6
Nozzles – secondary side	
Steam nozzle dia, in.	24
Vent nozzle dia, in.	1-1/2
Drain nozzle dia, in.	1-1/2
Drain nozzle dia, in.	1
Level sensing nozzle dia, in.	1
Thermowell dia, in.	1-1/2
Manway ID, in.	16
Feedwater nozzle dia, in.	3
Auxiliary feedwater nozzle dia, in.	3
Handholes dia, in.	6
Inspection Openings, dia, in.	3

TABLE 5.1-6

Reactor Coolant Pump Design Data

Item	Data-per-unit
Number of pumps	4
Design pressure, psig	2500
Design temperature, °F	650
Operating speed (nominal), rpm	1185
Pumped fluid temperature, °F	70 to 596
Developed head, ft	358
Capacity, gpm	90,670
Hydraulic efficiency, %	86
Seal water injection (Max.), gpm	10 per pump
Seal water return (Max.), gpm	6 per pump
Overall unit height, ft-in.	25-7
Water volume, cu ft	98
Motor stator frame diameter, ft	8
Pump-motor moment of inertia, $lb\text{-}ft^2$	74,625
Motor data	
Type	Squirrel-cage induction, single speed
Voltage, volts	13,800
Phase	3
Frequency, Hz	60
Starting	Across-the-line
Input (hot reactor coolant), kw/amps	5400/260
Input (cold reactor coolant), kw/amps	7200/347
Service factor	1.15 (in 50^{o}C ambient)

TABLE 5.1-7

Reactor Coolant System Piping Design Data

Reactor Inlet Piping

Pipe ID, in.	28
Design pressure/temperature, psig/°F	2500/650
Hydrotest pressure, psig	3125
Minimum thickness, in.	2-1/4
Coolant volume (hot-system total), ft^3	884
Dry weight, system total, lb (estimated)	106,450

Reactor Outlet Piping

Pipe ID, in.	36
Design pressure/temperature, psig/°F	2500/650
Hydrotest pressure, psig	3125
Minimum Thickness	
Original Piping (from Reactor to Straight Leg Before Flowmeter):	
Straight Leg:	2 5/8”
Elbow Intrados:	3 3/8”
Elbow Extrados:	2 7/8”
Replacement Piping (Flowmeter to ROTSG)	
Straight Leg:	2 11/16”
U-Bend:	3 11/16”
Coolant volume (hot-system total), $ft.^3$	1460
Dry weight, system total lb (estimated)	270,900

Pressurizer Surge Piping

Pipe size, in.	10 Sch 140
Design pressure/temperature, psig/°F	2500/670
Hydrotest pressure, psig	3125
Coolant volume, hot, ft^3	20
Dry weight, lb (estimated)	5000

Pressurizer Spray Piping

Pipe size, in.	2-1/2 Sch 160
Design pressure/temperature, psig/°F	2500/650
Hydrotest pressure, psig	3125
Coolant volume, hot, ft^3	2
Dry weight, lb (estimated)	650

Reactor Head Vent

Pipe size, in.	2-1/2 Sch 160
Design Pressure/Temperature, psig/°F	2500/650
Hydrotest Pressure, psig	3125

TABLE 5.1-7 (Continued)

Reactor Coolant System Piping Design Data

Function	No.	Dia. in.	Material
On reactor inlet piping			
High pressure injection	4	2-1/2 Sch 160	carbon steel, SS clad (1)
Pressurizer spray	1	2-1/2 Sch 160	Stainless steel
Drain/letdown	1	2-1/2 Sch 160	carbon steel, SS clad and buttered
Drain	3	2-1/2 Sch 160	carbon steel, SS clad and buttered
Thermowell	4	3/8	Inconel
Fast response temp conn	4	0.623	Inconel
Pressure taps	2	3/4 Sch 160	SS
On reactor outlet piping			
Decay	1	12 Sch 140	carbon steel, SS clad and buttered
Vent	2	1 Sch 160	Inconel UNS N06690 Thermally Treated (TT)
Conn on flow meters	4	3/4 Sch 160	Inconel UNS N06690 TT
Pressure sensing	4	3/4 Sch 160	Inconel UNS N06690 TT
Thermowell	2	3/8	Inconel UNS N06690 TT
Fast response temp conn	4	0.623	Inconel UNS N06690 TT
Surge line	1	10 Sch 140	carbon steel, SS clad and buttered
Level indicating	1	3/4 Sch 160	Inconel
On pressurizer surge piping			
Drain	1	1 Sch 160	Stainless steel
On pressurizer spray piping			
Auxiliary spray	1	1-1/2 Sch 160	stainless steel
Spray valve bypass	2	1/2 Sch 160	stainless steel

(1) With stainless steel safe end added stress relief.

TABLE 5.1-8

Transient Cycles Design Life [1]

No.	Description (ASME Category)	Design Cycles
1	Heatup from cold shutdown and cooldowns to cold shutdown	
	1A- Heatup from 70F to 8% Full Power (Normal) [2, 5]	240
	1B- Cooldown from 8% Full Power (Normal) [2, 5]	240
	1C-Natural Circulation Cooldown (Emergency)[3, 6]	20
2	Power change 0 to 15% and 15% to 0% (Normal)	1,440
3	Power loading 8% to 100% power (Normal) [4]	1,800
4	Power unloading 100% to 8% power (Normal) [4]	1,800
5	10% Step Load Increase (Normal)	8,000
6	10% Step Load Decrease (Normal)	8,000
7	Step Load Reduction (100% to 8%)- (Upset)	
	7A-Resulting from turbine trip	160
	7B-Resulting from electrical load reduction	150
8	Reactor Trip (Upset)	
	8A-Low RC flow directly causes Rx trip (Upset)	40
	8B-High RC outlet temperature, high RC pressure or overpower trip-assumes a turbine trip occurs without automatic control system action. (Upset)	160
	8C-High RC pressure resulting from loss of feedwater (Upset)	88
	8D-Other trips, including the following (Upset):	112
	(1) Any reactor trip which meets the definition of another transient classification (e.g., Transients 11, 15, 16, and 17) will also be recorded under 8D.	
	(2) Any reactor trip which does not fit into any other category will be classified 8D.	
	8E-Similar to 8A but RC pumps are tripped (emergency)[3, 6]	20
9	Rapid Depressurizations	
	9A-Rapid RCS Depressurization (Upset)	40
	9B-Rapid Depressurization, trip RC Pumps (Emergency)[3, 6]	10
10	Change of reactor coolant flow (typical change of flow transient is loss of one RCP) without Reactor Trip (Upset)	20
11	Rod withdrawal accident (Upset)	40
12	Hydrotests (Test)	
	12A-RCS Components Except OTSG Secondary (includes 5 shop tests)	20
	12B-OTSG Secondary (includes 10 shop tests)	35
13	Deleted (formerly Steady State Power Variations)	Not Applicable
14	Control Rod Drop (Upset)	40
15	Loss of Station Power (Upset)	40
16	Steam line failure (Faulted)	1
17	Steam generator boiling dry	
	17A-Loss of feedwater to one steam generator (Upset)	20
	17B-Stuck open turbine bypass valve (Emergency)[3]	10

TABLE 5.1-8 (Continued)

Transient Cycles Design Life [1]

No.	Description (ASME Category)	Design Cycles
18	Loss of feedwater heater (Upset)	40
19	Feed and bleed operations (Normal) [4]	4,000
20	Makeup and Pressurizer spray transients	
	20A-Makeup flow Transient 1 (Normal) [4]	30,000
	20B-Makeup flow Transient 2 (Normal) [4]	4.0E+6
	20C-Spray Valve/Pressurizer Spray Nozzle (Normal) [4]	20,000
21	Loss of coolant accident (LOCA) (Faulted)	1
22	Test Transients	
	22A1-High pressure injection system (Normal) [12]	40
	22A2-HPI System Pressure Isolation Integrity Test	13 (Nozzles 1-1 and 1-2) 40 (Nozzles 2-1 and 2-2)
	22B-Core flooding check valve (Normal)	240
23	Steam generator filling, draining, flushing and cleaning (Normal)	
	23A-Steam generator secondary side filling	
	Condition 1 [7, 11]	120
	Condition 2 [8, 11]	120
	23B-Steam generator primary side filling	
	Condition 1 [9, 11]	120
	Condition 2 [10, 11]	120
	23C-Steam generator flush[11]	40
	23D-Steam generator chemical cleaning[11]	20
24	Hot functional testing (Normal)	1
25	Decay Heat Removal Swapping Transient[13]	20

Footnotes:

1. Table 5.1-8 includes thermal design cycles only. Component design calculations also consider mechanical loads including 650 cycles of the maximum probable earthquake (aka, Operational Basis Earthquake); the reactor cavity seal plate is designed for 50 OBE cycles.

2. Reactor cavity seal plate limited to 50 HU/CD cycles.

3. ASME Classification is Emergency and is not required to be considered for calculation of peak stress and cumulative usage (ASME III, NB-3224.4).

4. Transient cycles not counted due to large number of design cycles.

5. Auxiliary feedwater bolted nozzles are limited to 875 transient cycles (An RCS heatup and cooldown is considered to be one transient cycle. Bolting/unbolting of the nozzles is considered to be one transient cycle. Also, an event or procedure that initiates auxiliary feedwater to the steam generator is analyzed for a transient cycle.).

6. For reactor vessel head vent line only.

TABLE 5.1-8 (Continued)

Transient Cycles Design Life [1]

7. Primary side: ≤200 °F, 0 - 485 psig; Secondary side:" ≥140 °F, 0 psig; Feedwater: 50 - 225 °F

8. Primary side: ≤120 °F, 0 - 485 psig; Secondary side: ≥60 °F, 0 psig; Feedwater: 50 - 225 °F

9. Primary fill water: 50 °F, 0 psig; Secondary side: 140 °F, 0 psig

10. Primary fill water: 140 °F, 0 psig; Secondary side: 50 °F, 0 psig

11. Transient is not counted as it is not a fatigue significant event.

12. Transient is not applicable to Davis-Besse. High pressure injection pumps recirculate back to the Borated Water Storage Tank during the High Pressure Injection System Test and therefore, no inventory is added to the Reactor Coolant System.

13. The significant transients which affect the restrictor and weld of the core flood nozzles are heatup and cooldown (transient numbers 1A and 1B), core flooding system periodic test (transient number 22B), and decay heat removal (DHR) swapping (transient number 25). For transient number 25, the transient cycles are not counted. The DHR Swapping Transient was established to address historical practices related to the DHR train swap. Current Davis-Besse procedures dictate that the RCPs are run during plant cooldown to approximately 160°F RCS temperature. The DHR trains are not swapped until the RCS temperature has been significantly reduced and therefore, a DHR Swapping Transient does not occur.

TABLE 5.1-9

Residual Elements Content and Irradiation Embrittlement Sensitivity
of A533 Steel Plates and Weldments – Reference for Figure 5.1-7

Ref	Grade and Class	Type	Specimen orientation and location	Weight %		Irrad Temp, °F	Neut-fluence (fiss spect), 1019 n/cm2 > 1 MeV	Incr In Cv 30 ft-lb tr temp, °F
				Cu	P			
1	B-1	Plate	RW 1T	0.14	0.009	550	2.3	125
1	B-1	Plate	RW 1/4T	0.14	0.009	550	2.3	120
1	B-1	Plate	RW 1/2T	0.14	0.009	550	2.3	120
1	B-1	Plate	RW 1T	0.14	0.010	550	2.3	80
1	B-1	Plate	RW 1/4T	0.14	0.010	550	2.3	95
1	B-1	Plate	RW 1/2T	0.14	0.010	550	2.3	95
1	B-1	Plate	RW 1/4T	0.19	0.010	550	0.5	140
1	B-1	Plate	RW 1/4T	0.19	0.010	550	1.7	190
1	B-1	Plate	RW 1/4T	0.09	0.008	550	0.22	0
1	B-1	Plate	RW 1/4T	0.09	0.008	550	2.0	80
1	B-1	Plate	WR 1/4T	0.09	0.008	550	2.0	90
1	B-2	Plate	RW 1/4T	0.09	0.008	550	0.5	35
1	B-2	Plate	RW 1/4T	0.09	0.008	550	2.0	75
1	B-2	Plate	RW 1T	0.09	0.008	550	2.0	65
1	B-2	Plate	RW 1/2T	0.09	0.008	550	2.0	60
1	B-1	Plate	RW 1/4T	0.12	0.008	550	1.7	70
1	B-1	Plate	RW 1/4T.	0.11	0.008	550	1.7	85
1	B-1	SA Weld	WL 1/4T	0.22	0.015	550	1.7	200
1	B-1	HA%	1/4T	--	--	550	1.7	145 (max)
1	B-1	HA%	1/4%	--	--	550	1.7	70 (min)
1	C-2	Plate	RW 1/4T	0.20	0.006	550	1.4	180
1	C-2	SA Weld	WL 1/4T	0.27	0.016	550	1.4	205
1	B-1	Plate	RW 1/4T	0.12	0.008	550	1.4	50
1	B-1	E. Weld	WL 1/4T	0.19	0.008	550	1.8	165
2	B-1	Plate	RW 1/4T	0.13	0.008	550	2.8	140
2	B-1	Plate	RW 1/4T	0.03	0.008	550	2.8	40
2	B-2	Plate	RW 1/4T	0.13	0.008	550	2.8	125
2	B-2	Plate	RW 1/4T	0.03	0.008	550	2.8	65

TABLE 5.1-9 (Continued)

Residual Elements Content and Irradiation Embrittlement Sensitivity of A533 Steel Plates and Weldments – Reference for Figure 5.1-7

Ref	Grade and Class	Type	Specimen orientation and location	Weight % Cu	Weight % P	Irrad temp, °F	Neut-fluence (fiss spect), 10^{19} n/cm^2 > 1 MeV	Incr In Cv 30 ft-lb tr temp, °F
2	B-1	Plate	WR 1/4T	0.03	0.008	550	3.1	70
3	B-1	Plate	WR 1/4T	0.14	0.008	550	2.6	165
3	B-1	Plate	RW 1/4T	0.14	0.008	550	2.7	170
3	B-1	Plate	WR 1/4T	0.18	0.008	550	2.8	200
3	B-1	SA Weld		0.23	0.011	550	2.5	270
4	B-1	Plate	RW 1T	0.09	0.008	550	0.5	0
4	B-1	Plate	RW 1T	0.09	0.008	550	2.4	60
4	B-1	Plate	RW 1/4T	0.09	0.008	550	0.5	0
4	B-1	Plate	RW 1/4T	0.09	0.008	550	2.4	85
4	B-1	Plate	WR 1/4T	0.09	0.008	550	0.5	0
4	B-1	Plate	WR 1/4T	0.09	0.008	550	2.4	105
5	B-1	Plate	RW 1T	0.18	0.012	555	0.47	70
5	B-1	Plate	RW 1/4T	0.18	0.012	560	0.94	95
5	B-1	Plate	RW 1/2T	0.18	0.012	560	1.05	130
5	B-1	Plate	RW 3/8T	0.14	0.012	550	0.52	50
5	B-1	SA Weld	WL 1/12T	0.15-0.33	0.013-0.017	565	1.15	185
5	B-1	E. Weld	WL 1/4T	--	0.006	555	1.12	75
6	B-1	SA Weld	WL 1/4T	0.23	0.011	550	2.5	270
7	B-2	SA Weld	WL 1/4T	0.05	0.01	550	2.7	10

[1] J. R. Hawthorne and U. Potapovs, Initial Assessments of Notch Ductility Behavior of A533 Pressure Vessel Steel with Neutron Irradiation, NRL Report 6772, Naval Research Laboratory, November 22,1968.

[2] J. R. Hawthorne, “Demonstration of Improved Radiation Embrittlement Resistance of A533B Steel Through Control of Selected Residual Elements,“ Irradiation Effects on Structural Alloys for Nuclear Reactor Applications, ASTM STP 484, Amer. Society Testing Mats. (1970) pp. 96-127.

TABLE 5.1-9 (Continued)

Residual Elements Content and Irradiation Embrittlement Sensitivity
of A533 Steel Plates and Weldments – Reference for Figure 5.1-7

[3] J. R. Hawthorne, "Post-Irradiation Dynamic Tear and Charpy-V Performance of 12-In. Thick A533B Steel Plates and Weld Metal," Nuc Eng & Design, 17 (1971), pp 116-130.

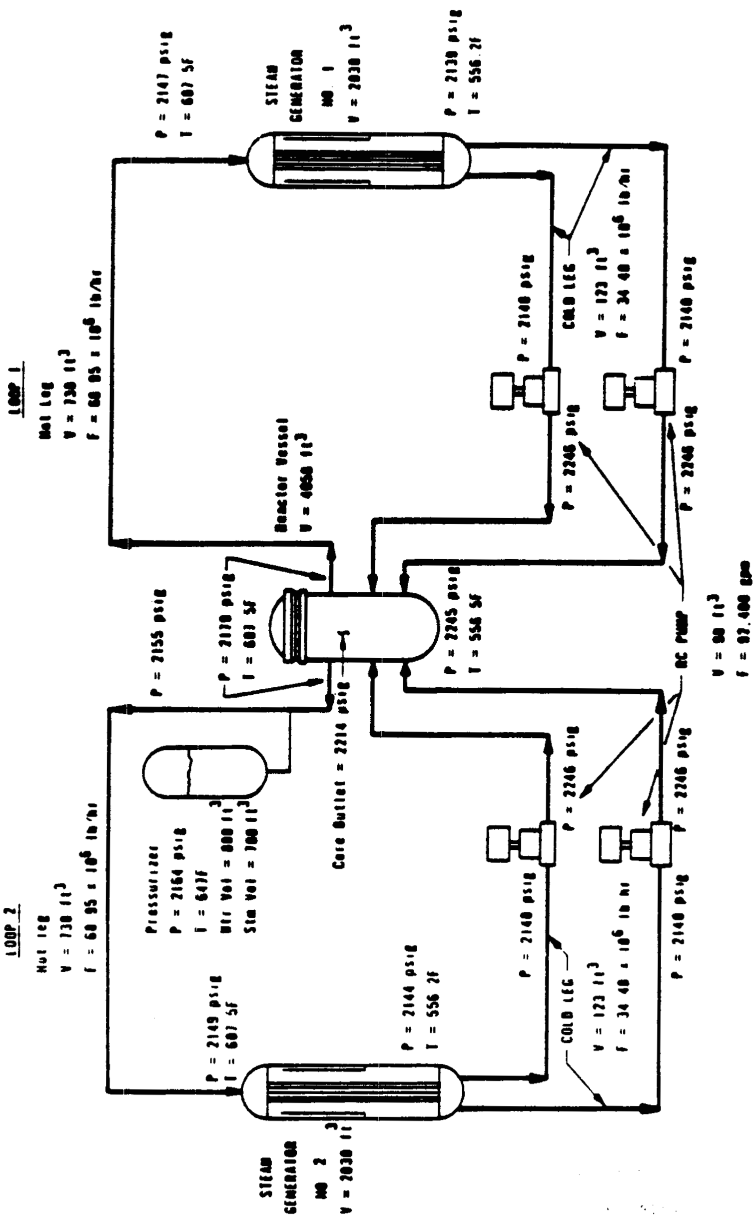
[4] L. E. Steel, et al., Irradiation Effects on Reactor Structural Materials, Quarterly Progress Report, August-October 1968, NRL Memorandum Report 1937, Naval Research Laboratory.

[5] R. G. Berggren and W. L. Stelzman, "Radiation Strengthening and Embrittlement in Heavy Section Plate and Welds", Nuc Eng Design, 17 (1971), pp 103-115.

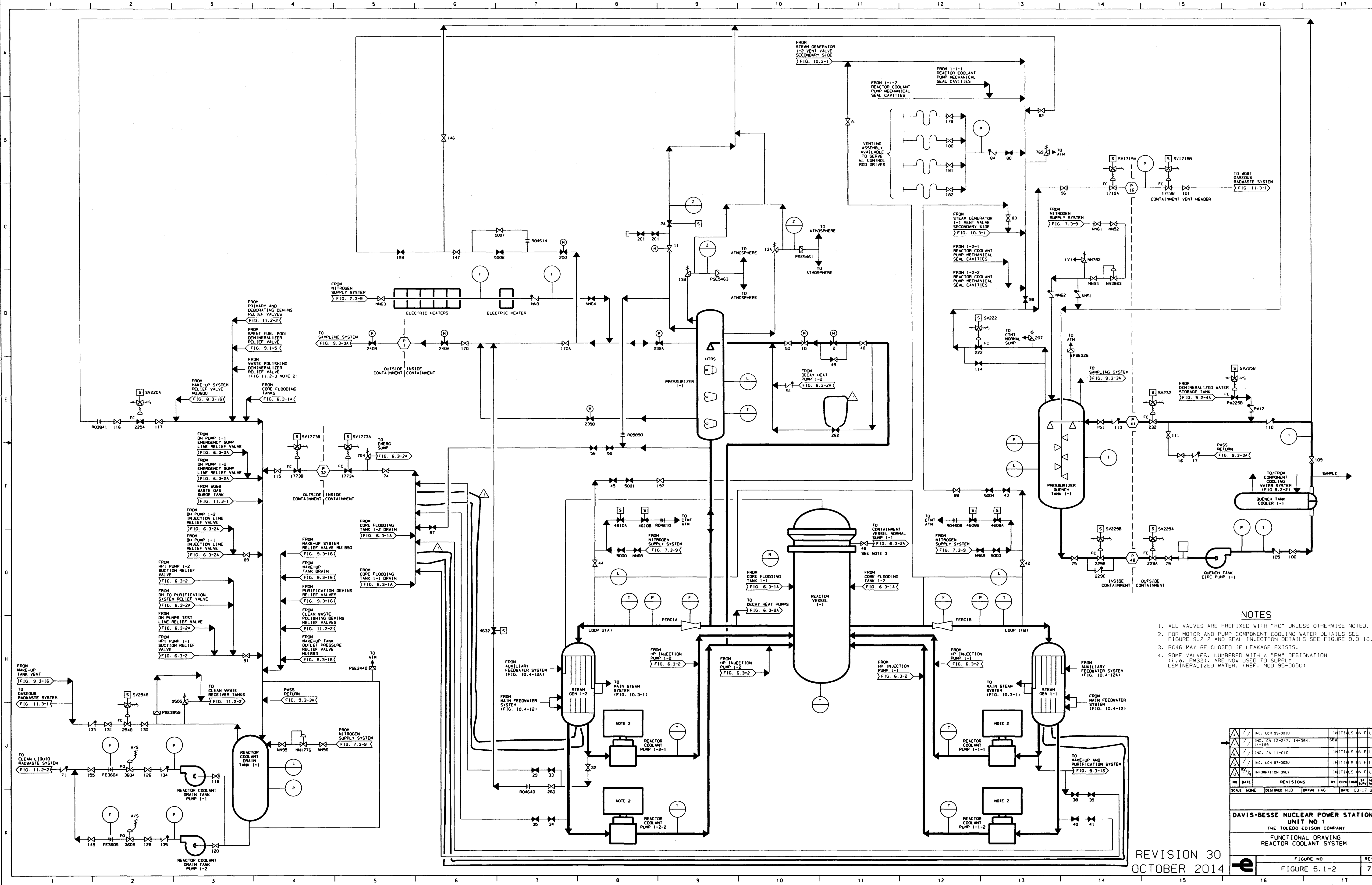
[6] J. R. Hawthorne, "Post-Irradiation Dynamic Tear Performance of 12-Inch A533B Submerged Arc Weldment," Irradiation Effects on Reactor Structural Materials, Quarterly Progress Report, August-October 1970, T. T. Claudson, Ed., WHAN-FR-40 1, Hanford Engineering Development Lab., Richland, Washington, Jan. 1971.

[7] J. R. Hawthorne, "A Radiation Resistance Weld Metal for Fabricating A533B Reactor Vessels," Irradiation Effects on Reactor Structural Materials, Quarterly Progress Report, May 1 – July 31, 1971, NRL Memorandum Report 2328, Naval Research Laboratory, pp 10-12.

[8] L. E. Steel, Major Factors Affecting Neutron Irradiation Embrittlement of Pressure-Vessel Steels and Weldments, NRL Report 7176, Naval Research Laboratory, October 1970.



DAVIS-BESSE NUCLEAR POWER STATION
REACTOR COOLANT SYSTEM FLOW
DIAGRAM AT FULL POWER
STEADY STATE CONDITIONS
FIGURE 5.1-1 REVISION 0
JULY 1982



NOTES

1. ALL VALVES ARE PREFIXED WITH "RC" UNLESS OTHERWISE NOTED.
2. FOR MOTOR AND PUMP COMPONENT COOLING WATER DETAILS SEE FIGURE 9.2-2 AND SEAL INJECTION DETAILS SEE FIGURE 9.3-16.
3. RC46 MAY BE CLOSED IF LEAKAGE EXISTS.
4. SOME VALVES, NUMBERED WITH A "PW" DESIGNATION (i.e. PW32), ARE NOW USED TO SUPPLY DEMINERALIZED WATER. (REF. MOD 95-0050)

NO	DATE	REVISIONS	BY	CHK	ENGR	SA SUPV	NED MGR
4	/ /	INC. UCN 99-001U	INITIALS ON FILE				
7	/ /	INC. CN 12-247, 14-094, 14-189	SBW				
6	/ /	INC. CN 11-010	INITIALS ON FILE				
5	/ /	INC. UCN 97-063U	INITIALS ON FILE				
0	05/21/91	INFORMATION ONLY	INITIALS ON FILE				

SCALE NONE | DESIGNED MJD | DRAWN PAG | DATE 03-17-90

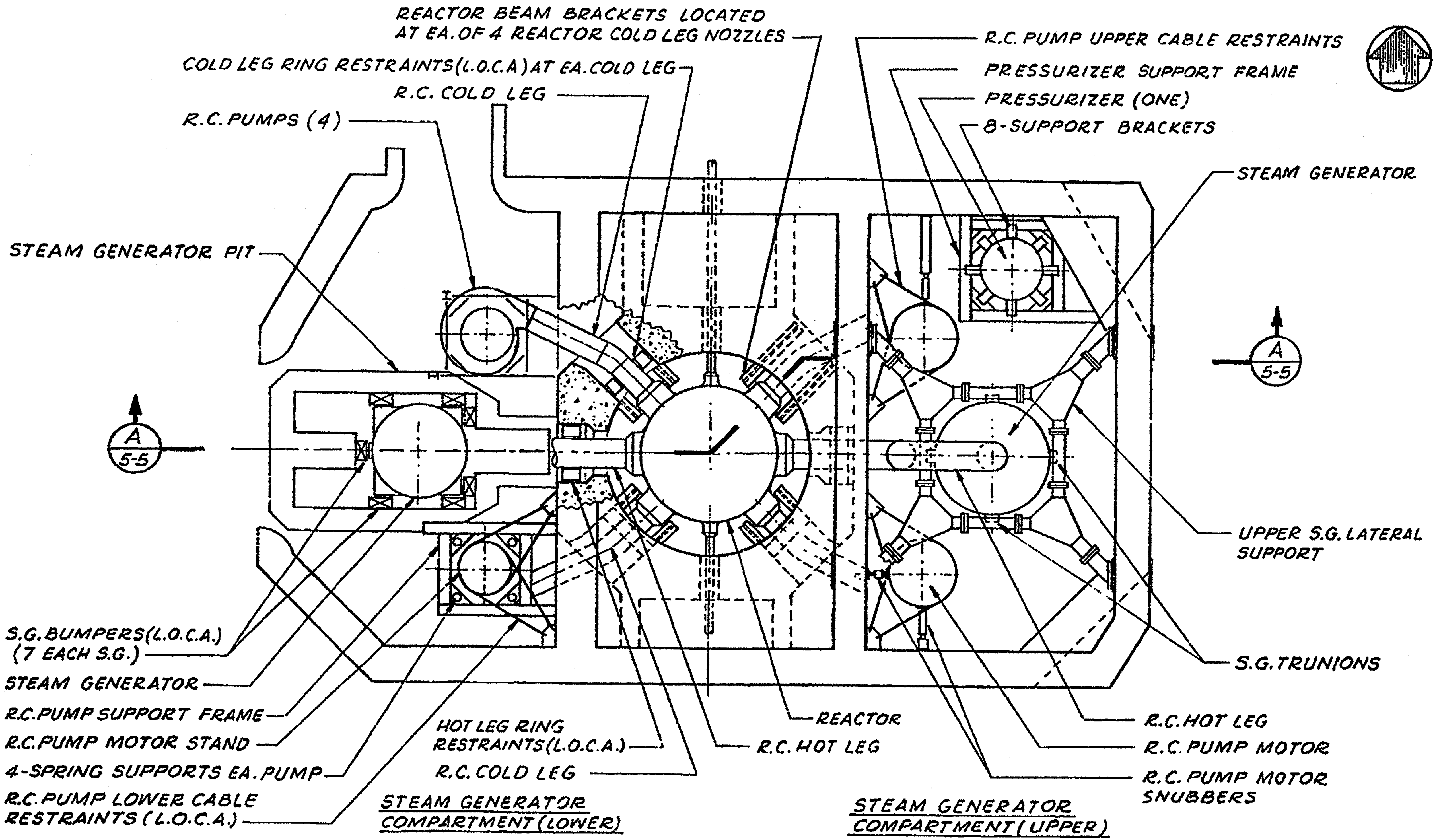
DAVIS-BESSE NUCLEAR POWER STATION UNIT NO 1
THE TOLEDO EDISON COMPANY

FUNCTIONAL DRAWING
REACTOR COOLANT SYSTEM

FIGURE NO: FIGURE 5.1-2 | REV: 7

REVISION 30
OCTOBER 2014

DFN=/G/USAR/UFIG512.DGN
DB 07-30-14



NOTES:

1. PIPE RESTRAINTS NOT SHOWN.
2. DELETED
3. WHIP RESTAINTS ARE NO LONGER REQUIRED TO BE INSTALLED. (REF. SECT. 3.6.2.2.1)

DAVIS-BESSE NUCLEAR POWER STATION

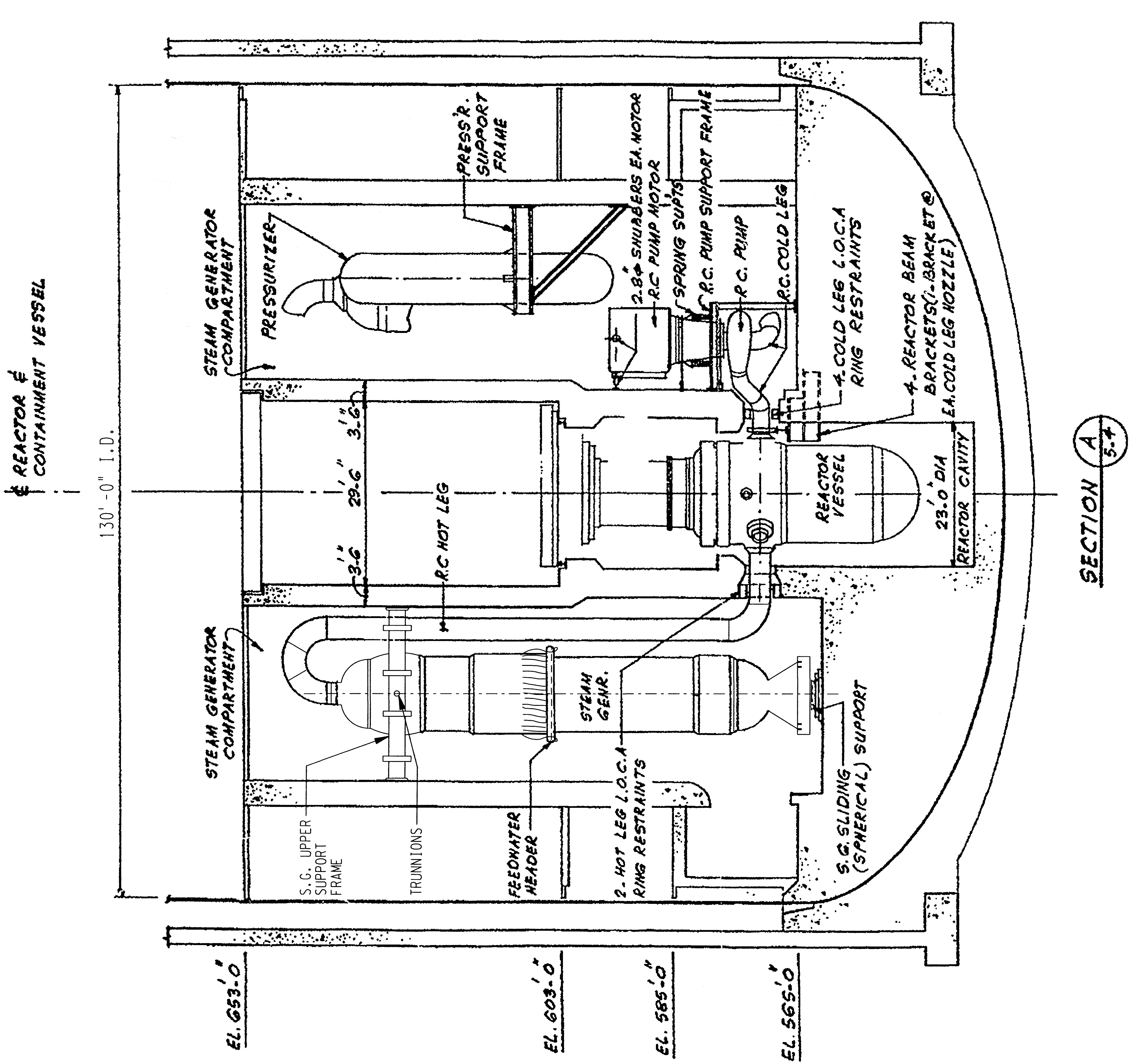
REACTOR COOLANT SYSTEM AND SUPPORTING STRUCTURES - PLAN

FIGURE 5.1-3

REVISION 30

OCTOBER 2014

DB 09-30-14 DFN:G/USAR/UFIG513.DGN/TIF



NOTES:

1. PIPE RESTRAINTS NOT SHOWN.
2. DELETED
3. WHIP RESTAINTS ARE NO LONGER REQUIRED TO BE INSTALLED. (REF. SECT. 3.6.2.2.1)

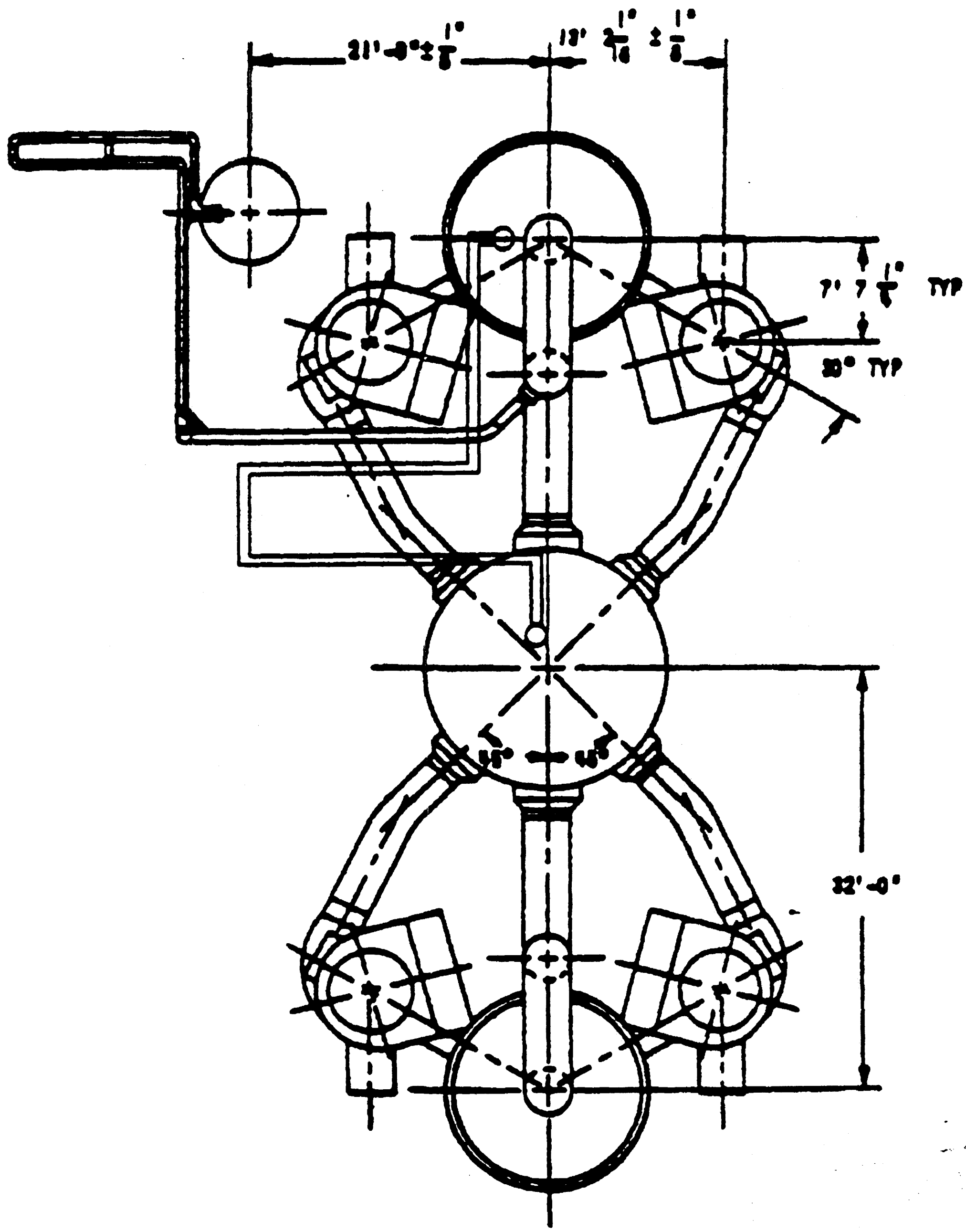
DAVIS-BESSE NUCLEAR POWER STATION

REACTOR COOLANT SYSTEM AND SUPPORTING STRUCTURES - ELEVATION

FIGURE 5.1-4

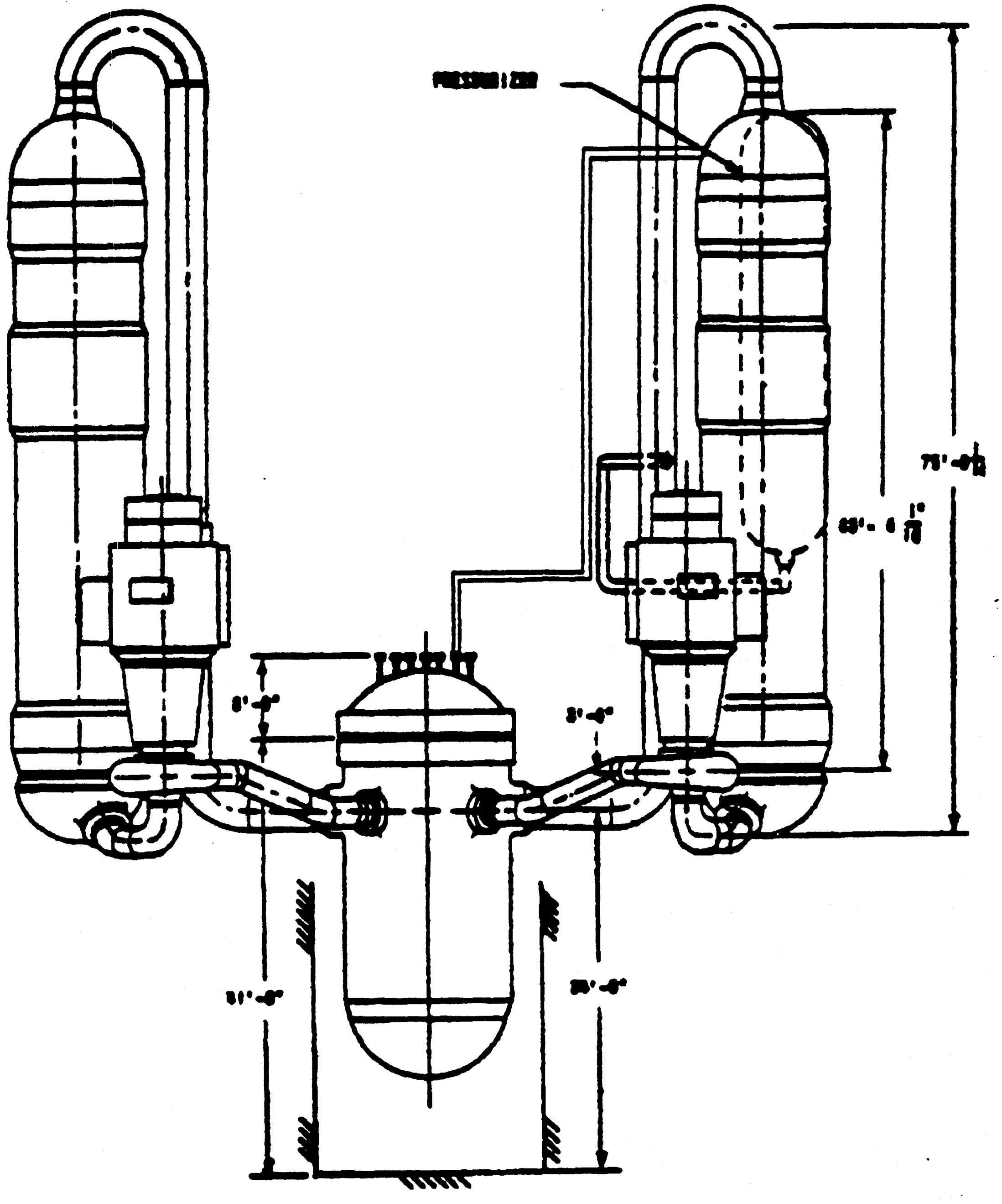
REVISION 30
OCTOBER 2014

DB 07-16-14 DFN:G/USAR/UFIG514.DGN/TIF



DAVIS-BESSE NUCLEAR POWER STATION
REACTOR COOLANT SYSTEM
ARRANGEMENT - PLAN

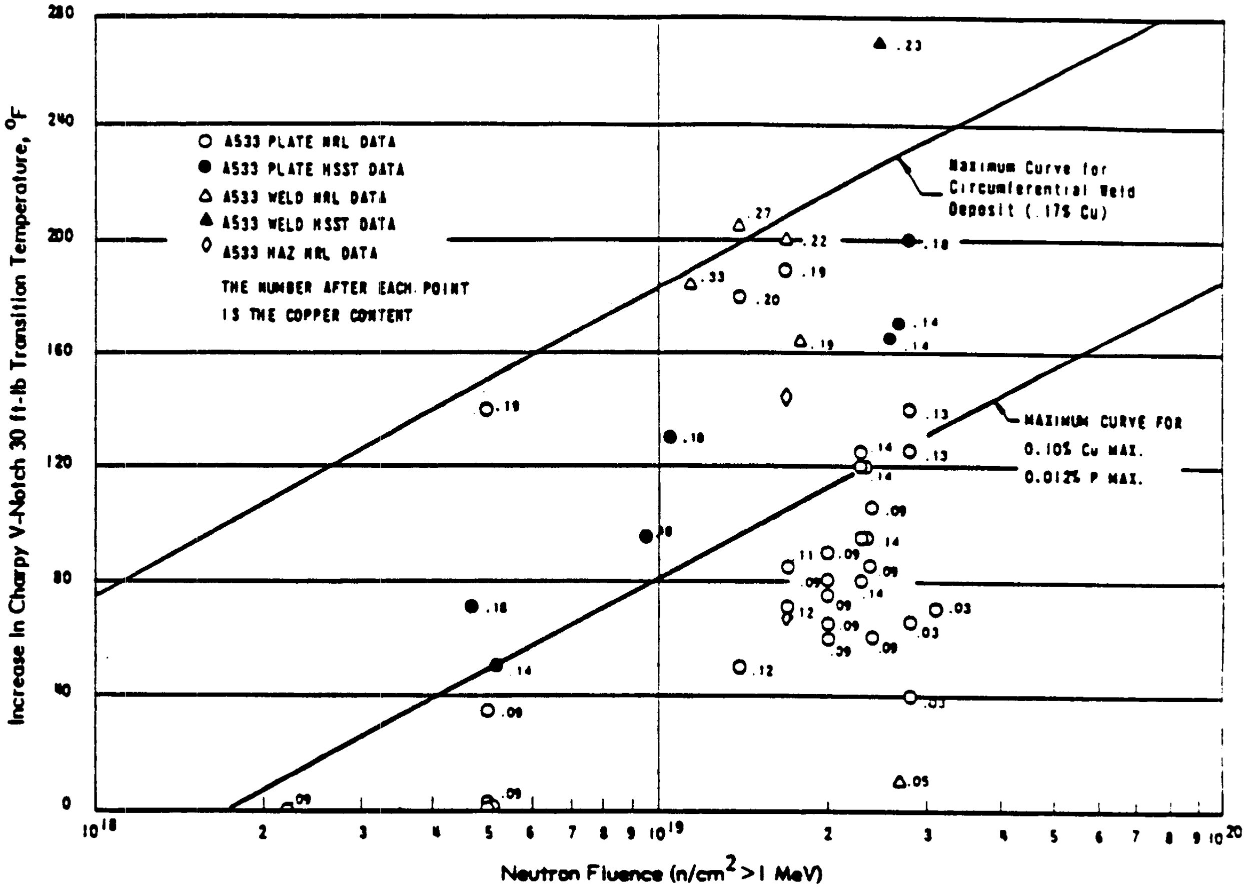
FIGURE 5.1-5 REVISION 9
JULY 1989



DAVIS-BESSE NUCLEAR POWER STATION
REACTOR COOLANT SYSTEM
ARRANGEMENT - ELEVATION

FIGURE 5.1-6

REVISION 9
JULY 1989



DAVIS-BESSE NUCLEAR POWER STATION
MAXIMUM PREDICTED TRANSITION
TEMPERATURE SHIFT VS NEUTRON FLUENCE
IRRADIATED AT 550F
FIGURE 5.1-7

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY (RCPB)

The integrity of the RCPB pressure vessels, including the RC pump casing, depends on four major factors: (1) design and stress analysis, (2) quality control, (3) proper operation, and (4) additional safety factors. The special care and detail used in implementing these factors in pressure vessel manufacture are briefly described in the following three subsections.

These pressure vessels are designed according to the requirements of the codes listed in Table 5.2-1. As a result, both primary and secondary stresses and the fatigue life of the vessels are completely evaluated. In establishing the fatigue life of these pressure vessels, the design cycles from Table 5.1-8 and the fatigue evaluation curves of the codes are used.

Since the applicable codes require a complete stress analysis, the designer must possess the necessary analytical tools, i.e., the solutions to the basic mathematical equations on the theory of elasticity. Babcock & Wilcox has confirmed the theory of plates and shells by measuring strains and rotations on the large flanges of actual pressure vessels; they were found to be in agreement with those predicted by theory. B&W has also conducted laboratory deflection studies of thick shell-and-ring combinations to define the accuracy of the theory, and is using computer programs developed from these test data.

The analytical procedure considers all steady-state, transient, and emergency (accident) conditions. A detailed design and an analysis of every part of each pressure vessel in the RC system are prepared as follows:

1. The vessel's size and configuration are set to meet the process requirements, the thickness requirements due to pressure and other structural dead and live loads, and the special fillet contour and transition taper requirements at nozzles, etc., as required by the applicable codes.

2. The vessel's pressure and temperature design transients as given in Table 5.1-8 are used in determining the pressure loading and the temperature gradient and their variations with time throughout the vessel. The pressure and temperature of the RC system are maintained within safe limits by the Reactor Protection System as described in Section 7.2. The resultant combinations of pressure loading and thermal stresses are calculated. Computer programs are used in this analysis.

3. The calculated stresses through the vessel's wall are compared with the allowable stresses of the applicable codes. These codes give safe stress level limits for the different types of stress applied: membrane stress (to ensure adequate tensile strength of the vessel), secondary stress (to ensure a vessel that will not progressively deform under cyclic loading), and peak stress (to ensure a vessel of maximum fatigue life).

   A design report is prepared and submitted to the authorized inspector and enforcement authorities. This report defines the design basis, loading conditions, etc., and summarizes the conclusions to permit independent checking by interested parties.

4. The reactor vessel was analyzed to demonstrate that it can safely accommodate the rapid temperature change associated with the postulated operation of the Emergency Core Cooling System (ECCS) at the end of the vessel's design life. Reference (29) evaluates the reactor vessel integrity during a small steam line

break, which creates a pressurized thermal shock condition. This transient generates the greatest level of stress in the reactor vessel. The analysis assumes that the water initially injected from the Borated Water Storage Tank (BWST) is at 35 °F. The results show that the integrity of the reactor vessel is not violated.

The reactor vessel's compliance with 10CFR50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" has been assessed in accordance with the methods specified in that regulation. The Pressure and Temperature Limits Report (PTLR) reports the required RT_{PTS} values for the reactor vessel beltline welds and forgings. The vessel meets the 10CFR50.61 screening for the most limiting weld and forging material at a fluence equal to 52 Effective Full Power Years of Operation (EFPY).

Note that stresses and fatigue for replacement reactor vessel closure head (added by ECP 10-0469) are analyzed considering an elevated temperature of 630 degrees F for the fluid in the closure head. Despite the increase temperature for the fluid in the closure head, the integrity of the replacement reactor vessel closure head will be maintained.

The following modes of failure were considered:

Ductile Yielding – the criterion for this mode of failure is that there shall be no gross yielding across the vessel wall using the minimum yield strength specified in the ASME Code, Section III. Note that the reactor vessel closure head (added by ECP-10-0469) is analyzed in accordance with the ASME B&PV Code, Section III, 1989 Edition with no Addenda. The analysis considered the maximum combined thermal and pressure stresses through the vessel wall thickness as a function of time during the safety injection. Comparison of the calculated stresses and the material yield stress indicates that local yielding may occur in the inner 8.0% of the vessel wall thickness.

Brittle Fracture – The margin of safety against brittle fracture is controlled as specified in Appendix G and H of 10CFR50 and Regulatory Guide 1.2, particularly with regard to specific guidelines for the treatment of heat-up and cool-down conditions and for analysis of the thermal shock transient. B&W is also a participant in the Joint NRC – Industry Review Committee established for the ORNL – NRC Thermal Shock Test Program.

Reference (29) documents the thermal shock analysis analyzed to ensure the reactor vessel's integrity will be maintained through out the life of the plant. The analysis was performed with a vessel fluence equal to 52 Effective Full Power Years (EFPY) of operation. The results demonstrate that the vessel will not fail. The vessel's compliance with 10CFR50.61 is reported in the Pressure and Temperature Limits Report (PTLR). The results show that the vessel is well below the applicable 10CFR50.61 screening criteria for the most limiting forging and weld material at 52 EFPY.

Fatigue – Particular attention is given to fatigue evaluation of the pressure vessels and to factors that affect fatigue life. The fatigue criteria of the ASME Code, Section III (1968) are the bases of designing for fatigue. Note that the reactor vessel closure head (added by ECP-10-0469) is analyzed in accordance with the ASME B&PV Code, Section III, 1989 Edition with No Addenda. They are based on fatigue tests of pressure vessels sponsored by the NRC and the Pressure Vessel Research Committee. The stress limits established for the pressure vessels are dependent on the temperature at which the stresses are applied.

NDTT – As a result of fast neutron absorption in the reactor vessel material opposite the core region, the ductility of the reactor vessel material will change. The effect is an increase in the NDTT. This NDTT shift is factored into the unit startup and shutdown procedures so that full operating pressure is not attained until the temperature of the reactor vessel is above the DTT. Below the DTT, the total stress in the vessel wall due to pressure and to the associated heatup and cooldown transient is restricted to limits for safe operation. These stress levels define an operating coolant pressure temperature path or envelope for a stated heatup or cooldown rate that must be followed during normal operation.

Additional Reactor Vessel Safety Factors – Additional methods and procedures used in designing the reactor vessel, which are considered conservative and provide an additional margin of safety, are as follows:

a. Use of the minimum specified yield strength of the material instead of actual values.

b. The design shift in NDTT as given in Subsection 5.2.3.8 is based on maximum predicted flux levels at the reactor vessel's inside wall surface, whereas the bulk of the reactor vessel material will experience a considerably lower exposure to radiation and consequently a lower change in NDTT over the life of the vessel.

c. Results from the method of neutron flux calculations, as described in Subsection 4.3.2.10, have been checked closely with experimental results and found to include an appropriate factor to account for azimuthal and uncertainty variation in the nvt at the reactor vessel wall.

   The foregoing descriptions of the design, fabrication, and operating procedures of the vessel present a basis for continued confidence in the integrity of the RC system components and support the conclusion that a rupture of the reactor vessel is not credible.

5.2.1 Design Criteria, Methods, and Procedures

5.2.1.1 Performance Objective

Steam Output – The RC system design information presented in this section is for an original design reactor power level of 2772 MWt with an additional 17 MWt input from the RC pumps, transferring a total of 2789 MWt to the steam generators. License Amendment No. 278 increased rated thermal power to 2817 MWt, resulting in a total of 2834 MWT transferred to the steam generators. The system is capable of producing a total steam flow of 11.76 million lb/hr at 2772 MWt and 12.13 million lb/hr (valves wide open) at 2817 MWt.

Transient Performance – The RC system follows step or ramp load changes under automatic control without safety valve or turbine bypass valve action as follows:

a. Step load changes – increasing load steps of 10% of full power in the range between 20 and 90% of full power and decreasing load steps of 10% of full power between 100 and 20% of full power are acceptable.

b. Ramp load changes – increasing load ramps of 5% per minute in the range between 15 and 90% of full power, or decreasing load ramps of 5% per minute

from 100 to 15% of full power are acceptable. From 90 to 100% of full power, increasing ramp load changes of 3% per minute are acceptable.

The combined actions of the control system and the turbine bypass system permit a 40 % load rejection without code safety valve action.

Partial Loop Operation – The RC system design permits operation with less than four RC pumps in operation. The steady-state operating power levels for combinations of RC pumps operating are as follows:

Reactor coolant pumps operating	Rated power, %
4	100
3	75
1 pump per loop	49

While operation on two reactor coolant pumps is provided for in the design of the reactor coolant system, power operation with only two reactor coolant pumps running is not allowed by Davis-Besse License Condition C.3.a.

The protective system that allows partial loop operation is described in Section 7.2.

5.2.1.2 Component Design Temperatures, Pressures, and Seismic Loads

Design and Test Pressures – The RC system components are designed structurally for an internal pressure of 2,500 psig, and the design and test pressures for the system components are shown in Tables 5.1-1a and 5.1-1b.

Design Temperature – The RC system components are designed for temperatures as shown in Tables 5.1-1a, 5.1-1b & 5.1-3 through 5.1-7.

Design Seismic Loads – The seismic loads used in the design of components in the RC system are contained in the design specification or report for each component.

5.2.1.3 10CFR50 Compliance

Table 5.2-1 shows compliance with the rules of 10CFR50, Section 50.55a, “Codes and Standards.”

5.2.1.4 Code Case Interpretations

Table 5.2-2 delineates ASME and ANSI code case interpretations applied to components of the components of the Reactor Coolant Pressure Boundary (RCPB).

5.2.1.5 Design Transients

Transients used in the design and fatigue analysis of the components of the RCPB are listed in Table 5.1-8. The transients are categorized relative to normal, upset, emergency, and faulted conditions as defined in the ASME Section III Code. The associated stress limits and loading conditions are given in Tables 5.2-3 and 5.2-4. The loading combinations are as follows:

Case I – Design Load Plus Maximum Probable (Smaller) Earthquake – This is a design condition and should meet the stress limits in Tables 5.2-3 and 5.2-4 (see normal/upset). The reactor must be capable of continued operation for this loading combination.

Case IA – Normal and Upset Loads Plus Maximum Probable (Smaller) Earthquake – This is a normal/upset condition and should allow continued operation of the reactor. The stress limits appear in Tables 5.2-3 and 5.2-4. It should be noted that along with the transients classified normal and upset in Table 5.1-8, a design mechanical load is used (i.e., dead weight).

Case IB – Emergency Loads – Transients classified Emergency in Table 5.1-8 have been considered. This is an emergency condition according to the ASME Code Section III. The stress limits are given in Tables 5.2-3 and 5.2-4.

Case II – Design Loads Plus Maximum Possible (Larger) Earthquake Loads – In establishing stress levels for this case, a “no-loss-of-function” criterion applies, and stress values higher than given in Case I can be allowed. The condition is defined as a faulted condition according to paragraph N-412 (t) (4) of Section III. The stress limits are those specified for items a and b of Case III.

Loss-of-Coolant Loads – A loss-of-coolant accident (LOCA) coincident with a seismic occurrence is analyzed to ensure that reactor shutdown and emergency core cooling can be initiated and maintained. It should be noted that per the criteria in USAR Section 3.6.2, dynamic effects from a postulated pipe rupture in the RCS can be excluded. The following loading case is considered:

Case III – Design Loads Plus Maximum Possible (larger) Earthquake Loads Plus Pipe Rupture Loads (Including Jet Reactions) – The design stress limits are based on one of the following:

a. Two-thirds of the ultimate strength of the material. The rationale for this design allowable stress of Case III is given in Topical Report BAW-10008, Part 1, Rev. 1, for type 304 stainless steel. This limit is used for all reactor internals, including bolts.

b. Stress limits in accordance with paragraph N-417.11 of Section III for faulted conditions (with the exception of 90% of the plastic instability loads).

c. A limit-type analysis to show that the structure maintains its integrity to allow the reactor to be shutdown and heat to be removed by the emergency core cooling system for an indefinitely long period.

d. Steam Generator tubes and tube to tubesheet joints are evaluated based on ASME Section III Appendix F.

5.2.1.6 Pump and Valve Classification

Valves and components within the RC pressure boundary which are classified as active or inactive are listed in Table 5.1-1b. The maximum allowable stress limits are in accordance with Class 1 ASME Code. Stresses in the valve shall not exceed the allowable working stress limits as set forth by the ASME Code Section III, Class 1. Stresses in the structural portion may be increased to 120% of Code allowables, but only if the primary stresses also remain below the yield stress. The allowable design criteria seat leakage for active valves is 10 cc per hour per inch of nominal diameter.

All Velan check valves on DB-1 were identified under specification 7749-M-212Q in response to IE Bulletin No. 79-04 dated March 30, 1979, regarding weights used in seismic category I piping stress analyses for Velan 3”, 4” and 6” swing check valves. The detailed results are attached to the Serial No. 1-63 letter dated May 2, 1979.

5.2.1.7 Pump and Valve Stress Criteria

Valves within the RCPB are not required to have detailed stress analysis performed for emergency and faulted conditions. Valves were analyzed for 3g horizontal and 3g vertical seismic loads combined with plant upset condition with maximum stress not to exceed 120% of ASME Pump and Valve Code allowables.

The stress criteria associated with the emergency and faulted operation condition categories for pumps are specified in Table 5.2-4.

5.2.1.8 Pipe Rupture Criteria

Pipe rupture criteria are discussed in Subsection 3.6.2.

5.2.1.9 Use of Plastic Instability Analysis

The plastic instability and limit analysis methods of ASME Section III have not been used in conjunction with elastic system dynamic analysis.

5.2.1.10 Environmental Protection for RC System

The principal components of the RC system are protected against environmental factors to which the system may be subjected.

5.2.1.10.1 Fire Protection

The extent to which the station is protected against fires is discussed in Subsection 9.5.1.

5.2.1.10.2 Flooding Protection

The extent to which the entire station site is protected against flooding is discussed in Section 3.4.

5.2.1.10.3 Missile Protection

Missile protection criteria are discussed in Section 3.5.

5.2.1.10.4 Seismic Effects

The seismic design of the RC system and its major components is discussed in Sections 3.2, 3.7, and 3.9.

5.2.1.11 Method of Stress Analysis

Each component is analyzed using a number of analytical and mathematical models to evaluate the pressure, thermal, seismic, and LOCA loading stresses at each of the areas where a nozzle

or discontinuity exists. However, the basic model and method of analysis is a discontinuity or interaction analysis with a model consisting of series of free-body elements for which influence coefficients and free-body thermal and pressure motions are calculated using classical shell theory equations based on “Beams on Elastic Foundation” by Hetenyi, “Theory of Plates and Shells” by Timoshenko, etc. Once these quantities are known, two compatibility equations (rotation and displacement) are written at each juncture of the discontinuity model.

The compatibility equations are then combined into a total matrix which is solved for the redundant shears and moments at each juncture. These redundant forces are those that are necessary to ensure continuity of the geometry. Using these forces, principal stresses and, ultimately, stress intensities are calculated which are compared to the Code allowable. This comparison ensures that the component design meets the Code criteria and is structurally adequate to perform its intended function.

Summary tabulations of the calculated primary plus secondary stress intensities for the pressurizer, steam generator, RC system piping, and reactor vessel are shown in Tables 5.2-5, 5.2-6, 5.2-7 and 5.2-8, respectively. A detailed stress report for each component was available before station operation. Each design report contains the analyses that evaluate the applicable pressure, thermal, seismic, and LOCA load stresses in accordance with the applicable specifications and codes.

Revised stress analyses have been performed for the Pressurizer Surge Line (including nozzles) to account for additional stress resulting from thermal stratification and striping in the surge line. The revised analyses were performed in accordance with the requirements of ASME Section III, 1986 Edition. Topical Report BAW 2127 (Reference 14), Final Submittal for Nuclear Regulatory Commission Bulletin 88-11 ‘Pressurizer Surge Line Thermal Stratification’ with Supplements 2 (Reference 15) and 3 (Reference 16) describe the analysis method and demonstrate compliance with the Code and regulatory requirements. The NRC has reviewed and approved the methodology used to perform this revised stress analysis (Reference 17).

Revised Stress analysis have been performed for RCS nozzle locations to provide for mitigation of Alloy 600 component items and dissimilar metal welds.

Revised stress analyses have been performed for the replacement once-through steam generators (including MFW and AFW components shipped with the replacement OTSG) in accordance with the requirements of ASME Section III, Class 1, 2001 Edition with 2003 Addenda. The analyses were conducted using both finite element methods and classical techniques. The results of the analysis are documented in the stress analysis reports for the replacement once-through steam generators.

5.2.1.12 Faulted Condition Stress Criteria

The design stress criteria for faulted condition loadings are given in Tables 5.2-3 and 5.2-4.

5.2.1.13 Class I Systems and Stress Levels

Design pressure, dead load, and earthquake stress at changes in flexibilities in the primary piping system (elbows and nozzles) are tabulated in Table 5.2-9. The joint numbers are shown in Figures 5.2-1, 5.2-2, and 5.2-3.

5.2.1.14 Pump Valve Stress Evaluation

The stress criteria for valves within the RCPB are in accordance with the ASME Pump and Valve Code. Analytical methods and the associated acceptance limits are in accordance with Class A requirements of ASME Section III.

Owing to the irregular shape of the pump casings, their analysis falls outside the scope of ASME Section III. They were analyzed using a finite element computer code for specific load cases with limits as given in Table 5.2-4.

5.2.1.15 Pump Critical Speed Criteria

Critical speed calculations have been performed which confirm that the critical speed of the RC pumps is in excess of 20 % above the design speed of the pumps. The bearing loading was determined by model testing, and the integrity of the bearing was determined by analysis at the worst loading condition that the bearing encounters.

5.2.1.16 Valve Qualification Test Program

All active valves whose operability is relied on to perform a safety function were given a shop hydrostatic pressure test and a seat leakage test, and were cycled for proper operation. After the valves have been installed in the system they are given functional tests to ensure that they are operable.

5.2.2 Overpressure Protection

5.2.2.1 Pressure Relieving Device Locations

Pressure-relieving devices for the reactor coolant system are located as shown in Figure 5.1-2. Pressure-relieving devices for the primary side of auxiliary or emergency systems interconnecting with the primary system are located as shown on the following figures:

Core Flooding System: Figure 6.3-1A

High Pressure Injection System: Figure 6.3-2

Decay Heat Removal/Low Pressure Injection System: Figure 6.3-2A

Makeup and Purification System: Figure 9.3-16

5.2.2.2 Pressure Relieving Device Design Criteria

The design and installation criteria for the mounting of pressure-relieving devices (safety valves and relief valves) within the RCPB are as follows:

a. Individual nozzles are provided for each safety valve on the pressurizer. The pressurizer safety valve nozzle is analyzed using a Bijlaard type of analysis. (Reference: Welding Research Bulletin 107, August 1965.) The allowable stress limits are given in Table 5.2-4. Maximum allowable loads for the safety valve nozzle are determined to be within design limits.

b. Manifolding of piping from the pressurizer to the safety valves is not permitted.

c. The safety valve is analyzed to determine the maximum allowable loads it can accept without causing seat leakage or without exceeding allowable stresses. The allowable loads are determined by calculating the deflection of the valve body and internals. These deflections are compared to the clearances predicted to exist without the applied loads and the calculated deflections must be less than the predicted clearances.

5.2.2.3 Overpressure Protection

B&W topical report BAW-10043 discusses overpressure protection for the RC system. Areva Document 51-9004090-005 “Measurement Uncertainty Recapture Power Uprate Summary Report” states that a turbine trip analysis was performed using NRC approved methods (Areva NP Document No. 43-10193PA-00). This analysis modeled an initial core power level of 3025 MWt (109.1% of 2772 MWt) and used the current installed main steam safety valve capacity. This analysis confirmed that the peak OTSG pressure was less than the ASME code allowable. Consequently, the small increase in pressure due to the MUR power uprate is bounded. The pressurizer is equipped with two spring-loaded code safety valves and one electromatic relief valve to relieve high pressure conditions in the reactor coolant system. Backpressure is considered in the selection of the safety valve. The valve is selected so that choked flow exists in all valves during all operating transients. The characteristics of these valves are described below:

Pressurizer Code Safety Valves

Identifying numbers	PSV-RC13-1, PSV-RC13-2
Size	4x6 inch
Design flow capacity each	336000 lbs/hr
Set pressure (nominal)	2500 psig
Set pressure tolerance	±1%

Pressurizer Pilot Operated Relief Valves

Number of valves	1
Size	2 1/2 x 4 inch
Set pressure	2450 psig
Set pressure tolerance	±4 psig

The secondary system is equipped with 18 spring-loaded code safety valves. The inlet side of all valves are 6 inch nominal diameter.

The relief capacity of two of the nine Main Steam Safety Valves per steam generator is 583,574 lbs/hr each (at 1155 psig, i.e. 110% of the 1050 psig system design pressure), and the capacity of the remaining seven valves is 845,759 lbs/hr each. The design pressure for the main steam piping is 1050 psig, however the design pressure for the replacement OTSG is 1150 psig. At least two of the safety valves per generator must be set to relieve at 1050 psig, while the other seven may be set to relieve at pressures up to 1100 psig. However, Technical Specifications allow main steam safety valves to be removed from service during power operations. Removal from service of the 1050 psig setpoint main steam safety valves may affect the ARTS arming setpoint. Refer to Section 7.4.1.4.1.2. Re-ratioing the Steam Generator ratio controller to operate with a planned ΔTc and unbalanced main feedwater flowrates is not allowed when one or more main steam safety valves is inoperable.

The total relieving capacity (at 1155 psig, i.e., 110% of the 1050 psig design pressure) of all safety valves on both main steam lines is 14,175,000 lbs/hr which is 120 % of the total secondary system flow of 11,760,000 lbs/hr at 2772 MWt. A maximum safety valve setpoint pressure of 1100 psig assures main steam system pressure remains below 1155 psig as required by the ASME Boiler and Pressure Vessel Code, 1971 Edition. All Main Steam Safety Valves shall have a ± 1% as-left setpoint tolerance in accordance with Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition. The acceptance criterion for Technical Specification Surveillance Requirement testing is determined in accordance with the Inservice Testing Program.

The plant analyzed in BAW-10043 is a 2789-MWt plant, similar size as Davis-Besse. The steam relief capacity is sized based on the thermal power rating of the plant and not on the number of steam safety valves. The number of valves quoted in BAW-10043 is 22. The 18 valves on Davis-Besse are sized to provide the steam relief capacity documented in BAW-10043 (2789 MWT). Per Areva Document 51-9004090-005, Safety Analysis confirmed the installed capacities and lift setpoints of the RCS and Main Steam relief valves to be valid for the MUR Conditions.

The Framatome Overpressure Protection Report, Reference 39, considers four different upset conditions; Control rod withdrawal, Turbine trip, Complete loss of power and Loss of feedwater flow. Reference 40 evaluates the impact of the replacement OTSG on these upset conditions. Because the replacement OTSG heat transfer characteristics and other parameters affecting the analysis are essentially the same, it is concluded that no changes to the overpressure protection are needed as a result of the replacement OTSG final design.

The design and operation of Davis-Besse Unit 1 are such that until the end of licensed operation for Davis-Besse Unit 1, no single equipment failure or single operator error will result in 10CFR50 Appendix G limitations being exceeded and that no common equipment failure would both cause a pressure transient and render the mitigating equipment inoperable. The pressure-temperature limit curves provide ample margin to allow the reactor coolant system to be lined up to the Decay Heat Removal System and its attendant pressure relief capacity providing ample protection to the Reactor Coolant System.

Where operator action is required to assure that Appendix G limitations are not exceeded, this action is attained through Administrative Controls and/or system alarm functions to alert the operator that such action is required through established procedural instructions. The only case in which operator error could result in an overpressurization event is the inadvertent dumping of a Core Flooding Tank. As discussed below, the operator would have to make two errors for this to occur. All other postulated events are mitigated by design. Two operator errors or two single failures would be necessary to exceed the pressure-temperature limits of Appendix G.

During shutdown, before decay heat removal system operation can be initiated, suction valves DH11 and DH12 must be opened. This can be done when system pressure decreases below the Decay Heat isolation valve interlock setpoints described in Section 7.6.1.1.2. This prevents opening the valves while the RCS pressure is above the design pressure rating of the decay heat removal system piping. Power is restored to the motor operators, the valves are opened, and then control power is removed from the motor operators. Position control of these valves, as well as the capability to remove control power from the motor operators, is available in the control room. Also, an alarm is provided in the control room any time DH11 or DH12 is open and power has not been removed from the operators. The instrumentation for the alarm is safety grade. Valve position indication is available from the control room whether or not power

is provided to the motor operators. Removing control power from the motor operators once the valves have been opened assures that the system function is not affected by inadvertent valve closure, and it also assures a relief path through relief valve PSV 4849, located in the suction line to the decay heat removal pumps, should a pressure transient occur.

Plant cooldown and depressurization will continue with DH11 and DH12 open and incapable of inadvertent closure. The pressurizer steam bubble will be replaced with a nitrogen bubble when the Reactor Coolant System pressure is decreased to approximately 30 psig. Water solid conditions are precluded at all times by, procedural controls, except for hydrostatic testing.

During plant heatup and repressurization, a steam bubble is formed in the pressurizer and the nitrogen vented when the RCS pressure is greater than 50 psig. When RCS temperature is greater than 280 °F, control power is restored to the motor operators of DH11 and DH12 and the valves are closed. Once the valves are closed, power is removed to prevent inadvertent opening. To ensure both valves are closed before system repressurization, interlocks are provided that trip off pressurizer heaters when pressure increases above setpoint and one of the valves is not closed. If both valves are not closed, pressurization by use of the heaters can not occur above the PSV 4849 setpoint of 320 psig. Also, if power is provided to the motor operator of DH11 and DH12, an automatic closure signal is sent to the valves if RCS pressure increases above the reset points of the valve interlocks, as described in Section 7.6.1.1.2. Valve closure will occur if power is available to the motor operators. The modifications and verifications of the interlocks between the pressurizer heaters and the DHR valves DH11 and DH12 and the changes in Technical Specifications are present in FCR 77-391 and its supplements.

The normal decay heat removal valves are seismically qualified and designed to Quality Group A. The control system is designed to withstand physical damage or loss of function caused by earthquakes and missiles; the control system for each valve receives control power from a separate essential supply. As discussed in USAR Subsection 7.6.2.1, the control system is designed to meet the intent of IEEE 279-1971, Sections 4.11 through 4.15 not being applicable to this control system. The relief valve (PSV 4849) is seismically qualified and designed to Quality Group B. Its functioning is not dependent on electric power or on air supply. It is dependent on system pressure alone and is, therefore, a passive component.

For a discussion of the overpressurization protection provided for the decay heat removal system, see Section 9.3.5.5.1.

In the event that Decay heat Removal pressure relief valve PSV 4849 becomes inoperable when the RCS is below 280 °F, two actions are employed to protect against overpressurization from inadvertent HPI system actuation or RCS makeup valve failure (full open) with a makeup pump operating. They are:

1. Operating procedures and technical specifications require that the HPI system be disabled when RCS temperature is below 280 °F.

2. Technical specifications limit the water level in the pressurizer for a given RCS pressure to a pressure where the RCS pressure-temperature limit will not be exceeded if a makeup pump delivers the entire contents of the makeup water tank into the RCS.

The dumping of a Core Flooding Tank was not considered because either (1) power is removed from the Core Flooding Tank isolation valves once it is closed upon plant cooldown and depressurization or (2) the tank is depressurized. Procedures define the specific action required

in either case. Other Postulated occurrences (makeup control valve failing open, loss of DHR System cooling, all pressurizer heaters energizing) do not produce a pressure excursion as severe as that produced by the two HPI pumps. Although the pressurizer, by procedure, cannot be solid, for the purpose of analysis it was considered to go solid during the transient.

As noted above, in order to ensure that the Core Flooding Tanks do not dump into the Reactor Coolant System, one option available to the operator is the removal of power from the Core Flooding Tank isolation valves once they are closed. To this end, the unit includes the following features, as discussed in USAR Subsection 6.3.2.15:

Position switches on each Core Flooding Tank valve actuate open and close valve position indicators for each valve. The indicators are located in the control room.

Two separate alarms, one for each valve, are actuated if a valve is open and reactor coolant pressure is reduced to a value that could cause emptying of the Core Flooding Tanks; these alarms alert the operator to an impending situation where he could inadvertently discharge the Core Flooding Tanks during station shutdown.

The isolation valves are closed prior to depressurizing the reactor coolant system below 675 psig.

Power is removed from the valves after depressurizing the Reactor Coolant System prior to initiating decay heat removal. With power removed, the possibility of the valves opening and causing either the pressure-temperature limits of the RC system or the design pressure limits of the DHR System to be exceeded, is precluded.

Assuming that the unit is undergoing cooldown from the hot shutdown condition, the following events will take place:

As RC pressure decreases below 675 psig, the alarms are actuated in the control room if the operator has not closed the valves prior to this pressure. The operator would then close the valves to deactivate the alarms. Failure to close the valves would require a double operator error. First, the operator must fail to follow the procedure which specifically instructs him to close the valve. Second, the alarms resulting from the open valve at a pressure below 675 psig would have to be ignored by the operator. After closing these valves, power is removed from the valves.

When power is removed from the valves, in the cases described above, the breaker of the combination starter of each isolation valve is manually tripped open and padlocked. The tripped position of the breakers is monitored by essential indication on the main control board by one blue indicating light for each breaker. For a discussion of the analysis showing the over-pressurization protection provided in the event that both HPI pumps are inadvertently started, see Section 9.3.5.5.1.

In order for the operator to monitor the reactor coolant system temperature and pressure, there are a variety of control room readouts available, as indicated in USAR Table 7.5-1. Among those listed are the following:

Measured Parameter	Type of Readout	Range
RC loop pressure (2 sensors in each loop)	Linear scale indicator Recorder Station computer output Audio-visual alarm indication	0-2500 psig
(1 sensor in loop 2)	Linear scale indicator	0-500 psig
RC loop inlet temperature (2 sensors in each loop)	Linear scale indicator Digital Indicator Station computer output	50-650 °F

This instrumentation (above) is used in service during long periods of cold shutdown as well as during startup and shutdown operations. The Technical Specifications require that the reactor coolant system (with the exception of the pressurizer) temperature and pressure be determined to be within limits at least once per 30 minutes during heatup and cooldown operations.

The present design and operation provides the overpressurization protection necessary to assure that no single equipment failure or single operator error results in Appendix G limitations being exceeded, and that no common equipment failure would both cause a pressure transient and render the mitigating equipment inoperable.

5.2.3 Material Considerations

5.2.3.1 Material Selection

All materials used in the RC System have been selected to meet the expected environmental and service conditions. Materials for the RCPB components are equivalent to or better than those listed in Table 5.2-10. The chemical composition for the reactor vessel beltline region is given in Subsection 5.2.3.8 and Table 5.2-15.

5.2.3.2 Materials Exposed to Reactor Coolant

All materials exposed to the reactor coolant exhibit corrosion resistance for the expected service conditions. The materials used, as given in Table 5.2-10 are 304SS, 316SS, Inconel (Ni-Cr-FE), or weld deposits with corrosion resistance equivalent to or better than the other materials listed. These materials were chosen because they are compatible with the reactor coolant. The RCPB contains no furnace-sensitized, wrought austenitic stainless steel. Sensitized stainless steel weld overlay (cladding) is permitted.

Weld Overlays (both Optimized and Full Structural), Half Nozzle Repairs, and Nozzle Replacements have been implemented at RCS Nozzle locations utilizing Alloy 690/52(M) material to provide for mitigation of Alloy 600 component items and dissimilar metal welds. These materials were chosen based on their resistance to primary water stress corrosion cracking (PWSCC).

Pressurizer Half Nozzle Repairs and Full Nozzle Replacement (Thermowell) designs results in a portion of the carbon steel pressurizer nozzle bore to be exposed to reactor coolant and have been evaluated for the expected service conditions with acceptable results (Reference 23 and 24).

The RC System is a controlled addition, closed loop which is not conducive to contaminant introduction. In addition, purification systems are provided to maintain the contaminant levels within the limits specified in Table 9.3-4 and Table 9.3-5. The materials in the RC System are not adversely affected by expected contaminants or radiolytic products.

5.2.3.3 Material Compatibility With Insulation and Environment

The RCPB insulation is metal-reflective insulation fabricated from stainless steel. Limited sections of “Nukon” piping blankets are installed on the reactor vessel head continuous vent line. Both of the insulations are compatible with the material of construction and the environmental atmosphere resulting from reactor coolant leakage.

5.2.3.4 Reactor Coolant Additives

The water chemistry specifications for the reactor coolant and steam generator feedwater as given in Tables 9.3-4 and 9.3-5 respectively provide an environment that is compatible with the reactor coolant materials (Table 5.2-10) and the core materials (Zirconium alloys and Inconel). The pH of the coolant is controlled by the addition of lithium (^{7}Li) hydroxide to minimize corrosion of the system surfaces in contact with the coolant solution. In turn, coolant activity and radiation levels of the components are minimized. Zinc may be added to the coolant to reduce the radiation source term and for mitigation of Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 600 components. Application of zinc is effective in reducing metal release from RCS materials and in reducing cobalt pickup on these materials due to zinc’s ability to displace cobalt from the resident RCS oxide films. Hydrogen is added to the coolant during critical operation to combine chemically with the oxygen produced by radiolysis of the water. Dissolved hydrogen concentration in the RC is maintained from 25-50 cc/Kg of H_2O. (This corresponds to a hydrogen partial pressure in the Makeup Tank of 7.3 to 29.3 psig). During noncritical, less than 200 °F, operation, hydrazine may be added to react chemically with any oxygen that may be present. The additions of hydrogen and hydrazine minimize the effect of oxygen on the corrosion of the reactor coolant system surfaces at the expected service conditions. Hydrogen peroxide may be added to the Reactor Coolant System (RCS) to induce forced oxidation when RCS cold leg temperature is less than 180°F with at least one reactor coolant pump in operation; or when the RCS is on decay heat cooling with no reactor coolant pumps in operation and the decay heat cooler outlet temperature is ≤140°F, thus allowing the use of purification demineralizers to clean up ionic species liberated in the Reactor Coolant System.

5.2.3.5 Fracture Toughness Criteria

The RCPB ferritic materials were ordered and tested in accordance with the requirements of the 1968 edition of Section III of the ASME Code, including all addenda through Summer 1968. The 1968 edition does not require the determination of the Reference Temperature RT_{NDT}, as defined by the Drop Weight nil-ductility transition temperature NDT and by the Charpy V-notch 50 ft-lbs/35 mils of lateral expansion. The 1968 edition does not require a Charpy V-notch upper shelf energy for specimens oriented in the weak direction equal to or higher than 75 ft-lbs for the reactor vessel beltline materials. Also, the 1968 edition does not require that materials

for bolting and other fasteners with nominal diameters exceeding 1 inch shall meet the minimum requirement of 25 mils lateral expansion and 45 ft-lbs in terms of Charpy V-notch tests.

Materials for the replacement reactor vessel closure head added by ECP-10-0469 were ordered and tested per requirements of ASME B&PV Code 1989 Edition No Addenda, Section III.

All the RCPB ferritic base materials meet the Charpy V-notch energy value requirements listed in Section III at a temperature of + 40 °F or lower. For weld deposits, the transition temperature was obtained by performing Charpy V-notch impact tests during procedure qualification on weld deposits using the same flux and filler wire combinations as used for the production welds. All weld deposits meet Charpy V-notch energy values required by Section III at a temperature of +40°F or lower.

The preceding paragraphs in Section 5.2.3.5 do not apply to the replacement Hot Leg piping spool pieces. Materials for these items were ordered and tested in accordance with ASME B&PV Code, Section III Class 1, 2001 Edition 2003 Addenda and 10 CFR 50 Appendix G, “Fracture Toughness Requirements”.

5.2.3.6 Materials Toughness Properties

The available beltline region materials have been impact tested in accordance with Appendix G to 10CFR50. The materials identified to be part of the beltline region are from top to bottom, the upper circumferential weld, the upper shell forging, the middle circumferential weld, and the lower shell forging. The results of these tests are presented in Table 5.2-15.

The materials are described by heat number or weld qualification number, type (e.g., SA 508 Class 2), and reactor vessel location. The unirradiated impact properties include the drop weight temperature (T_{NDT}), the temperatures at which the lower bound Charpy test curves exhibit 50 ft-lbs and 35 mils of lateral expansion, the Charpy V-Notch Upper Shelf energy (Cv-USE), and the reference temperature (RT_{NDT}). Because of the unavailability of material, these properties were conservatively estimated for the weld metal used in the upper and lower circumferential welds. These values are also included in Table 5.2-15. The table also lists the copper, phosphorus, and sulfur content and the end-of-service predicted neutron fluence (E>1 Mev, n/cm^2) at the 1/4 T and 3/4 T vessel wall locations for each material. These fluence values are used to predict the end-of-service impact properties.

As discussed in subsection 5.2.3.5 the ferritic materials of the reactor coolant pressure boundary were ordered, tested and met the requirements of the 1968 Edition and 1968 Summer Addenda of the ASME Code. Other than the beltline region materials, these materials are not available for testing, and the RT_{NDT} for each class of material must be estimated. Note that the reactor vessel closure head (added by ECP-10-0469) is analyzed in accordance with ASME B&PV Code, Section III, 1989 Edition with No Addenda. Note that the RT_{NDT} of the SA-508 Class 3 ferritic material for this reactor vessel closure head is -50 degrees F.

Based on an evaluation of data from tests conducted on ferritic materials for which the new fracture toughness requirements were applied, the initial RT temperature is conservatively estimated as follows:

Material	Estimated RT Temperature NDT
SA 508 class 2	+60 °F or Drop Weight NDT
SA 533 grade B class 1	+40 °F
Carbon steel plates	+10 °F
Piping	+40 °F
Weld metal	+20 °F
HAZ of SA 508 class 2	+30 °F
HAZ of SA 533 grade B class 1	+10 °F
HAZ of carbon steel plates	-20 °F

Since the impact properties of the beltline region materials of a reactor vessel changes throughout its lifetime, periodic adjustments are required on the pressure-temperature limitations. The magnitude of the adjustment, in terms of temperature, is proportional to the shift of the RT_{NDT} temperature caused by the neutron fluence. So it is important to know either by predicting or by the results of the material surveillance program, the radiation induced ΔRT_{NDT} of the beltline region materials. The original estimates of ΔRT_{NDT} used B&W's design curves for predicting the radiation-induced ΔRT_{NDT} presented in B&W Topical report BAW-10100A "Compliance with 10CFR50, Appendix H, for Oconee Class Reactors." Changes in RT_{NDT} are now determined using the methods and values described in the Pressure and Temperature Limits Report.

It is also necessary to know, by prediction or from the results of the material surveillance program, the drop of the Charpy upper shelf energy (Cv-USE) level caused by irradiation. The drop in Cv-USE is important because the RT_{NDT} is based on the temperature shift of the Charpy test curves measured at the 50 ft-lbs level or at the 35 mils of lateral expansion level, whichever temperature shift is greater. If the Cv-USE level drops below the 50 ft-lbs level or the 35 mils of lateral expansion level, the adjusted (or irradiated) RT_{NDT} is not defined. Originally, the effects of radiation on the Charpy Upper Shelf Energy level of the beltline region materials were estimated using the curves in BAW-10100A. The radiation-induced drop in Cv-use is given in terms of the % of upper shelf energy reduction, and is plotted as a function of copper content and neutron fluence. These curves predict the maximum drop in upper shelf energy with 97.5 % confidence. Equivalent reductions in upper shelf energy levels are observed for weld and base metal when the base metal has approximately 0.05 % higher copper content than the weld metal. The derivatin of the design curves and additional explanation of their use is included in BAW-10100A.

The current upper shelf energy evaluation for the limiting reactor vessel weld is determined in BAW-2192PA (Ref.31). The NRC accepted the results of that evaluation in License Amendment 278.

The preceding paragraphs in Section 5.2.3.6 do not apply to the replacement Hot Leg Upper spool pieces. Materials for these items were ordered and tested (including RT_{NDT} determination) in accordance with ASME B&PV Code, Section III Class 1, 2001 Edition, 2003 Addenda and 10 CFR 50 Appendix G, "Fracture Toughness Requirements".

5.2.3.7 Analysis for Heatup and Coldown Pressure Temperature Limits

The heatup and cooldown pressure-temperature limits of the Davis-Besse plant were established using the methods described in B&W Topical Report BAW-10046A. Rev 2; ASME Code Section XI, Appendix G, 1995 Edition with Addenda through 1996 as modified by ASME Code Cases N 588 and N 640; and BAW-2308, Revisions 1 A and 2 A. The methods described in these documents allowed the determination of the P T limits in accordance with Appendix G to 10CFR50. These limits were determined for the following loading conditions:

a. Preservice system hydrostatic tests

b. Inservice leak and hydrostatic tests

c. Normal operation, including heatup and cooldown

d. Reactor core operation

Requirements for the Reactor Coolant System pressure-temperature limits are included in the Technical Specifications and the Pressure and Temperature Limits Report.

5.2.3.8 Transition Temperature

Originally, the prediction of the increase in transition temperature of the beltline region materials due to irradiation-induced embrittlement was based on the copper and phosphorus contents and the calculated maximum neutron fluence levels. As can be seen in Table 5.2-15, the center circumferential weld, (WF-182 1) has the highest copper and phosphorus contents of the beltline region materials exposed to the maximum fluence ($1.68 \times 10^{19} n/cm^2$,E>1 Mev). Based on the design curves presented in BAW-10100A, the end-of-service adjusted RT_{NDT} of this material is estimated to be about 260 °F.

Further research has determined that copper and nickel affect the shift in RT_{NDT} as described in NRC Regulatory Guide 1.99, Revision 2. The Pressure and Temperature Limits Report provides the current end-of-life fluence data, discusses the current methodology to compute the adjusted RT_{NDT} (ART) and provides ART values.

B&W Report, BAW-2325 Revision 1, discusses Davis-Besse's compliance with 10 CFR 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events. Through the prescribed methods of 10 CFR 50.61, it has been determined that the reactor vessel will not exceed the established reference temperature for pressurized thermal shock (RT_{PTS}) screening criteria prior to the expiration date of the Davis-Besse Operating License.

5.2.3.9 Non-Stabilized Stainless Steels

AISI Type 304 and 316 stainless steels are the non-stabilized grades of austenitic stainless steels with a carbon content greater than 0.03%, are used for components of the Reactor Coolant System. All grades of austenitic stainless steels (AISI Type 3xx series) were required to be furnished in the fully solution-heat-treated condition before fabrication or assembly into components or systems. The solution heat treatment varied according to the applicable material specifications. Any inspection to verify susceptibility to intergranular attack was made only when inspection was a part of the applicable material specification. Austenitic stainless steels that were exposed to hot forming were subjected to the same requirements as the original furnished material after completion of hot forming. Such tests as A-262 practice E were

not considered meaningful in relation to actual PWR operating conditions. The unstabilized austenitic stainless steels with carbon content greater than 0.03% that are used in Davis-Besse are exposed to reactor coolant that meets the water chemistry requirements of Tables 9.3-4 and 9.3-5.

The use of furnace-sensitized wrought stainless steel in components of the RCPB has been eliminated. Whenever stainless steel components are welded to ferritic materials, Inconel buttering of the ferritic material, followed by a stress-relief heat treatment, has preceded joining of the two components with Inconel weld metal. Unstabilized austenitic stainless steel materials that are exposed to temperatures in the heat range of 800 to 1500 °F, as a result of heat treatment or hot forming operations, are retested for susceptibility to intergranular attack according to the same test criteria as required for the original furnished material.

All stainless steel materials that are used in systems for reactor shutdown and emergency core cooling (piping, castings, etc.) are supplied in the solution annealed condition and that rapid cooling is used from the solution annealing temperature. Forming operations in the 800 to 1500°F range have been prohibited. Preheat and interpass temperatures for the welding of stainless steels have been limited to 350°F maximum.

As an example, four welding processes are used to weld stainless steel internals: (1) automatic submerged arc (ASA), (2) gas metal arc (GMA), (3) gas tungsten arc (GTA), and (4) shielded metal arc (SMA). Nitrogen is used in place of argon or helium gas as a purge gas in the welding process.

a. Automatic Submerged Arc (ASA) – A welding process in which the heat for welding is supplied by an arc developed between the bare-metal, consumable electrode and the workpiece. The arc is shielded by a layer of granular and fusible flux, which blankets the molten weld metal and the base metal near the joint and protects the molten weld metal from atmospheric contamination. The ASA welding parameters are set, so that the maximum heat input is 90,000 joules/inch.

b. Gas metal Arc (GMA) – An arc welding process in which the heat for welding is generated by an arc between a consumable electrode and the work metal. The bare solid electrode becomes the filler metal as it is consumed. The welding area is protected from atmospheric contamination by a gaseous shield provided by a steam of inert gas or a mixture of inert gases fed through the electrode holder. The GMA welding parameters are set, so that the maximum heat input is less than 84,000 joules/inch.

c. Gas Tungsten Arc (GTA) – An arc welding process in which the heat is produced between a nonconsumable electrode and the work metal. The welding area is protected by an inert gaseous shield. The GRA welding parameters are set, so that the maximum heat input is 60,000 joules/inch.

d. Shielded Metal Arc (SMA) – A welding process in which the heat for welding is generated by an arc established between a flux-covered consumable electrode and the work. The welding area is protected from atmospheric contamination by an inert gas shield obtained from combustion and decomposition of the flux covering. The SMA welding parameters are set, so that the maximum heat input is 60,000 joules/inch.

B&W QC personnel periodically check the weld deposits to ensure that the procedures are actually followed. The weld data sheets register the heat input range for the process for each weld. B&W's experience indicates that the welding processes used by B&W do not produce severe sensitization (as defined by ASME A393) in the weld deposits or heat-affected zones.

Hot cracking of welds in stainless steels used in systems for reactor shutdown and emergency core cooling is avoided by specifying that stainless steel welding materials contain 8 to 25% ferrite. The ferrite content is measured by plotting on the Schaeffler diagram the actual analysis of each heat of bare wire or consumable insert, or the all weld metal analysis of each batch of covered electrodes. This method ensures adequate ferrite content of production welds in stainless steel. Welding procedures and welders are qualified in proper accordance with Section IX of the ASME Code, and the procedure limitations are strictly enforced during production welding. Preheat and interpass temperatures used during production welding of stainless steel are limited to 350 °F maximum, assuring that the heat input is kept low.

Insofar as practicable, components such as valves are furnished in the solution-heat-treated condition. Where welding was required during manufacturing, welding procedures were reviewed to ensure that heat input and interpass temperatures were within the proper range to avoid severe sensitization of austenitic stainless steel. Castings were fully heat-treated following weld repair.

5.2.3.10 Delta Ferrite Control

The chemistry of austenitic stainless steel welds in the primary system was selected on the basis that the predicted microstructure of the weld deposit eliminates the probabilities of microfissuring. Generally, all austenitic stainless steel weld metals except for type 16-8-2 filler metal have delta ferrite contents within the range of 5 to 15% as calculated by the Schaeffler diagram or as determined by the magnetic permeability of an undiluted weld pad. Welding procedures are in accordance with the applicable section of the ASME Code.

Sensitization in stainless steels used in systems for reactor shutdown and emergency core cooling is prevented by ensuring that the ferrite content of stainless steel castings and welds is above 5%. This has been done by specifying that all stainless steel castings contain between 5 and 15% ferrite and all welding materials contain between 8 and 25% ferrite. The ferrite content is measured, using the Schaeffler diagram.

5.2.3.11 Compliance With Guides for RC Pump Flywheels

During the Eighth refueling outage a reactor coolant pump motor and flywheel were replaced. The motor was manufactured to the same design specifications as the original motor. The flywheel supplied with the replacement motor differs from the flywheel described in Appendix 5A in material and testing, but does not affect the compliance with GDC-4 and Safety Guide 14. For more detail on the flywheels see Appendix 5A.

5.2.4 RCPB Leak Detection System

The Reactor Coolant System leakage detection system includes the following:

a. The containment atmosphere particulate radioactivity monitoring system.

b. The containment sump level/flow monitoring system.

c. The containment atmosphere gaseous radioactivity monitoring system.

The containment atmosphere particulate radioactivity monitoring system incorporates monitors which are seismic, redundant, supplied with essential power, and Q-listed.

The containment sump level and flow monitoring system design includes containment vessel normal sump level indication in the control room. Flow rates are obtained by monitoring pump run time and multiplying by a flow rate. Instrumentation and power supplies are not essential. The transmitter is mounted seismically. There are redundant normal sump levels and alarms with control room indication. They are non-essential and non-seismic.

Wide and narrow range sump level indicators have been added to the containment sump design.

The containment atmosphere gaseous radioactivity monitoring system utilizes monitors which are seismically qualified and supplied with essential power. They are described is USAR Subsection 12.1.4.

All systems used for reactor coolant system leakage detection can be calibrated during operation except the sump levels.

The Davis-Besse Unit 1 RCS Leak Detection Systems meet the intent of the regulatory positions provided in Regulatory Guide 1.45 and therefore the prerequisites for applying GDC-4 are satisfied.

5.2.4.1 Methods of Determining Leakage

Portions of the system are located within the secondary shielding and are inaccessible during reactor operation. Coolant leakage to the CV atmosphere is in the form of liquid and vapor. The liquid drains to the containment-vessel normal sump, and the vapor is condensed in the containment air coolers and also reaches the CV normal sump via a drain line from the coolers.

The CV radiation monitors provide positive indication in the control room of RC leakage. Analyses of RC inventory trends and CV normal sump level changes also provide positive indication of RC system leakage to the CV.

The water inventory balance method for an accurate determination of RCS leakage and the guidance for locating and classifying leakage are described in Surveillance Test, RCS Water Inventory Test.

5.2.4.2 Indication of Leakage

Changes in the rate of increase in CV normal sump water level are an indication of total containment leakage. The level alarms on the station computer and annunciator and level indicator are located in the control room.

Makeup to the RC System as a result of leakage is initially supplied from the makeup tank inventory. Monitoring of the makeup tank level provides a direct indication of RC leakage.

Changes in the RC leakage rate in the CV cause changes in the control room indication of the CV atmosphere gas activity.

5.2.4.3 Leak Detection System Adequacy

The sensitivity and response time of the reactor coolant pressure boundary leakage detection systems vary depending on the detection method. The containment particulate monitor is the most sensitive radiation detection instrument available for detecting reactor coolant leakage into the containment.

The particulate monitor sensitivity to an increase in reactor coolant leakage rate depends on the magnitude of the normal leakage into the containment. The sensitivity is, greatest when the normal leakage is low.

The containment airborne particulate monitor measures the buildup of particulate on a fixed filter and measures the sample flow which produces the particulate buildup. If normal reactor coolant leakage results in air particulate activity above the air particulate monitor's detectability threshold, the instrument can be adjusted to alarm on leakage increases above the baseline value.

The containment radioactive gas monitor is inherently less sensitive than the containment air particulate monitor and will function if significant reactor coolant gaseous activity existed from fuel cladding defects.

The containment radioactive gas monitor is capable of detecting gross gaseous radioactivity in concentrations as low as 2.6×10^{-6} μCi/cc of containment air (See Table 11.4-1). Assuming a reactor coolant gaseous activity of 2.75 μCi/cc at the beginning of core life (4 EFPD), corresponding to about 0.1 % fuel defects, a leak of 1.0 gpm would be found at 44 minutes after the leak began with the radioactive gas monitor. Under these circumstances, this instrument is a useful backup to the air particulate monitor.

Leakage to the Component Cooling Water (CCW) System from the Reactor Coolant System heat exchangers inside the containment is detected by radiation detectors and surge tank level detectors in the CCW system. (See Subsection 9.2.2, Component Cooling Water System.)

5.2.4.4 RCPB Leakage Crack Analysis

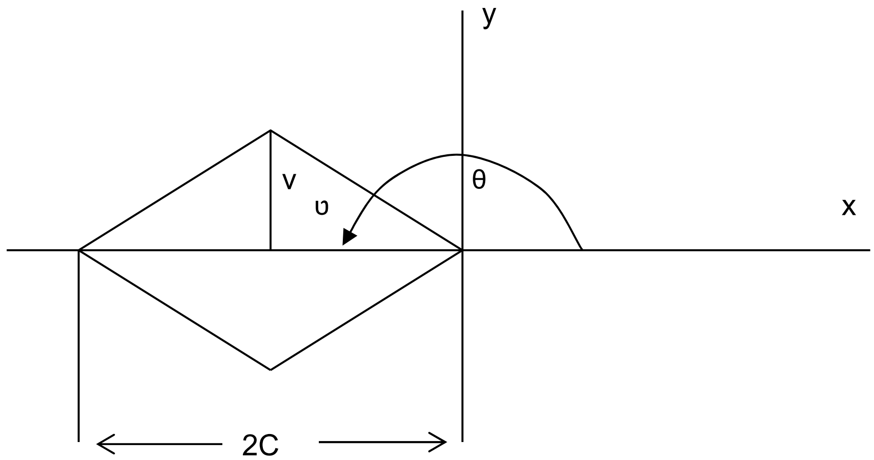
The following information is furnished to determine the maximum allowable leakage rate from unidentified sources in the RCPB:

a. The length of a through-wall crack that would leak at the rate of the proposed limit, as a function of wall thickness.

b. The ratio of that length of a critical through-wall crack based on the application of the principles of fracture mechanics.

c. The mathematical model and data used in such analysis.

1. To determine the crack size required for a leakage rate of 1 gpm, the following crack form was assumed:



The displacement (v) in the y-direction is found to be

$$v = \frac{K_I}{G}\left[\frac{\sqrt{r}}{\sqrt{2\pi}}\sin\frac{\theta}{2}\right]\left[2 - 2\upsilon - \cos^2\left[\frac{\theta}{2}\right]\right]$$

where:

$K_I = \sigma y\sqrt{\pi C}$

σ_Y = yield strength

υ = Poisson's ratio

G = shear modulus

Therefore, for θ = 180 degrees, r = C and v = $(\sigma_Y / G\sqrt{2})(2 - 2\upsilon)$

The velocity (U) out of the crack may be shown to be

$$U = \left[\frac{(P_1 - P_2)(2g)(144)}{\rho\left[\frac{fT}{d} + K\right]}\right]^{1/2}$$

where:

P_1 = operating pressure (2200 psia),

P_2 = containment vessel pressure (15 psia),

ρ = density, lbm/ft^3,

f = friction factor (assumed 0.06),

T = thickness of shell or pipe considered, in.,

d = equivalent diameter of crack, in.,

K = shock factor for expansion and contraction (1.5),

g = Newton's Constant 32.2 $\frac{\text{lbm} - \text{ft}}{\text{lbf} - \text{sec}^2}$

For the 28-inch-ID cold leg piping,

OD = 33-1/8 in.,

ID = 28-5/8 in.,

T = 2.25 in. (excluding thickness of cladding),

ρ = 46.4 lbm/ft^3 at 557°F,

σ_Y = 30.4 x 10^3 psi at 570 °F (106 Grade C),

G = 11.5 x 10^6 psi,

υ = 0.3.

Assuming a crack length of 1-1/2 inches (C = ¾ in.)

V = 0.00192 in.,

A = (4)(C)(V/2) = 2CV = 0.00296 in^2,

$$d = \frac{4A}{WP} = \frac{4(2CV)}{4\left(C^2 + V^2\right)^{1/2}} \cong 2V = 0.00394\text{in.}$$

where A is flow area and WP is wetted perimeter. Therefore,

$$U = \left[\frac{(2200 - 15)(2)(32.2)(144)}{46.4\left[\frac{(0.06)(2.25)}{0.00394} + 1.5\right]} \right]^{1/2} = 110.5\text{fps}$$

Flow rate in gallons/minute = 1.0 gpm.

Therefore, a crack length of 1-1/2 inches in the cold leg piping may be expected to leak at the rate of 1 gpm. Similarly, for the reactor vessel a crack length of 1-5/8 inches was determined.

2. Based on the answer to item 1 and the analysis described in item 3, ratios of through-wall crack length for the leak limit to the critical through-wall crack are as follows:

Irradiated reactor vessel shell:

$$\text{Ratio} = \frac{1.625\text{in.}}{10.9\text{in.}} = 0.15$$

Primary piping cold leg:

$$\text{Ratio} = \frac{1.50\text{in.}}{8.7\text{in.}} = 0.17$$

3 a. Consider a through-wall crack in the irradiated reactor vessel shell. Because of the large diameter of the shell and the wall thickness, the mathematical model considered most reasonable is for a through-wall crack in a flat plate. The equation is

$$K = \sigma(\pi C)^{1/2}$$

where:

K = stress intensity factor,

C = ½ crack length,

σ = membrane stress.

When K reaches a critical value called K_{IC}, the fracture toughness of the material, the crack length (2C) is the critical crack length.

For the reactor vessel, the conditions are:

Material, SA508CL2.

Temperature, 570 °F, $\sigma_Y(570) = 42.2$ ksi.

$\sigma = 29.0$ ksi (from contract design reports).

$K_{IC} = 120 \text{ ksi } \sqrt{\text{in}}$.

The K_{IC} value is actually for SA533CL1 plate, but SA508CL2 is an equivalent forging specification. Since no fracture toughness information is available on SA508CL2 material, the use of K_{IC} for SA533CL1 has been justified by comparing Charpy, dropweight, and NDT data for both materials.

Rearranging the foregoing equation and substituting K_{IC} for K,

$$C = \frac{K_{IC}^2}{\pi\sigma^2} = \frac{(120)^2}{(29)^2} = 5.45 \text{ in.}$$

Therefore, the critical crack length = 2C = 10.9 inches.

3 b. Consider a through-wall crack in the cold leg of the primary piping. An equation considered applicable for high-toughness materials is

$\sigma^* = \sigma_h M$

where:

σ^* = flow stress of material, $1.04\sigma_Y + 10$ ksi,

σ_h = membrane hoop stress,

M = stress magnification factor, $\left[1 + 1.61\frac{C^2}{Rt}\right]^{1/2}$

σ_Y = yield stress,

C = ½ critical crack length,

R = average radius of pipe,

t = pipe wall thickness.

For the primary piping the conditions are as follows:

Material, SA106 GR C.

Temperature, 670 °F σ_Y (570°) = 30.425 ksi.

ID = 28-5/8 in.

t = 2-1/4 in.

R = 15-7/16 in.

$\sigma_h = \sigma_Y$ = 30.425 (assumed as worst case).

Rearranging this equation and making appropriate substitutions,

$M = \sigma^*/\sigma_h$

$$1 + 1.61\frac{C^2}{Rt} = \left[\frac{1.04\sigma_Y + 10}{\sigma_Y}\right]^2$$

$$1 + \frac{1.61\,C^2}{15.4375(2.25)} = \left[\frac{1.04(30.425) + 10}{30.425}\right]^2$$

$$1 + \frac{1.61\,C^2}{15.4375(2.25)} = 1.873$$

$$C^2 = \frac{0.873(15.4375)2.250}{1.61}$$

$C^2 = 18.834$

C = 4.35.

Therefore, the critical crack length = 2C = 8.7 in.

The flow stress (σ^*) equation is valid if

$$\frac{(K_C / \sigma_Y)^2}{C} \geq 5$$

yielding

$$K_C \geq \sigma_Y [5C]^{1/2}$$

$$\geq 30.425\ [5(4.35)]^{1/2} = 142$$

where K_C = critical stress intensity factor (ksi $\sqrt{in.}$)

K_C data for SA106 GR C material are not available. Therefore, data on SA106 GR B material, some of which could be classified as SA 106 GR C, were used. The lowest K_C value for the SA106 GR B material that could be classified as GR C is 168 ksi $\sqrt{in.}$

5.2.4.5 Maximum Allowable Leakage

The maximum allowable leakage rate for the RC system is specified in the Technical Specifications. The RC makeup system is normally capable of providing up to 140 gpm of makeup. The ratio of the normal makeup capacity to the leakage limit is 14.

5.2.4.6 Leakage Detection Sensitivity

The sensitivities of the leak detection methods are as follows:

a. Containment vessel normal sump – The sump contains 30 gallons per inch of height. Any significant increase in sump level will be detected in the control room.

b. Reactor coolant inventory – The pressurizer liquid level is automatically maintained at a constant level. Any loss of reactor coolant results in a decrease in makeup tank level. The makeup tank capacity is 31 gallons per inch of height, and each gradation on the level indicator represents 1 inch of tank fluid height. A 1-gpm leak would take approximately 30 minutes to be detected.

c. Leakage detection sensitivities for containment radiation monitors is discussed in Section 5.2.4.3.

d. Component cooling water radioactivity monitor – The CCW radioactivity monitor includes two detector channels as described in Subsection 11.4.2.2.3. Based on a reactor coolant system cesium-137 activity of 2.6 X 10^{-2} μCi/cc for 0.1% failed fuel, a leakage rate of 1 gpm from the RCS to the CCWS would be detected in about 15.6 minutes.

5.2.4.7 Leakage Identification

The station design has provisions for identifying the source of reactor coolant leakage, or at least for systematically eliminating several sources from consideration, during normal operation.

A review of potential causes of reduction in inventory of reactor coolant indicates that fluid losses could be postulated at the following areas, given as examples, with means of identifying and the adequacy of these means included in the listing:

a. RC pump seals – an RC pump mechanical seal leak could be identified by one or a combination of the following indications:

1. A high level indication on the RCP seal leakage flow detector.

2. High seal water temperature at the outlet of the RC pump seal return to seal return cooler.

3. High level in the reactor coolant pump standpipe mechanical seal.

b. Pressurizer relief valves – reactor coolant inventory reduction as a result of seat leakage through the pressurizer pilot-operated relief valve (PORV) may be identified by either of these indications:

   1. High temperature downstream of the relief valve.

   2. Increased level in the pressurizer quench tank that provides an indication of the magnitude of the leak

c. Safety injection and decay heat removal lines connected to the RC system – back-leakage through a safety injection or decay heat removal line could be identified as the point of RC leakage through the following items:

   1. Level change in Core flooding Tank(s) will provide an indication of the magnitude of the leak.

   2. Contact temperature of safety injection lines located outside the CV.

   3. Increase in radioactivity concentration in safety injection fluid as determined by sample analysis.

   4. Increase in pressure in safety injection and Decay Heat Removal System lines located outside the containment vessel, or relief valve discharge on connected low-pressure systems.

d. Steam generator primary-to-secondary leakage can be identified as the cause of a reduction in RC inventory by one or a combination of the following:

   1. Sampling secondary side for radioactivity.

   2. Measuring N^{16} levels in the main steam lines upstream of the isolation valves.

   3. High activity as monitored in the condenser air ejector vent lines.

   4. Indication of radioactivity, boric acid, or conductivity in condensate, e.g., from main steam line drain traps, as indicated by laboratory analysis.

   5. An indication of the magnitude of the leak is determined by secondary side radioactivity measurements, e.g., tritium balance method.

e. RC System drain lines – leakage could be identified as possibly originating from flow through one or more RC drain valves by a combination of observing RC drain tank level and noting the contact temperature change of the RC drain header piping leading to the RC drain tank.

f. Makeup and Purification System – a reduction in RC inventory could be caused by leakage from portions of this system, most of which are located outside the CV. Leakage originating from this source can be confirmed through observation of sump levels, sump pump operation, valve stem leakage, pump leakoff connections, and visual inspection of portions of the piping.

g. RC System pipe or vessel connections – Increased containment air cooler normal heat load, increased drainage collection rate, and increased airborne activity would be symptoms of a significant leak directly to the CV atmosphere. The precise location of the leak could best be determined by visual inspection for escaping steam or water or by the presence of boric acid crystals deposited near the leak.

   Leakage from the vessel connections in the incore monitoring instrumentation region can be detected by the Containment Leakage Detection System. The system, tradename FLÜS, is a non-safety related system that provides early detection of leakage by monitoring humidity levels in the region between the reactor vessel insulation and the reactor vessel near the incore nozzle area.

The limits on RC system leakage are given in Technical Specifications.

5.2.4.8 Shutdown Due to Excessive Leakage

Criteria for shutdown of the reactor in the event that the reactor coolant leakage rate is exceeded are given in the Technical Specifications.

5.2.4.9 Testing of Leakage Detection System

The sensitivity and operability of the RC system inventory and CV normal sump methods of leak detection are determined by the checks, tests, and calibrations conducted as part of the preventive maintenance program. Pump flow rate tests are performed for the CV normal sump leak detection system.

Surveillance testing requirements for RC system leakage are given in the Technical Specifications.

5.2.5 Inservice Inspection and Testing Program

Inservice Inspection Program for the Davis-Besse Nuclear Power Station Unit No. 1 was submitted in compliance with 10CFR50.55a(g). The Inservice Inspection Program for the fourth 10-year interval has been developed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI of the 2007 Edition with the 2008 Addenda as modified by 10 CFR 50.55a or relief granted in accordance with 10 CFR 50.55a.

The Inservice Testing Program for the Davis-Besse Nuclear Power Station Unit No. 1 was submitted in compliance with 10 CFR 50.55a(f). The Inservice Testing Program for the fourth 10-year interval has been developed in accordance with the requirements of the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) of the 2004 Edition through the 2006 Addenda as modified by 10 CFR 50.55a or relief granted in accordance with 10 CFR 50.55a.

5.2.6 Loose Parts Monitoring

Any loose parts would accumulate in the bottom of the reactor vessel and in the top of each steam generator. These three areas are monitored acoustically by the loose parts monitoring system (LPM), which provides the operator an immediate audible and visual alarm of any moving, loose parts in these areas. The LPM includes facilities whereby pre-recordings of sounds normal to the reactor can be played for comparison. In addition, the recording facilities are capable of recording sounds from any of the monitored points for analysis and record.

TABLE 5.2-1

Reactor Coolant System Codes and Classifications

Component	Order Date	Code Class	Code
Reactor vessel	10-28-68	A	ASME[1][7]III 1968 Edition w/Addenda thru Summer '68
Reactor Vessel Closure Head	NA	A	ASME III 1989, Edition w/No Addenda
CRDM (PASNY, PGE & WPPSS SOURCED)	NA	A	ASME III, 1974 Edition w/No Addenda
CRDM (TVA SOURCED)	NA	A	ASME III, 1971 Edition w/Summer 1972 Addenda
Steam Generators (tubeside)	11-27-2007	A	ASME[1][6][10] III 2001 Edition, 2003 Addenda
(shellside)	11-27-2007	A	ASME[1][6][10] III 2001 Edition, 2003 Addenda
RC pump casing	05-15-70	A	ASME[1] III 1968 Edition w/Addenda thru Winter '68
Pressurizer	10-28-68	A	ASME[1] III 1968 Edition w/Addenda thru Summer '68
Pressurizer safety valves	01-15-71	I	Draft ASME[1] Pump and Valve Code, November '68
Pressurizer Pilot-Operated Relief Isolation Valve	NA	I	ASME[1] III 1974 Edition w/Addenda thru Summer '76
Pressurizer spray line isolation valve	01-16-91	I	ASME III 1986 Edition, No Addenda
Relief valve	01-15-71	I	Draft ASME[1] Pump and Valve Code, November '68
Motor			IEEE[2], NEMA[3], and ANSI[4]
Loop isolation valves			ASME[1] III 1971
2 1/2" & larger	11-11-71		
2" & smaller	02-15-72		
Other valves			ASME[1] III 1971 or later
2 1/2" & larger	11-11-71		NRC endorsed Code year
2" & smaller	or later		w/addenda
R.C. Piping	10-28-68	1	ANSI[5 x 8] B31.7 Draft, February '68 w/Errata, June '68
Replacement R.C. Piping	11-27-2007	1	ASME[1][10] III 2001 Edition, 2003 Addenda
Reactor Head Vent		1	ASME[1] III 1971
Other piping	06-23-71		ASME[1] III 1971

TABLE 5.2-1 (Continued)

Reactor Coolant System Codes and Classifications

(1) American Society of Mechanical Engineers, Boiler and Pressure Vessel Code.

(2) Institute of Electrical and Electronics Engineers.

(3) National Electrical Manufacturers Association.

(4) American National Standards Institute No. C50.2-1955 and C50.20-1954.

(5) American National Standards No. B31.7 dated February 1968, including Errata sheet dated June 1968.

(6) Tubeside Faulted conditions for the Replacement OTSG are evaluated based on ASME Section III NB-3225 and Appendix F-1331.1, 2001 Edition, 2003 Addenda.

(7) The reactor vessel closure head has been replaced with a new closure head provided by AREVA Inc. This reactor vessel closure head is constructed in accordance with the requirements of the ASME B&PV Code, Section III, 1989 Edition with No Addenda.

(8) Calculation C-ME-099.20-004, Rev. 0, RCS Piping Stress Report Summary, also shows that the RCS piping meets the design requirements of USAS B31.7-1969.

(10) The replacement OTSGs (including MFW and AFW components) and Replacement Hot Leg Spools are designed and fabricated in accordance with Class 1 requirements of the ASME Boiler and Pressure Vessel Code, Section III, 2001 Edition, 2003 Addenda.

TABLE 5.2-2

Code Case Interpretations

Code Case	Title
1332-4	Requirements for Steel Forgings, Sections III and VIII, Division 2 (ASME)
1335-5	Requirements for Bolting Material, Section III (ASME)
1337-5	Requirements for Special Type 403 Modified Forgings or Bars, Section III (ASME)
1401	Welding Repairs to Cladding of Class A Section III Vessel After Final Post-Weld Heat Treatment (ASME)
1407-1	Time of Examination for Class A Section III Vessels (ASME)
1440	Use of ASME Code for Pumps and Valves for Nuclear Power – Class I and II Safety Valves, Section III (ASME)
1492	Post-Weld Heat-Treatment Sections I, III, VIII, Divisions 1 and 2 (ASME)
N-411	Alternative Damping Valves for Seismic Analysis of Piping, an alternative to Table N-1230-1, Appendix N of ASME Section III Division 1
N-474-1	Design Stress Intensity and Yield Strength Values for UNS N06690 with a Minimum Specified Yield Strength of 35 ksi, Class 1 Components Section III, Division 1
N-474-2	Design Stress Intensity and Yield Strength Values for UNS N06690 with a Minimum Specified Yield Strength of 35 ksi, Class 1 Components Section III, Division 1
N-389-1	Alternative Rules for Repairs, Replacements or Modifications, Section XI, Division 1
	- ASME Section XI, 1989 Edition, Paragraph IWB-4230, Tube or Tubesheet Hole Plugging by Fusion Welding
N-504-2	Alternate Rules for Repair of Classes 1, 2 and 3 Austenitic Stainless Steel Piping and Section XI, Division 1
N-504-3	Alternate Rules for Repair of Classes 1, 2 and 3 Austenitic Stainless Steel Piping and Section XI, Division 1)
N-588	Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels.
2142-2	F-Number Grouping for Ni-Cr-Fe Filler Metals Section IX (Applicable to all Sections, including Section III, Division 1, and Section XI)
N-638-1	Similar and Dissimilar Metal Welds Using Ambient Temperature Machine GTAW Temper Bead Technique, Section XI, Division 1
N-638-2	Similar and Dissimilar Metal Welds Using Ambient Temperature Machine GTAW Temper Bead Technique, Section XI, Division 1
N-638-3	Similar and Dissimilar Metal Welds Using Ambient Temperature Machine GTAW Temper Bead Technique, Section XI, Division 1
N-638-4	Similar and Dissimilar Metal Welds Using Ambient Temperature Machine GTAW Temper Bead Technique, Section XI, Division 1
N-640	Alternative Reference Fracture Toughness for Development of P-T Limit Curves

TABLE 5.2-2 (Continued)

Code Case Interpretations

Code Case	Title
N-740-2	Dissimilar Metal Weld Overlay for Repair or Mitigation of Class 1, 2, and 3 Items
N-770	Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated With UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities
N-474-2	Design Stress Intensities and Yield Strength Values for UNS N06690 with a Minimum Specified Yield Strength of 35 KSI, Class 1 Components, Section III, Division 1
2143-1	F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode Section IX
N-416-3	Alternative Pressure Test Requirements for Welded or Brazed Repairs, Fabrication Welds or Brazed Joints for Replacement Parts and Piping Subassemblies, or Installation of Replacement items by Welding or Brazing, Classes 1,2, and 3 Section XI, Division 1
N-729-1	Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure Retaining Partial Welds Section XI, Division 1
1337-7	Special Type 403 Modified Forgings or Bars, Section III
1337-10	Special Type 403 Modified Forgings or Bars, Section III

TABLE 5.2-3

Loading Conditions and Stress Limits for Code Class 1 Pressure Piping per ANSI B31.7

Loading Condition	Stress Intensity Limits	
Normal & Upset	Primary	a) $P_m \le S_m$ b) $P_L + P_b \le 1.5\ S_m$
	Secondary	a) $P_e \le 3.0\ S_m$ b) $P_L + P_b + P_e + Q \le 3.0\ S_m$ c) $P_L + P_b + P_e + Q + F \le S_e$
Emergency	Primary	a) $P_m \le 1.2\ S_m$ or S_y [(1)] b) $P_L \le 1.8\ S_m$ or $1.5\ S_y$ or $0.8\ C_L$ c) $P_L + P_b \le 1.8\ S_m$ or $1.5\ S_y$ or $0.8\ C_L$
	Secondary Stresses – Evaluation not required	
Faulted	Primary	a) $P_m \le$ b) $P_L \le$ c) $P_L + P_b \le$ → $2/3\ S_u$
	Secondary Stresses – Evaluation is not required	

Where:
(1) Use the larger value.

C_L = Collapse load defined in ASME Boiler and Pressure Vessel Code, Section III
F = Peak stress for fatigue evaluation.
P_b = Primary bending stress.
P_e = Expansion stress intensity.
P_L = Primary local membrane stress.
P_m = Primary general membrane stress.
Q = Self-equilibrating stress.
Nominal value of expansion stress.
S_m = Allowable stress.
S_e = Allowable stress from the application design fatigue curve.
S_y = Minimum yield strength at applicable temperature.
S_u = Ultimate strength of material at temperature.

TABLE 5.2-4

Loading Conditions and Stress Limits:
Code Class I Pressure Vessels[1]

Condition	Stress Intensity Limits
1. Normal	(a) $P_m \leq S_m$ (b) P_m (or P_L) + $P_b \leq 1.5\ S_m$ (c) P_m (or P_L) + P_b + Q $\leq 3.0\ S_m$ (d) P_m (or P_L) + P_b + Q + F $\leq\ S_e$
2. Upset	(a) $P_m \leq S_m$ (b) P_m (or P_L) + $P_b \leq 1.5\ S_m$ (c) P_m (or P_L) + P_b + Q $\leq 3.0\ S_m$ (d) P_m (or P_L) + P_b + Q + F $\leq\ S_e$
3. Emergency	(a) $P_m \leq 1.2\ S_m$ or S_y whichever is larger (b) P_m (or P_L) + $P_b \leq 1.5\ (1.2\ S_m)$ or $1.5\ S_y$ whichever is larger or $0.8\ C_L$
4. Faulted	(a) $P_m \leq C_L$ or - 2/3 S_u (b) P_m (or P_L) + $P_b \leq C_L$ or 2/3 S_u

Where:

P_m = primary general membrane stress intensity.
P_L = primary local membrane stress intensity.
P_b = primary bending stress intensity.
Q = secondary stress intensity.
S_m = allowable stress value from ASME B&PV Code, Section III, Nuclear Vessels.
S_y = minimum specified material yield (ASME B&PV Code, Section III).
C_L = collapse load defined in ASME B&PV Code, Section III.
S_u = ultimate strength of material at temperature.
S_e = allowable stress from the application design fatigue curve.

[1] Original Construction Loading Conditions and Stress Limits for Code Class I Pressure Vessels. See appropriate calculation for construction code limits used for current calculations

TABLE 5.2-5

Primary Plus Secondary Stress
Intensity Summary for Pressurizer Components[2]

Area	Stress Intensity, psi	Allowable Stress, 35_m, psi (Operating Temperature)
Spray Nozzle	33,200	70,000
Surge Nozzle	48,000	51,000
External Supports	35,010	55,200
Heater Bundle Core Plate	39,100	80,100
Closure Studs	58,100	81,800
Shell	43,000	55,200

Cumulative Fatigue Usage Factors Summary for Pressurizer Components[2]

Area	Usage Factor[1]
Spray Nozzle	0.00
Surge Nozzle	0.34
External Supports	0.10
Heater Bundle Cover Plate	0.05
Closure Studs	0.59
Shell	0.03

(1) As defined in Section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

(2) Original construction values, see appropriate calculation for current values.

TABLE 5.2-6

Primary Plus Secondary Stress Intensity
Summary for Steam Generator[3]

Location	Stress Intensity, psi	Allowable stress, 3 S , psi	Cum. Fatigue usage factor[1]
Primary nozzles			
Inlet	26,700	52,000	0.01
Outlet	34,000	52,000	0.01
FW nozzle	42,900	63,000	0.40
Steam outlet nozzle	16,000	53,400	0.0
Aux. FW nozzle	19,300	53,400	0.0
Support skirt	121,500[2]	80,000	0.96[2]
Tubesheet	35,000	80,000	0.13
Tubesheet to shell			
Primary side	35,000	80,000	0.11
Secondary side	25,000	70,000	0.01
Secondary side shell	28,480	56,000	0.0

[1] As defined in section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

[2] From elastic-plastic analysis.

[3] Original construction values, see appropriate calculation for current values.

TABLE 5.2-7

Primary Plus Secondary Stress Intensity
Summary for piping Components[3]

Area	Stress intensity, psi	Fatigue usage factor[1]	Allowable stress, $3S_m$, psi[2]
Hot leg elbow	51,300	0.02	55,200
Hot leg straight	41,600	0.02	58,200
Cold leg elbow	44,800	0.02	56,640
Cold leg straight	31,900	0.01	59,670
Surge line elbow	67,400[1]	0.05	50,100
Surge line straight	48,400	0.0	50,100
Surge line to hot leg juncture	76,000[1]	0.87	50,100
Spray line elbow	--	--	--
Spray line straight	--	--	--
Spray line to cold leg juncture	--	--	--
Decay heat line to hot leg juncture	--	--	--
HPI line to cold leg juncture	--	--	--
Reactor Head Vent	--	--	--

(1) Elastic-plastic evaluation performed for justification.

(2) Operating temperature.

(3) Original construction values, see appropriate calculation for current values.

TABLE 5.2-8

Primary Plus Secondary Stress Intensity Summary for Reactor Vessel Components[3]

Area	Stress Intensity, psi	Fatigue usage factor[1]	Allowable stress, $3S_m$, psi[2]
Control rod housing	59,800	0.05	60,000
Head flange	63,300	0.03	80,100
Vessel flange	36,900	0.03	80,100
Closure studs	100,600	0.70	107,700
Primary nozzles			
Inlet	57,600	0.13	80,100
Outlet	32,400	0.01	80,100
Core flooding nozzle	57,500	0.08	80,100
Bottom head to shell	28,100	0.01	80,100
Bottom Instrumentation			
Nozzle belt to shell	33,400	0.01	80,100

(1) As defined in section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

(2) Operating temperature.

(3) Original construction values, see appropriate calculation for current values.

TABLE 5.2-9

Primary Piping Stresses [1]

Joint No.	Design Pressure + X + Y eq. psi	or	Dead Load + Z + Y eq. psi
Hot Leg			
111	14,798.93		15,653.41
122	23,373.75		25,011.56
130	20,187.62		21,111.49
52	20,368.08		21,162.87
125	23,641.39		25,277.11
139	12,468.74		13,392.62
Upper CL			
56	17,187.24		18,254.64
45	29,176.66		29,760.38
31	27,909.79		28,279.87
38	19,402.52		21,128.67
69	18,795.17		21,844.04
83	10,372.45		12,007.60
Upper CL			
113	17,218.78		18,361.54
127	30,479.07		31,667.54
142	27,963.28		28,466.09
144	19,421.81		21,102.63
143	18,807.92		21,814.35
138	10,376.79		12,001.42
Lower CL			
105	25,694.00		25,286.43
79	22,617.62		23,171.83
61	38,572.41		35,234.16
43			
Lower CL			
88			
102	38,579.35		35,142.84
118	22,691.25		23,252.83
131	25,064.26		23,468.01
Surge Line			
54	12,187.31		11,285.70
36	18,915.06		17,549.82
23	18,227.96		17,386.25
3	28,106.76		18,330.00
16	21,544.76		17,781.40
14	18,368.38		18,446.63
13	18,319.02		18,791.76
15	18,190.04		16,694.94
5	18,167.98		16,981.31
7	20,136.04		21,028.74

[1] Original construction values, see appropriate calculation for current values.

TABLE 5.2-10

Reactor Coolant Pressure Boundary (RCPB)

Component	Section	Material
Reactor vessel	Pressure plate	SA-533, Grade B, Class 1
	Closure Head Pressure Forging	SA-508 Class 3
	Pressure forgings	SA-508 Class 2
	Cladding	18-8 stainless steel
	Thermal shield and internals	SA-276 type 304 SA-240 type 304 SA-473 type 304 SA-312 type 304
	Nozzles	SA-508 Class 2
	Control rod drive mechanism, closure head continuous vent line, and incore instrument penetrations	Inconel-SB167

Weld material for all manganese-molybdenum base metal is Mn-Mo-NiF-60.

Additional weld material includes:

Safe end on terminal end of core flooding nozzle is SA336F8M.

Weld for safe end to core flooding nozzle is SB-195 E-Ni-Cr-Fe-3 (INCO 182T) or SB-304 ER-Ni-Cr. -3 (INCO 82T).

Weld for instrument nozzle to lower head SB-195 E-Ni-Cr-Fe-3 (INCO 182T).

Weld for control rod to adaptor SB-304 ER-Ni-Cr. -3 (INCO 82T).

Weld material for the reactor vessel closure head to CRDM nozzle J-grove weld is ERNiCrFe-7 or ENiCrFe-7. Weld material for the reactor vessel closure head to continuous vent line nozzle J-groove weld is ER-NiCrFe-7/7A.

Component	Section	Material
Pressurizer	Plate	SA-516 GR70, SA-533 GR B, SA-240 type 304
	Forgings	SA-508 Class 1, SA-182 type F316 SA-336 Class F8M, SB-166
	Bolting	SA-320 Grade L43
	Welds	SFA-5.5, SFA-5.9, SFA-5.11, SFA-5.14, SFA-5.17

TABLE 5.2-10 (Continued)

Reactor Coolant Pressure Boundary (RCPB)

Component	Section	Material
Steam Generator	Plate	SA-533 Type B Cl 1
	Forgings	SA-508 Gr. 3 Cl 2
	Seamless tube	SB-163 UNS N06690
	Bolting	SA-193 Gr. B7 and B8, SA-194 G. 7, ASTM F-436 Type 1
	Weld	SFA-5.4, SFA-5.9, SFA-5.11, SFA-5.14
Reactor Coolant Piping	Plate	SA-516 Gr 70
	Forgings	SA-182 types F316 and F316L, SB-166 Ni-Cr-Fe, SA-105 Gr II SA-508 Gr. 1A, SB-166 UNS N06690 TT
	Seamless pipe and tube	SA-106 Gr C, SA-376 type 316, SA-336 C1 F8M
	Fittings	SA-403 Gr WP 316
	Weld	SFA-5.4, SFA-5.5, SFA-5.9, SFA-5.11, SFA-5.14, SFA-5.17 SFA-5.23, SFA-5.28
	Bolting	SA-564, Gr 630, SA-194 Gr 7 or Gr 2H SA-453, Gr 660 Condition B, SA-320, Gr L43, ASTM 461 – Gr 630
Reactor Coolant Pumps	Materials in contact with reactor coolant :	
	Bolts and Studs	SA-193 BS Inconel X-750 HTH
	Castings	SA-351 Gr CF8M
	Forgings	SA-182 and SA-461, SA-403 SA-638 Gr 660, and SA-479, A-182 Gr FXM-19
	Plates	SA-240
	Piping	SA-312
	Tubing	SA-213 and SB-167
	Bars	SA-276 and SA-479
	Materials not in contact with reactor coolant:	
	Plates	SA-515, Gr 70
	Castings	SA-216, Gr WCB
	Tubing	SB-407
	Insulation	Type 304 stainless steel
Valves	Pressure-containing parts	(S)A-351, Gr CF8M, and Gr CF3M; (S)A-182, F316, and F347
	Wire	AMS-5698A
	Welds	SA-371 and SA-398

TABLE 5.2-10 (Continued)

Reactor Coolant Pressure Boundary (RCPB)

Component	Material				
	Body	Disc	Seat	Stem	Bonnet
Seal injection flow isolation valve					
Pump seal return isolation valve	ASTM-A182 F316	Stellite		ASTM-A461 Type 630	
Letdown cooler inlet valve	ASTM-A182 F316	ASTM-A182 F316	ASTM-A182 F316	ASTM-461 Type 630	ASTM-AB2 F316
Letdown cooler outlet valve	ASTM-A182 F316	ASTM-A182 F316	ASTM-A182 F316	ASTM-461 Type 630	ASTM-A182 F316
Letdown cooler	(Tubing) SA-213-T316L	(Manifold) SA-182-T316L	(Manifold sleeves) SA-182-T316L		
High-pressure injection valves	ASTM-A182	Stellite No. 6	ASTM-A461	ASTM-A182 Type 630	F316
Seal return isolation valve	ASTM-A182	Stellite No. 6 (wedge)		ASTM-A461 Type 630	
Makeup isolation valve	ASTM-A182 F316	ASTM-A182 F316	ASTM-A182 F316	ASTM-461 Type 630	ASTM-A182 F316
Letdown isolation valve	ASTM-A182 F316	ASTM-A182 F316	ASTM-A182 F316	ASTM-461 Type 630	ASTM-A182 F316
Pressurizer spray control valve	ASTM-182 F316	Stellite No. 6	Stellite	ASTM-A461 Type 630	ASTM-A182 F316
Pressurizer spray isolation valve	ASTM-A182 F316	ASTM-A182 F316	ASTM-A182 F316	SA-638 Type 660	ASTM-A182 F316
Pressurizer relief isolation valve	SA-182 F316	SA-351 CF3M	SA-351 CF3M	SA-564 Type 630	SA-182 F316

TABLE 5.2-10 (Continued)

Reactor Coolant Pressure Boundary (RCPB)

Component	Material				
	Body	Disc	Seat	Stem	Bonnet
Pressurizer pilot-operated relief valve	ASME SA182 F316	SS 347/348 Stellite 6 H.F. Chrome Plated	ASME 479 316 Stellite 6 H.F.		
Pressurizer safety relief valves	ASTM-A351 CF8M	ASTM-A477 GR-651	ACI-CF8M	ASTM-A276 Type 410	ASTM-105 GR II
LP injection valve	ASTM-A182 F316	(Wedge) ASTM-A351 CF8M	ASTM-A351 CF8M	ASTM-A461 Type 630	
DH removal outlet valve	ASTM-A182 F316	ASTM-A182 F316	ASTM-A351 CF8M	ASTM-A461 Type 630	

TABLE 5.2-12

Design Flow Distribution

Pump Combination	2-1 Flow	2-2 Flow	1-1 Flow	1-2 Flow
4 pumps	88,000	88,000	88,000	88,000
3 pumps	92,091	92,091	117,928	-38,397
1 pump/loop	120,421	-33,612	120,421	-33,612

TABLE 5.2-13

Reactor Coolant System Parameters [1]

Total core power output, MWt	2772
Design system flow, 10^3 Gpm/pump	88.0
Design core flow available for heat transfer 10^6 lb/h	123.4
Reactor vessel inlet temperature (100% power) °F	555.4*
Reactor vessel outlet temperature, (100% power) °F	608.6
Core flow area available for heat transfer, ft^2	49.2
Reactor coolant system pressure drop, ft	338
Unrecoverable core pressure drop, psi	17.2
Average core coolant velocity, ft/sec	15.7
Cold leg coolant velocity, ft/sec	45.9
Hot leg coolant velocity, ft/sec	61.0
Minimum core DNB ratio, W-3 (100% power)	1.79
Minimum core DNB ratio, W-3 (112% power)	1.41

At the end of a cycle, the average reactor coolant temperature, T_{AVE}, may be reduced by 12 °F (see section 5.1.2)

[1] Original design values, see appropriate calculation for current values.

TABLE 5.2-14

Fabrication Inspections

Components	Radiographic	Ultrasonic	Dye Penetrant	Magnetic Particle	Eddy Current
1. Reactor Vessel					
1.1 Forgings					
1.1.1 Flanges		X(1)		X	
1.1.2 Studs, bar		X			
1.1.3 Studs after final machining				X	
1.1.4 Nozzle shell forgings		X(1)		X	
1.1.5 Main nozzle forgings		X(1)		X	
1.1.6 Dutchman forging		X(1)		X	
1.1.7 CRD mechanism adaptor		X	X		
1.1.8 CRD mechanism housing		X	X		
1.1.9 Closure Head Continuous Vent Nozzle		X	X		
1.2 Plates					
1.2.1 Shell plate		X(1)		X	
1.3 Instrumentation tubes		X	X		
1.4 Closure O-rings		X	X		
1.5 Weldments					
1.5.1 Circumferential main seams	X			X	
1.5.2 CRD mechanism adaptor to shell			X		
1.5.3 CRD mechanism adaptor to flange	X		X		
1.5.4 Closure Head Continuous Vent Nozzle Lower Nozzle to Upper Nozzle	X		X		
1.5.4 Main nozzles	X			X	
1.5.5 Instrumentation nozzle connection			X		
1.5.6 Nozzle safe-ends, weld deposit		X	X		
1.5.7 Temporary attachment after removal				X	
1.5.8 All accessible welds after hydrotest			X or	X	
1.5.9 O-ring closure weld	X		X		
1.5.10 Cladding, sealing surfaces		X(2) (+)	X		
1.5.11 Cladding, all others		X(3) (+)	X		

TABLE 5.2-14 (Continued)

Fabrication Inspections

Components	Radiographic	Ultrasonic	Dye Penetrant	Magnetic Particle	Eddy Current
2. Steam generator					
2.1 Tubesheet					
2.1.1 Forging		X(1) (8)		X(8)	
2.1.2 Cladding		X(2) (+) (8)	X(8)		
2.2 Heads					
2.2.1 Forging		X(1) (8)		X(8)	
2.2.2 Cladding		X(3) (+) (8)	X(8)		
2.3 Shell					
2.3.1 Forging		X(1) (8)		X(8)	
2.4 Tubes (4)		X(8)			X(8)
2.5 Nozzles (forgings)		X(1) (8)	X(8) or	X(8)	
2.6 Studs, bar		X(8)		X(8)	
2.7 Studs after final machining				X(8)	
2.8 Weldments					
2.8.1 Deleted					
2.8.2 Deleted					
2.8.3 Shell, circumferential	X(8)	X(8)		X(8)	
2.8.4 Cladding, sealing surfaces		X(2) (+) (8)	X(8)	X(8)	
2.8.5 Cladding, all other		X(3) (+) (8)	X(8)	X(8)	
2.8.6 Nozzle to shell	X(8)	X(8)		X(8)	
2.8.7 Level sensing connections		X(8)	X(8)	X(8)	
2.8.8 Instrument connections			X(8)	X(8)	
2.8.9 Conical Support and base		X(+) (8)		X(8)	
2.8.10 Tube-to-tubesheet (5)			X(8)		
2.8.11 Temporary attachment after removal			X(8) or	X(8)	
2.8.12 After hydrostatic test (all accessible welds)		X(8)	X(8) (9)	X(8)	
2.8.13 Lifting lugs				X(8)	
3. Pressurizer					
3.1 Heads					
3.1.1 Plate		X(1)		X	
3.1.2 Cladding		X(3) (+)	X		

TABLE 5.2-14 (Continued)

Fabrication Inspections

Components	Radiographic	Ultrasonic	Dye Penetrant	Magnetic Particle	Eddy Current
3.2 Shell					
3.2.1 Forging		X(1)		X	
3.2.2 Plate		X(1)		X	
3.2.3 Cladding		X(3) (+)	X		
3.3 Header bundles					
3.3.1 Cover plate		X		X	
3.3.2 Diaphragm		X	X		
3.3.3 Studs, bar		X			
3.3.4 Studs and nuts after final machining				X	
3.3.5 Heaters					
3.3.5.1 Tubing		X	X(+)		
3.3.5.2 Positioning of heater element in tube	X				
3.4 Nozzle (forgings)		X(1)		X	
3.5 Weldments					
3.5.1 Shell, longitudinal as deposited by submerged arc	X			X	
3.5.2 Shell, longitudinal as deposited by electroslag	X	X		X	
3.5.3 Shell, circumferential	X			X	
3.5.4 Cladding, sealing surfaces		X(2) (+)	X		
3.5.5 Cladding, all other		X(3) (+)	X		
3.5.6 Nozzle to shell	X			X	
3.5.7 Nozzle safe ends (if weld deposit)		X	X		
3.5.8 Nozzle safe end (if forging or bar)	X		X		
3.5.9 Instrumentation and vent connections			X		
3.5.10 Support brackets		X		X	
3.5.11 Heater guide tube pad		X	X		
3.5.12 Temporary attachment after removal				X	
3.5.13 All accessible welds after hydrotest				X	
4. Piping					
4.1 Pipe					
4.1.1 Forgings		X(1)		X	
4.1.2 Cladding		X(3) (+)	X		

TABLE 5.2-14 (Continued)

Fabrication Inspections

Components	Radiographic	Ultrasonic	Dye Penetrant	Magnetic Particle	Eddy Current
4.2 Bends					
4.2.1 Plate		X(1)		X(+)	
4.2.2 Cladding		X(3) (+)	X		
4.3 Nozzle forgings		X(1)		X	
4.4 Weldments					
4.4.1 Longitudinal	X			X	
4.4.2 Circumferential	X			X	
4.4.3 Cladding, elbows		X(3) (+)	X		
4.4.4 Cladding, straight		X(3) (+)	X		
4.4.5 Nozzles to run pipe	X			X	
4.4.6 Thermowell connections			X		
5. Reactor Coolant Pumps					
5.1 Castings	X		X		
5.2 Forgings		X	X		
5.3 Weldments					
5.3.1 Circumferential	X		X		
5.3.2 Piping connections			X		
6. Valves (see note 6 and 7)					
6.1 Castings	X		X		
6.2 Forgings		X	X		

(1) 100% scanning for longitudinal-wave technique and 100% shear-wave technique.
(2) Test of cladding defects and of bond to base metal.
(3) Test of cladding bond to base metal (spot check).
(4) An additional eddy current test, in excess of code requirements, is also performed
(5) Also gas leak test – B&W requirement.
(+) Additional B&W requirement.
(6) Valve RC10 (Forging) received an RT examination in lieu of a UT examination
(7) ASME Section III – Paragraph NB-2510b exempts volumetric examination for forged valves between two (2) and four (4) inches nominal pipe size. Valve RC11 received a surface examination only.
(8) The replacement once through steam generator is inspected and tested as specified in the Certified Design Specification in accordance with the replacement OTSG Inspection and Test Plans (ITPs).
(9) PT Inspection of the following connections rather than MT: blowdown, drain, vent, sampling, temperature sensing, and level sensing.

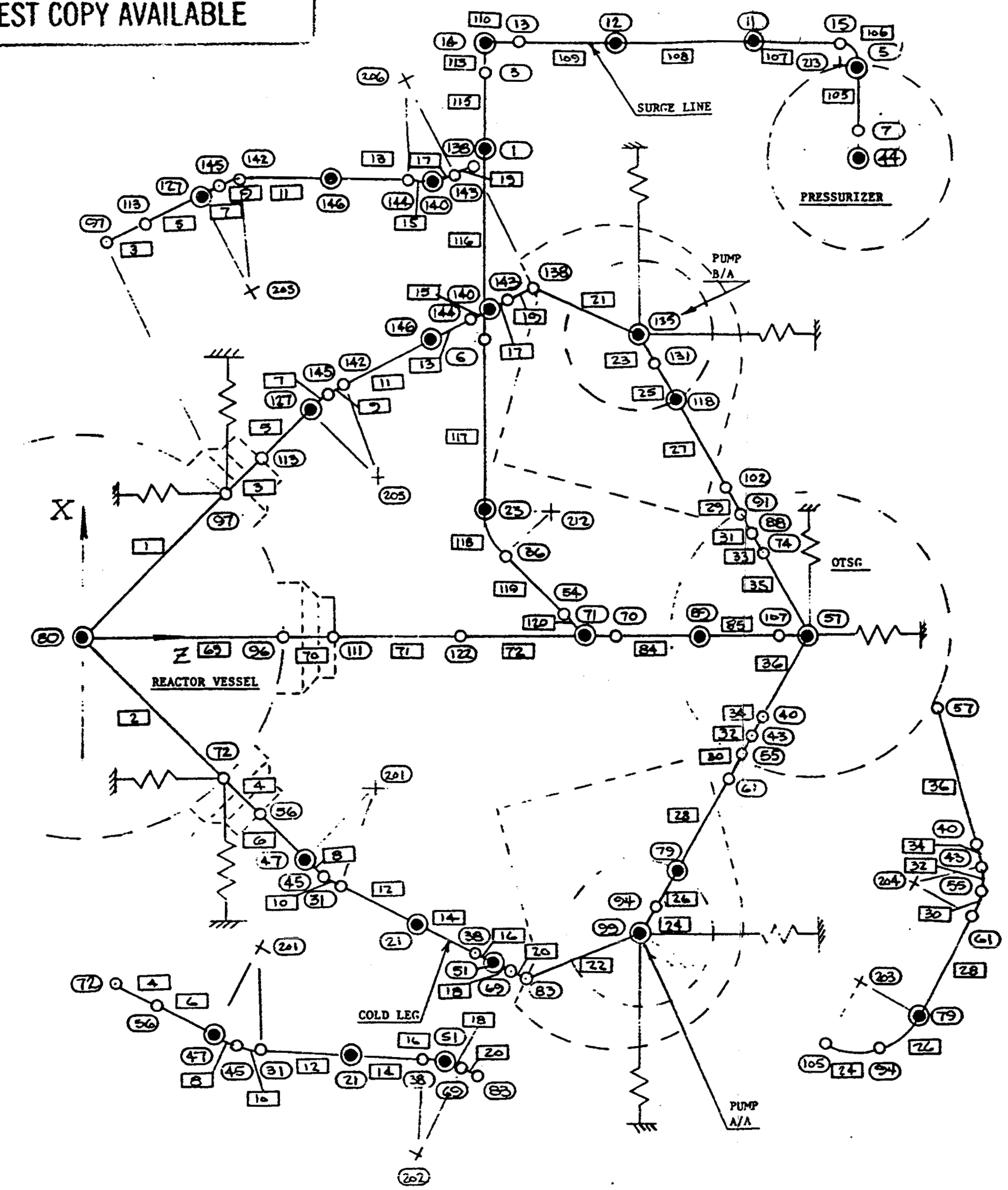
TABLE 5.2-15

Properties of Reactor Vessel Materials and Identification of Beltline Region Materials (historical)

Notes	Mtl. Ident. Heat No.	Type	Shell Region Location	Distance Core Midplane in Weld Centerline cm	Drop Wt. T NDT °F	CVN Transverse 50 ft-lb °F	CVN Transverse 35 mils L. E. °F	CVN Transverse USE Ft-lb	RT NDT °F	*** Chemistry Cu %	*** Chemistry P %	*** Chemistry S %	** End-of-Service Neutron Fluence, n/cm^2 (32 FPY, E>1 Mev) @1/4T	** End-of-Service Neutron Fluence, n/cm^2 (32 FPY, E>1 Mev) @3/4T
	123Y317 (ADB 203)	SA508 Cl. 2	Nozzle Belt		50	61	58	134	50	.04	.007	.009	3.2×10^{18}	1.0×10^{18}
2	123X244 (AKJ 233)	SA508 Cl. 2	Upper* Shell		20	30	22	144	20	.04	.004	.006	1.68×10^{19}	3.8×10^{18}
1	5P4086	SA508 Cl. 2	Lower* Shell		50	27	4	118	50	.02	.011	.011	1.68×10^{19}	3.8×10^{18}
4	5P4086 HAZ	Weld HAZ	Lower* Shell		-20	22	31	124	-20	.02	.011	.011	1.68×10^{19}	3.8×10^{18}
3	WF182-1	Weld	Middle* Circum. (100%)	Center	-20	62	14	81	2**	.18	.014	.015	1.68×10^{19}	3.8×10^{18}
	WF233	Weld	Upper* Circum. (91%)	195.6	-	-	-	(66)	(20)**	.22	.015	.016	4×10^{18}	8×10^{17}
	WF232	Weld	Upper* Circum. (9%)	195.6	-	-	-	(66)	(20)	.14	.011	.007	4×10^{18}	8×10^{17}
	WF233	Weld	Lower Circum. (88%)	-249	-	-	-	(66)	(20)	.22	.015	.016	$<5.7 \times 10^{17}$	-
	WF232	Weld	Lower Circum. (12%)	-249	-	-	-	(66)	(20)	.14	.011	.007	$<5.7 \times 10^{17}$	-

1 Surveillance Base Metal A
2 Surveillance Base Metal B
3 Surveillance Weld
4 Surveillance Weld HAZ, Base Metal A
* Beltline Region Materials
** See Pressure and Temperature Limits Report, Table 1, for updated data.

*** NOTE: The current chemistry values used in the determination of the pressure - temperature limits are based on AREVA Document: 32-9017744 – 003, “Davis-Besse ART Values at 52 EFPY,” 10/29/2009.



REACTOR COOLANT LOOP MATHEMATICAL MODEL (1)
(PLAN)

(1) Original construction model, see appropriate calculation for the current model.

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REACTOR COOLANT LOOP MATHEMATICAL MODEL (PLAN)

FIGURE 5.2-1

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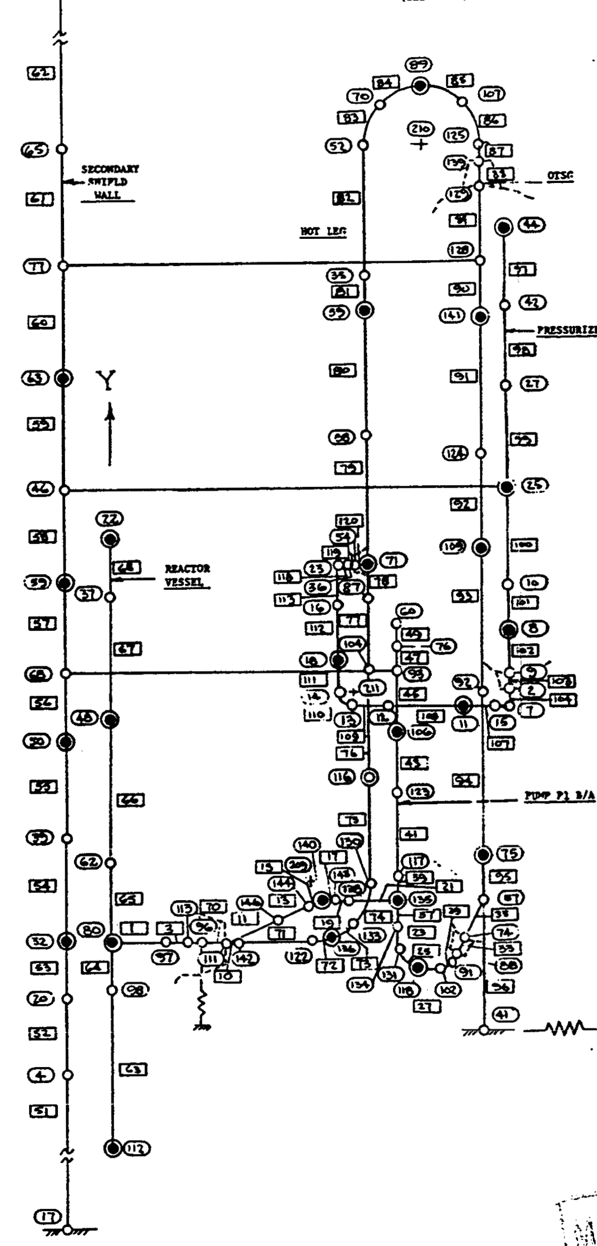
REACTOR COOLANT LOOP MATHEMATICAL MODEL (1)
(ELEVATION)



MARGINAL QUALITY DOCUMENT
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(1) Original construction model, see appropriate calculation for the current model.

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REACTOR COOLANT LOOP MATHEMATICAL MODEL (ELEVATION)

FIGURE 5.2-2

REVISION 16
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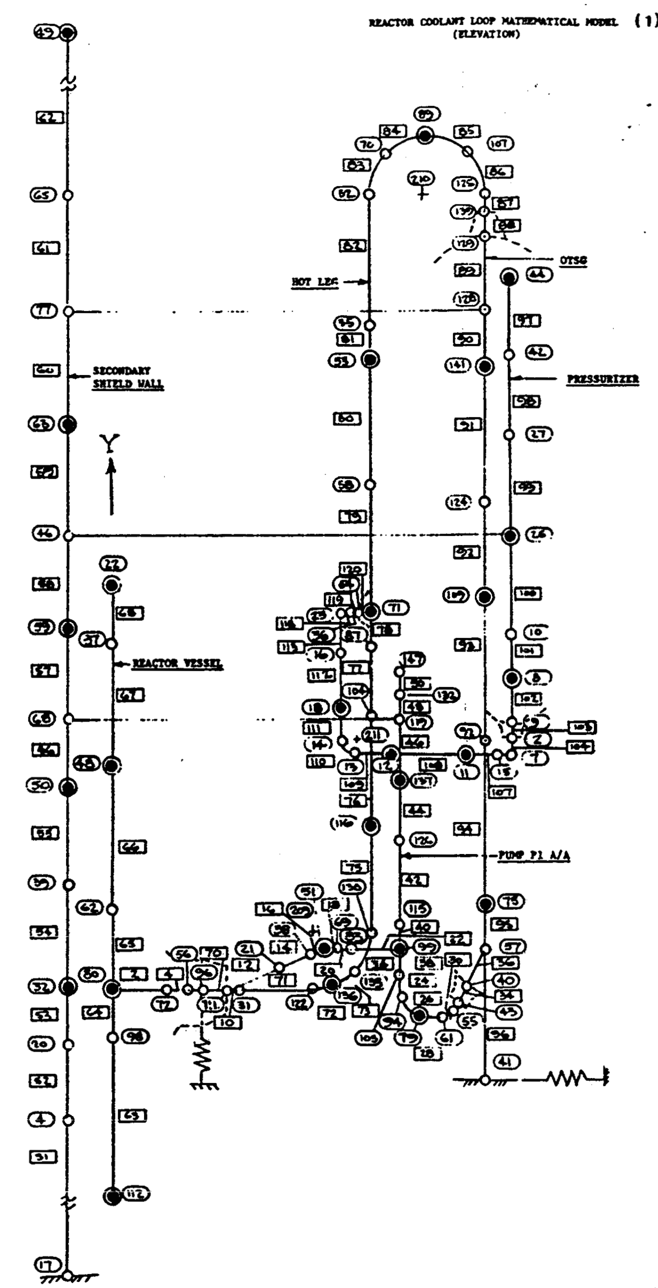

(1) Original construction model, see appropriate calculation for the current model.

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REACTOR COOLANT LOOP MATHEMATICAL MODEL (ELEVATION)

FIGURE 5.2-3

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5.3 THERMAL HYDRAULIC SYSTEM DESIGN

5.3.1 Bases of System Design

The RC System is designed to maintain a minimum pressure of 2135 psig at the core outlet during normal load changes such as steps and ramps up or down in power. The design also permits natural-circulation cooling of the system using the steam generators to remove decay heat, as a minimum, from normal operating conditions to the operating temperature and pressure at which the decay heat removal system may be placed in operation. In addition, the system is arranged to keep the reactor coolant volume at a minimum to limit the energy release during an accident.

The pressurizer steam and water volume are sized as follows:

a. The outsurge following a reactor trip from full power does not actuate the high-pressure injection system.

b. The system pressure following a reactor trip is sufficiently above the high-pressure injection actuation point, so that the quantity of makeup water needed to permit heater operation does not further reduce the pressure to the high-pressure injection actuation point when 50% of the makeup is assumed to be mixed with the saturated water in the pressurizer.

c. The insurge following a turbine trip does not fill the pressurizer to more than 90% of the total volume, assuming that the transient starts at the first high-level alarm point.

The pump flywheel provides continued flow following a loss of pump power, so that the reactor neutron power can be reduced before departure from nucleate boiling occurs.

System flow is adequate during both steady-state and anticipated operational transient conditions to ensure that the minimum core DNB ratio is greater than the design limit. Hydraulic stability is also ensured.

Nuclear peaking is controlled to prohibit excessively high peaks and the corresponding linear heat rates that could promote fuel melting.

5.3.2 Core Peaking Factors

A typical design core peaking distribution used for the thermal-hydraulic design is shown in Figure 5.3-1. This distribution was selected for reference design calculations since it produces a higher overall peaking condition and a more limiting minimum core DNB ratio than any of the distributions expected during core life. This reference distribution is applicable for all times during the fuel lifetime. Variations of power distributions are controlled so that the minimum core DNBR as calculated by the design distribution at the overpower will never be encroached during steady-state operation.

5.3.3 Thermal Hydraulic Design Characteristics

The analytical methods used to calculate the core thermal-hydraulic conditions are discussed in detail in Subsections 4.4.2.3, 4.4.2.8, and 4.4.3.4. Hydraulic calculations for the primary piping,

steam generators, and reactor vessel are discussed below. In all the calculations, the 1967 ASME Steam Tables have been used for the thermodynamic data.

a. Piping Losses:

Piping losses were calculated by standard methods using Moody's friction factors. Standard methods from the B&W calculation standards have been put on the digital computer with curves having been fit for ease and accuracy of computation.

b. Steam Generator Losses:

Steam generator losses were calculated using methods that have been verified in once-through steam generator test programs. See B&W Topical Report BAW-10027 for a discussion. Steam generator losses were calculated using CFD analysis for the breached holes which have tapered inlets as discussed in B&W Report PR-01, Rev. 1.

c. Reactor Vessel Losses:

Reactor vessel losses were calculated using methods that have been verified in vessel-model-flow-test programs. The flow tests are discussed in B&W Topical Report BAW-10037, Rev. 2.

Transient hydraulic work was analyzed using a combination of analog and digital computer programs. The analog simulation used to determine flow coastdown characteristics included descriptions of flow drop relations in the reactor coolant loops and the vendor-supplied pump zone maps to describe the pump flow characteristics as a function of speed and torque. Flow-speed, flow-torque, and flow-head relationships were solved by affinity laws. Reactor power, coolant flow, and inlet temperatures were input data to a digital program that determined the core thermal-hydraulic response.

5.3.4 RC Pump NPSH Requirements

The reactor coolant pump is not started until the pressure at the suction of the pump is at least 150 psig. This stipulation precludes operation below the minimum required NPSH for the pump. This applies to the plant under cold conditions, ambient temperature, and in no way affects the DNBR of the reactor core. This limit is established for reactor coolant pump protection.

5.3.5 BWR Power Flow Operating Map (Not Applicable)

5.3.6 Temperature Power Operating Map

Temperature-power operating maps with reactor flow and core inlet and outlet temperatures for various pump combinations as a function of load are given in Figures 5.3-2, 5.3-3, and 5.3-4. Figure 5.3-2 depicts reactor inlet, average, and exit coolant temperatures as a function of power during normal steady-state, four-pump operation. The average reactor coolant temperature of 582°F is maintained within specified tolerances by the Integrated Control System. The coolant temperatures shown in Figure 5.3-2 provide the basis for the thermal-hydraulic design analyses of Section 4.4. Specifically, at the 112% of 2772 MWt design overpower and starting with the nominal inlet and outlet temperatures of 553°F and 612°F, respectively, and a system pressure of 2200 psia, the hot channel DNB ratio is calculated assuming maximum design thermal hydraulic conditions (see Subsection 4.4.3.4.1) and is provided in the current cycle's reload

report (see Appendix 4B). Beginning in Cycle 16, at the end of a cycle (EOC), the average reactor coolant temperature, T_{AVE}, may be reduced by 12°F (less instrument error). This maneuver may result in a reactor coolant inlet temperature of approximately 547°F and will extend the Effective Full Power Life of the core through the negative Moderator Temperature Coefficient. For future cycles, the effects of the T_{AVE} reduction on the core mechanical, nuclear and thermal-hydraulic parameters as well as any potential effects on LOCA and non-LOCA analyses and/or consequences will be addressed by the cycle-specific reload report. (Reference USAR Appendix 4B)

Allowable variations from the nominal temperature and pressure at design overpower conditions are computed for each reload core. These analyses compute the combinations of reactor coolant temperatures and pressures at design overpower conditions that produce a hot channel DNB ratio equal to the critical heat flux correlation limit. The limits for these temperature-pressure combinations are given in the Safety Limit Technical Specifications. The critical heat flux correlation and corresponding limit that have been used in reload design analyses of the current cycle are provided in the reload report (Appendix 4B).

In addition to defining the thermal safety margins of the core for plant maneuverability purposes, the Technical Specification Safety Limit curve also provides the basis for setting the variable low pressure trip allowable values for the Reactor Protection System given in the Technical Specifications. The minimum coolant flowrates (4-pump and 3-pump) needed to ensure the validity of the pressure-temperature limits are provided in the Technical Specifications.

Table 15.2.5-4 provides the natural circulation flow capability as a function of the decay heat generation.

5.3.7 Load Following Capability

The NSSS is designed to respond rapidly to normal and abnormal load demands imposed by the operators, the dispatch system, or the grid. The load change capability is as follows:

Nominal Power Ramps:

	Positive ramp		Negative ramp
1.	0.5%/min between 0 and 15% power	5.	0.5%/min between 15 and 0% power
2.	5%/min between 15 and 20% power	6.	5%/rnin between 20 and 15% power
3.	5%/min between 20 and 90% power	7.	5%/min between 100 and 20% power
4.	3%/min between 90 and 100% power		

Load Following:

a. One daily power change cycle equivalent to 50% of full power at nominal ramp rates for 90% of the equilibrium core cycle.

b. Within the range from 0 to 90% of a core cycle, the reactor system is capable of a return to at least 15% power at peak xenon following a reactor trip. The power

increase from 15 to 100% power proceeds as xenon decays and is burned out by the existing power level.

Step Load Change:

a. Three cycles per day of up to 10% of full power.

b. Turbine trip without reactor trip. A reactor trip may occur following a turbine trip. See Section 7.4.1.4.

c. Full electrical load rejection without reactor trip. A reactor trip may occur on a loss of load at high power. See Section 15.2.7.3.

The transient capability is provided by the control system as described below.

5.3.7.1 Integrated Control System (ICS)

The basic requirement of the Integrated Control System (ICS) is to match the generated megawatts with the megawatt demand. The ICS does this by coordinating the flow of steam to the turbine and the rate of steam production. The flow of steam to the turbine is controlled by the throttle valves. The turbine header pressure is used as an index to determine whether the steam flow rate and the steam production rate are equal. The rate of steam production is controlled by varying the total amount of feedwater and neutron power and also by maintaining a proper ratio, between feedwater flow and neutron power, so that the proper steam conditions exist. Feedwater flow is controlled by the feedwater valves and pumps, and neutron power is controlled by the control rods in the reactor.

The ICS is designed to recognize certain conditions that are limiting on the unit. When such a condition exists, the ICS prevents operation above the limiting value by providing for a runback of the unit to a predetermined value at a rate that is a function of the existing condition. The conditions that create a load limiting situation and the corresponding limits are as follows:

Condition	Load limit % full power	Runback rate, %/min
Loss of one RC pump	75	20
Loss of two RC pumps, 1/loop	45	20
Loss of either feed pump	55*	20
Low deaerator level	55	20
High MFP discharge pressure	60	20

* It is permissible to operate the unit at a power limit up to 65 % while staying within rated speed and horsepower of the remaining main feedwater pump.

Above the low level limit, the reactor may be operated on manual or automatic control. In automatic control, the Reactor Coolant System average temperature is held within a narrow band around the setpoint.

The Reactor Coolant System temperature and the steam temperatures are shown as a function of steady-state power level in Figure 5.3-5.

Reactivity control is maintained by movable control rods (CRA), by soluble poison (boric acid) in the reactor coolant, and by burnable poison. The control rods are used to regulate power level

and to compensate for the immediate reactivity changes (Doppler) associated with a change in power level for operations above low level limit. The regulating CRA bank is allowed to vary within a specified position range which is a function of power level. A boron feed and bleed controller is provided as an integral part of the ICS.

5.3.7.2 "Feed and Bleed" Capabilities

Following a change in reactor power, the reactivity of the core changes with time as a result of variations in xenon concentration.

After a power reduction, the xenon concentration increases to a maximum value (peak xenon) and then begins to decrease to a new equilibrium value corresponding to the new power level.

Upon return to initial power, the xenon concentration decreases rapidly to a value less than the initial power equilibrium value and then slowly returns to the initial power equilibrium value.

The extent of the peak xenon poisoning, the minimum xenon concentration after return to initial power, and the time required for the xenon concentration to recover to the initial power equilibrium value vary depending on the initial, reduced, and final level and on the operating time at these power levels.

The capability of the boron control system to maintain the control rods in their nominal position (feed and bleed) depends on several factors, including the following:

1. Boron concentration at time of transient.

2. Time at which dilution is initiated.

3. Dilution rate (boron depletion).

4. Time at reduced power.

5. Feed and bleed system capacity.

For a given load change, the maximum change in xenon reactivity and the time to make the change determines the change in boron concentration and the dilution rate. The feed and bleed control system is designed to accommodate the 50% load change capability mentioned in Subsection 5.3.7.

Steady-State Power Operations:

During the first few days of core life the boron is decreased from its Beginning Of Life (BOL) concentration to compensate for the buildup of xenon and samarium poisons to their steady-state values. Until these reactor poisons reach equilibrium, the boron concentration is decreased at a higher rate than for the remainder of the cycle.

Fuel burnup causes the core's excess reactivity to deplete during the fuel cycle. A routine boron dilution is necessary to keep the control rods within their specified position range. This dilution is performed with the aid of the boron feed and bleed control station and is required about twice a week at full-power operation with equilibrium xenon and samarium.

To illustrate an ideal example, consider a 50% load rejection from equilibrium conditions at 100%: the control rods ramp in to their nominal position at 50% power, the xenon concentration begins to build up, and dilution is begun at a given rate. At first, since the xenon buildup rate is greater than the boron depletion rate, the control rods withdraw to maintain 50% power. As the xenon buildup rate decreases, the control rods insert to their nominal position. At peak xenon a demand for return to power is made. If the dilution rate is such that the core concentration is reduced sufficiently to compensate for xenon, then a return to power at a nominal ramp is possible. In the higher flux range, xenon decays rapidly, causing a corresponding increase in reactivity. Inserting the control rods to their normal 100% power position partially offsets this increase; however, to maintain the rods within their specified position range, sufficient boron is added to the core to replace that which was depleted and to compensate for xenon undershoot. After the minimum xenon concentration is reached, the dilution process returns the core to its initial boron concentration as equilibrium xenon builds in.

On reaching the time in core life when a “peak return” at nominal ramp rates is not possible, the reactor ramps to some lower power level and then slowly recovers as xenon is burned out.

5.3.8 Transient Effects

The ICS, as described in Section 7.7, maintains the power balance between the reactor system and the turbine system during steady-state and transient conditions. The transient effects of several normal and abnormal transient conditions are described below.

5.3.8.1 Increasing Power Transients

With an increase in power demand, the integrated control system initiates control rod motion, change in feedwater flow, and change in turbine governing valve position. At the same time, the RC system average temperature tends to decrease owing to the increased heat transfer from the reactor coolant to the secondary system. On a decrease in reactor coolant temperature, the control system moves rods to return the system average temperature to normal (within the deadband limits). The decrease in RC average temperature also causes an outsurge in the pressurizer resulting in a decrease in pressure. This, in turn, energizes the pressurizer heaters to maintain system operating pressure.

The extent of rod motion, decrease in pressure, and time to return to equilibrium loop conditions are functions of ramp rate and time duration or the size of step increases.

5.3.8.2 Decreasing Power Transients

Decreasing power transients also consist of both step and ramp decreases in turbine generator load. For a decreasing power transient, the reduction in steam flow and increase in reactor coolant temperature call for rod insertion. A pressurizer insurge occurs, compressing the steam and increasing system pressure. The pressure increase actuates the pressurizer spray valves which spray reactor inlet water into the steam space to condense steam and maintain the pressure at the desired value. During severe load drops the turbine bypass valves open to assist in stabilizing steam pressure. A reactor trip may occur on a loss of load at high power due to the addition of the Anticipatory Reactor Trip System (ARTS) and the raising of the Pilot Operated Relief Valve (PORV) setpoint. See Section 15.2.7.3.

5.3.8.3 Changes in Reactor Coolant Flow

Changes in coolant flow result from tripout of one or more reactor coolant pumps, a locked rotor, and startup of an idle pump. (See Chapter 15.)

5.3.8.4 Large Load Changes

Large load changes, such as turbine trip, electrical load rejection, and complete loss of station power, are described in Chapter 15.

The transient cycles for which the reactor vessel is designed are described in Subsection 5.2.1.5.

5.3.9 Thermal Hydraulic Characteristics

Table 5.2-13 summarizes the thermal-hydraulic characteristics of the RC system.

1.021	1.434	1.607	1.573	1.407	1.128	0.542	0.432
1.434	1.538	1.613	1.557	1.385	1.137	0.758	0.461
1.607	1.613	1.588	1.484	1.321	1.001	0.752	0.400
1.573	1.557	1.484	1.334	1.103	0.558	0.510	
1.407	1.385	1.321	1.103	0.895	0.643	0.364	
1.128	1.137	1.001	0.559	0.643	0.437		
0.542	0.758	0.752	0.510	0.364			
0.432	0.461	0.400					

ASSEMBLY RADIAL PEAK

(1) VALUES USED FOR THE CURRENT CYCLE ARE CONTAINED IN THE APPLICABLE RELOAD LICENSING CALCULATIONS.

DAVIS-BESSE NUCLEAR POWER STATION
TYPICAL DESIGN REACTOR FUEL ASSEMBLY POWER
DISTRIBUTION FOR 1/4 CORE (1)
FIGURE 5.3-1

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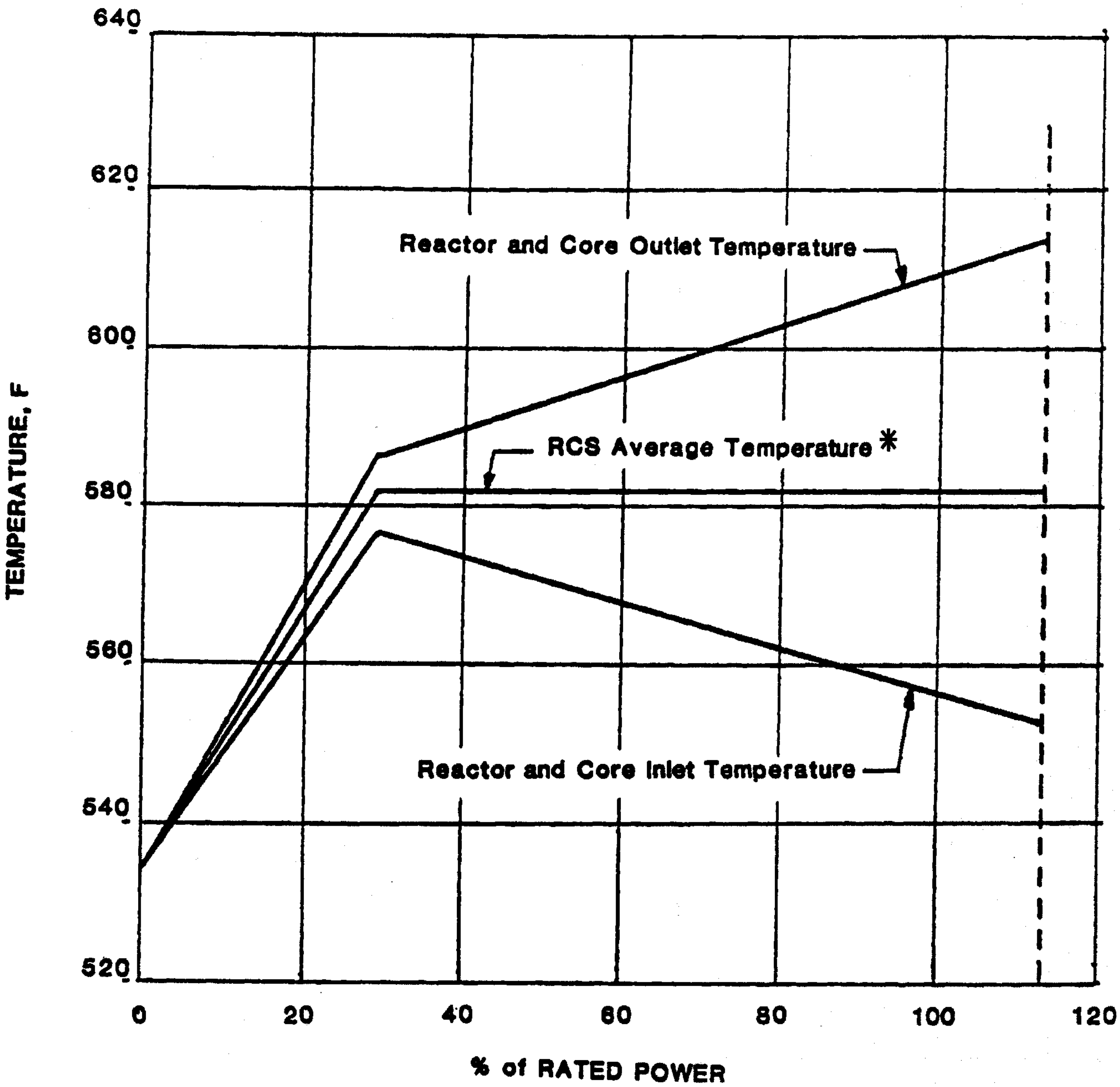
NOVEMBER 2002



* At the end of a cycle, the average reactor coolant temperature, T_{AVE}, may be reduced by 12° F (see section 5.3.6).

DAVIS-BESSE NUCLEAR POWER STATION
TEMPERATURE-POWER OPERATING MAP WITH
FOUR RC PUMPS OPERATING
FIGURE 5.3-2

REVISION 27
JUNE 2010

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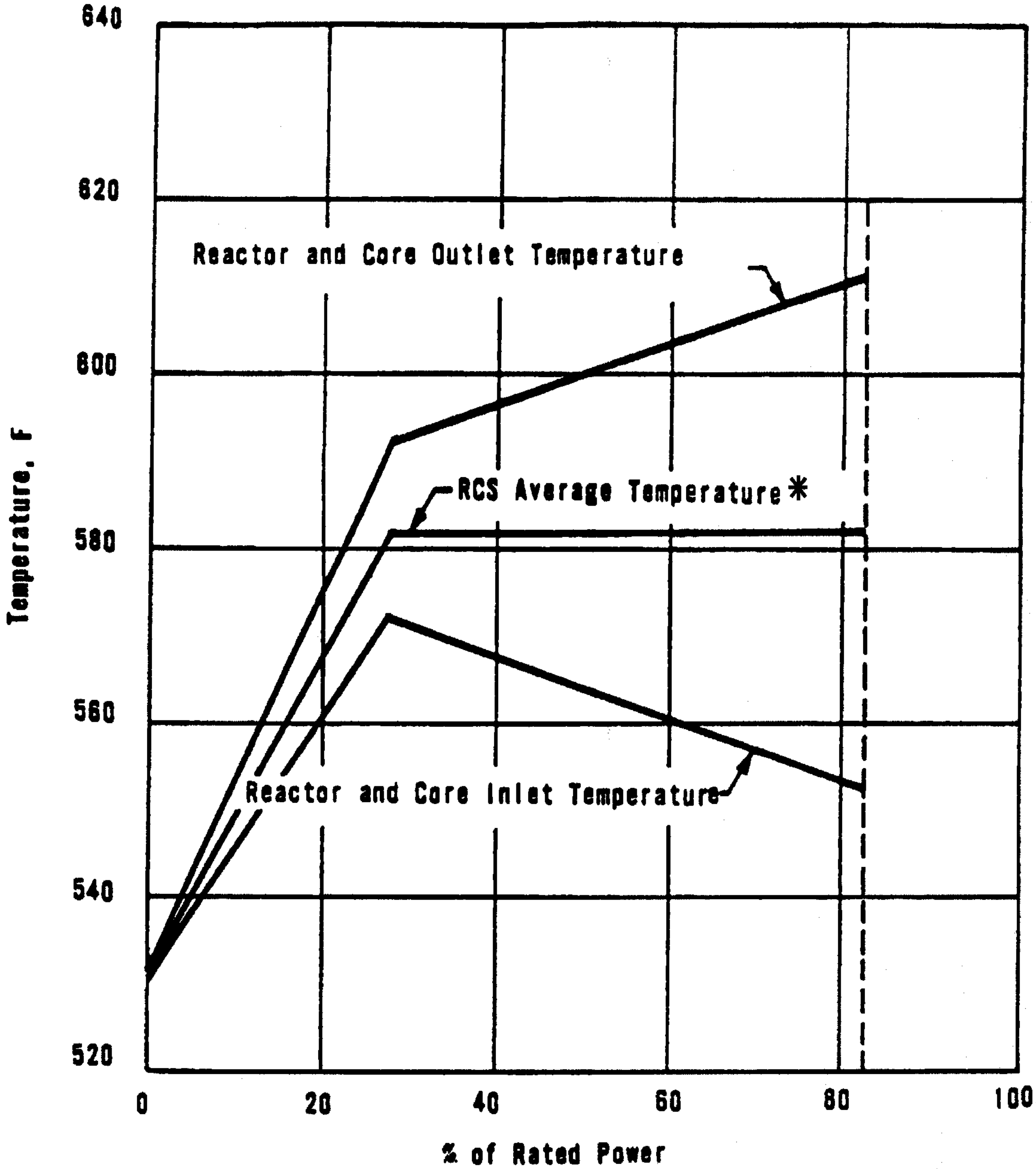
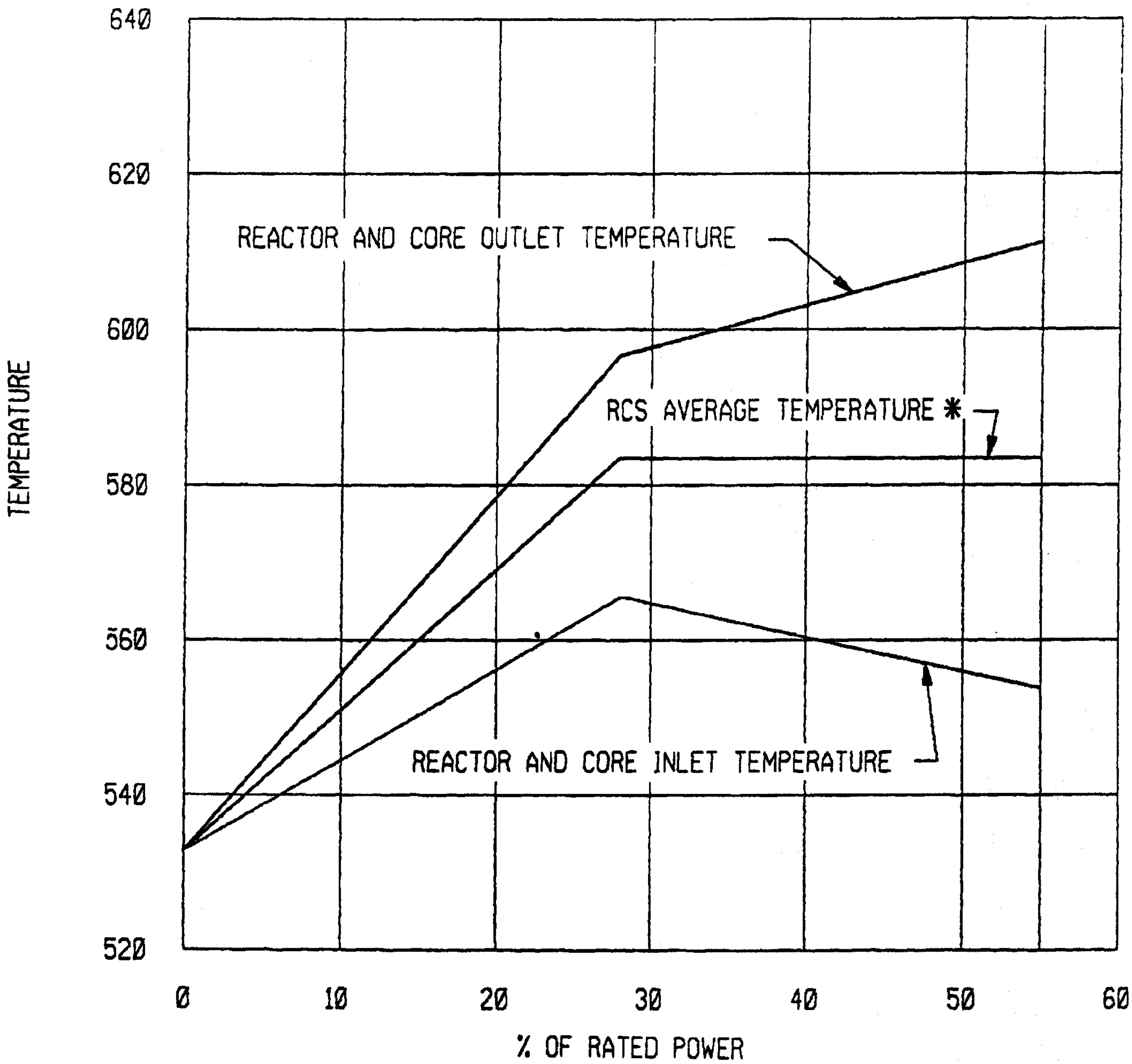


* At the end of a cycle, the average reactor coolant temperature, T_{AVE}, may be reduced by 12° F (see section 5.3.6).

DAVIS-BESSE NUCLEAR POWER STATION
TEMPERATURE-POWER OPERATING MAP WITH
THREE RC PUMPS OPERATING
FIGURE 5.3-3

REVISION 27
JUNE 2010

DB 03-12-10 DFN:/G/USAR/UFIG533.DGN/CIT



* At the end of a cycle, the average reactor coolant temperature, T_{AVE}, may be reduced by 12° F (see section 5.3.6).

DAVIS-BESSE NUCLEAR POWER STATION

TEMPERATURE-POWER OPERATING MAP

TWO RC PUMPS OPERATING

(ONE PER LOOP)

FIGURE 5.3-4

REVISION 27
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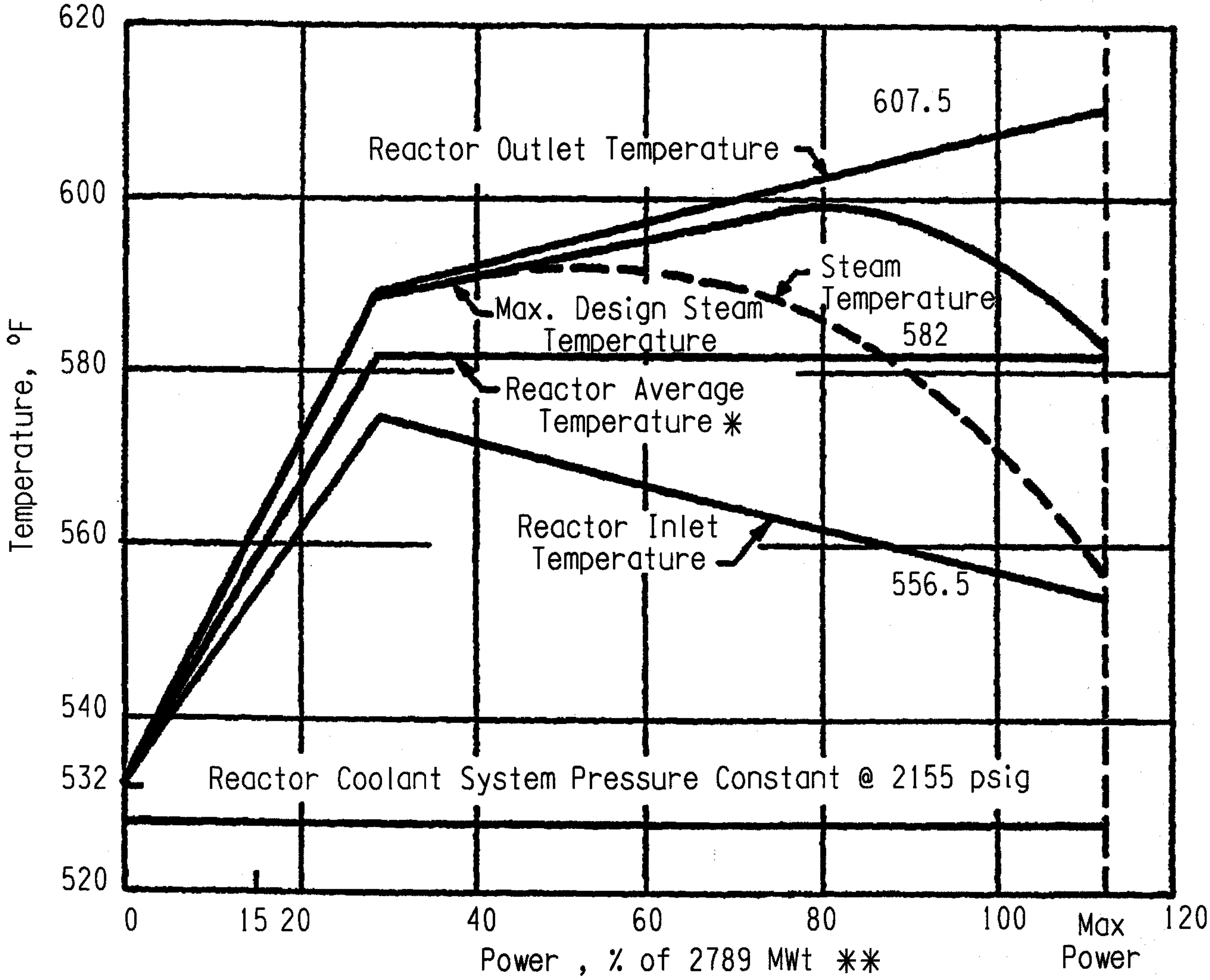
DB 03-12-10 DFN:/G/USAR/UFIG534.DGN/CIT



✱ At the end of a cycle, the average reactor coolant temperature, T_{AVE}, may be reduced by 12° F (see section 5.3.6).

✱✱ License Amendment No. 278 increased rated core thermal power from 2772 MWt to 2817 MWt, resulting in the total power transferred to the steam generators increasing from 2789 MWt to 2834 MWt.

DAVIS-BESSE NUCLEAR POWER STATION

REACTOR COOLANT AND STEAM TEMPERATURE VS LOAD

FIGURE 5.3-5

REVISION 27
JUNE 2010

DB 03-12-10 DFN:/G/USAR/UFIG535.DGN/TIF

5.4 REACTOR VESSEL AND APPURTENANCES

5.4.1 Design Bases

The reactor is designed as a Class A vessel in accordance with ASME Code, Section III, for the design conditions presented in Table 5.1-3 and for the transients listed in Table 5.1-8. The design life is 40 years for all metal parts except the closure seal gaskets.

Service pressurization limits are established to be consistent with material changes in the core region resulting from radiation effects. The assembled reactor vessel is designed to withstand earthquake(s) in accordance with the requirements stated below.

5.4.1.1 Maximum Probable (Smaller) Earthquake

The assembled reactor vessel is designed to withstand the Maximum Probable Earthquake and to maintain its functional and structural integrity as a Class I structure in accordance with these requirements:

a. The horizontal and vertical ground accelerations are considered to act simultaneously.

b. Earthquake loadings are determined by dynamic analysis using the specified maximum probable earthquake.

c. All significant modes of vibration are included in the analysis. For determination of resultant shears and moments, the modes are considered on square root of the sum of the squares basis.

d. Seismic loads are combined with other specified normal loads, and the resultant stresses comply with requirements of the ASME Code, Section III.

5.4.1.2 Maximum Possible (Larger) Earthquake

In addition to the analysis described above, a dynamic analysis is performed using response spectra of a greater intensity than used for the specified maximum probable earthquake.

The assembled reactor vessel is designed to withstand conditions as determined from the results of this analysis without loss of function and to permit a safe orderly shutdown of the station.

5.4.1.3 Simultaneous Maximum Possible Earthquake and Loss-of-Coolant Accident

The assembled reactor vessel is designed to withstand a simultaneous occurrence of the maximum possible earthquake and loss-of-coolant accident. The stresses resulting from this simultaneous loading condition may exceed the material yield strength so long as safe shutdown of the reactor is not jeopardized.

5.4.2 General Description

The reactor vessel consists of a cylindrical shell, a spherically dished bottom head, and a ring flange to which a removable reactor closure head is bolted. Supports and related structural components on the reactor vessel nozzles support the vessel and attach to structural steel,

which is embedded in the building's foundation. The spherically dished reactor closure head is manufactured with an integral ring flange which mates with and is bolted to the vessel with large-diameter studs. All internal surfaces of the vessel are clad with stainless steel weld deposit.

Two concentric metallic O-rings provide the pressure integrity seal between the closure head and the vessel's flanges. Pressure taps at the annulus between the two O-rings are used to monitor for inner o-ring seal leakage. The reactor vessel's ring flange includes an internal ledge to support the core and the internal structural components.

The vessel has two outlet nozzles through which reactor coolant is transported to the steam generators, and four inlet nozzles through which reactor coolant re-enters the reactor vessel. The smaller nozzles between the reactor coolant nozzles serve as inlets for decay heat cooling and emergency cooling water injection (core flooding and low-pressure injection engineered safety functions). The reactor coolant and the control rod drive penetrations are located above the top of the core to maintain a flooded core in the event of a rupture of a reactor coolant pipe or of a control rod drive pressure housing.

The bottom head of the vessel is penetrated by incore instrumentation nozzles. The closure head is penetrated by flanged nozzles which provide for attaching the control rod drive mechanisms. The closure head is also penetrated by the continuous vent line nozzle, which attaches the continuous vent line piping to the closure head.

Guide lugs welded to the reactor vessel's inside wall limit a vertical drop of the reactor internals and core to 1/2 inch or less and prevent rotation failure. These lugs provide shock support for the internals and control the motion of the lower end of the core support assembly while under the influence of horizontal seismic loads.

The reactor vessel shell material is protected from fast neutron flux and gamma heating effects by a series of water annuli and stainless steel thermal shields between the core and the vessel's wall.

Metal reflective insulation is used on the exterior of the vessel from the closure flange down to and including the exterior of the bottom head dome. Removable metal reflective insulation panels enclose the top head closure flange and studs. Metal reflective insulation is used on the closure head dome.

Primary coolant system interconnecting piping is carbon steel with internal stainless steel cladding except for the 2-1/2" reactor head continuous vent piping which is stainless steel. There are no isolation or check valves to introduce pressure transients from valve operation.

The incore instrument penetrations are joined by field-welds to pipes that terminate in bolted sealing flanges located in a shielded area at a higher elevation in the CV. These tubes contain incore detector assemblies for measuring neutron flux and temperatures in the reactor core. The incore detector assemblies are subject to the reactor coolant pressure.

The reactor vessel is protected from overpressure by the Reactor Protection System high-pressure trip and pressurizer ASME code safety valves located on the pressurizer's top head.

5.4.3 Design Evaluation

The reactor vessel was designed, fabricated, and erected in accordance with proven and recognized design codes and quality standards applicable for the specific function or classification. It is designed for a pressure of 2500 psig at a temperature of 650°F. The corresponding operating pressure of 2155 psig allows an adequate margin for normal load changes and operating transients. The vessel is designed to meet the codes listed in Table 5.2-1.

5.4.3.1 Pressure Vessel Safety

The integrity of the reactor vessel depends on four major factors: (1) design and stress analysis, (2) quality control, (3) proper operation, and (4) additional safety factors.

The reactor pressure vessel is designed to the requirements of the ASME Code, Section III. A stress analysis of the entire vessel is conducted under both steady-state and transient operations. The result is a complete evaluation of both primary and secondary stresses and the fatigue life of the entire vessel.

A detailed design and analysis of every part of the vessel is prepared as follows:

a. The size and configuration of the vessel are set to meet the process requirements, the thickness requirements for pressure and other structural dead and live loads, and the special fillet contour and transition taper requirements at nozzles, etc., as required by ASME Section III.

b. The pressure and temperature design transients for the vessel (Table 5.1-8) are used in determining the pressure loading and temperature gradient and their variations with time throughout the vessel. The resultant combinations of pressure loading and thermal stresses are calculated. Computer programs are used in this development.

c. The stresses through the vessel are evaluated using the allowable stresses given in ASME Section III. This code gives safe stress level limits for all types of applied stress. These are membrane stress (to ensure adequate tensile strength of the vessel), membrane plus primary bending stress (to ensure a distortion-free vessel), secondary stress (to ensure a vessel that does not progressively deform under cyclic loading), and peak stresses (to ensure a vessel of maximum fatigue life).

   A design report is prepared and submitted to the authorized inspector and enforcement authorities. This report defines in detail the design basis, loading conditions, etc., and summarizes the conclusions to permit independent checking by interested parties.

d. The core flood nozzle shown in Figure 4.2-5 contains a flow restrictor. The restrictor and attachment weld are designed in accordance with ASME Section III. The significant transients which affect the restrictor and weld are Reactor Coolant System heatup and cooldown, including the core flooding system periodic test transient and decay heat removal initiation. All transients are considered as normal operating conditions and are considered in determining thermal stresses and the fatigue usage factor. The fatigue analysis includes a strength reduction factor of two on the weld per ASME Section III. The weld has been designed to

withstand a differential pressure of up to 2250 psi which may occur because of a core flooding line LOCA. A dynamic magnification factor of two was applied to the pressure to account for instantaneous application. Based on these criteria, the average shear stress in the weld yields a safety margin of 1.4. These assumptions and safety margin are sufficient to ensure the structural integrity of the nozzle, restrictor, and weld for all operating and faulted conditions. During the core flooding transient, the maximum Δp across the nozzle is expected to be approximately 200 psi. This is a factor of greater than 11 less than the design loading assumptions. During operation of the Decay Heat System, the Δp loads on the restrictor are insignificant.

5.4.4 Reactor Vessel Tests

The reactor vessel was designed, manufactured, and tested in accordance with Section III of the ASME Code and was subjected to non-destructive testing as listed below and in Table 5.2-14.

a. All pressure boundary plates were ultrasonically inspected using both normal and shear wave techniques.

b. All pressure boundary forgings were ultrasonically inspected.

c. Carbon and low-alloy base material was magnetic-particle inspected after quenching and tempering.

d. All accessible welds were inspected by either magnetic particle or liquid penetrant methods after hydrostatic testing.

e. The external surface of the entire vessel, including weld seams, is magnetic-particle inspected after final shop hydrotesting.

f. All cladding is inspected by the liquid-penetrant method.

g. All cladding is ultrasonically tested for bond after stress relief at the highest stress relief temperature to be used.

h. After fabrication, welds are inspected by radiography, liquid-penetrant, or magnetic-particle methods.

i. Stud forgings are inspected for flaws by two ultrasonic inspections axial and radial. In addition to the ultrasonic tests, magnetic-particle inspection is performed on the finished studs.

5.4.5 Heatup and Cooldown Rates

The maximum normal heatup rate from cold shutdown to normal operating temperature is 50°F/hr. The maximum cooldown rate is 100°F/hr from normal operating temperature to 270°F and 50°F/hr from less than 270°F to cold shutdown. These rates are valid until the end of the operating license. These rates apply for the entire RC system, including the reactor vessel. With this maximum normal limit on the system, the reactor vessel thermal loadings are well within the vessel's design limits.

For a rapid depressurization of the RCS (which may be desirable for isolating a steam generator tube leak), the RCS cooldown rate limit is 235°F/hr from normal operating temperature to 500°F. The maximum allowable emergency cooldown rate for the RC system from normal operating temperature to a cold shutdown temperature is 100°F/hr.

5.4.6 In-Place Annealing

The design of the reactor pressure vessel has been reviewed to determine the feasibility of in-place annealing. The results indicate that there is nothing that precludes in-place annealing.

5.4.7 Material Surveillance Program

The Davis-Besse 1 Reactor Vessel Material Surveillance Program (RVSP) meets the requirements of Appendix H to 10CFR50. The materials selected for surveillance monitoring are those identified as being the materials which first control the operating limitations of the reactor vessel during its service life. The RVSP materials are listed in Table 5.2-15. The selection procedure for these materials is described in B&W Topical Report BAW-10100A.

There are a total of six (6) specimen capsules provided in the Davis-Besse 1 surveillance program. The type and number of specimens and the frequencies for removing and testing these specimens is as described for the modified program in B&W Owners Group Report, BAW-1543A. Five of the specimen capsules were irradiated beginning with initial operation, while the remaining capsule was inserted in the Surveillance Specimen Holder Tube (SSHT) following the third Davis-Besse 1 cycle. The withdrawal and insertion schedule is based on the latest approved supplement to B&W Topical Report BAW 1543A.

AREVA Topical Report BAW-2241 P-A (Ref. 30) describes an analytical method of predicting fast fluence in the pressure vessel. The fluence is used to calculate the reference nil-ductility temperature (RT_{NDT}) as a function of fast neutron fluence (E>1MeV). This information will affect operating limitations with respect to normal heatup and cooldown procedures for reactor vessel. Requirements for these pressure-temperature limits are included in the Technical Specifications and the Pressure and Temperature Limits Report.

As described in BAW-1875A, the B&W Owner Group cavity dosimetry program, neutron fluence will be determined by dosimetry located in the cavity between the reactor vessel and the concrete shield. After an initial comparison between the internal specimen capsules and the external dosimetry, the external dosimetry may be used to determine the remaining vessel life when all internal specimen capsules have been removed.

The method of neutron fluence (E>1MeV) calculation used to determine the surveillance specimen withdrawal schedule and current predictions of reactor vessel irradiation is described in Subsection 4.3.2.10. As discussed in Subsection 4.2.2.2, Subsection 6, a total of six SSHT's are installed. These SSHT locations have two lead factors depending on their azimuthal location with respect to reactor vessel axis (see Figure 4.2-5) which is either 11° or 26.5°. The two 26.5° locations were chosen as being in the lowest available azimuthal flux location to minimize the flux lead factor available for specimen capsules. The original flux lead factor calculations using the methods and model described in B&W Topical Report BAW-10100A showed the flux lead factor to be 2.3 from the specimen capsule to the location of highest azimuthal flux on the reactor vessel inner wall for the 26.5° location and 4.1 for the 11° location. The updated fluence model used for the surveillance specimen withdrawal schedule shows the lead factors for these SSHT locations to be 3.9 and 5.4, respectively, from the specimen capsule to the reactor vessel inner wall location of highest flux. This difference from the original

calculations is attributed to the calculational model changes, particularly in the inclusion of the capsule purturbation effect, the use of realistic thermal conditions and the use of an updated azimuthal flux variation. The actual inservice lead factor for the 26.5° SSHT location will vary depending on the actual azimuthal peaks for the particular core cycle design. Plant specific fluence analyses, accounting for the actual core design, were performed when the Davis-Besse 1 surveillance capsules were tested.

See the Pressure and Temperature Limits Report for current status of the specimen capsules.

5.4.8 Special Fabricating Processes

The cladding process is the only special process used for the vessel.

Intergranular separations (underclad cracking) in low alloy steel heat-affected zones under austenitic stainless steel weld cladding were detected in SA-508, Class 2 reactor vessel forgings manufactured to a coarse grain practice and clad by high-heat-input submerged-arc welding processes. As described in Table 5.2-15 of the USAR, some of the Davis-Besse RPV materials are ASME SA 508, Class 2 low alloy steel. This includes the forgings for the upper shell flange, nozzles (except instrumentation, CRDM nozzles and the closure head continuous vent line nozzle), nozzle belt, upper & lower shells, and transition piece (Dutchman).

Babcock & Wilcox (B&W) conducted an intensive investigation of underclad cracking in the 1970s, consisting of testing and analysis. Results of the investigation showed the subject flaws are present only in A-508, Class 2, forgings manufactured to a coarse grain practice and clad by high-heat-input submerged-arc welding processes. The maximum discontinuity depth observed throughout the industry (0.156 in.) was used in the fracture mechanics analysis summarized in Topical Report BAW-10013-A, “Study of Integranular Separations in Low-Alloy Steel Heat-Affected Zones under Austenitic Stainless Steel Weld Cladding,” October 1972. The flaw growth analysis was performed for a 40-year cyclic loading, and a 40-year end-of-life fluence value of 3×10^{19} n/cm^2 (E> 1.0MeV) was used to determine fracture toughness properties. The results of the fracture mechanics analysis demonstrated that the critical crack size required to initiate fast fracture is several orders of magnitude greater than the assumed maximum flaw size plus predicted growth due to design fatigue cycles.

Topical Report BAW-10013 was reviewed and approved by the staff. In the SER, dated October 11, 1972, the staff concurred with the B&W’s findings that the integrity of a vessel having flaws such as described in the subject report would not be compromised during the life of the plant.

5.4.9 Special Design and Fabrication Features

No special design and fabrication features are incorporated in the vessel to further improve its reliability and reduce its potential for failure.

5.4.10 Reactor Vessel Fabricator

The reactor vessel was fabricated by Babcock & Wilcox. The replacement reactor vessel closure head was fabricated by AREVA (formerly B&W). The B&W quality assurance program is described in FSAR Chapter 17.

5.4.11 Lifetime Design Transients

Lifetime design transients for the reactor vessel are presented in Table 5.1-8.

5.4.12 Vessel Materials and Fabrication Inspection

The materials of construction for the reactor vessel are given in Table 5.2-10, and the inspections conducted during fabrication are presented in Table 5.2-14.

5.5 COMPONENT AND SUBSYSTEM DESIGN

5.5.1 Reactor Coolant Pumps

5.5.1.1 Design Bases

The RC pump provides movement of coolant between the reactor vessel and the steam generator. The pump is designed in accordance with the ASME Code, Section III as applicable.

Included in the design considerations are all facets of station operation for a 40-year lifetime, which includes pre-operational testing, normal operation, abnormal operation, and accident conditions.

5.5.1.2 Description

The RC pumps are single stage, single suction, constant speed, vertical centrifugal pumps. The pump's sealing system consists of three mechanical seal assemblies arranged in a removable cartridge and a top vapor barrier standpipe to prevent reactor coolant fluid leakage to the atmosphere. The pump is shown in Figure 5.5-1, and principal design parameters are listed in Table 5.1-6. The performance characteristics of the pump are shown in Figure 5.5-2.

The pump casing consists of a bottom suction inlet passage, which delivers the reactor coolant to the pump impeller, a multivane diffuser and a collecting scroll which directs fluid out through a horizontal discharge nozzle. A water-lubricated, self-aligning radial hydrostatic bearing is located in the pump casing just above the main impeller. The pump casing is welded into the piping system, and the pump internals can be removed for inspection or maintenance without removing the casing from the piping.

Each pump has a separate, single-speed, top-mounted electric drive motor, which is connected to the pump by a removable shaft coupling. A driver mount is used as the transition piece between the motor and the pump, which, in conjunction with the removable shaft coupling, allows replacement of pump seals without removal of the drive motor.

The pump stuffing box is a unit containing a thermal barrier, recirculation impeller, shaft-seal heat exchanger, removable mechanical seal cartridge, and a top vapor barrier standpipe. Shaft sealing is accomplished with a mechanical seal system containing three face-type mechanical seals in series. Water for lubrication and cooling the seals is injected below the first mechanical seal, at a flow of 8 to 10 gpm into the pump. Most of the injection water flow passes into the pump casing through a close-running, spiral-grooved restriction bushing, and the remainder flows upward through the three mechanical seals. Three pressure-breakdown devices carry a small leakage flow in parallel with the face-type mechanical seals; they are designed to equalize the pressure drops so that each seal takes only approximately one-third of the total pressure. Leakage flow from the three pressure-breakdown devices leaves the pump through the controlled bleed off line to join the return to the injection water supply system. The standpipe provides a collection chamber for any leakage from the upper (third) mechanical seal and a connection for orderly disposal of the leakage. An RCP seal leakage flow detector provides external indication of this leakage.

Component cooling water is also furnished to the pump as a backup to the seal injection water system. The component cooling water flows to a heat exchanger mounted integrally with the pump. The heat exchanger is sized to provide enough cooling capacity to prevent excessive heating of the mechanical seals in the event that seal injection water is lost. The upper drain

chamber, which features an overflow drain to a closed disposal system, further prevents leakage to the CV if the performance of the mechanical seal should experience excessive deterioration.

The RC pump motors are vertical, squirrel-cage induction machines. Each motor has a two-piece bolted flywheel to increase the rotational energy, thus prolonging pump coastdown and ensuring a more gradual loss of main coolant flow to the core in the event that pump power is lost. The flywheel assembly is at the upper end of the rotor and includes an anti-reverse device to prevent back-rotation, thereby reducing starting time and rotor heating.

The motors are totally enclosed with air-to-water heat exchangers to provide a closed-circuit cooling air flow through the motor. Radial bearings are of the pivoted pad type and of segmented or split construction. The thrust bearing is a double-acting Kingsbury type of self-equalizing pivoted pad construction designed to carry the full thrust of the pump. A high-pressure oil system with separate pumps is provided with each motor to establish an oil film on the thrust bearings before starting. Once started, the bearings carry their own oil film, and the motor provides its own oil circulation through the oil cooler.

To eliminate the potential hazard resulting from a possible oil spill from the RCP motor lube oil system, piping and component enclosures, and upper and lower oil drip pans have been installed for each RCP motor to direct the spilled oil to a drain tank.

The upper oil pan is installed to collect the oil spill from the RCP motor’s oil cooler, oil lift pumps, and upper bearing oil fill and drain piping. To prevent oil splash due to the pipe rupture or gasket failure in the high pressure portion of the oil system, i.e., the oil cooler and lift pumps, a shield wall and cover have been provided in this area. Enclosures are installed over high pressure piping and components located outside this shield wall and the upper bearing oil level control assembly.

The lower oil drip pan is of conical type, installed along the motor shaft to contain any oil spill downward along the motor shaft and the oil collected from the lower bearing oil fill and drain connections.

Enclosures are also installed around the lower bearing oil level control assembly with a drain path to the lower oil drip pan.

Both oil drip pans have a drain connection piped to one of two drain tanks located at El.-565 feet - 0 inches. No valves will be installed on the drain line. Each drain tank serves two reactor coolant pumps.

Since the oil drip pans and the drain tanks are open to the containment atmosphere, they are not designed for any internal pressure. A minimum of welding has been used in the joint areas so that some of the plates are removable to provide access for maintenance when required. The drain tanks are seismically braced.

5.5.1.3 Evaluation

The total stress resulting from thermal expansion, pressure and mechanical and seismic loadings is considered in the design of the RC pumps. The total stresses expected in the pumps are within the maximum allowed by the ASME Code, Section III.

5.5.2 Steam Generators

5.5.2.1 Design Basis

The steam generators (SG) remove heat generated by the reactor by producing superheated steam in the secondary side. Table 5.1-5 lists the design basis values used in the design. The steam generators are designed in accordance with ASME III for Class A vessels.

Also, the design considers any structural effect as a result of vibration, whether induced by operational or environmental conditions. The design consideration includes all facets of station operation for a 40-year lifetime which includes preoperational testing, normal operation, abnormal operation, hydrostatic tests, and accident conditions.

Design changes to provide the dual level setpoint control (in lieu of a single setpoint when auxiliary feedwater is required) have been made to maintain adequate decay heat removal and pressurizer level indication during anticipated events and to maintain consistency with the small break LOCA analysis (as described in Chapter 6).

SG level is but one factor that contributes to maintaining pressurizer level on scale during anticipated events. Auxiliary feedwater addition rates and secondary steam pressure control also effect pressurizer level. The dual level setpoint control provides an integrated system to coordinate steam generator level and auxiliary feedwater addition rates. Measures to provide proper equipment operation which affect secondary pressure have been taken. However, equipment malfunctions cannot be prevented with absolute certainty. If pressurizer level indication is lost (for any reason) during anticipated events, this occurrence is considered an operational inconvenience and not a safety problem. The operator can rely on RC system pressure to assure that a level of water in the pressurizer is maintained.

The "auto-essential" steam generator level control includes a dual setpoint. Following automatic actuation of auxiliary feedwater by the Steam and Feedwater Rupture Control System (SFRCS), steam generators level will be controlled to greater than 35 inches (indicated on the Startup Range) above the lower tube sheet if no SFAS Level 2 actuation occurs. For accident conditions where both auxiliary feedwater and SFAS Level 2 are automatically actuated (indicative of loss of coolant accident conditions), the auto-essential level control will regulate water addition of the steam generator to achieve and maintain an actual level above the lower tube sheet of at least 120 inches.

5.5.2.2 Description

The general arrangement of the steam generators is shown in Figure 5.5-3. The steam generator is a vertical, straight-tube-and-shell heat exchanger which produces superheated steam at approximately a constant pressure over the power range. Reactor coolant flows downward through the tubes, and steam is generated on the shell side. The high-pressure parts of the unit are the hemispherical heads, the tubesheets, and the straight Inconell tubes between the tubesheets. Tube supports hold the tubes in a uniform pattern along their length.

[1]Inconel, a trade name of an alloy manufactured by the International Nickel Company, also has substantial common usage as a generic description of a Ni-Fe-Cr alloy conforming to ASME Specification SB-163. It is in the latter context that reference is made here.

The shell, the outside of the tubes, and the tubesheets form the boundaries of the steam-producing section of the vessel. Within the shell, the tube bundle is surrounded by a baffle, which is divided into two sections. The upper part of the annulus between the shell and baffle is the superheater outlet, and the lower part is the feedwater inlet-heating zone. Vents, drains, instrumentation nozzles, and access ports (manways, handholes, and inspection openings) are provided on the shell side of the unit. The reactor coolant side has access ports (manways and inspection openings). The replacement OTSG has a flat lower primary head hence it does not require a drain nozzle for the bottom head. The reactor coolant side of the unit is vented by a vent connection on the reactor coolant inlet pipe to each unit. The unit is supported by a base support platform and a base support stool attached to the bottom head. The base support platform rests on a sliding support and provides the required freedom of movement to accommodate thermal expansion of the RC system.

Reactor coolant water enters the steam generator at the upper plenum, flows down the Inconel tubes while transferring heat to the secondary shell-side fluid, and leaves through the lower plenum. Figure 5.5-4 shows the flow paths and steam generator heating regions.

Four heat-transfer regions exist in the steam generator as feedwater is converted to superheated steam. Starting with the feedwater inlet, these are Feedwater Heating, Nucleate Boiling, Film Boiling, and Superheated Steam.

a. Feedwater Heating:

   Feedwater is heated to saturation temperature by direct contact heat exchange. The feedwater entering the unit is sprayed into a feed/heating annulus (downcomer) formed by the shell and the baffle around the tube bundle. The steam that heats the feedwater to saturation is drawn into the downcomer through the aspirating port by condensing action of the relatively cold feedwater. At 100% power, the aspirating steam is approximately 15% of the total main steam line flow. The downcomer provides the last stage of feedwater preheating as the feedwater is heated to saturation temperature corresponding to the steam generator pressure in the downcomer. Some additional heating is also supplied by conductive heat transfer through the lower baffle. At the bottom of the tube bundle, a zone of esseptially saturated liquid exists.

   The momentum of the downward directed feedwater stream and the gravity head of the liquid in the downcomer provide the driving head for the steam generator. This head in the downcomer balances the gravity head of the boiling mixture in the tube bundle and the frictional losses in: (1) the downcomer (primarily the orifice plate), (2) the tube bundle (primarily the tube support plates), and (3) the aspirator port.

b. Nucleate Boiling:

   Saturated water enters the tube bundle, and the steam-water mixture flows upward on the outside of the Inconel tubes counter-current to the reactor coolant flow. The vapor content of the mixture increases almost uniformly until Departure from Nucleate Boiling (DNB) is reached, and then film boiling and superheating occur. The quality at which transition from nucleate boiling to film boiling occurs is a function of pressure, heat flux, and mass velocity. The length of the boiling zone varies depending on the power level of the reactor and the thermal-hydraulic conditions in the region. Since the length of the boiling zone varies, the length of

the superheating zone also varies. This affects the amount of superheat added to the steam before it leaves the steam generator.

c. Film Boiling:

Dry saturated steam is produced in the film boiling region at the upper portions of the tube bundle.

d. Superheated Steam:

Saturated steam is raised to final temperature in the superheater region.

Figure 5.5-5 is a plot of heating surface versus load. Note that an approximately constant minimum level is held below approximately 28% load. The amount of surface (of length) of the nucleate boiling section and the film boiling section is proportional to load. The surface available for superheating varies inversely with load; i.e., as load decreases the superheater section gains from the nucleate and film boiling regions. Mass inventory in the steam generator increases with load as the length of the heat-transfer regions varies.

Various "levels" are measured in the steam generator. These level measurements are actually differential pressure (dP) measurements across different physical regions of the steam generator. These dPs have contributions from the mass of water and steam and the flow inducted frictional losses between the level taps. The dP contribution from flow inducted frictional losses are due primarily to losses at the orifice plate, the tube support plates, and the tube surfaces. This flow inducted dP varies with the square of the velocity of the fluid in the steam generator, which varies with the plant's power level.

The various levels measure are Operate Range, Startup Range, and Full Range.

a. Operate Range

The operate range has a lower tap approximately 103.8" above the lower tubesheet upper surface in the downcomer (above the orifice plate). The upper tap is in the tube region just above the aspirator port at approximately 395.8" above the lower tubesheet upper surface. The operate range measures the dP between the taps and converts this to a percentage of the dP which would exist between the taps if the entire space was filled with saturated water at the temperature being measured in the lower downcomer. Therefore, the operate range is generally interpreted as a percentage by volume of the water in the downcomer above the lower tap.

b. Startup Range

The startup range has a lower tap approximately 7.8" above the lower tubesheet upper surface. The upper tap is the same tap used by the operate range. The startup range indicates the head of the water and steam mass and the frictional losses (primarily at the tube support plates) in the tube region below the aspirator port (boiling region).

c. Full Range

The full range lower tap is the same tap used by the startup range. The upper tap is at the upper tubesheet lower surface. The full range is generally used when placing the steam generator in wet lay-up.

Above approximately 28% load, steam generator level is allowed to rise as required to maintain reactor coolant average temperature constant. However, steam generator levels for a given power level are dependent on the amount of fouling in the steam generator. Steam generator level limits are defined in the Technical Specifications.

Reactor coolant and steam temperatures versus reactor power are shown in Figure 5.5-6. As shown, both turbine header pressure and average reactor coolant temperature are held constant over the load range from approximately 28 to 100% of rated power. Constant turbine header pressure is obtained by a variable two-phase boiling length (see Figure 5.5-5) and by the regulation of feed flow to obtain proper secondary mass inventory for the steam generator. In addition to average reactor coolant temperature, reactor coolant flow is also constant. The difference between reactor coolant inlet and outlet temperatures increases proportionately as load is increased. Steam generator saturation pressure and temperature are approximately constant, resulting in a variable outlet steam temperature. Beginning in Cycle 16, at the end of a cycle (EOC), the average reactor coolant temperature, T_{AVE}, may be reduced by 12°F (less instrument error). This maneuver may result in a reactor coolant inlet temperature of approximately 547°F and will extend the Effective Full Power Life of the core through the negative Moderator Temperature Coefficient. For future cycles, the effects of the T_{AVE} reduction on the core mechanical, nuclear and thermal-hydraulic parameters as well as any potential effects on LOCA and non-LOCA analyses and/or consequences will be addressed by the cycle-specific reload report. (Reference USAR Appendix 4B)

Figure 5.5-7, a plot of temperature versus tube length, shows the temperature differences between shell and tube throughout the steam generator at rated load. The excellent heat-transfer coefficients permit the use of a secondary operating pressure and temperature that are close enough to the reactor coolant average temperature to permit a straight-tube design.

The shell temperature is controlled by direct-contact steam that heats the feedwater to saturation, and the shell is bathed with saturated water from the feedwater inlet to the lower tubesheet.

In the superheater section, the tube wall temperature approaches the reactor coolant fluid temperature since the steam film heat-transfer coefficient is considerably lower than the reactor coolant heat-transfer coefficient. By baffle arrangement in the superheater section, the shell section is bathed with superheated steam above the steam outlet nozzle, further reducing temperature differentials between the tubes and the shell.

Tables 9.3-4 and 9.3-5 provide specifications for steam generator water quality.

5.5.2.3 Evaluation

The steam generator is a straight tube-and-shell design. The tubes are expanded (to a partial depth) into the tubesheet. The tubes are seal welded to the tube-sheet near the tube ends. There are currently no repair methods qualified in the Steam Generator program for the replacement OTSG. Any tubes found to be defective must be plugged. Figure 5.5-3 is a model of this type of generator. The principal design parameters are given in Table 5.1-5. The steam

generator is designed and manufactured in accordance with the Class 1 requirements of Section III, ASME Boiler and Pressure Vessel Code as given in Table 5.2-1.

For consideration of the interactions between the steam generators and the Steam and Feedwater Rupture Control System (SFRCS) see USAR Section 7.4.1.3.

The natural circulation test at Davis-Besse 1 (TP800.04) demonstrated that a 35-inch (indicated) steam generator level of AFW provides adequate natural circulation for decay heat removal.

Transitions from solid natural circulation to reflux boiling and back to solid natural circulation may cause slug flow in the hot leg piping. The loads imposed on the tubes of the Steam Generator (SG) during the postulated “slug flow” have been conservatively evaluated and found to be acceptable. Based on very conservative assumptions, the end loading on each tube will be 21.5 lbf compared to a theoretical buckling load of about 476.3 lbf.

It was assumed for this analysis that a water level has been established in the hot leg piping and inside the tubes of SG. The transient consists of a “front” of solid water impinging on the primary face of the upper tubesheet. The flow was assumed to be equal to full 100% power flow (about 70,000,000 lb/hr). The load is assumed to be a suddenly - applied load. The upper tubesheet is conservatively assumed to offer no resistance to the load and the lower tubesheet is assumed to be fixed so that the entire load is absorbed by tubes directly under the primary inlet nozzle. The flow is assumed to not follow the diffuser so that the velocity impinging on the tubesheet is the same as the velocity in the 36-inch nozzle. Hot leg temperature is assumed to be 605°F.

The velocity in the 36-inch pipe would be 64.4 ft/sec. By use of the momentum equation, the steady-state force on the upper tubesheet due to the velocity would be 16,080 lb_f. Assuming a suddenly-applied load, the momentary force would be 32,160 lb_f. There are about 1500 tubes in a 36-inch diameter circle. Thus the 32,160 lb_f will result in 21.5 lb_f per tube. Since the cross-sectional area of each tube is 0.070 in, the momentary axial compressive stress in these tubes would be 307 psi.

The research and development program for the once-through steam generator employing direct-contact heat exchange feedwater heating has been completed. The results of this work and of vibration testing on steam generator tubes are reported in B&W Topical Report BAW-10027, “Once-Through steam Generator Research and Development Report.” Most of this work is directly applicable to the steam generator design of Davis-Besse. Portions of the results pertaining to tube loads during transients are superseded by the results contained in reference 20. The reference 20 analysis also describes tubesheet tube hole dilation that occurs during transients.

The replacement OTSG Flow Induced Vibration Report Reference 37 shows that through the design life, the replacement OTSGs will be capable of operation at all specified conditions with freedom from flow-induced or turbulence-induced vibrations which result in significant tube degradation. For the replacement OTSG results pertaining to tube loads during transients are contained in Reference 33 and results concerning tube hole dilation are contained in Reference 38. The maximum diametrical expansion of the tubesheet hole is 0.0009336”.

The design includes the effects of differences in the coefficient of thermal expansion for the tubes (Inconel) and the shell (SA-508 Gr. 3 C1. 2). During normal operation, the mean temperature of the tube is greater than that of the shell. The steam generator has been evaluated for all operating conditions, including a loss of reactor coolant and a rupture of a

steam outlet line. The tubes have been checked for gross yielding (tubes colder than shell, resulting in tension stresses) and for critical buckling (tubes hotter than shell, resulting in compression stresses). All of these conditions have been evaluated, and it has been shown that the design meets the criteria of Section III, ASME Boiler and Pressure Vessel Code as given in Table 5.2-1.

In the unlikely event that reactor coolant is lost, the primary side will drop to near atmospheric pressure, the secondary side will maintain operating pressure (1050 psi), and the tubes will cool faster than the secondary shell. The effect of this condition on the tubesheet was evaluated, and the results indicate that the criteria of Section III are met. The effect of the temperature difference between the secondary shell and the tubes was checked along with the pressure conditions. The tubes meet the criterion of gross yielding and show that the primary-to-secondary boundary is maintained. The results of actual pressure tests on 5/8-inch-OD, 0.034-inch-wall Inconel tubing justify the design pressure condition of 1050 psi on the external surface of the tubes. Collapse under external pressure was shown to occur at 4950 psig. This provides a significant factor of safety against collapse under the 1050 psi accident conditions for loss of reactor coolant. It is concluded that the primary-to-secondary boundary (tubes and tubesheet) can maintain its integrity during a LOCA. The results of this work are reported in BAW-10027.

Following the rupture of a secondary pipe, the secondary side will drop to atmospheric pressure, and, assuming that the primary side maintains pressure, the secondary shell will cool faster than the tubes. The effect of this condition on the tubesheet was evaluated, and the results indicate that the criteria of Section III are met. The effect of the temperature difference between the secondary shell and the tubes was checked along with the pressure conditions. The results show that the tubes meet the criterion of critical buckling even though some of the tubes undergo insignificant permanent deformation. The use of tube support plates throughout the secondary side provides the tubes additional strength against buckling. The basic design criterion of Section III for the tubes is a pressure differential of 2500 psi applied to the inside diameter of the tubes at design temperature. Therefore, a rupture of the secondary side would not significantly increase the pressure stresses in the tubes beyond that normally considered in their design. The effect of fluid dynamic forces on the steam generator's internals under this condition has been simulated by a 37-tube laboratory boiler. The results show that the integrity of the reactor coolant boundary is maintained under the most severe mode of secondary blowdown. It can be concluded that the primary-to-secondary boundary (tubes and tubesheet) can maintain its integrity in the event of a secondary pipe rupture.

The effect of maximum possible (large) earthquake conditions plus the primary and secondary ruptures has been considered to ascertain whether the pressure boundary can maintain its integrity. It was found that maximum possible earthquake conditions contribute insignificant loadings as compared to the rupture loadings. Therefore, it can be stated that the integrity of the primary and secondary boundary can be maintained under the conditions of simultaneous maximum possible earthquake plus primary and secondary rupture.

Report BAW-10027 describes a simulated primary pipe rupture test performed on a model steam generator. The test demonstrated that a rupture of the reactor coolant pressure boundary does not induce failure of the tubes in any respect. This was verified by subsequent hydro-test of the unit:

a. The steam generator tubes are sized for the reactor coolant side pressures and temperatures acting alone using methods specified in ASME Section III, Article NB-3000. The design pressure is 2500 psig and the design temperature is 650°F.

Other design data are specified in Tables 5.1-5 and 5.2-10. The supporting effect of secondary side pressure is not considered in sizing the tubes. The tubes are designed for the system transients specified in Table 5.1-8. The applicable design stress intensity limits associated with Cases 1 through III of Subsection 5.2.1 are as follows:

Ni-Cr-Fe SB-163

$S_{ultimate}$ = 80,000 psi

S_{yield} = 35,000 psi at 100°F

S_m = 23,300 psi

Case	Stress Limit	Limit Value
I	$P_m \leq 1.0\ S_m$	23,300 psi
	P_1 plus $P_b \leq 1.5\ S_m$	35,000 psi
	P_1 plus P_b plus P_e plus $Q \leq 3.0\ S_m$	70,000 psi
II	$P_m \leq 2/3\ S_u$	42,200 psi at 630°F
	P_1 plus $P_b \leq 2/3\ S_u$	42,200 psi at 630°F
III	$P_m \leq 2/3\ S_u$	42,200 psi at 630°F
	P_1 plus $P_b \leq 2/3\ S_u$	42,200 psi at 630°F

The design conditions are conservative values of the maximum expected pressure, temperatures and operating transients.

For the replacement OTSG the Reference 32 and 33 analysis shows the structural integrity of the primary-to-secondary boundary (tubes and tubesheet) is maintained during design, normal, upset, emergency and faulted conditions, such as LOCA and secondary pipe rupture. The actual pressure tests for the Alloy 600 tubing as documented in the BAW-10027 report are bounding for the Alloy 690 tubing because the ultimate stress and the yield stress for Alloy 690 tubing is higher than for Alloy 600 tubing.

b. The margin for tube wall thinning that can be tolerated to prevent collapse of the tube during the postulated condition of a break in the reactor coolant pressure boundary is 0.014 inch. A tube uniformly reduced in wall thickness by 0.014 inch could withstand the secondary side pressure of 1050 psig without exceeding the stress limits identified in Case III. The tube wall margin available was calculated from data based on collapse tests of representative tube samples. The minimum thickness of virgin tubewall is 0.034 inch. Uniform, over all thinning of the tube wall is highly unlikely - localized pits or cavities are more probable. In the presence of corrosion pits or cavities on the tube's external surface, wall thicknesses as little as 0.0002 and 0.009 inch for cavities 0.010 and 0.100 inch in diameter, respectively, can withstand the external pressure conditions existing during a reactor coolant pipe break. These conditions contribute to tube leaks from the primary to secondary side before a significant number of tubes can develop general thinning greater than 0.014 inch. Tube leaks are detected by increased secondary system activity. Leaking tubes are then located and plugged per the Steam Generator Program. The replacement OTSG has Alloy 690 tubing, which has pitting

resistance equal to or better than that of the Alloy 600 tubing used in the original OTSG.

Collapse pressure tests on representative tube specimens proved that the mode of collapse was ductile without subsequent leakage. The minimum collapse pressure on a virgin tube sample was 4950 psig at a temperature of 650°F. The specimens were hydrotested with an internal pressure of 3600 psig after collapse, and no leaks were detected. Other tube samples were dye-penetrant examined after collapse, and none was found to have cracks. The collapse tests for the Alloy 600 tubing are bounding for the Alloy 690 tubing because the ultimate stress and the yield stress for Alloy 690 tubing is higher than for Alloy 600 tubing.

c. For the replacement OTSG, excess material of 0.003 inch over the design tube minimum wall thickness is available in the tube wall.

Steam generator tubes that are found to be leaking may be plugged. There are currently no approved tube repair methods listed in the Steam Generator Program.

Tube fouling tests, also described in BAW-10027, were conducted with the addition of insoluble corrosion products, iron and nickel oxide, and the soluble materials, sodium hydroxide, chloride, and sulfate to evaluate once-through steam generator (OTSG) tube fouling on performance and determine the steam solubility of several commonly encountered water soluble materials. The results of these tube fouling tests showed that the insoluble corrosion product oxides would deposit on OTSG surfaces. However, local deposition of these solids will not occur due to the flow characteristics in the OTSG and the free flow of water and steam through the broached tube support plates. These deposits will not, of themselves, lead to tube deterioration from wastage because of their general nature and will not cause intergranular attack without some corrosive agent, such as sodium hydroxide. The solubility of sodium hydroxide in superheated steam at OTSG operating conditions was found to be far in excess of the maximum total solids limits specified for the feedwater. Thus, caustic will not accumulate or concentrate in the OTSG during operation.

The removal of corrosion product oxides from the OTSG surfaces is accomplished by a fill-soak and drain technique using a solvent (ammoniated EDTA at 200°F) that is aggressive to corrosion product oxides rather than to the materials of construction. This effect was evaluated using Ni-Cr-Fe alloy tubing at velocities to 15 fps with impingement on the tube samples and a temperature of 200°F. In addition, a mixture of suspended corrosion products (Cr_2O_2 + NiO + Fe_3O_4) was added to the solution to investigate its abrasiveness when introducing the solvent. The penetration rate observed per an eight hour period ranged from 0.0003 to 0.002 mils.

One of the requirements placed on the solvent during its development was that it be thermally decomposable and that the decomposition products would have no adverse effects on the tube material, especially in the area of the lower tubesheet crevice. To confirm that the solvent developed met this requirement, one of the test boilers tubed with the Ni-Cr-Fe alloy and having the same lower tubesheet configuration as a full size OTSG was intentionally fouled with corrosion product oxides. The boiler was then chemically cleaned with the subject solvent which returned the boiler to its pre-fouled performance condition. Destructive examination of the boiler did not show any adverse effects of the solvent on materials of construction. The solvent was found to decompose to CO_2, NH_3 and water, and there is no accelerated long term corrosion if small amounts of the residual solvent remained in the OTSG on restart after cleaning.

5.5.2.4 Tests and Inspections

All pressure boundary materials and associated fabrication procedures meet the requirements of the specified code. Table 5.2-14 lists the inspections to be performed on pressure boundary material.

5.5.2.5 Radiation Effect

The expected level of secondary side radiation is discussed in Chapter 11.

5.5.3 Reactor Coolant Piping

5.5.3.1 Design Basis

The reactor coolant piping provides a flow between the reactor vessel and the steam generators for removal of heat from the reactor vessel. Included in the piping run are reactor coolant pumps which circulate the reactor coolant fluid. Design data is presented in Table 5.1-7. The reactor coolant piping is designed in accordance with ANSI B.31.7 for Class I piping. Replacement Hot Leg Upper piping spool pieces are designed in accordance with ASME B&PV Code, Section III Class 1, 2001 Edition, 2003 Addenda.

The design also considers any structural effect as a result of vibration, whether induced by operational or environmental conditions. The design consideration includes all facets of station operation for 40-year lifetime, which includes preoperational testing, normal operation, abnormal operation, and accident conditions.

The requirements for design, fabrication, and testing of shop fabricated reactor coolant piping assemblies and associated equipment are described in the Purchase Order Technical Specification (Document No. 7749-M-509-1-2) for reactor coolant piping.

For the replacement RCS Hot Leg upper piping spool pieces the requirements for design, fabrication and testing are described in the Certified Design Specification TS-4890 Rev.1

5.5.3.2 Description

The general arrangement of the reactor coolant piping is shown in Figures 5.1-5 and 5.1-6. Materials are listed in Table 5.2-10.

The RC piping connects the components of the reactor coolant system and contains pressurized water at the system pressure during normal operating conditions. Borated water heated in the reactor vessel is transferred through the reactor outlet piping to the steam generator primary inlet. The coolant is returned to the reactor through the reactor inlet piping via the reactor coolant pumps. Included as part of the RC piping is the surge line, which transmits reactor coolant between the pressurizer and reactor outlet piping. Also included as a part of the RC piping is the spray line, which connects the reactor inlet piping and the pressurizer. It transports reactor coolant for use in pressure control by emitting a spray for depressurization during insurges to the pressurizer.

In addition to the pressurizer's surge and spray piping connection, the piping has welded connections for pressure taps, temperature elements, vents, drains, decay heat removal, and emergency core cooling high-pressure injection water. A thermal sleeve provided in the high-

pressure injection connection to the reactor coolant inlet piping satisfies the thermal transient design analysis, which indicates that such protection is required.

Design, stress analyses, quality control, and operating limits for the reactor coolant piping provide a level of system integrity equivalent to that provided for the pressure vessels of the RC system.

All interior surfaces of the interconnecting piping are clad with stainless steel to eliminate corrosion problems and to reduce contamination of the coolant.

5.5.3.3 Evaluation

The total stresses resulting from thermal expansion, pressure, and mechanical and seismic loadings are considered in the design of the RC piping. The total stresses expected in the original piping are within the maximum allowed by ANSI Code, B31.7 for Nuclear Power Piping; the total stresses expected for the Replacement Hot Leg sub-assemblies are within the requirements of ASME B&PV Code, Section III Class 1, 2001 Edition, 2003 Addenda. Connections have thermal sleeves, when required, to limit stresses from thermal shock to acceptable values.

Revised stress analyses have been performed for the Pressurizer Surge Line (including nozzles) to account for additional stress resulting from thermal stratification and striping in the surge line. The revised analyses were performed in accordance with the requirements of ASME Section III, 1986 Edition. Topical Report BAW 2127 (Reference 14), Final Submittal for Nuclear Regulatory Commission Bulletin 88-11 “Pressurizer Surge Line Thermal Stratification” with Supplements 2 (Reference 15) and 3 (Reference 16) describe the analysis method and demonstrate compliance with the Code and regulatory requirements. The NRC has reviewed and approved the methodology used to perform this revised stress analysis (Reference 17).

MPR Corporation performed a scoping type of review to evaluate the validity of the concrete walls and a “belt-like” component used for hot leg support at DB 1. This review was limited to the design and analysis of support structure components and did not cover actual fabrication. Based on a check of pertinent drawings, the “belt-like” component is a LOCA pipe whip restraint for the hot leg, located at the first elbow downstream from the reactor vessel. Accordingly, the scoping type of review covered this restraint and, for completeness, each of the other hot and cold leg LOCA pipe whip restraints, their embedments, and the associated concrete walls. The review did not cover the steam generator supports, which were the subject of a previous review. Specifically, the following areas were reviewed:

a. Design drawings of the reactor coolant piping restraints, their embedments, and the steam generator compartment walls.

b. Design basis LOCA dynamic loads applied to the piping restraints.

c. Design analyses of the restraints, embedments and walls.

Based on this review, it is determined that the design of the reactor coolant piping restraints, their embedments, and the associated steam generator walls is adequate for the dynamic loads experienced during a LOCA.

5.5.3.4 Tests and Inspections

All pressure boundary materials and associated fabrication procedures meet the requirements of the specified code. Table 5.2-14 lists the inspections to be performed on pressure boundary material.

Inspection of all normally accessible safety-related piping, including both redundant trains, was completed September 21, 1979 in response to NRC IE Bulletin 79-14, dated July 2, 1979, Revision 1, dated July 18, 1979, Supplement 1, dated August 15, 1979, and Supplement 2, dated September 7, 1979, regarding the inspection program to verify that the DB-1 seismic analysis input for safety related piping systems conforms to the actual field configuration. Preliminary evaluation of the discrepancies discovered by the inspection has been completed and the results indicated that none of these discrepancies adversely affect system operability as described in the Serial No. 1-93 letter.

5.5.3.5 Additional Considerations

Associated with the deposition of stainless steel is the requirement for considering stress-corrosion cracking. This cracking is primarily a function of stress, environment, and the material involved. During design of the system, materials that are exposed to the reactor coolant were selected for compatibility with the reactor coolant under system design conditions. The corrosion resistance of these materials is maintained by proper design, fabrication, and operations practice. The stress levels encountered by the materials during the life of the system are reduced through the application of proper design and fabrication techniques. In addition, restrictive water chemistry requirements and cleaning methods provide a controlled environment for RC system materials. This combination of proper material selection, reduction of stresses, and strict control of the environment provides assurance of freedom from stress-corrosion cracking.

The design of supporting structures, restraints, insulation, and in-service inspection techniques considers the location of each RC system weld requiring in-service inspection. Pressure boundary welds are shown on applicable drawings and are confirmed on as-built drawings to ensure that the locations of the welds are identified.

5.5.4 Main Steam Line Flow Restrictors

Main steam line flow restrictors are not applicable to the Davis-Besse Unit No. 1 design.

5.5.5 Main Steam Line Isolation System

Instrumentation is provided on the main steam lines to detect a rupture and to initiate closure of the turbine stop valves and main steam isolation valves (see Figure 10.3-1). A non-return valve on each steam line limits blowdown of the unaffected steam generator before the turbine stop valves close if the rupture is in the Auxiliary Building.

5.5.6 Reactor Core Isolation System

Not applicable to B&W NSSS.

5.5.7 Decay Heat Removal System

The design bases, description, evaluation, tests and inspections, and leakage (radiological) considerations for the Decay Heat Removal System are discussed in Subsection 9.3.5.

5.5.8 Reactor Coolant Makeup and Purification System

The design bases, description, evaluation, tests and inspections, and leakage (radiological) considerations for RC Makeup and Purification system are discussed in Subsection 9.3.4.

5.5.9 Main Steam Line and Feedwater Piping

The design of the main steam lines and feedwater piping is discussed in Section 10.3 and Subsection 10.4.7.

5.5.10 Pressurizer

5.5.10.1 Design Basis

The pressurizer provides a capability for maintaining the RC system at a pressure to prevent boiling of the reactor coolant. The pressurizer is designed in accordance with ASME III for Class A vessels. Design data are presented in Table 5.1-4.

The design also considers any structural effect as a result of vibration, whether induced by operational or environmental conditions. The design consideration includes all facets of station operation for a 40-year lifetime which includes preoperational testing, normal operation, and accident conditions.

5.5.10.2 Description

The pressurizer is a vertical-cylindrical vessel that is connected to the reactor outlet piping by the surge piping. The general arrangement is shown in Figure 5.5-8. The electrically heated pressurizer establishes and maintains RC pressure within prescribed limits and provides a surge chamber and a water reserve to accommodate changes in reactor coolant volume during operation. The water volume is based on the capability of the system to experience a reactor trip and not uncover the low-level sensors in the lower shell and to maintain the pressure high enough to preclude activation of the HP injection system. The steam volume also provides this capability. The steam volume is based on the capability of the system to experience a turbine trip and not cover the level sensors in the upper shell. The vessel is protected from thermal effects by a thermal sleeve in the surge line connection and by a distribution baffle on the surge pipe inside the vessel.

Two ASME Code relief valves are connected to the pressurizer to relieve system overpressure. An additional pilot-operated relief valve limits the lifting frequency of the code relief valves. Each of two ASME code safety valves discharge steam into a separate tee opening directly into the containment vessel. A rupture disk is installed between each code safety outlet flange and the discharge tee in order to limit the spread of contamination in containment due to code safety leakage. The pilot-operated relief valve (PORV) discharges to the pressurizer quench tank within the containment vessel.

The Reactor Coolant System (RCS) High Point Vent System described below provides vents on each of the two hot legs and on the pressurizer to vent non-condensable gases from the high

points of the RCS to assure that core cooling during natural circulation will not be inhibited. Each vent line will be controlled by two valves from the control room using individual hand-switches and each valve has positive position indication in the control room. The two hot leg vent lines have restrictive orifices sized such that the flow rate does not exceed the RCS makeup system capability. The vent on the pressurizer is also designed and sized such that the inadvertent opening of both valves could not cause the RCS to depressurize when all pressurizer heaters are energized. The two hot leg vents are routed to the containment atmosphere in an unobstructed area. The pressurizer vent is routed to the quench tank. The vent on each hot leg is controlled by two solenoid operated valves. The valves are Nuclear Class 1 and are seismically and environmentally qualified to the criteria of the DB-1 FSAR. The valves are powered from Class 1E power supplies. The two valves on Loop 1 line receive Channel 1 AC and Channel 1 DC power, respectively. The two valves on Loop 2 receive Channel 2 AC and Channel 2 DC power, respectively. The AC valves utilize control penetrations, while the DC valves utilize power penetrations through the containment vessel. The valves fail closed on loss of power.

Valve HV-239A and valve HV-200 are powered from a Channel 2 Class 1E supply. To utilize the high point vent on the pressurizer, these valves would be opened from the control room.

All taps for the new vent lines and all associated valving and instrumentation will be located above the maximum credible water level in the containment vessel and will be protected against damage from adjacent systems. Redundancy of one reactor coolant loop vent is provided by the other hot leg vent. The primary vent path for the pressurizer is the installed Pilot Operated Relief Valve (PORV). The additional pressurizer vent described here will provide a back-up capability for that path.

Replaceable electric heater bundles in the lower section and a water spray nozzle in the upper section maintain the steam and water at the saturation temperature corresponding to the desired RC system pressure. During outsurges, as the pressure in the reactor decreases, some of the water in the pressurizer flashes to steam to maintain pressure. Electric heaters are actuated to restore the normal operating pressure. During loss of offsite power events, pressurizer heaters can be used to facilitate natural circulation. The current design for emergency power supply provides manual loading of sufficient pressurizer heater capacity, needed to compensate for pressurizer ambient heat loss, to each emergency diesel generator to meet the requirements of subsection 2.1.1 of NUREG-0578. This permits additional time to regain offsite power and reduces the need for alternative means of establishing natural circulation or challenges to Emergency Core Cooling Systems. During insurges, as pressure in the reactor system increases, a water spray from a reactor inlet line condenses steam and thus reduces pressure. Spray flow and heaters are controlled by the pressurizer’s pressure controller. The pressurizer spray nozzle contains a thermal sleeve which protects it from cold water.

There are three redundant differential pressure type level transmitters to monitor the pressurizer level at the Davis-Besse Nuclear Power Station Unit 1 (DB-1) during insurge/outsurge operation.

Two of these three level transmitters are safety grade (Class 1E) transmitters. Each of these two essential transmitters is supplied by redundant essential 1E power supplies. These two essential transmitters feed two level indicators each (one per channel on the main control board located in the control room and one per channel on the Auxiliary Shutdown Panel). In addition to the above essential equipment, another level transmitter is fed by a non-essential uninterruptible power supply through a non-essential inverter.

The output of the above three transmitters goes through a selector switch; any one of the three can be selected through the operation of this switch. This selected output is temperature-compensated, and the resultant compensated level signal is sent to the pressurizer level recorder and an indicating controller.

The pressurizer level indication to the operator is further supported by three redundant computer points fed by the above three transmitters.

It should be emphasized that in the event of flooding of the containment vessel caused by a loss-of- coolant accident, the availability of these three level transmitters is not affected by water level. This is due to the raised loop configuration at Davis-Besse Unit 1 that locates the lowest pressurizer level transmitter more than 15 feet above the conservatively calculated containment maximum water level.

To eliminate abnormal buildup or dilution of boric acid within the pressurizer and to minimize cooldown of the coolant in the spray and surge lines, a bypass flow is provided around the pressurizer spray control valve. This continuously circulates 0.75-3.0 gpm (Reference 21 and 22) of reactor coolant from the heat transport loop. A sampling connection to the liquid volume of the pressurizer is provided for determining the boric acid concentration. A steam space sampling line provides a capability for sampling and/or venting accumulated gases.

During cooldown and after the decay heat removal system is placed in service, the pressurizer can be cooled by circulating water through a connection from the discharge of a decay heat removal cooler to the pressurizer spray line. When the Reactor Coolant System pressure has been decreased sufficiently, the pressurizer steam bubble is replaced with a nitrogen bubble.

5.5.10.3 Evaluation

The total stresses resulting from thermal expansion, pressure, and mechanical and seismic loadings are considered in the design of the pressurizer. The total stresses expected in the pressurizer are within the maximum allowed by the ASME III Code. Connections have thermal sleeves, when required, to limit stresses from thermal shock to acceptable values.

5.5.10.4 Test and Inspections

All pressure boundary materials and associated fabrication procedures meet the requirements of the specified code. Table 5.2-14 lists the inspections to be performed on pressure boundary material.

The RCS high point vents and pressurizer vent are tested and maintained in accordance with procedures and the Technical Requirements Manual (TRM). The TRM is incorporated by reference into the USAR.

5.5.11 Pressurizer Quench Tank and Cooler

Steam released from the pressurizer pilot-operated relief valve (PORV) is discharged into the pressurizer quench tank. The tank is partially filled with water, and the vapor phase is a predominantly nitrogen atmosphere at or near the containment vessel ambient condition. Steam is discharged under the water level to condense and cool by mixing with the water through sparger nozzles. Following the steam release, the tank's contents are cooled to normal temperature by recirculating the water through the pressurizer quench tank, recirculating pump,

and quench tank cooler. The level of the tank is controlled by draining through the quench tank cooler to the reactor coolant drain tank in case of high level or by filling the tank from the demineralized water storage tank in case of low level.

The pressurizer quench tank was originally sized with a 10% margin based on the maximum requirement to condense and cool a discharge of pressurizer steam from two Code Safety Valves and the PORV at full power.

The quench tank is protected against overpressure by the pressure relief valve RC207 (setpoint 90 psig) and a rupture disc designed to rupture at a pressure of 100 psig. In addition, the capability exists for venting the tank to the gaseous radwaste system by opening valve RC222 upon high pressure alarm PAH P710 annunciation (set at 70 psig). A deflector shield was added to the pressurizer quench tank rupture disc flange in order to divert any steam flow away from the wiring adjacent to the quench tank. The pressurizer quench tank and cooler are shown on Figure 5.1-2.

The pressurizer quench tank (T003) is constructed of stainless steel throughout. Design features of the tank are as follows:

Number	1
Type	Vertical, cylindrical
Material	SS
Volume (nominal), gal	6700
Design temperature, °F	350
Design pressure, psig	100
Sparger assembly	
Capacity, lb/hr	805,000
Design pressure, psig	700
Design temperature, °F	500
Rupture disc capacity, lb/hr at 100 psi	805,000

Design features of the Pressurizer Quench Tank Cooler (E036) are as follows:

Quantity	1
Type	horizontal "U" tube
Performance Heat Load	cools Quench Tank contents from 280°F to 120°F within four hours
Batch Volume	600 ft^3
Batch Flowrate	100 gpm (tube side)
CCW Flowrate	50 gpm (shell side)
CCW Inlet Temp	95°F
Design Press/Temp	150 psig/350°F (tube side) 150 psig/200°F (shell side)

Codes & Standards	ASME III Class 3	(tube side)
	ASME VIII	(shell side)
	TEMA C	
	Seismic Class II	
Materials	304 SS	tube side
	CS	shell side

5.5.12 Reactor Coolant Drain Tank and Pumps

The Reactor Coolant Drain Tank (RCDT) T014 serves as a receiver for effluent obtained from various drains and code relief valves as depicted in USAR Fig 5.1-2.

The RCDT and RCDT Pump are shown on USAR Figure 5.1-2.

The radioactive liquid contents of the RCDT are pumped to the Clean Liquid Radwaste System for further processing. The RCDT functions outside the RCS pressure boundary.

The RCDT pumps operate automatically, in sequential order, depending on the level present within the RCDT. Non-condensable gases or gases displaced as the RCDT is being filled are diverted automatically to the Gaseous Radwaste System for processing. Flooding of the gas header is prevented by a level controlled isolation valve.

The RCDT is protected against overpressure by three passive operating relief devices. The first two, pressure relief valve PSV-2555 and rupture disc PSE-3959 discharge to the Clean Waste Receiver Tanks (T015-1 and T015-2). The third relief device is rupture disc PSE 2440 which discharges to atmosphere in the RCDT room. Rupture disc PSE-2440 also provides vacuum protection for the RCDT. Nitrogen is supplied as needed by N_2 regulator PCV 1776 to maintain pressure in the RCDT.

Level indication and level alarms are provided to the control room by level indicator LI 1721 and the plant computer respectively.

The RCDT is constructed of stainless steel throughout. Design features of the tank are as follows:

Quantity	1
Type	Vertical Cylindrical
Volume (nominal), gal	690
Design Pressure, psig	50
Design Temperature, °F	300
Material	304SS
Design Code	ASME III, Class C

Design features of the RCDT pumps P46-1 and P46-2 are as follows:

Equipment Number	P46-1	P46-2
Type	Centrifugal	Centrifugal, Canned Rotor
Rated Capacity, gpm	70	70
Design Temperature, °F	200	200
Material	316 SS	316 SS
Design Code	Mfg. Std.	ASME III, Class 3

5.5.13 Valves

All line valves in the RCPB are weld-end valves except for the root valve for the electrically actuated relief valve on the pressurizer, which is flanged. The flanged valve has ANSI B16.5, standard raised faced flanges.

Applicable codes, stress criteria and classification for RCPB valves are noted in Section 5.2.

5.5.14 Safety and Relief Valves

The pressurizer safety valves are bellows sealed, balanced, spring-loaded safety valves. The valve is provided with a supplementary back-pressure balancing piston to counteract the maximum back pressure effects arising in the event of bellows failure.

The pilot-operated relief valve (PORV) on the pressurizer is an electrically controlled, pilot-operated, pressure loaded relief valve.

Modifications have been made to assure the reduction of the likelihood of automatic actuation of the pressurizer PORV during anticipated transients. At Davis-Besse Unit 1, the high reactor coolant pressure trip is set at ≤2355 psig. The PORV opening setpoint was raised to 2450 psig. The PORV utilizes magnetic reed switches located in the pilot valve solenoid assembly to provide the operator better status of the position of the PORV.

5.5.15 Component Supports

5.5.15.1 Design Basis

The component supports maintain the position of the component so that allowable stresses and/or deflections for the component and attached piping are not exceeded because of deadweight, thermal expansion, seismic forces, and piping ruptures. Per the criteria in USAR Section 3.6.2, dynamic affects from a postulated pipe rupture in the RCS can be excluded. However, pipe rupture of other piping (i.e., Main Steam) must be included. The design requirements and stress limits for the imposed loadings are given in Subsection 3.8.2.3.4.

Supports for Nuclear Class 1 piping and valves are designed as indicated in Subsection 3.9.2.3.

Structural steel supports for the reactor vessel, steam generators, reactor coolant pumps, and pressurizer are designed as indicated in Subsection 3.8.2.3.4.

Loading combinations and stress limits for the reactor coolant pump snubbers are as follows:

Loading combinations: Pipe rupture and SSE
Stress limits: 0.9 of material yield stress

5.5.15.2 Description

Reactor Vessel:

The reactor vessel is supported by four pads which are integrally forged on the reactor coolant inlet nozzles. Each support pad bears on a support shoe which rests on the vessel support structure. The support shoe is a structural member that transmits the support loads to the supporting structure, the primary shield. The supports restrain seismic and dead weight lateral,

vertical and rotational movement of the reactor vessel and still allow thermal growth by permitting radial sliding at each support. LOCA restraint rings are mounted in the shield wall.

Steam Generators:

The steam generator is supported and/or restrained at three locations to take lateral and vertical loads imposed by dead weight, thermal expansion, earthquake, and pipe rupture. The steam generator rests on a sliding support containing Lubrite plates. The sliding support is attached to a base support platform and conical support. The bottom plate of the sliding support is fastened to the foundation by anchor bolts. Unrestrained lateral and rotational motion in any direction is permitted by this support. The support and its attachments resist vertical loads. Lateral support is provided at the upper tubesheet. Four trunnions, bolted to the upper tubesheet, fit into a structural frame which extends to the secondary shield wall. Vertical and radial thermal expansion is unrestrained. The support allows rotation about an axis which is perpendicular to an axis drawn through the steam generator and the reactor vessel. The support resists lateral loads applied in any direction and resists torsional effects of load application. Lateral seismic and pipe rupture loads are shared by the upper and lower lateral supports.

Additional lateral support is provided for the steam generator through the conical support stool and a base support platform. The support stool is bolted to the top plate of the base support platform. The additional lateral support consists of four guides attached to the base support platform’s exterior. The four guides are then reacted by the building bumpers. These guides are designed to resist lateral shear and torsional effects of the load application.

Pressurizer:

The pressurizer is supported from lugs attached to the vessel’s wall and located symmetrically around the circumference of the vessel near the operating center of gravity. The lugs are bolted to a structural steel frame which is attached to the secondary shield wall. The supports allow vertical and radial thermal expansion and resist lateral, vertical, and rotational movement resulting from seismic and pipe rupture loadings.

Primary Piping:

Restraints are located along the primary piping to prevent a failure of the reactor coolant pressure boundary from leading to a failure of the secondary (steam-feedwater) pressure boundary or from reducing the minimum performance capabilities of engineered safety features below those specified in the safety analysis. The restraints are functionally designed to preclude impairment of the normal operation of the station.

Pumps and Motors:

Each pump and motor assembly is supported vertically by four spring supports and the primary cold leg piping. The spring supports are attached to the motor support stand and to a structural steel frame extending from the floor and the shield wall.

Two hydraulic snubbers provide seismic restraint for each assembly. These horizontally mounted snubbers are attached to the motor near the top and to brackets or frames extending from concrete walls.

Restraints at the top and bottom of the motor support stand limit lateral LOCA movements. However, based on the criteria in USAR Section 3.6.2.2.1, the dynamic effects from a

postulated pipe rupture in the RCS can be excluded and these whip restraints are no longer required to be installed. The structural steel frame to which the spring supports are attached limit vertical LOCA movements.

5.5.15.3 Evaluation

The dead weight, thermal expansion, seismic, and LOCA loadings imposed on the supports from the components were considered in the design of the supports. The stress limits are given in Section 3.8.

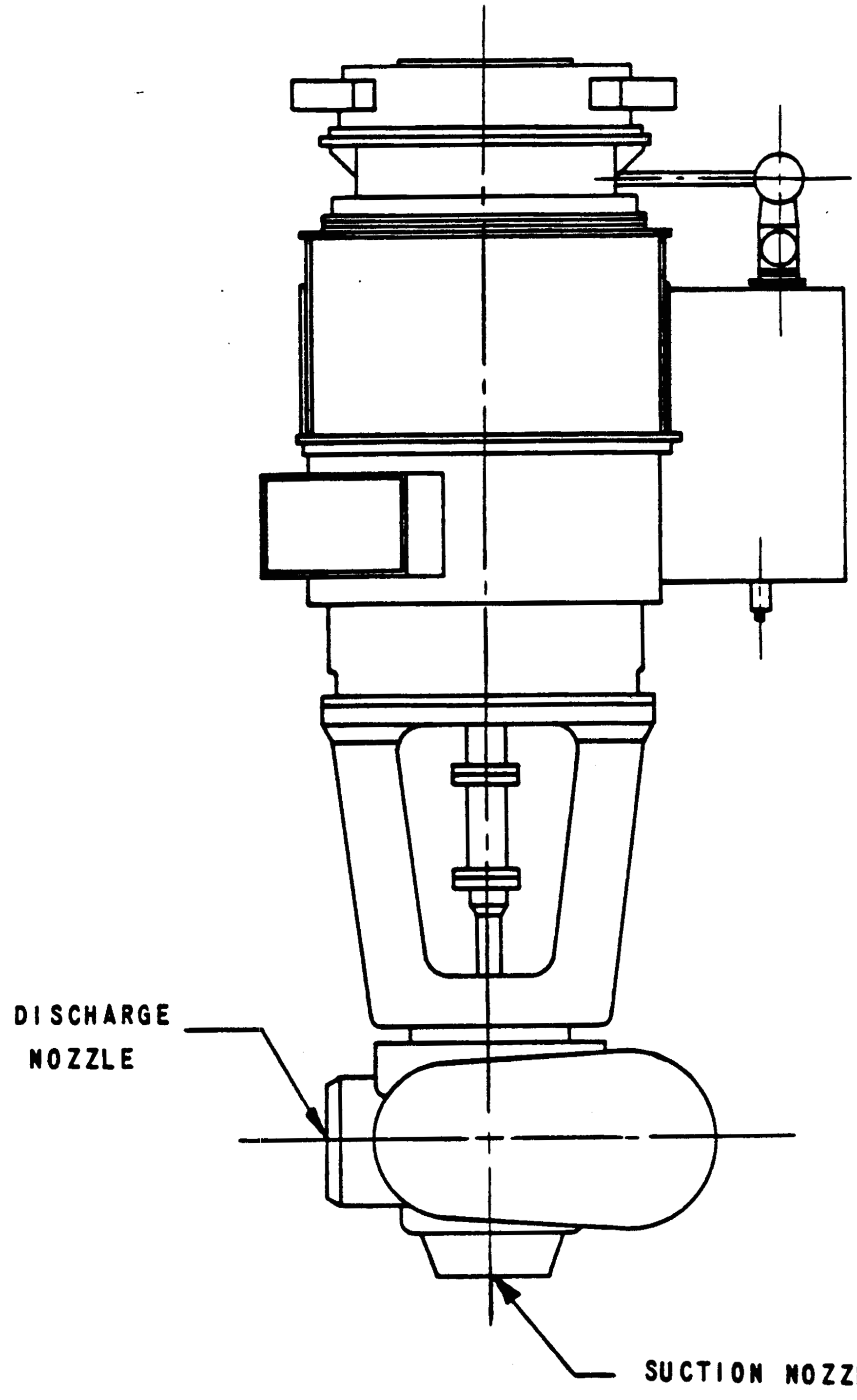
MPR Associates conducted a scoping-type of design review on steam generator supports and concluded that the design of the SG supports at DB-1 is adequate for dynamic loads. This review is historic information applicable to the original OTSG. Dynamic loads due to a large break LOCA are not applicable to the replacement OTSG due to leak-before-break.

5.5.15.4 Test and Inspections

The tests and inspections for the supports that are integral with the components are the same as those for the associated component as applicable.

5.5.16 Reactor Vessel Head Vent

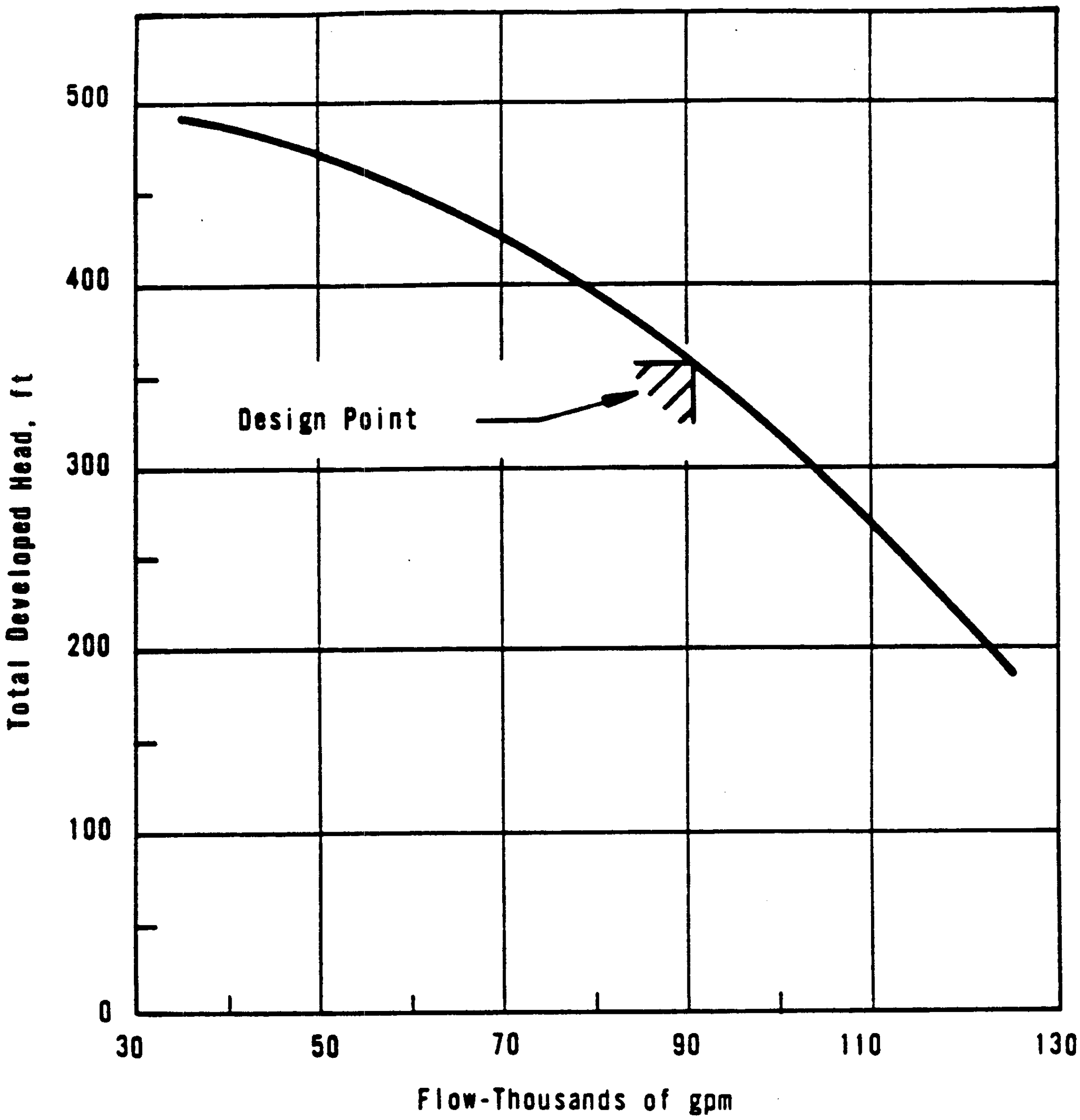
The Reactor Vessel Continuous Head Vent consists of piping attached to the Reactor Vessel which terminates at a connection at the top of the Steam Generator No.2. The purpose of the system is to allow any non-condensible gases or steam which may collect in the Reactor Vessel upper head region, during accident conditions, to vent to the hot leg high point of Steam Generator No. 2. The gases can then be removed via the high point vents and the steam can be condensed. A thermocouple is also provided in the line to measure fluid temperature.



DAVIS-BESSE NUCLEAR POWER STATION
REACTOR COOLANT PUMP

FIGURE 5.5-1

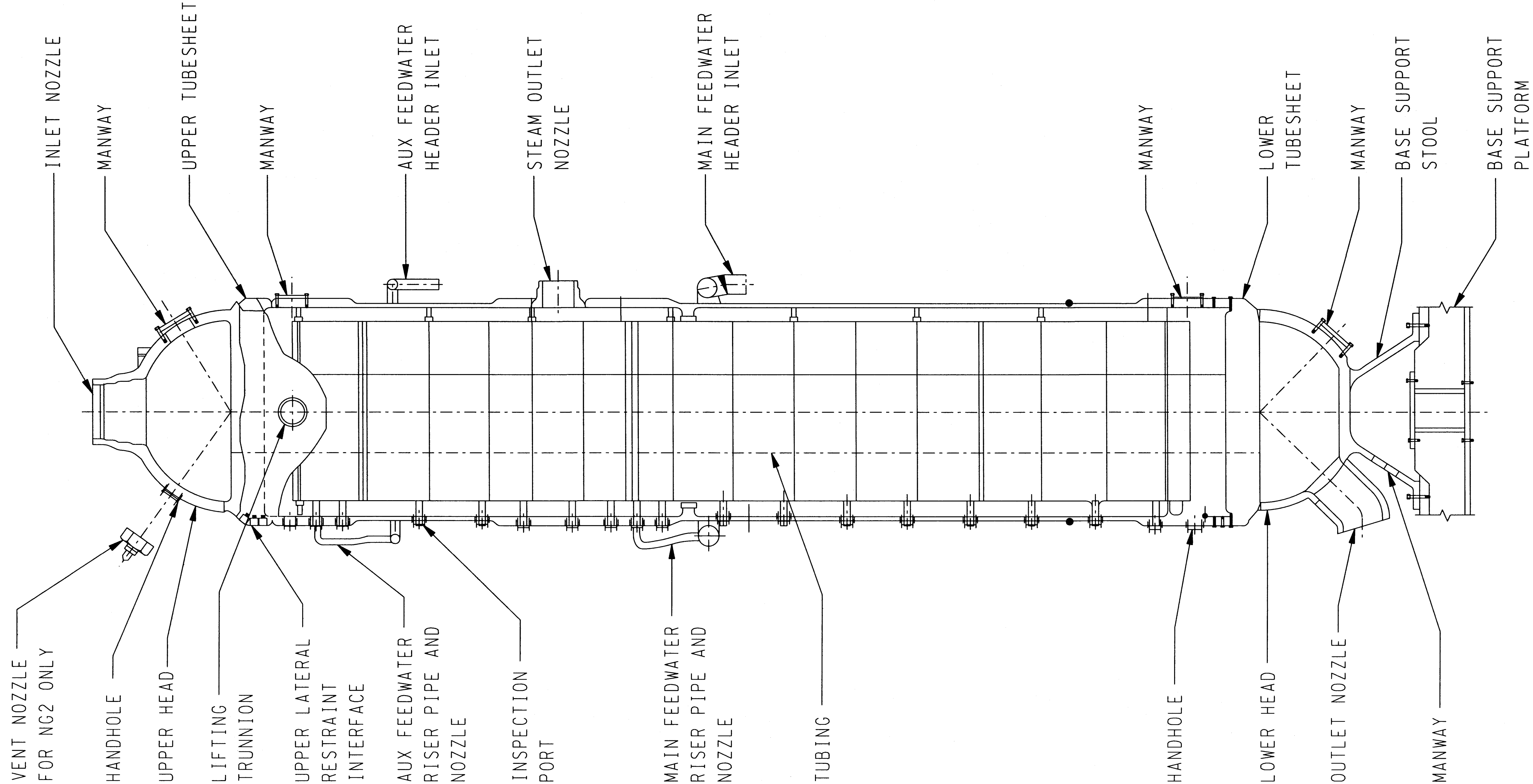
REVISION 0
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATI
REACTOR COOLANT PUMP
PERFORMANCE CURVE

FIGURE 5.5-2

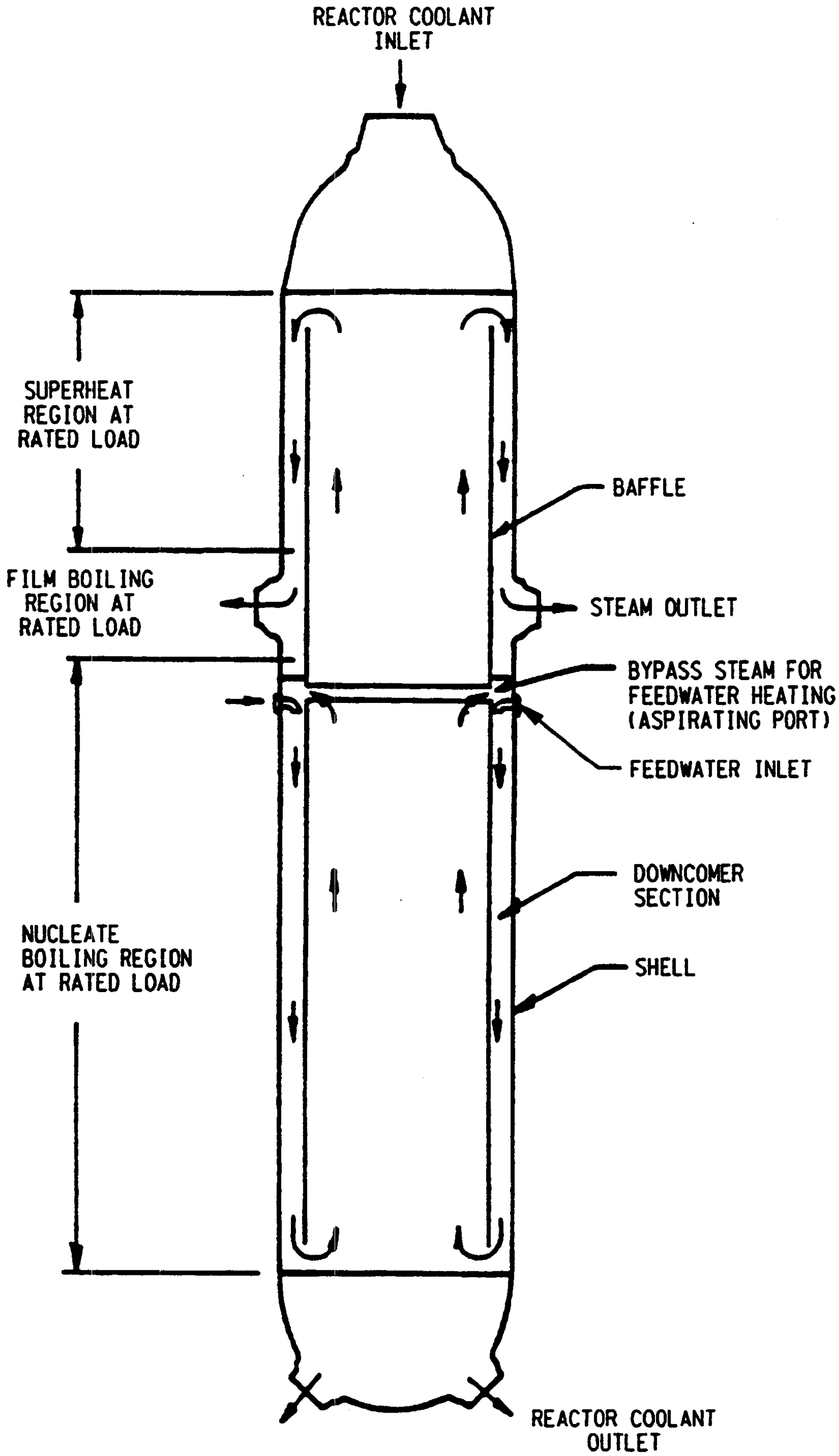
REVISION 0
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION
ARRANGEMENT OF DAVIS-BESSE REPLACEMENT ONCE THROUGH STEAM GENERATOR (OTSG)
FIGURE 5.5-3

REVISION 30
OCTOBER 2014

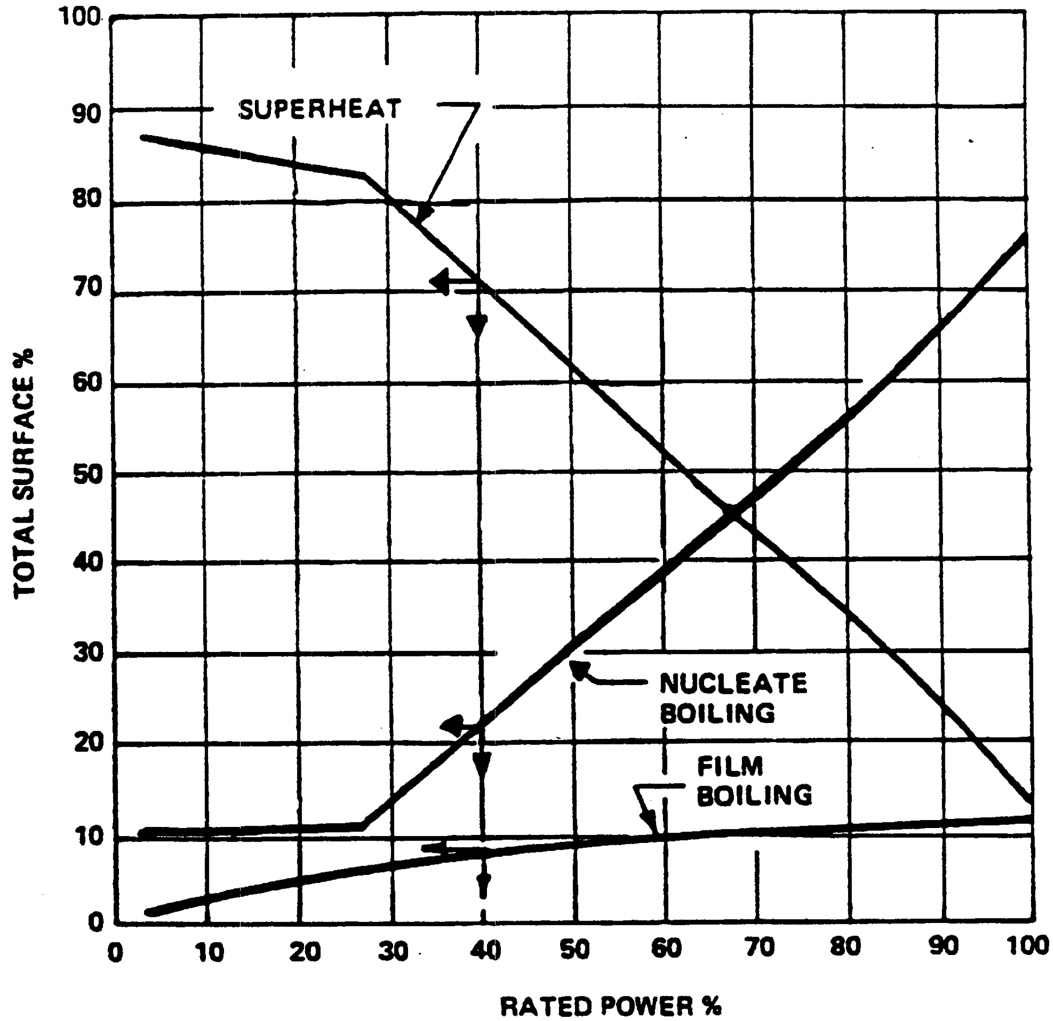
DB 07-15-14 DFN:G/USAR/UFIG553.DGN



DAVIS-BESSE NUCLEAR POWER STATION
STEAM GENERATOR HEATING REGIONS

FIGURE 5.5-4

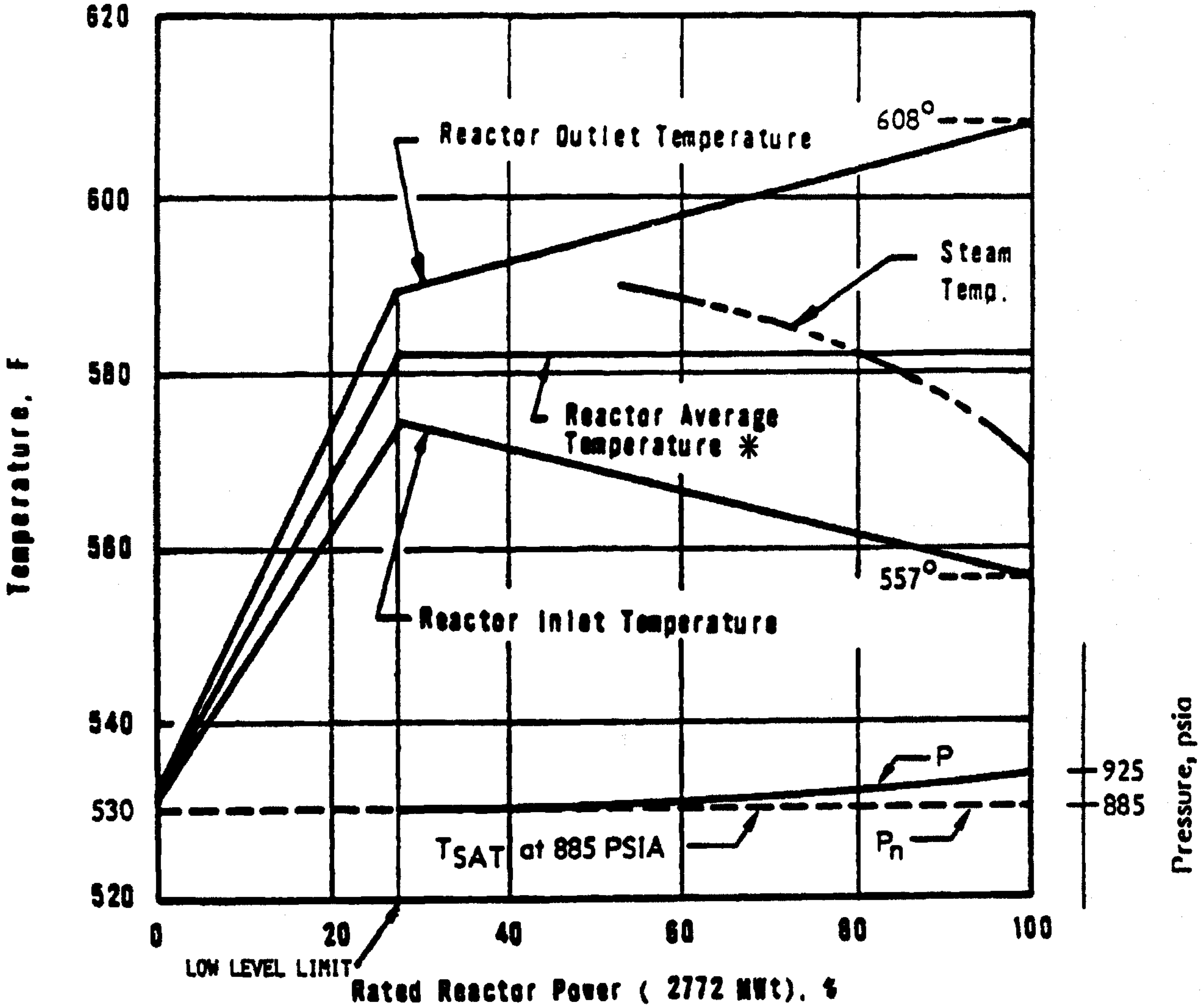
REVISION 20
DECEMBER 1996



DAVIS-BESSE NUCLEAR POWER STATION
STEAM GENERATOR HEATING SURFACE
VS POWER

FIGURE 5.5-5

REVISION 20
DECEMBER 1996



P Steam Generator Outlet Pressure

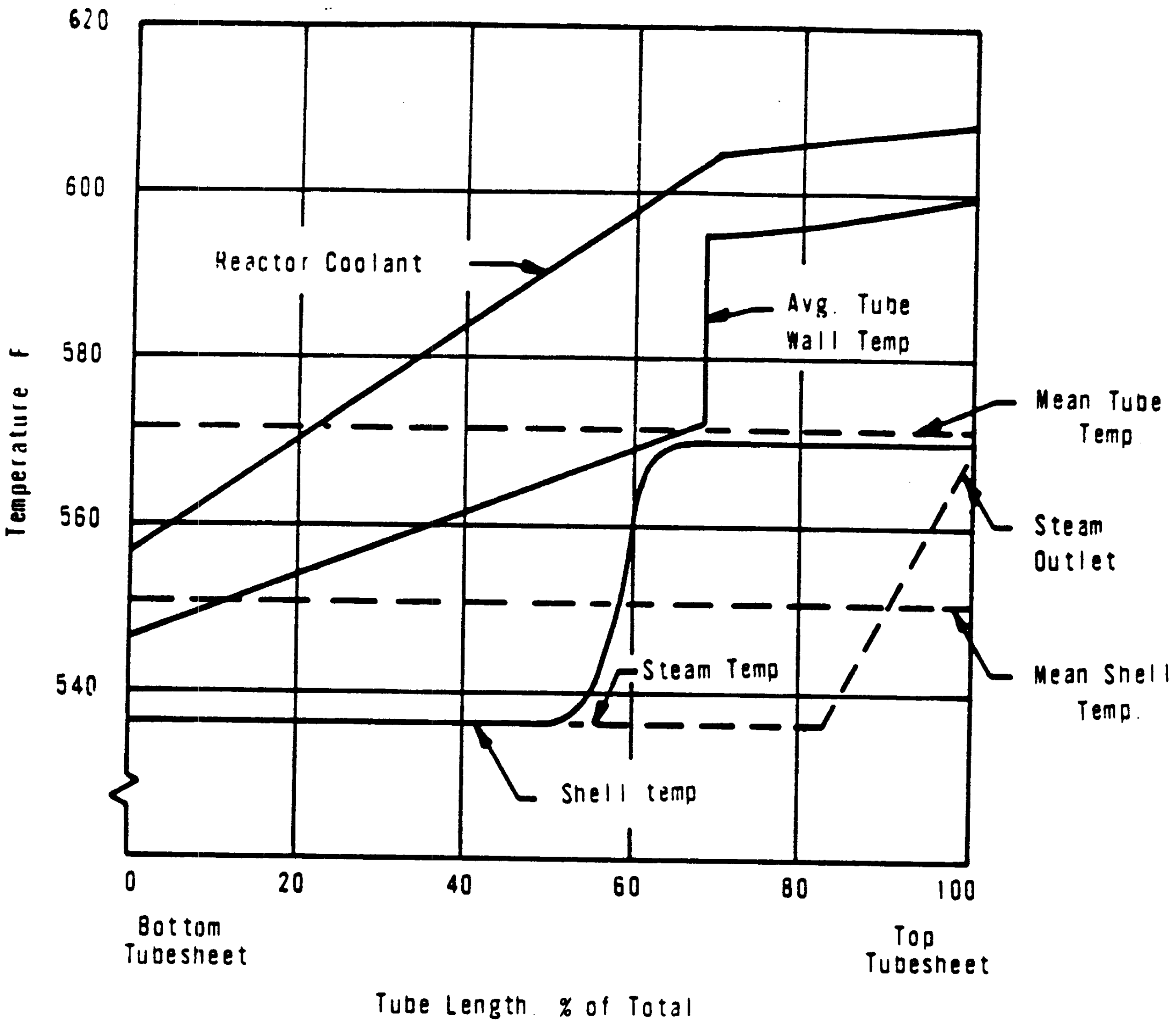
P_n Turbine Header Pressure

* At the end of a cycle, the average reactor coolant temperature, T_{AVE}, may be reduced by 12° F (see section 5.3.6).

DAVIS-BESSE NUCLEAR POWER STATION
REACTOR AND STEAM TEMPERATURES
VS REACTOR POWER
FIGURE 5.5-6

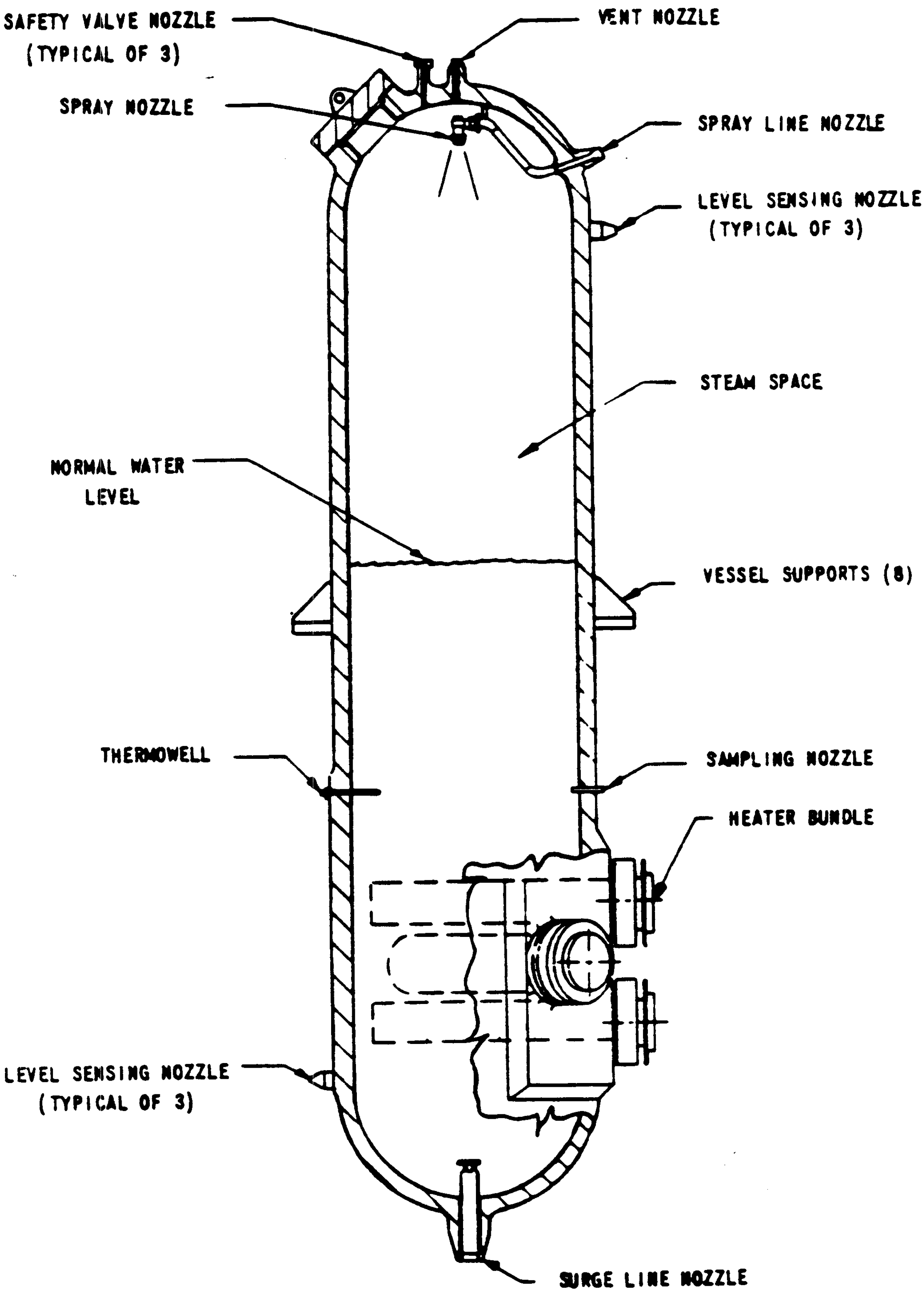
REVISION 27
JUNE 2010

DB 03-12-10 DFN:/G/USAR/UFIG556.DGN/CIT



DAVIS-BESSE NUCLEAR POWER STATION
STEAM GENERATOR TEMPERATURES

FIGURE 5.5-7 REVISION 0
JULY 1982



DAVIS-BESSE NUCLEAR POWER STATION
PRESSURIZER

FIGURE 5.5-8 REVISION 0
JULY 1982

5.6 INSTRUMENTATION APPLICATION RC SYSTEM

Instrumentation in the RC system is provided for the measurement and control of process variables as required for proper reactor operation and protection.

The following process variables are measured by redundant instruments:

a. Reactor coolant flow by four sensors in each loop.

b. Reactor coolant pressure by two narrow-range sensors in each loop.

c. Reactor coolant temperature by two sensors in each loop.

These signals are sent to the reactor protection system (RPS).

The current RC system flow indication scheme has been modified to meet the single failure criterion with regard to pressure sensing lines to the flow differential pressure transmitters. Specifically the design provides for a tee connection just downstream (approximately 4 inches) of the existing pressure tap root valves for both the high pressure and low pressure sensing taps. Downstream of these tees each set of redundant lines are routed to two flow transmitters. The redundant sets of sensing lines are routed independently, including a new penetration of the secondary shield wall independent from the existing set. These new sensing lines are stainless steel and seismically supported. Two of the flow transmitters receiving these new sensing lines have a minimum separation of 3 feet from the other two transmitter locations.

The pressure of the reactor coolant is also measured by four redundant, wide-range sensors whose signals are provided to the SFAS and two extended range sensors for post accident monitoring.

The following process variables are measured and used as input to the ICS for station control:

a. Reactor coolant flow in each loop is provided from the RPS. The signal is temperature-compensated before going to the ICS. The RC flow (temperature-compensated) is indicated in the control room and is summed and sent to the ICS.

b. Reactor coolant narrow-range temperature is measured by two redundant sensors in both the outlet and inlet legs in both loops which are monitored by a Smart Analog Selector Switch (SASS). The signals are used for the generation of differential and average temperature readings for input to the ICS. All reactor coolant temperature parameters, measured or derived, are recorded or indicated in the control room.

c. The reactor coolant narrow range pressure, reactor coolant flow, reactor coolant outlet and inlet leg narrow range temperature in both loops, and reactor coolant outlet and inlet leg wide range temperature in both loops are used by the ICS to calculate Core Thermal Power (CTP).

The following process variables are measured and used for other RC system control functions:

a. Two selectable narrow-range RC pressure transmitters, through the RPS, provide a signal to control the pressurizer heaters, pressurizer spray valve, and pressurizer

pilot operated relief valve. The pressure is recorded and actuates alarms in the control room.

b. The pressurizer level is measured by three redundant, manually selectable level sensors. The signal selected is compensated for temperature (NRC IE Bulletin 79-21 response, DB-81-013 letter dated February 24, 1981). The compensated signal is transmitted to the control room to be recorded, and actuates alarms. The compensated signal is also used for pressurizer level control and interlock to the pressurizer heaters. Redundant indicators, outside the control room, are provided for essential indication of the pressurizer level.

c. The pressurizer temperature is measured by two redundant, manually selectable temperature sensors. The selected signal is transmitted to the control room for temperature indication. The selected signal also provides the input for pressurizer level temperature compensation.

d. The RC wide-range loop inlet temperature is measured by two redundant, manually selectable temperature sensors in each loop. Both signals in each loop provide temperature indication in the control room, and the selected signal in each loop supplies input for RC pump interlocks.

The design and logic of this instrumentation is discussed further in Chapter 7.

The hot leg resistance temperature detectors (RTD) along with reactor coolant pressure indication indicate whether the Reactor Coolant System (RCS) is sub cooled or saturated. This determination allows operator actions designed to return the system to a subcooled condition if found otherwise.

Should these actions be ineffective, the incore thermocouples or the hot leg RTDs indicating superheated temperatures for the existing RCS pressure would provide a positive indication that the core is partially uncovered and inadequately cooled. Those indications would lead to subsequent operator actions designed to depressurize the RCS to reach the low pressure injection system setpoint in order to reflood the core region. These operator actions will recover the core and begin to recover RCS inventory. The existing instrumentation will show when the core is recovered by a decrease in incore thermocouple readings. The reactor coolant saturation meter automatically provides an RCS saturation evaluation to the operator.

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APPENDIX 5A

SAFETY EVALUATION OF RC PUMP MOTOR FLYWHEELS

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SAFETY EVALUATION OF RC PUMP MOTOR FLYWHEELS

LIST OF TABLES

LIST OF FIGURES

5A.1.0 Introduction And Statement Of Compliance With Safety Guide 14

Safety Guide 14 contains a statement of an acceptable method for implementing the requirements of General Design Criterion 4. This method is summarized in the second paragraph of the discussion (Part B) of the Guide. Compliance with this paragraph of Safety Guide 14, as quoted below, is presented as justification of the safety design of these flywheels.

Reactor Coolant pump flywheels are of a simple geometric shape, and normally are made of a ductile material. Their quality can be closely controlled and their service conditions are not severe; therefore, the use of suitable material and adequate design and service inspection can provide a sufficiently small probability of a flywheel failure that the consequences of failure need not be protected against.

5A.1.1 Design Objectives

Implementation of the design objectives for flywheel integrity was developed during the 18-month period between the purchase of the RC pump motor (2/9/70) and the issuance of Safety Guide 14. By the time Safety Guide 14 was issued, the flywheels were partially fabricated, and it was not practical to consider using some of the detailed guidelines in the regulatory position of the Guide.

These design objectives are listed below together with the related paragraphs of this evaluation.

a. Provide an adequate structural design for the flywheel. Paragraph (5.0) "Design Conditions of Flywheel Integrity," relates to the structural design.

b. Provide flywheel material that is suitably ductile. Paragraphs 2.0 and 5.1 contain information relative to the ductility of the material.

c. The quality of the flywheels shall be closely controlled. Paragraphs 2.0 and 5.1 contain information relative to quality control.

d. Provide an adequate program for inservice inspection. Paragraph 5.2 describes the inservice inspection program.

5A.2.0 Flywheel Description

As shown in Figure 5A.2-1, the flywheel basically consists of two discs bolted together with three keyways to key it to the upper end of the motor shaft. The dimensions of the discs are summarized below.

	Thickness, in.	Bore diameter, in.	Outer diameter, in.
Upper disc	5.94	9.38	72.0
Lower disc	4.56	9.43	65.0

The flywheel discs were fabricated from vacuum-degassed, low-alloy steel plates specified to ASTM A533-69A, Grade B, Class 1. Two discs were flame-cut from one steel plate with allowance of exclusion of flame-affected metal (1/2 inch on inner bore radius removed by machining after flame-cutting). In addition to the test requirements of ASTM A533-69A, Grade B, Class 1, Charpy V-notch impact tests were also conducted in accordance with ASTM A-370. A minimum of three Charpy V-notch impact tests were also conducted in accordance with

ASTM A-370. A minimum of three Charpy V-notch impact tests from each plate, oriented parallel and normal to the principal rolling direction, were tested at 10°F. The acceptance criteria required that the average energy value of the three Charpy specimens be equal to or higher than 30 ft-lb and that no energy value be less than 25 ft-lb. The results of the Charpy V-notch tests are as follows:

		Charpy V-notch Properties			
		Longitudinal		Transverse	
Heat no.	Applicable flywheels	Temp, °F	Energy, ft-lb	Temp, °F	Energy, ft-lb
C5507-3	2 lower discs	10	105, 93, 92	10	40, 48, 49
C9633-3	2 upper discs	10	85, 105, 102	10	28, 43, 30
C9747-34	2 upper discs	10	91, 75, 86	10	55, 60, 63
C9878-1L	2 lower discs	10	62, 70, 55	10	55, 62, 62

These four heats of material met the following requirements with regard to tensile properties:

Tensile strength, psi	80,000 to 100,000
Yield strength, min., psi	50,000

5A.3.0 Flywheel Operating Conditions

5A.3.1 Temperature

Since missiles are a safety consideration only when the reactor coolant and containment pressures and temperatures are at nearly normal operating levels, the minimum design temperature for this analysis was approximately +120°F, which is the normal operating temperature of the containment.

5A.3.2 Speeds and Transients

Once the turbine-generator is brought to speed, the RC pump is powered by the main generator. The normal operating speed of the pump is about 1189 rpm with a synchronous speed of 1200 rpm. The 1189-rpm speed (say, 1200 rpm for simplicity) corresponds to a turbine-generator speed of 1800 rpm. Since the turbine-generator is subject to overspeed during a loss of load, the RC pumps are also subject to an overspeed condition. The maximum speed of the turbine-generator for this transient is 120% of normal operating speed, and the pump speed lags that of the turbine-generator. Hence, for all practical purposes, the maximum speed of the flywheel is 120% of normal. For conservatism, however, 125% of operating speed (1500 rpm) is selected as the maximum possible operating speed of the flywheel.

5A.4.0 Overspeed Test

At the time these motors were tested there were no overspeed test requirements for the flywheel; however, all motors were run at overspeed with the flywheels in place. Two of the four motors were run at 115% of rated speed, and two were run at 125% of rated speed.

5A.5.0 Design Conditions of Flywheel Integrity

5A.5.1 Provisions Against Non-Ductile Failure

5A.5.1.1 Fracture Toughness Criterion

As described in Paragraph 1.2, the materials of construction of the pump flywheels were ordered prior to the effective date of Safety Guide 14. Consequently, the materials of construction were not specified to comply with the fracture toughness requirements of that Guide. However, the materials have adequate toughness for protection against non-ductile failure and meet the principal fracture toughness criteria of Safety Guide 14. The fracture criterion used to demonstrate the adequate toughness and compliance with the Guide is as follows:

> The minimum, static fracture toughness of the material at the normal operating temperature of the flywheel should be equivalent to a critical stress intensity factor, K_{IC} of at least 150 ksi $\sqrt{in.}$

This criterion is described in Subsection 5.4.1.1 of the Pump Flywheel Integrity (PWR) Regulatory Standard Review Plan of the U.S. Nuclear Regulatory Commission.

5A.5.1.2 Compliance With Fracture Criterion

To determine the temperature at which the flywheel materials of construction will exhibit a K_{IC} of at least 150 ksi $\sqrt{in.}$, the reference nil-ductility temperature (RT_{NDT} defined in paragraph NB-2300 of the 1972 Summer Addenda of Section III of ASME Code, 1971 Edition) must be established. Since the test data required for the exact determination of the RT_{NDT} were not required when the flywheel materials were ordered, the RT_{NDT} must be estimated. In order to estimate RT_{NDT}'s that are appropriately conservative, B&W has collected and evaluated data from tests conducted on SA 533, Grade B, Class 1, steel plates. The collected test data are from tests conducted in accordance with the fracture toughness requirements of the 1972 Summer Addenda of Section III of the ASME Code.

Table 5A-1 shows the measured and estimated RT_{NDT}'s of SA 533, Grade B, Class 1, steel plates of various section thicknesses. The data illustrate that the RT_{NDT}, is not always controlled by the drop weight T_{NDT}. On 6 out of 13 of the tested heats, the Charpy impact tests were performed at temperatures higher than T_{NDT} + 60°F in order to meet the 50 ft-lb minimum energy level requirement. The highest RT_{NDT} of all the SA 533B steel plates is 40°F. Based on these data, an RT_{NDT} of 40°F can conservatively be assigned to the flywheel materials of construction. The table also shows the conservatism of the estimated temperature. This is the difference between the estimated 40°F and the measured temperature; the range of conservatism is from 0 to 80°F.

In Figure A-4200-1 of the 1974 Edition of Section XI of the ASME Code, the K_{IC} of A 533, Grade B, Class 1 type materials is related to temperature (T) and to the RT_{NDT} . The normal operating temperature of the pump flywheel is defined in Section 3.1 as 120°F, and the RT_{NDT} of the flywheel materials has been estimated as 40°F. Figure A-4200-1 illustrates that the K_{IC} exceeds 150 ksi in. when T- RT_{NDT} is equal to 80°F. This demonstrates that the flywheel materials have adequate toughness at the normal operating temperature of the flywheel. It is believed that showing compliance with the fracture criterion described above eliminates the necessity for a fracture mechanics analysis to demonstrate that the pump flywheel materials have adequate toughness for protection against non-ductile failure during any postulated accident conditions.

5A.5.2 Pertinent Stresses

As mentioned earlier, the flywheel consists of two discs bolted together (see Figure 5A.2-1) and keyed to the upper end of the motor shaft. If these two discs were not bolted together, they obviously could be analyzed separately. However, considering the following items, one can deduce that the nominal stresses to the inertial forces can be calculated conservatively by ignoring these bolts:

a. The upper disc is larger in diameter than the lower disc; hence, the radial displacement of the upper disc owing to the inertial forces at a given radius would be greater than that of the lower disc at the same radius.

b. The existence of the bolts tends to restrict this differential radial displacement, which means that the belts would pull radially inward on the upper disc and radially outward on the lower disc.

c. This inward pull on the upper disc would reduce the inertial stresses (compared to a free disc) on the upper disc and increase those on the lower disc. Although the "net" nominal stresses on the lower disc would be higher than if the bolts were not present, it is clear that they would not exceed the nominal stresses for the upper disc. Neglecting the effect of the bolts and the lower disc should provide a somewhat conservative estimate of the nominal inertial stresses of the flywheel.

Referring now to the upper disc and neglecting the bolt holes and keyways, the nominal tangential and radial stresses may be determined from the following equation:

$$\sigma_t(r) = \frac{\rho w^2 R_o^2}{g} \frac{3+v}{8} \left[1 + \frac{R_i^2}{R_o^2} - \frac{1+3v}{3+v} \left[\frac{r^2}{R_o^2} \right] + \frac{R_i^2}{r^2} \right]$$

$$\sigma_R(r) = \frac{\rho w^2 R_o^2}{g} \frac{3+v}{8} \left[1 + \frac{R_i^2}{R_o^2} - \frac{r^2}{R_o^2} - \frac{R_i^2}{r^2} \right]$$

where:

w = angular velocity, rad/sec,

R_o = disc outside radius, in.,

R_i = disc inside radius, in.,

v = Poisson's ratio,

g = acceleration of gravity in./sec^2,

ρ = density of steel, lb/in.^{3}.

The combined primary stress at the normal operating speed (1189 rpm) is 12,400 psi (25% of the minimum yield strength) and is due to the tangential stress (σ_t) from the centrifugal forces since no interference fit loading is applicable for these flywheels, which are installed on the shaft by light pressed fit. The combined primary-stress at the designed overspeed (125% of normal) of the flywheel is 19,400 psi (37% of the minimum yield strength).

5A.5.3 Flywheel Integrity Considered for Other Safety Concerns

At startup, shear stresses in the flywheel keys and the motor shaft are negligible relative to their shear capability; as such, the flywheel will remain intact with the shaft.

With the motor short-circuited, the flywheel key and motor shaft shear stresses are well below their respective shear capacities, and, as before, the flywheel will, remain intact with the shaft.

Assuming an impeller seizure, the pump shaft would fail in torsion just below the coupling of the motor. The motor would continue to run at the same speed, and the flywheel would maintain its integrity since it would still be supported on a shaft with two bearings.

In the unlikely event of a shaft seizure, the flywheel keys would be the first to fail, and the flywheel would remain in place and rotate on the motor shaft.

The critical frequencies in the bending and torsional modes for the rotating components of the pump are well above seismic excitation frequencies.

The flywheel is designed so that the primary stresses do not exceed 25% of the minimum specified yield strength at normal operating speed. In fact, based on the concepts developed by Griffith and Irwin, the bursting speed of the flywheel has been calculated to be 3900 rpm, more than three times the normal operating speed.

5A.6.0 Nondestructive Testing

5A.6.1 Preservice Inspection

After flame-cutting the disc blanks from the steel plates, the plates were subjected to a 100% volumetric ultrasonic inspection of the flat surface by straight and angle beams. The ultrasonic inspection of these flywheels met or exceeded the requirements specified in Section III of the ASME Code, Case 1338-4. The angle-beam shear wave inspection used a 3%, 1-inch-long calibration notch.

The 100% volumetric inspection made normal to the surface can detect flaws parallel to the surface. Such flaws having an area greater than approximately 0.5 in.^{2} are recorded. The combination of the normal and angle-beam UT made from the flat surface is capable of detecting the more severe defect - that oriented in a radial plane. Such a defect is subject to the inertial tangential stresses.

After rough machining of the disc blanks, mating sides of the upper and lower discs were examined by the dye-penetrant method to an 18-inch diameter.

Also, after the discs are bolted together and final-machined, the bore, keyways, and exposed surfaces are dye-penetrant examined to an 18-inch diameter. Both penetrant examinations are made in accordance with the ASME Code, Section III, Paragraph N-322.

5A.6.2 Inservice Inspection

Per Davis-Besse Technical Specifications the inservice inspection program for each flywheel complies with Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. Amendment 232 of the Technical Specifications took exception to positions 1 and 2 of Section C.4.b of the Regulatory Guide, based upon Westinghouse Topical Report WCAP-14535A, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination," dated November 1996. The inservice inspection program for each flywheel includes the following:

a. Each reactor coolant pump flywheel shall be inspected at least once every 10 years, by conducting either an ultrasonic examination of the volume from the inner bore of the flywheel to the circle of one-half the outer radius, or a surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of the disassembled flywheel.

b. The examination procedures and acceptance criteria are in accordance with positions 3, 4, and 5 of Section C.4.b of Regulatory Guide 1.14, Revision 1, 1975.

5A.7.0 Replacement Flywheel Analysis

During the Eighth refueling outage a reactor coolant pump motor and flywheel were replaced. The motor was manufactured to the same design specifications as the original motor. The flywheel supplied with the replacement motor differs from the flywheel described in this Appendix in material and testing, but does not affect compliance with GDC-4 and Safety Guide 14.

The flywheel material was changed from a vacuum processed plate material to a vacuum processed forging. This change in material did not affect the nominal composition of the flywheel material nor did it change the tensile strength of the flywheel material. This change in material forming technique does not affect the material density. Based on the above, the moment of inertia added by the flywheel, and therefore the flywheel loads are unchanged from that described in this Appendix. The strength of the flywheel material are equivalent to that originally installed. Based on the above flywheel installed is not subject to ductile failure in service.

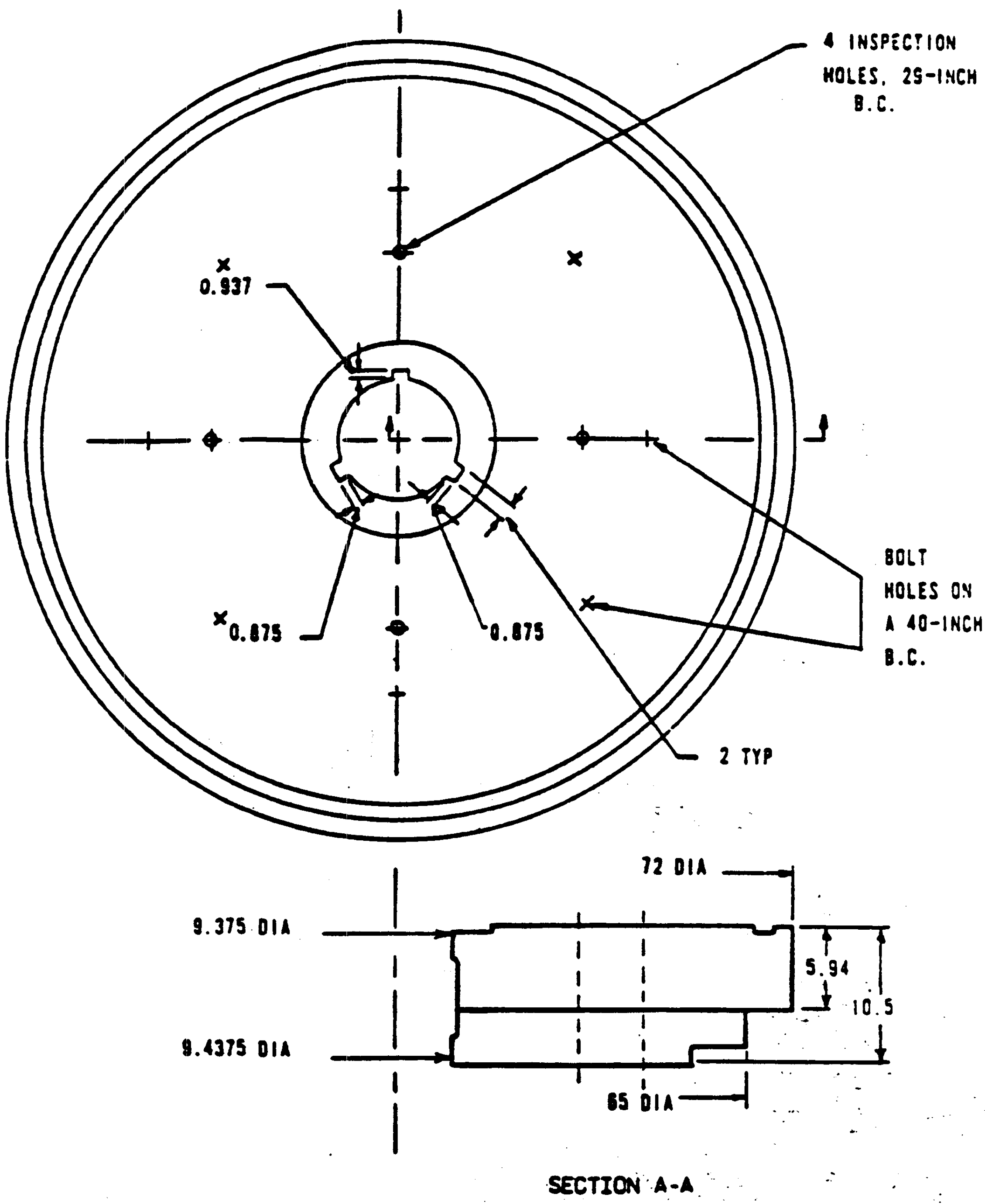
The testing changes were limited to the method used to demonstrate material ductility sufficient to prevent non ductile fracture. The testing performed demonstrated that the flywheel material has a critical stress intensification factor that exceeds the 150 ksi $\sqrt{\text{inch}}$ described in this Appendix as the screening criteria for nonductile fracture. Based on the above, the flywheels are not subject to nonductible failure in service.

Compliance with GDC-4 and Safety Guide 14 is ensured by: adequate structural design and suitable material with adequate tensile strength to preclude ductible failure; material processing, inspection and quality control to assure no unacceptable defects exists; material ductility sufficient to protect against nonductile failure; and continued compliance is assured by providing an inservice inspection plan.

TABLE 5A-1

Measured vs Estimated RT_{NDT} of SA533, Grade B, Class 1 Plates

Case	T_{NDT} by drop weight, °F	Charpy V-notch properties Temp, °F	Energy, ft-lb	Lat. exp, mils	Measured RT_{NDT}, °F	Estimated RT_{NDT}, °F	Conservatism of estimated RT_{NDT}, Δ°F
1	+10	+80	71,74,65	51,54,51	+20	+40	+20
2	-40	+20	71,80,81	60,62,68	-40	+40	+80
3	0	+60	55,57,55	40,38,38	0	+40	+40
4	-30	+30	85,104,112	62,69,77	-30	+40	+70
5	0	+90	55,60,60	43,49,44	+30	+40	+10
6	-30	+30	52,64,75	41,42,55	-30	+40	+70
7	-40	+20	59,68,79	39,52,55	-40	+40	+80
8	+20	+100	52,55,61	38,43,48	+40	+40	0
9	-40	+70	73,75,77	56,57,58	+10	+40	+30
10	+10	+80	51,51,60	43,42,41	+20	+40	+20
11	-20	+70	63,71,75	52,57,59	+10	+40	+30
12	+10	+70	112,113,62	76,79,45	+10	+40	+30
13	+10	+70	56,63,50	58,47,38	+10	+40	+30



DAVIS-BESSE NUCLEAR POWER STATION
FLYWHEEL GEOMETRY

FIGURE 5A.2-1

REVISION 0
JULY 1982