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5.0 REACTOR COOLANT SYSTEM

5.1 SUMMARY DESCRIPTION

The Reactor Coolant System (RCS) shown in Figure 5.1-1 consists of 3 similar heat transfer loops connected in parallel to the reactor pressure vessel. Each loop contains a reactor coolant pump, steam generator, and associated piping and valves. In addition, the system includes pressurizer, a pressurizer relief tank, interconnecting piping, and instrumentation necessary for operational control. All of the above components are located in the Reactor Building.

During operation, the RCS transfers the heat generated in the core to the steam generators where steam is produced to drive the turbine generator. Borated demineralized water is circulated in the RCS at a flowrate and temperature consistent with achieving the required reactor core thermal-hydraulic performance. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber (boron) used in chemical shim control.

The RCS pressure boundary provides a barrier against the release of radioactivity generated within the reactor and is designed to ensure a high degree of integrity throughout the life of the plant.

RCS pressure is controlled by the use of the pressurizer where water and steam are maintained in equilibrium by electrical heaters and water sprays. Steam is formed (by the heaters) or condensed (by the pressurizer spray) to minimize pressure variations due to contraction and expansion of the reactor coolant. Spring loaded safety valves and power operated relief valves are mounted on the pressurizer and discharge to the pressurizer relief tank, where the steam is condensed and cooled by mixing with water.

1. Reactor Coolant Pressure Boundary

   The extent of the RCS is defined as:

   a. The reactor vessel including control rod drive mechanism housings.

   b. The reactor coolant side of the steam generators.

   c. Reactor coolant pumps.

   d. A pressurizer attached to one of the reactor coolant loops.

   e. Safety and relief valves.

f. The interconnecting piping, valves, and fittings between the principal components listed above.

g. The piping, fittings, and valves leading to connecting auxiliary or support systems up to and including the second isolation valve (from the high pressure side) on each line.

2. RCS Components

a. Reactor vessel

The reactor vessel is cylindrical with a welded hemispherical bottom head and a removable flanged and gasketed hemispherical upper head. The vessel contains the core, core supporting structures, control rods, and other parts directly associated with the core.

The vessel has inlet and outlet nozzles located in a horizontal plane just below the reactor vessel flange but above the top of the core. Coolant enters the vessel through the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom and flows up through the core to the outlet nozzles.

b. Steam generators

The steam generators are vertical shell and U-tube evaporators with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel.

c. Reactor coolant pumps

The reactor coolant pumps are identical single-speed centrifugal units driven by air cooled, three-phase induction motors. The shaft is vertical with the motor mounted above the pump. A flywheel on the shaft above the motor provides additional inertia to extend pump coastdown. The inlet is at the bottom of the pump; discharge is on the side.

d. Piping

The reactor coolant loop piping is specified in sizes consistent with system requirements.

The hot leg inside diameter is 29 inches and the inside diameter of the cold leg return line to the reactor vessel is 27-1/2 inches. The piping between the steam generator and the pump suction is increased to 31 inches in inside diameter to reduce pressure drop and improve flow conditions to the pump suction.

e. Pressurizer

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads. Electrical heaters are installed through the bottom head of the vessel while the spray nozzle, relief, and safety valve connections are located in the top head of the vessel.

f. Pressurizer relief tank

The pressurizer relief tank is a horizontal, cylindrical vessel with hemispherical ends. Steam from the pressurizer safety and relief valves is discharged into the pressurizer relief tank through a sparger pipe under the water level. This condenses and cools the steam by mixing it with water that is near ambient temperature.

g. Safety and relief valves

The pressurizer safety valves are of the totally enclosed pop type. The valves are spring loaded, self actuated, and with back pressure compensation features. The power operated relief valves limit system pressure for large power mismatch. They are operated automatically or by remote manual control. Remotely operated valves are provided to isolate the inlet to the power operated relief valves if excessive leakage occurs.

3. RCS Performance Characteristics

Tabulations of important design and performance characteristics of the RCS are provided in Table 5.1-1.

a. Reactor coolant flow

The reactor coolant flow, a major parameter in the design of the system and its components, is established with a detailed design procedure supported by operating plant performance data, by pump model tests and analyses, and by pressure drop tests and analyses of the reactor vessel and fuel assemblies. Data from all operating plants have indicated that the actual flow has been well above the flow specified for the thermal design of the plant. By applying the design procedure described below, it is possible to specify the expected operating flow with reasonable accuracy.

Three (3) reactor coolant flowrates are identified for the various plant design considerations. The definitions of these flows are presented in the following paragraphs and the application of the definitions is illustrated by the system and pump hydraulic characteristics on Figure 5.1-2.

b. Best estimate flow

The best estimate flow is the most likely value for the actual plant operating condition. This flow is based on the best estimate of the reactor vessel, steam generator, piping flow resistance, and on the best estimate of the reactor coolant pump head flow capacity, with no uncertainties assigned to either the system flow resistance or the pump head. System pressure drops, based on best estimate flow, are presented in Table 5.1-1. Although the best estimate flow is the most likely value to be expected in operation, more conservative flowrates are applied in the thermal and mechanical designs, as noted on Figure 5.1-2.

c. Thermal design flow

Thermal design flow is the basis for the reactor core thermal performance, the steam generator thermal performance, and nominal plant parameters used throughout the design. To provide the required margin, the thermal design flow accounts for flow resistance uncertainties in the reactor vessel, steam generator, and piping as well as uncertainties in the reactor coolant pump head and the methods used to measure flowrate. The combination of these uncertainties, which include a conservative estimate of the pump discharge weir flow resistance, is equivalent to increasing the best estimate RCS flow resistance by approximately 19 percent. The intersection of this conservative flow resistance with the best estimate pump curve, as shown in Figure 5.1-2,

establishes the maximum recommended thermal design flow. This procedure provides a flow margin for thermal design of approximately 4.85 percent for initial plant operation with Model D3 steam generators. The current thermal design flow is selected such that adequate margin exists between the thermal design flow and the best estimate flow for the Delta-75 steam generators with 10% tube plugging. The thermal design flow is confirmed each cycle. Tabulations of important design and performance characteristics of the RCS, as provided in Table 5.1-1, are based on the thermal design flow. RN 09-022

d. Mechanical design flow

Mechanical design flow is the conservatively high flow used in the mechanical design of the reactor vessel internals and fuel assemblies. To assure that a conservatively high flow is specified, the mechanical design flow is based on a reduced system resistance (90 percent of the original best estimate) and on increased pump head capability (107 percent of best estimate). The intersection of this flow resistance with the higher pump curve, as shown on Figure 5.1-2, establishes the mechanical design flow. The resulting mechanical design flow of 107,100 gpm is approximately 3.98 percent greater than the original best estimate flow of 103,000 gpm. RN 09-022

Design pump overspeed, due to a turbine generator overspeed of 20 percent, results in a peak reactor coolant flow of 120 percent of the mechanical design flow.

e. Flows with one pump shutdown

The design procedure for calculation of flows with one pump shutdown is similar to the procedure described above for calculating flows with all pumps operating. For the case where reverse flow exists in the idle loop, the system resistance incorporates the idle loop with a locked rotor pump impeller reverse flow resistance as a flow path in parallel with the reactor vessel internals. The thermal design flow uncertainty includes a conservative application of parallel flow uncertainties (reactor internals high, idle loop low) as well as the usual component, pump, and flow measurement uncertainties, thereby resulting in a conservatively low reactor flowrate for the thermal design. The mechanical design flow uncertainty is increased slightly to account for the slightly higher uncertainties at the higher pump flows. Original thermal design, best estimate, and mechanical design flows for the Virgil C. Summer Nuclear Station with 2 out of 3 pumps operating are summarized in Table 5.1-1.

4. Interrelated Performance and Safety Function

The interrelated performance and safety functions of the RCS and its major components are listed below:

a. The RCS provides sufficient heat transfer capability to transfer the heat produced during power operation and during the initial phase of plant cooldown to the steam and power conversion system.

b. The system provides sufficient heat transfer capability to transfer the heat produced during the subsequent phase of plant cooldown and cold shutdown to the residual heat removal system.

c. The system heat removal capability under power operation and normal operational transients, including the transition from forced to natural circulation, shall assure no fuel damage within the operating bounds permitted by the reactor control and protection systems.

d. The RCS contains the water used as the core neutron moderator and reflector and as a solvent for chemical shim control.

e. The system maintains the homogeneity of soluble neutron poison concentration and rate of change of coolant temperature such that uncontrolled reactivity changes do not occur.

f. The reactor vessel is an integral part of the RCS pressure boundary and is capable of accommodating the temperatures and pressures associated with the operational transients. The reactor vessel also functions to support the reactor core and control rod drive mechanisms.

g. The pressurizer maintains the system pressure during operation and limits pressure transients. During the reduction or increase of plant load, reactor coolant volume changes are accommodated in the pressurizer via the surge line.

h. The reactor coolant pumps supply the coolant flow necessary to remove heat from the reactor core and transfer it to the steam generators.

i. The steam generators provide high quality steam to the turbine. The tube and tubesheet boundary are designed to prevent or control, to acceptable levels, the transfer of activity generated within the core to the secondary system.

j. The RCS piping serves as a boundary for containing the coolant under operating temperature and pressure conditions and for limiting leakage (and activity release) to the reactor building atmosphere. The RCS piping contains borated water which is circulated at the flowrate and temperature consistent with achieving the reactor core thermal and hydraulic performance.

5.1.1 SCHEMATIC FLOW DIAGRAM

The RCS is shown schematically in Figure 5.1-3. Included on this figure is a tabulation of principal pressures, temperatures, and the flowrate of the system under normal steady-state full power operating conditions. These parameters are based on the best estimate flow at the pump discharge. RCS volume under the above conditions is presented in Table 5.1-1.

5.1.2 PIPING AND INSTRUMENTATION DIAGRAMS

The piping and instrumentation diagrams of the RCS are shown on Figure 5.1-1 (Sheets 1 through 3). The diagrams show the extent of the systems located within the reactor building and the points of separation between the RCS and the secondary (heat utilization) system.

5.1.3 ELEVATION DRAWINGS

Elevation and layout drawings showing principal dimensions of the RCS, in relation to supporting and surrounding concrete structures, are shown by Figures 1.2-9 and 1.2-10.

TABLE 5.1-1

SYSTEM DESIGN AND OPERATING PARAMETERS | 02-01

Original System Design And Operating Parameters With Model D3 S/Gs

Plant Design Life, years	40
Nominal Operating Pressure, psig	2,235
Total System Volume Including Pressurizer and Surge Line, ft^3	9,410 (hot)
System Liquid Volume, Including Pressurizer Water at Maximum Guaranteed Power, ft^3	8,850 (hot)
Pressurizer Spray Rate, maximum gpm	700
Pressurizer Heater Capacity, kW	1,400
Pressurizer Relief Tank Volume, ft^3	1,300

Original System Thermal and Hydraulic Data with Model D3 S/Gs

	Original Design	
	3 Pumps Running	2 Pumps Running
NSSS Power, MWt	2,785	1,671
Reactor Power, MWt	2,775	1,666
Thermal Design Flows, gpm/loop		
Active Loop-	98,000	104,200
Idle Loop(1)	-	24,300
Reactor	294,000	184,100
Total Reactor Flow, 10^6 lb/hr	109.6	68.6
Temperatures, °F		
Reactor Vessel Outlet	618.7	602.5
Reactor Vessel Inlet	556.0	547.9
Steam Generator Outlet	555.8	547.7
Steam Generator Steam	540.2	535.8
Feedwater	435	380
Steam Pressure, psia	964	930
Total Steam Flow, 10^6 lb/hr	12.20	6.8
Best Estimate Flows, gpm/loop		
Active Loop	103,000	107,700
Idle Loop(1)	-	23,100
Reactor	309,000	192,300
Mechanical Design Flows, gpm/loop		
Active Loop	107,100	112,400
Idle Loop(1)	-	24,100
Reactor	321,300	200,700

TABLE 5.1-1 (Continued)

Current System Design and Operating Parameters with Delta-75 S/Gs

Plant Design Life, years	40	
Nominal Operating Pressure, psig	2235	
Total System Volume Including Pressurizer Surge Line, ft^3	9914	
System Liquid Volume, Including Pressurizer Water at Maximum Guaranteed Power, ft^3	9354	
Pressurizer Spray Rate, maximum gpm	700	
Pressurizer Heater Capacity, kW	1366.6	02-01
Pressurizer Relief Tank Volume, ft^3	1300	

Current System Thermal and Hydraulic Data with Delta-75 S/Gs
Current Design - 3 pumps Operating (3)

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	Low Temperature (4)		High Temperature (5)		
	0% SGTP	10% SGTP	0% SGTP	10% SGTP	
NSSS Power, MWt	2912	2912	2912	2912	
Reactor Power, MWt	2900	2900	2900	2900	
Thermal Design Flow, gpm/loop	92,600	92,600	92,600	92,600	
Total Reactor Flow, 10^6 lb/hr	106.3	106.3	104.0	104.0	
Temperatures, °F					
Reactor Vessel Outlet	607.3	607.3	621.9	621.9	RN 09-022
Reactor Vessel Inlet	536.7	536.7	552.9	552.9	
Steam Generator Outlet	536.3	536.3	552.6	552.6	
Steam Generator Steam	523.7	521.7	540.4	538.4	
Feedwater	440	440	440	440	
Steam Pressure, psia	839	824	966	950	
Total Steam Flow, 10^6 lb/hr	12.77	12.76	12.84	12.83	
Best Estimate Flow, gpm/loop	106,200 (7)	--	--	103,000 (8)	RN 09-022
Mechanical Design Flow, gpm/loop	107,100	107,100	107,100	107,100	
Core Bypass flow, %	8.9	8.9	8.9	8.9	

TABLE 5.1-1 (Continued)

SYSTEM DESIGN AND OPERATING PARAMETERS

	System Pressure Drops[2]		
	Original Design	Current Design[5]	
Reactor Vessel ΔP, psi	40.7	41.51	RN 09-022
Steam Generator ΔP, psi	37.9	46.52	
Hot Leg Piping ΔP, psi	1.8		
Pump Suction Piping ΔP, psi	4.5		
Cold Leg Piping ΔP, psi	1.9		
Total Piping ΔP, psi	8.2	8.22	RN 09-022
Pump Head, feet	293	298	

(1) Reverse direction in idle loop.

(2) System pressure drops are based on best estimate flow.

(3) Design allows hot full power T_{avg} to range from 572°F to 587.4°F

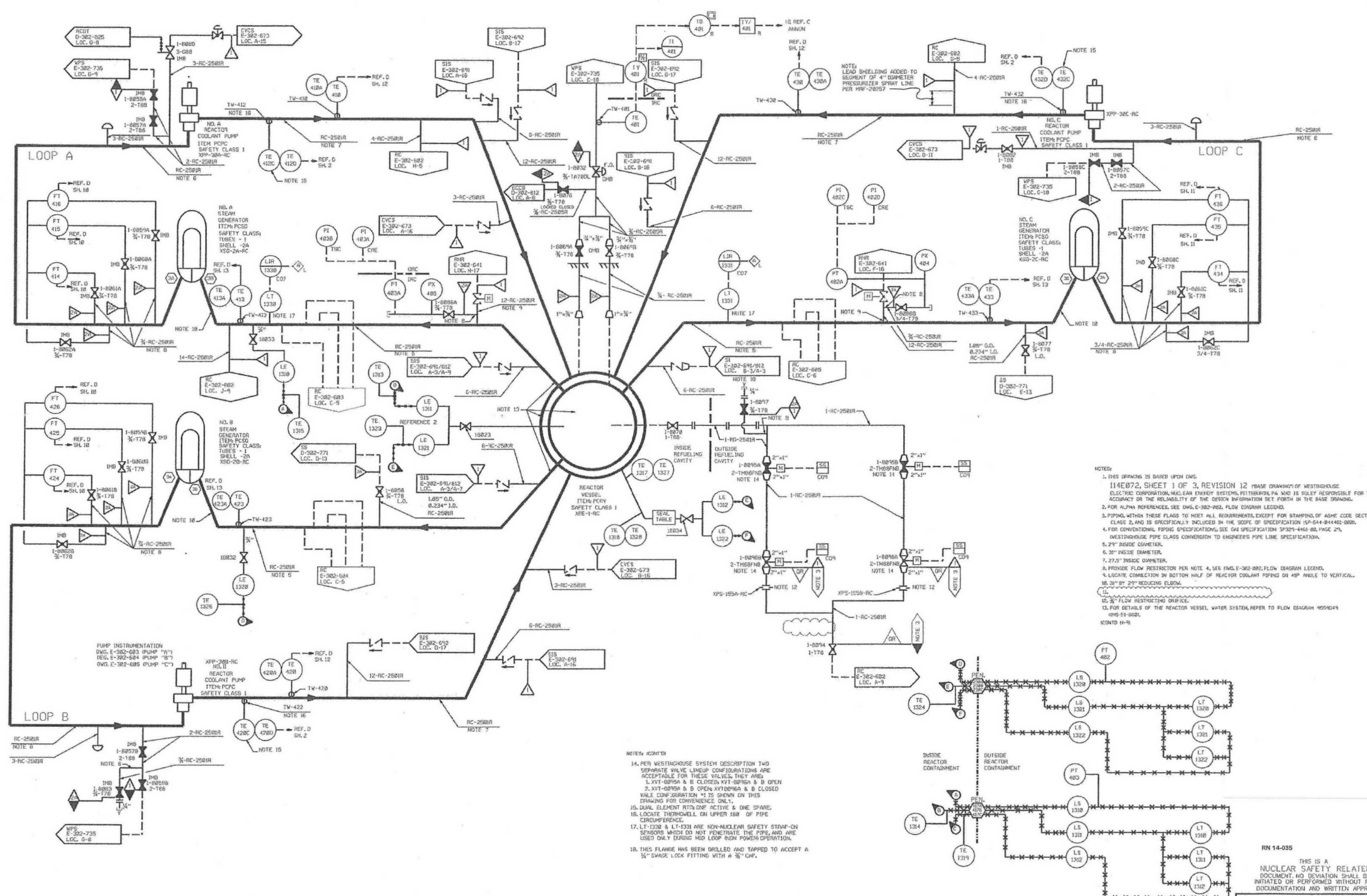
(4) At T_{avg} of 572°F

(5) At T_{avg} of 587.4°F

(6) Engineered Safeguards Design Rating

(7) Thimble plugs removed. RN

(8) Thimble plugs inserted. 09-022



NOTES:

1. THIS DRAWING IS BASED UPON DWG.
114E072, SHEET 1 OF 3, REVISION 12 ("BASE DRAWING") OF WESTINGHOUSE ELECTRIC CORPORATION, NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA WHO IS SOLEY RESPONSIBLE FOR THE ACCURACY OR THE RELIABILITY OF THE DESIGN INFORMATION SET FORTH IN THE BASE DRAWING.
2. FOR ALPHA REFERENCES, SEE DWG. E-302-002, FLOW DIAGRAM LEGEND.
3. PIPING WITHIN THESE FLAGS TO MEET ALL REQUIREMENTS, EXCEPT FOR STAMPING, OF ASME CODE SECTION III, CLASS 2, AND IS SPECIFICALLY INCLUDED IN THE SCOPE OF SPECIFICATION (SP-544-044461-000).
4. FOR CONVENTIONAL PIPING SPECIFICATIONS, SEE GAI SPECIFICATION SP329-4461-00, PAGE 29, (WESTINGHOUSE PIPE CLASS CONVERSION TO ENGINEER'S PIPE LINE SPECIFICATION).
5. 29" INSIDE DIAMETER.
6. 31" INSIDE DIAMETER.
7. 27.5" INSIDE DIAMETER.
8. PROVIDE FLOW RESTRICTOR PER NOTE 4, SEE DWG. E-302-002, FLOW DIAGRAM LEGEND.
9. LOCATE CONNECTION IN BOTTOM HALF OF REACTOR COOLANT PIPING ON 45° ANGLE TO VERTICAL.
10. 31" BY 29" REDUCING ELBOW.
11.
12. 3/8" FLOW RESTRICTING ORIFICE.
13. FOR DETAILS OF THE REACTOR VESSEL WATER SYSTEM, REFER TO FLOW DIAGRAM 4559D49 (1MS-51-088).

(CONTD H-9)

NOTES: (CONT'D)

14. PER WESTINGHOUSE SYSTEM DESCRIPTION TWO SEPARATE VALVE LINEUP CONFIGURATIONS ARE ACCEPTABLE FOR THESE VALVES. THEY ARE:
    1. XVT-8095A & B CLOSED; XVT-8096A & B OPEN
    2. XVT-8095A & B OPEN; XVT8096A & B CLOSED
    VALE CONFIGURATION #1 IS SHOWN ON THIS DRAWING FOR CONVENIENCE ONLY.
15. DUAL ELEMENT RTD: ONE ACTIVE & ONE SPARE.
16. LOCATE THERMOWELL ON UPPER 180 OF PIPE CIRCUMFERENCE.
17. LT-1330 & LT-1331 ARE NON-NUCLEAR SAFETY STRAP-ON SENSORS WHICH DO NOT PENETRATE THE PIPE, AND ARE USED ONLY DURING MID LOOP (NON POWER) OPERATION.
18. THIS FLANGE HAS BEEN DRILLED AND TAPPED TO ACCEPT A 1/4" SWAGE LOCK FITTING WITH A 3/16" CAP.

RN 14-035

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ESSENTIAL

DRAWING LEGIBILITY CLASS 1
SCE&G CAD ENHANCED

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
17	10/27/03	DDJ	CADD ENHANCED PER ECR-50239	MGR	DDJ
16	11/18/02	DDJ	REVISED PER ECR-70276	MGR	TOC
15	01/08/98	JTS	REVISED PER CGSS-97-0569	LEK	MGR
14	09/25/95	AVN	REVISED PER MRF-22640	MGR	JEW
19	09/17/14	DDJ	REVISED PER ECR72002	MGR	SC
18	8/30/05	RHM	REVISED PER ECR-70765	MGR	RHM

FSAR Figure 5.1-1, Sht. 1

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

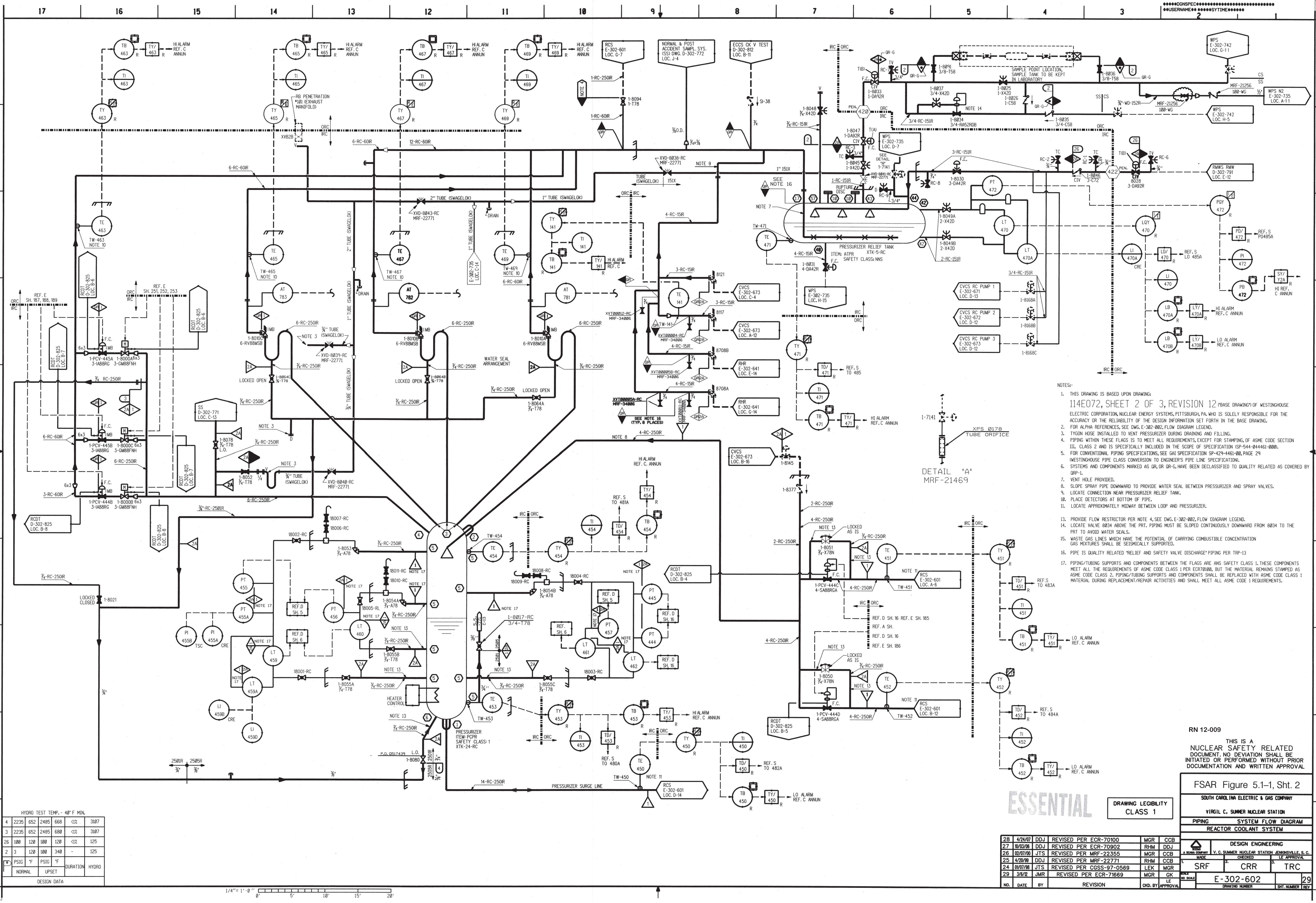
PIPING — SYSTEM FLOW DIAGRAM

REACTOR COOLANT

DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. AVN	2. MGR	3. DDS

SCALE: NO SCALE — DRAWING NUMBER: E-302-601 — SHT. NUMBER: — REV: 19



NOTES:-

1. THIS DRAWING IS BASED UPON DRAWING:
   114E072, SHEET 2 OF 3, REVISION 12 ("BASE DRAWING") OF WESTINGHOUSE ELECTRIC CORPORATION, NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA. WHO IS SOLELY RESPONSIBLE FOR THE ACCURACY OR THE RELIABILITY OF THE DESIGN INFORMATION SET FORTH IN THE BASE DRAWING.
2. FOR ALPHA REFERENCES, SEE DWG. E-302-002, FLOW DIAGRAM LEGEND.
3. TYGON HOSE INSTALLED TO VENT PRESSURIZER DURING DRAINING AND FILLING.
4. PIPING WITHIN THESE FLAGS IS TO MEET ALL REQUIREMENTS, EXCEPT FOR STAMPING, OF ASME CODE SECTION III, CLASS 2 AND IS SPECIFICALLY INCLUDED IN THE SCOPE OF SPECIFICATION (SP-544-044461-000).
5. FOR CONVENTIONAL PIPING SPECIFICATIONS, SEE GAI SPECIFICATION SP-429-4461-00, PAGE 29 (WESTINGHOUSE PIPE CLASS CONVERSION TO ENGINEER'S PIPE LINE SPECIFICATION).
6. SYSTEMS AND COMPONENTS MARKED AS QR, OR QR-G, HAVE BEEN DECLASSIFIED TO QUALITY RELATED AS COVERED BY QRP-1.
7. VENT HOLE PROVIDED.
8. SLOPE SPRAY PIPE DOWNWARD TO PROVIDE WATER SEAL BETWEEN PRESSURIZER AND SPRAY VALVES.
9. LOCATE CONNECTION NEAR PRESSURIZER RELIEF TANK.
10. PLACE DETECTORS AT BOTTOM OF PIPE.
11. LOCATE APPROXIMATELY MIDWAY BETWEEN LOOP AND PRESSURIZER.

13. PROVIDE FLOW RESTRICTOR PER NOTE 4, SEE DWG. E-302-002, FLOW DIAGRAM LEGEND.
14. LOCATE VALVE 8034 ABOVE THE PRT. PIPING MUST BE SLOPED CONTINUOUSLY DOWNWARD FROM 8034 TO THE PRT TO AVOID WATER SEALS.
15. WASTE GAS LINES WHICH HAVE THE POTENTIAL OF CARRYING COMBUSTIBLE CONCENTRATION GAS MIXTURES SHALL BE SEISMICALLY SUPPORTED.
16. PIPE IS QUALITY RELATED "RELIEF AND SAFETY VALVE DISCHARGE" PIPING PER TRP-13
17. PIPING/TUBING SUPPORTS AND COMPONENTS BETWEEN THE FLAGS ARE ANS SAFETY CLASS 1. THESE COMPONENTS MEET ALL THE REQUIREMENTS OF ASME CODE CLASS 1 PER ECR70100, BUT THE MATERIAL REMAINS STAMPED AS ASME CODE CLASS 2. PIPING/TUBING SUPPORTS AND COMPONENTS SHALL BE REPLACED WITH ASME CODE CLASS 1 MATERIAL DURING REPLACEMENT/REPAIR ACTIVITIES AND SHALL MEET ALL ASME CODE 1 REQUIREMENTS.

HYDRO TEST TEMP. - 40° F MIN.

4	2235	652	2485	668	<1%	3107
3	2235	652	2485	680	<1%	3107
26	100	120	100	120	<1%	125
2	3	120	100	340	-	125
#	PSIG	°F	PSIG	°F	DURATION	HYDRO
	NORMAL		UPSET			
DESIGN DATA						

RN 12-009

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ESSENTIAL

DRAWING LEGIBILITY CLASS 1

FSAR Figure 5.1–1, Sht. 2

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

PIPING SYSTEM FLOW DIAGRAM

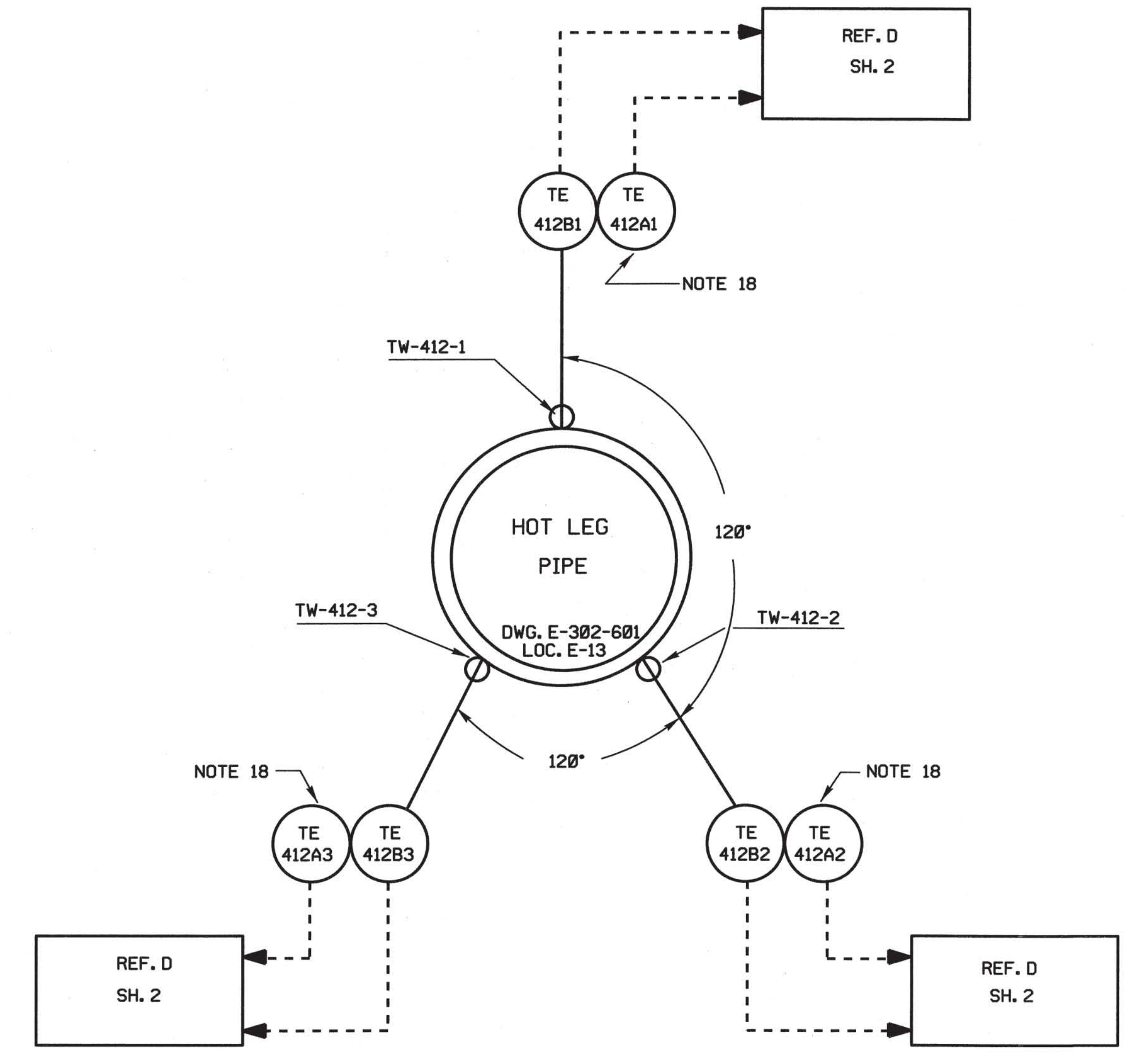
REACTOR COOLANT SYSTEM

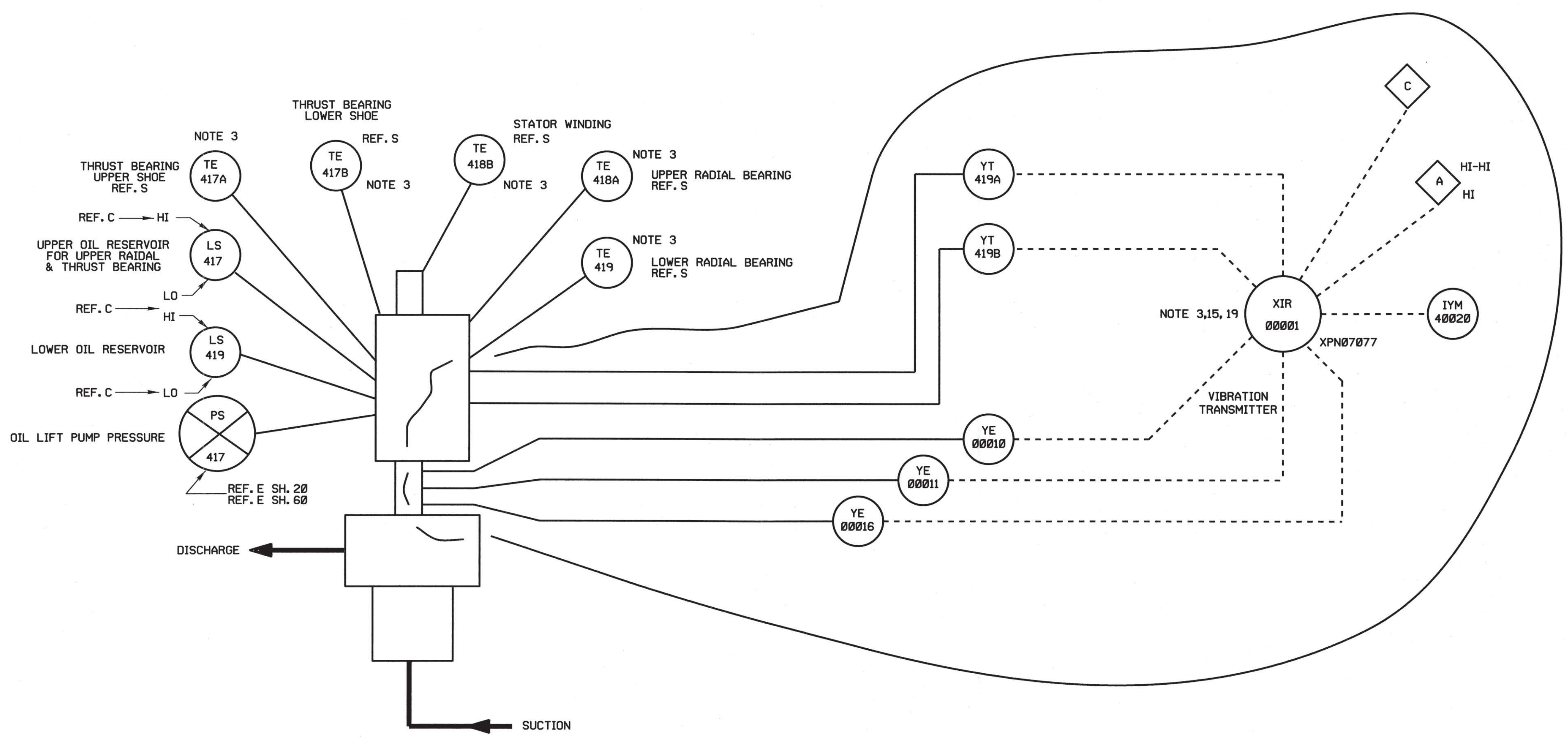
NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
28	4/24/07	DDJ	REVISED PER ECR-70100	MGR	CCB
27	10/03/06	DDJ	REVISED PER ECR-70902	RHM	DDJ
26	02/07/00	JTS	REVISED PER MRF-22355	MGR	CCB
25	4/20/99	DDJ	REVISED PER MRF-22771	RHM	CCB
24	01/07/98	JTS	REVISED PER CGSS-97-0569	LEK	MGR
29	3/6/12	JMR	REVISED PER ECR-71669	MGR	GK

SCE&G A SCANA COMPANY — DESIGN ENGINEERING — V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. SRF	2. CRR	3. TRC

SCALE: NO SCALE — DRAWING NUMBER: E-302-602 — SHT. NUMBER: — REV: 29





REACTOR COOLANT PUMP (A)
INSTRUMENTATION

NOTES:

1. THIS DRAWING IS BASED UPON DWG. 114E072, SHEET 3 OF 3, REVISION 12 (BASE DRAWING) OF WESTINGHOUSE ELECTRIC CORPORATION, NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA WHO IS SOLELY RESPONSIBLE FOR THE ACCURACY OR THE RELIABILITY OF THE DESIGN INFORMATION SET FORTH IN THE BASE DRAWING.
2. FOR ALPHA REFERENCES, SEE DWG. E-302-002, FLOW DIAGRAM LEGEND.
3. TEMPERATURE ELEMENTS AND TRANSMITTERS THAT PROVIDE INPUT TO THE PLANT COMPUTER.
4. FOR CONVENTIONAL PIPING SPECIFICATIONS, SEE GAI SPECIFICATION SP-329-4461-00, PAGE 29, (WESTINGHOUSE PIPE CLASS CONVERSION TO ENGINEER'S PIPE LINE SPECIFICATION).
5.
6.
7.
8.
9. ALL THERMOWELL CONNECTIONS SHALL HAVE REMOVABLE INSULATION.
10.
11.
12.
13.
14.
15. LOCAL STATION PROVIDED OUTSIDE CONTAINMENT FOR MONITORING PUMP VIBRATION.
16.
17.
18. DUAL ELEMENT RTD; ONE ACTIVE & ONE SPARE.
19. FOR LOCK ROTOR TRIP SEE DRAWING 208-082, SHEET RC18.

ESSENTIAL

RN 13-001

NUCLEAR SAFETY RELATED

FSAR Figure 5.1–1 SH. 3A

DRAWING LEGIBILITY
CLASS 1
SCE&G CAD ENHANCED

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION

PIPING — SYSTEM FLOW DIAGRAM
REACTOR COOLANT

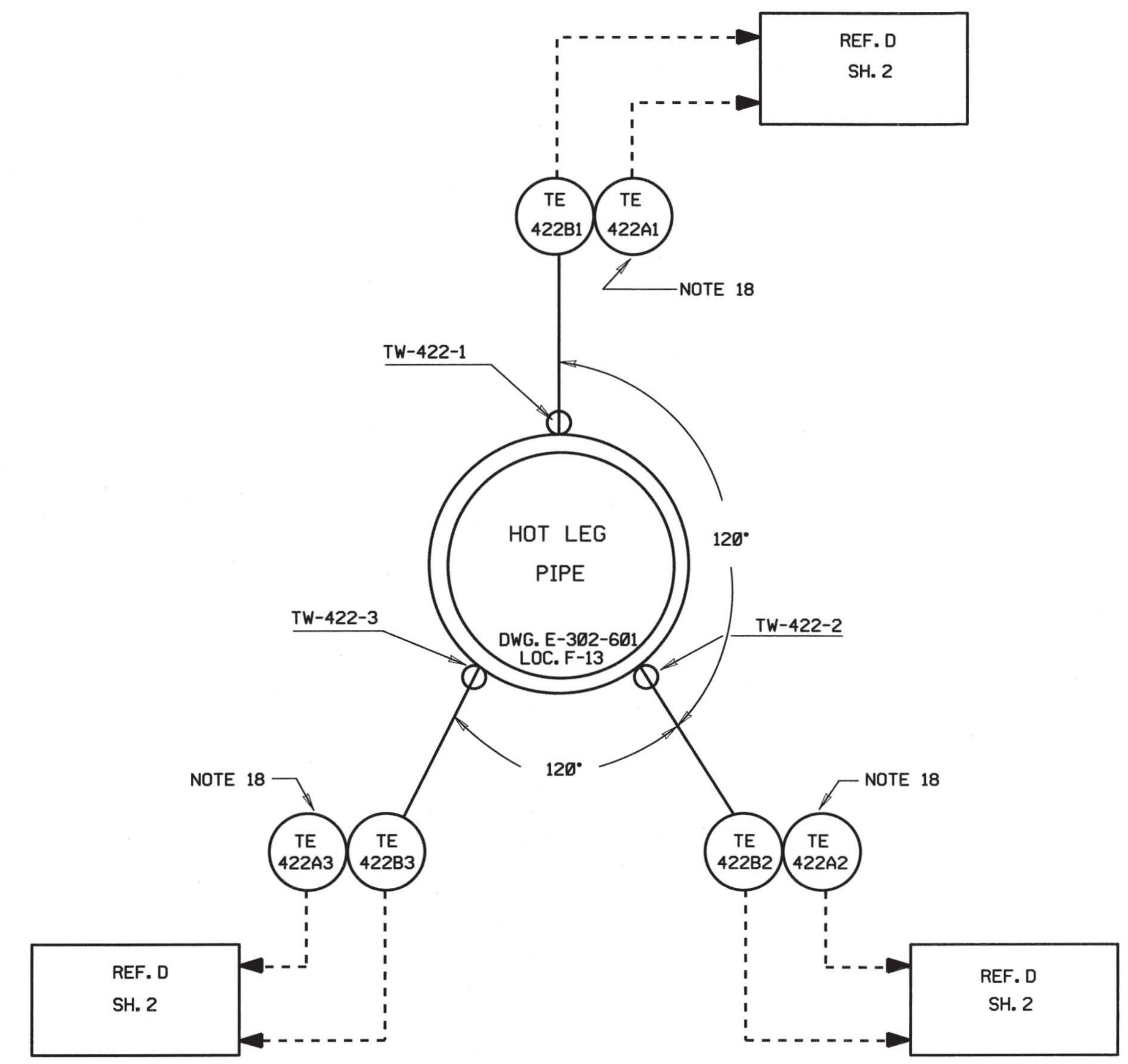
NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
7	5/11/11	RHM	REVISED PER ECR-50683	MGR	MM
6	5/16/03	JMR	CADD ENHANCED PER ECR-50239	MGR	DDJ
5	1/12/98	JMR	REVISED PER CGSS-97-0569	LEK	MGR

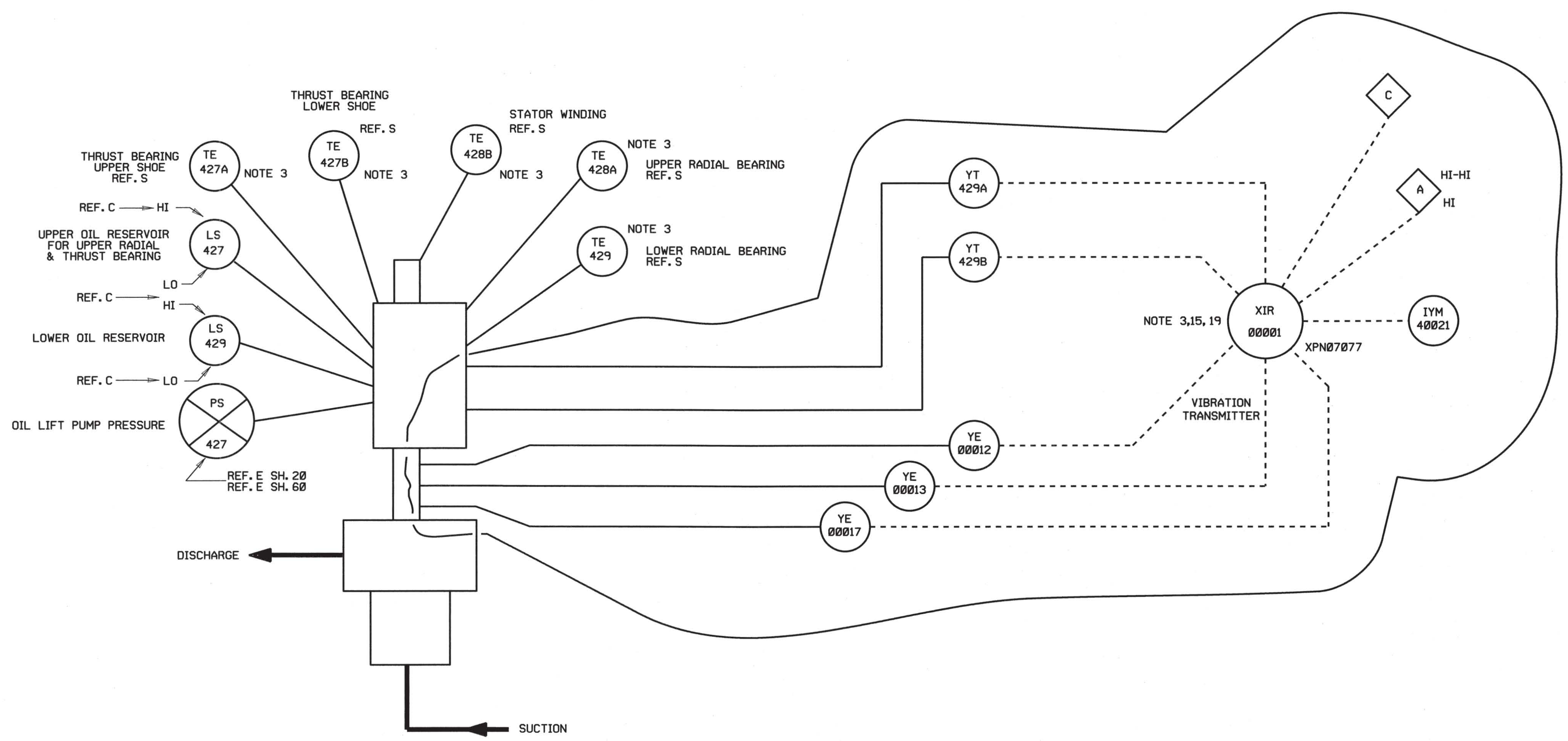
SCE&G A SCANA COMPANY
DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. JMR	2. LEK	3. MGR

SCALE NO SCALE — DRAWING NUMBER E-302-603 — SHT. NUMBER — REV 7

1/4" = 1'-0" 0' 5' 10' 15' 20'





REACTOR COOLANT PUMP B INSTRUMENTATION

NOTES:
1. THIS DRAWING IS BASED UPON DWG. 114E072, SHEET 3 OF 3, REVISION 12 ("BASE DRAWING") OF WESTINGHOUSE ELECTRIC CORPORATION, NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA WHO IS SOLELY RESPONSIBLE FOR THE ACCURACY OR THE RELIABILITY OF THE DESIGN INFORMATION SET FORTH IN THE BASE DRAWING.
2. FOR ALPHA REFERENCES, SEE DWG. E-302-002, FLOW DIAGRAM LEGEND.
3. TEMPERATURE ELEMENTS AND TRANSMITTERS THAT PROVIDE INPUT TO THE PLANT COMPUTER.
4. FOR CONVENTIONAL PIPING SPECIFICATIONS, SEE GAI SPECIFICATION SP-329-4461-00, PAGE 29, (WESTINGHOUSE PIPE CLASS CONVERSION TO ENGINEER'S PIPE LINE SPECIFICATION).
5.
6.
7.
8.
9. ALL THERMOWELL CONNECTIONS SHALL HAVE REMOVABLE INSULATION.
10.
11.
12.
13.
14.
15. LOCAL STATION PROVIDED OUTSIDE CONTAINMENT FOR MONITORING PUMP VIBRATION.
16.
17.
18. DUAL ELEMENT RTD; ONE ACTIVE AND ONE SPARE.
19. FOR LOCK ROTOR TRIP SEE DRAWING 208-082, SHEET RC18.

ESSENTIAL

DRAWING LEGIBILITY CLASS 1
SCE&G CAD ENHANCED

RN 13-001

NUCLEAR SAFETY RELATED

FSAR Figure 5.1-1 SH. 3B

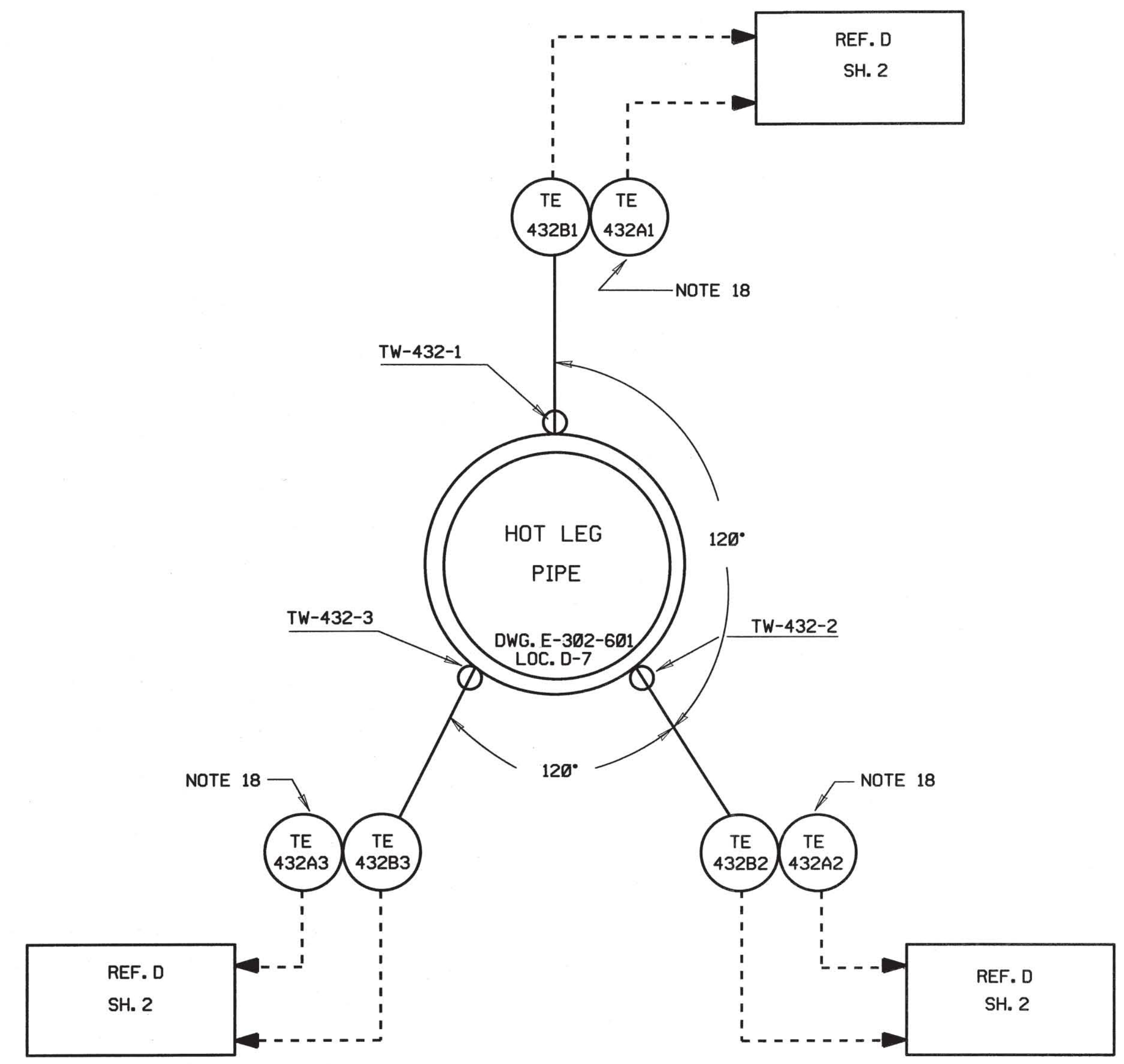
SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
PIPING — SYSTEM FLOW DIAGRAM — REACTOR COOLANT
DESIGN ENGINEERING

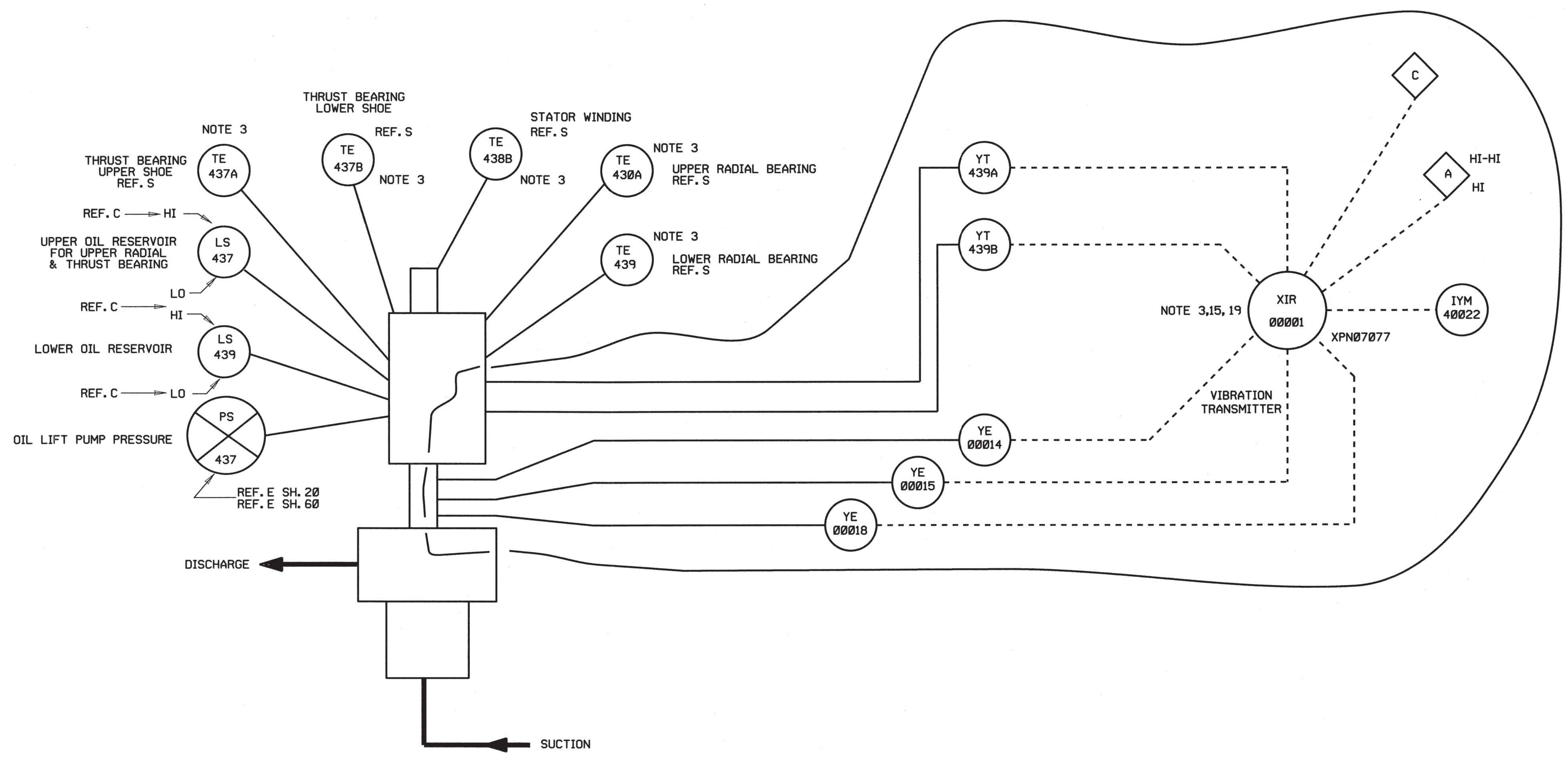
MADE	CHECKED	LE APPROVAL
JMR	LEK	MGR

DRAWING NUMBER: E-302-604 — SHT. NUMBER: 8

NO.	DATE	BY	REVISION	CKD. BY	APPROVAL
8	3/3/11	RHM	REVISED PER ECR-50683	MGR	MM
7	5/20/03	JMR	CADD ENHANCED PER ECR-50239	MGR	DDJ
6	1/13/98	JMR	REVISED PER CGSS-97-0569	LEK	MGR

1/4" = 1'-0" 0' 5' 10' 15' 20'





REACTOR COOLANT PUMP C
INSTRUMENTATION

NOTES:

1. THIS DRAWING IS BASED UPON DWG. 114E072, SHEET 3 OF 3, REVISION 12 (BASE DRAWING) OF WESTINGHOUSE ELECTRIC CORPORATION, NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA WHO IS SOLELY RESPONSIBLE FOR THE ACCURACY OR THE RELIABILITY OF THE DESIGN INFORMATION SET FORTH IN THE BASE DRAWING.
2. FOR ALPHA REFERENCES, SEE DWG. E-302-002, FLOW DIAGRAM LEGEND.
3. TEMPERATURE ELEMENTS AND TRANSMITTERS THAT PROVIDE INPUT TO THE PLANT COMPUTER.
4. FOR CONVENTIONAL PIPING SPECIFICATIONS, SEE GAI SPECIFICATION SP-329-4461-00, PAGE 29, (WESTINGHOUSE PIPE CLASS CONVERSION TO ENGINEER'S PIPE LINE SPECIFICATION).
5.
6.
7.
8. ALL THERMOWELL CONNECTIONS SHALL HAVE REMOVABLE INSULATION.
9.
10.
11.
12.
13.
14.
15. LOCAL STATION PROVIDED OUTSIDE CONTAINMENT FOR MONITORING PUMP VIBRATION.
16.
17.
18. DUAL ELEMENT RTD; ONE ACTIVE & ONE SPARE.
19. FOR LOCK ROTOR SEE DRAWING 208-082, SHEET RC18.

ESSENTIAL

RN 13-001

NUCLEAR SAFETY RELATED

FSAR Figure 5.1–1–3C

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

PIPING SYSTEM FLOW DIAGRAM
REACTOR COOLANT

DRAWING LEGIBILITY
CLASS 1
SCE&G CAD ENHANCED

SCE&G A SCANA COMPANY
DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

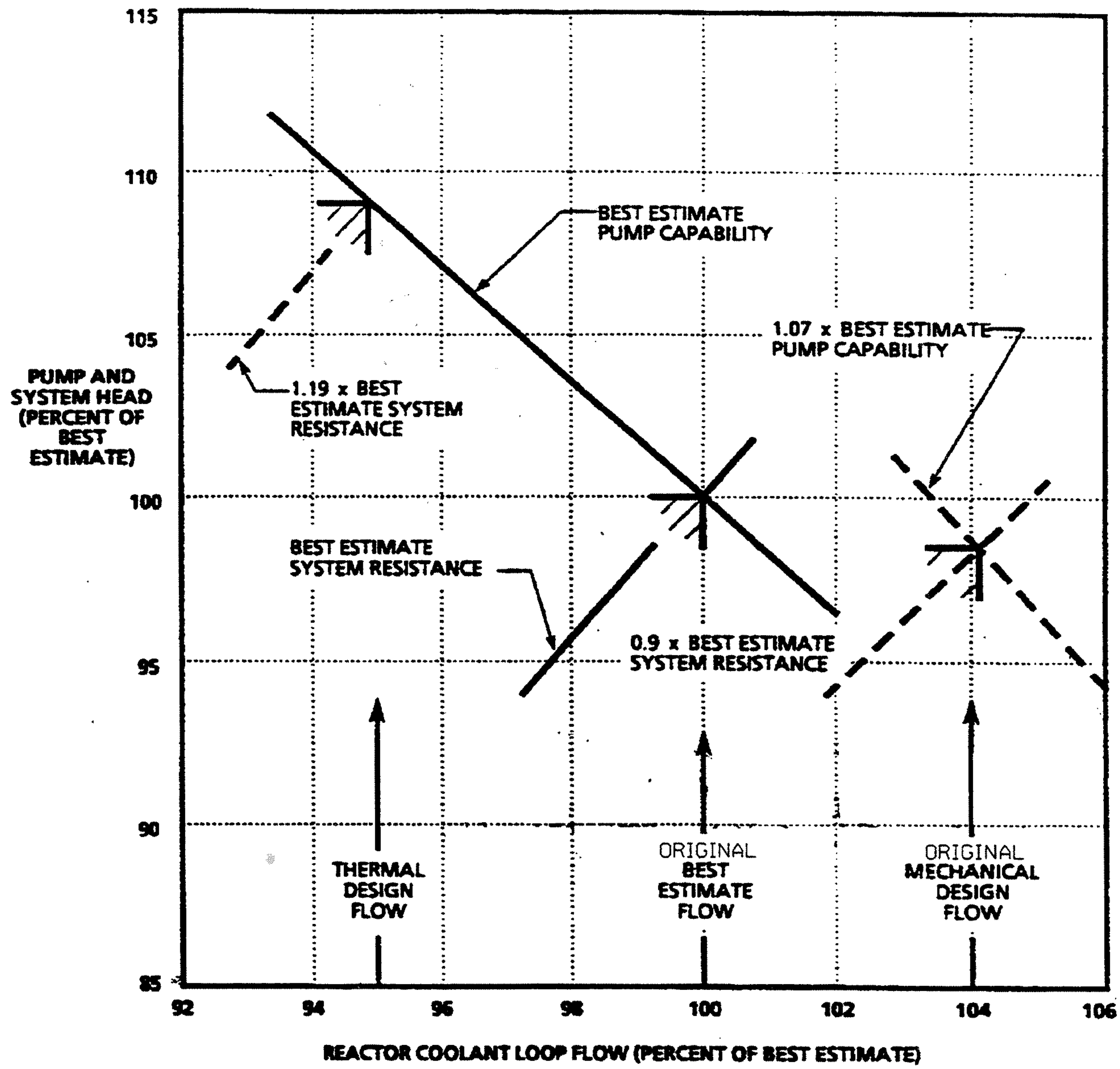
MADE	CHECKED	LE APPROVAL
1. JMR	2. LEK	3. MGR

SCALE NO SCALE

DRAWING NUMBER: E-302-605 | SHT. NUMBER: | REV: 7

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
7	5/11/11	RHM	REVISED PER ECR-50683	MGR	MM
6	5/27/03	JMR	CADD ENHANCED PER ECR-50239	MGR	DDJ
5	1/12/98	JMR	REVISED PER CGSS-97-0569	LEK	MGR

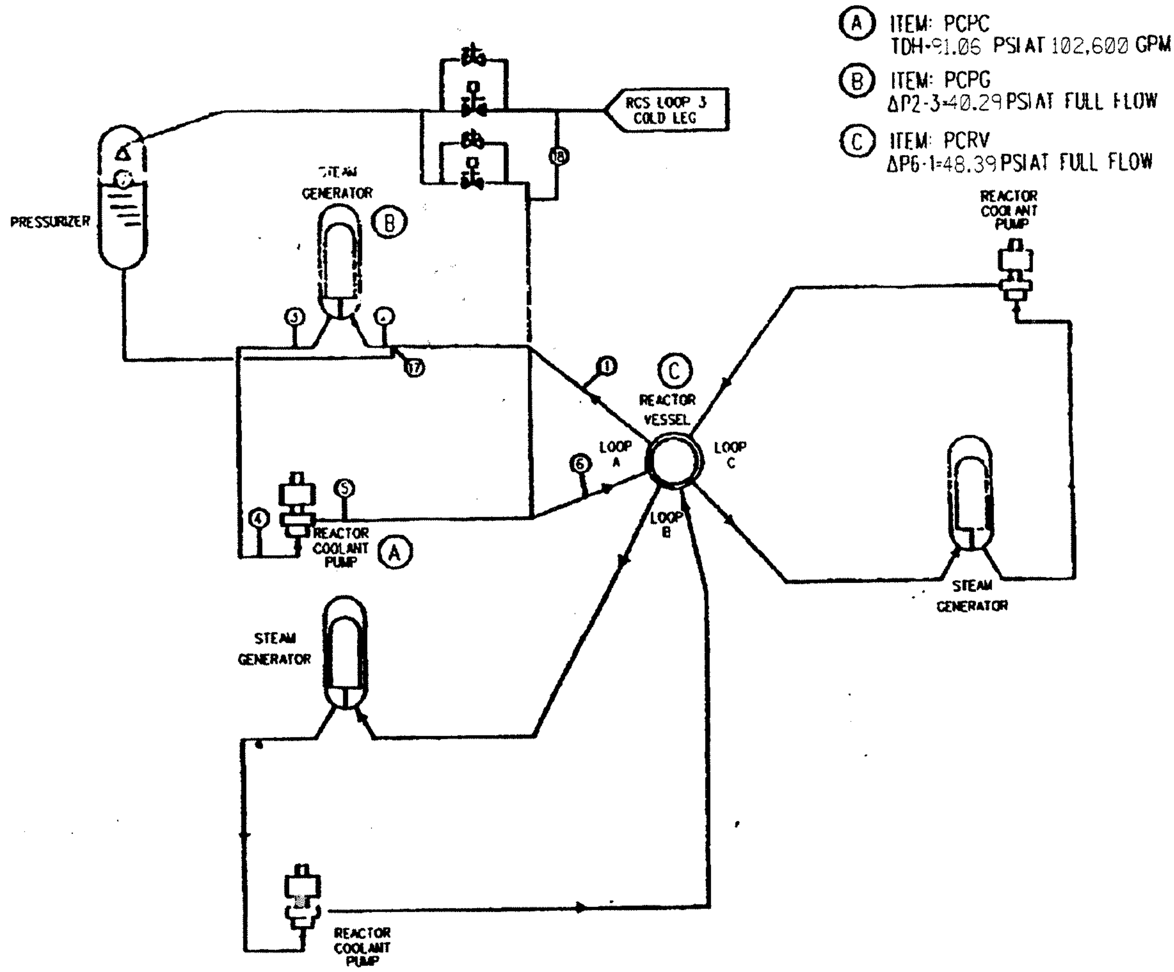
1/4" = 1'-0" 0' 5' 10' 15' 20'



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Pump Head Flow Characteristics
Original Design Consideration

Figure 5.1-2

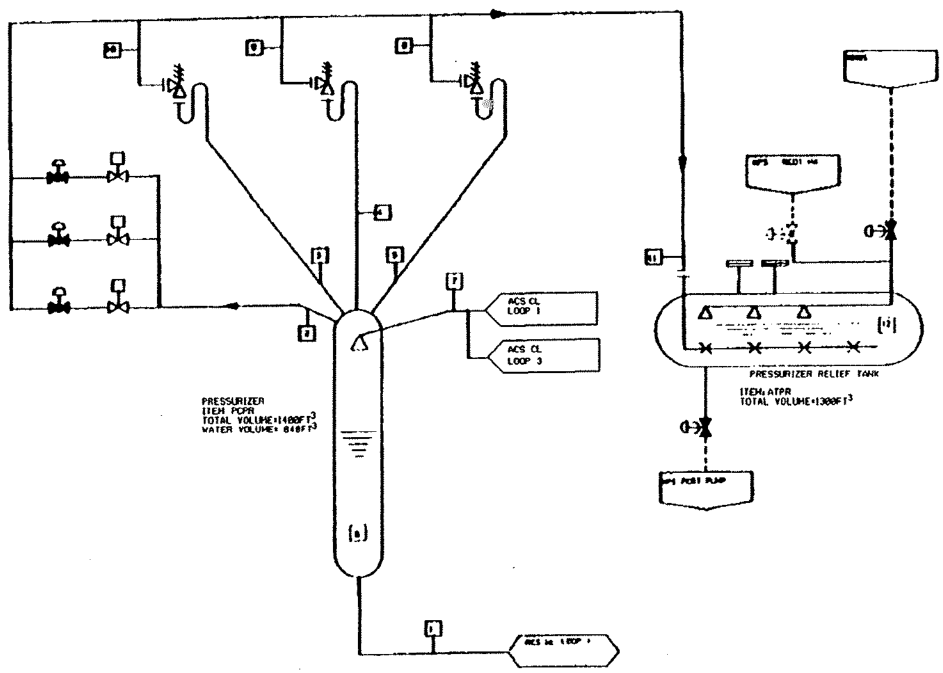


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Coolant System Process Flow Diagram (Sheet 1 of 3)

Figure 5.1-3

AMENDMENT 96-02
JULY 1996



NOTES:

PIPING SIZE AND CLASS HAVE NOT BEEN INDICATED.
REFER TO (W) PIPING AND INSTRUMENTATION DIAGRAM.

BASED ON Tavg=587.4°F

NORMAL STEADY STATE 100% POWER

LOCATION	1	2	3	4	5	6	7	8	9	10	11	12
FLOW GPM	2.5	0	0	0	0	-	2	0	0	0	0	-
PRESSURE PSIG	2235	2235	2235	2235	2235	2235	2235	3.0	3.0	3.0	3.0	3.0
TEMPERATURE	652	652	652	652	652	652	552.6	120	120	120	120	120

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Coolant System Process Flow Diagram (Sheet 2 of 3)

Figure 5.1-3

AMENDMENT 96-02
JULY 1996

ACCUMULATOR TANKS

NOMINAL TANK VOLUME - 1450 FT.^{3}

NOMINAL WATER VOLUME - 1000 FT.^{3}

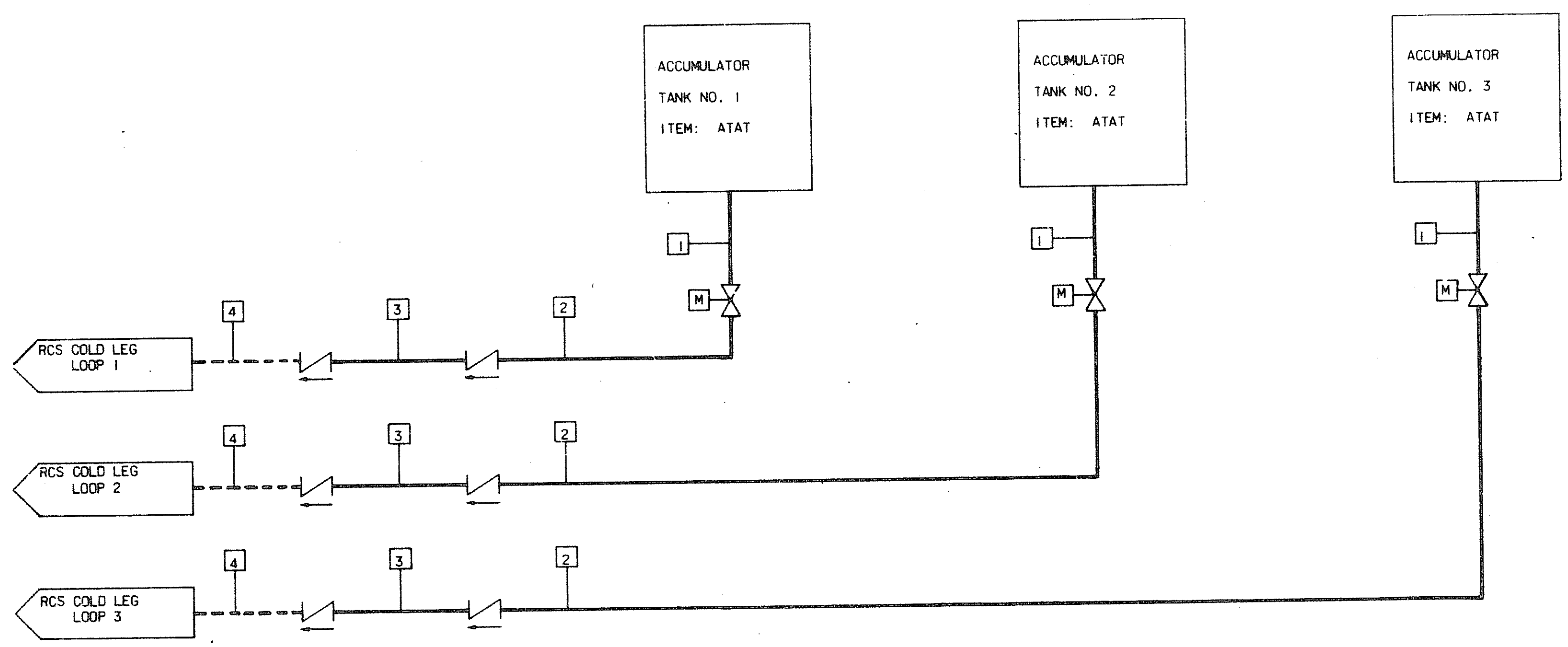
NOMINAL GAS PRESSURE - 640 PSIG

NORMAL TEMPERATURE - 120°F

NOTES:

1. PIPING SIZE AND CLASS HAVE NOT BEEN INDICATED. REFER TO W PIPING AND INSTRUMENTATION DIAGRAM.

2. ACCUMULATORS DISCHARGE TO THE REACTOR COOLANT SYSTEM DURING THE REACTOR COOLANT SYSTEM BLOWDOWN PRESSURE TRANSIENT. ACTUAL FLOW RATES DEPEND ON TYPE OF LOSS OF COOLANT ACCIDENT AND SUBSEQUENT PRESSURE TRANSIENT. FLOW DATA PRESENTED IS THE MAXIMUM FLOW THAT OCCURS FOR DOUBLE ENDED. GUILLOTINE COLD LEG LOSS OF COOLANT ACCIDENT.



TYPICAL ACCUMULATOR BLOWDOWN (DOUBLE-ENDED GUILLOTINE COLD LEG LOSS OF COOLANT ACCIDENT - NOTE 2)

LOCATION	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
FLOW - GPM	15000	15000	15000	15000																		
PRESSURE - PSIG	NOTE 2	NOTE 2	NOTE 2	NOTE 2																		
TEMPERATURE - °F	120	120	120	120																		

Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Emergency Core Cooling System
Process Flow Diagram
(Sheet 3 of 3)

Figure 5.1-3

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section presents the discussion of the measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) for the plant design lifetime. In this context, the RCPB is as defined in Section 50.2 of 10 CFR 50. In that definition, the RCPB extends to the outermost containment isolation valve in system piping which penetrates the containment and is connected to the reactor coolant system (RCS). Since other sections of this report already describe the components of these auxiliary fluid systems in detail, the discussion in this section is limited to the components of the RCS as defined in Section 5.1, unless otherwise noted.

For additional information on the components which are part of the RCPB (as defined in 10 CFR 50), and which are not described in this section, refer to the following sections:

Section 6.3 - For discussions of the RCPB components which are part of the Emergency Core Cooling System.

Section 9.3.4 - For discussions of the RCPB components which are part of the Chemical and Volume Control System.

Section 3.9.2 - For discussions of the design loadings, stress limits and analyses applied to ASME Code Class 2 and 3 components.

The phrase, RCS, as used in this section is as defined in Section 5.1. When the term RCPB is used in this section, its definition is that of Section 50.2 of 10 CFR 50.

5.2.1 DESIGN OF REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

5.2.1.1 Performance Objectives

The performance objectives of RCS are described in Section 5.1. The equipment code and classification list for the components within the RCS are given in Table 3.2-1.

The RCS, in conjunction with the reactor control and protection systems, is designed to maintain the reactor coolant at conditions of temperature, pressure and flow adequate to protect the core. The design requirement for safety is to prevent conditions of high power, high reactor coolant temperature or low reactor coolant pressure or combinations of these which could result in a departure from nucleate boiling ratio (DNBR) less than the design limit.

The RCS is designed to provide controlled changes in the boric acid concentration and the reactor coolant temperature. The reactor coolant is the moderator, reflector, and solvent for the chemical shim. As a result, changes in the coolant temperature or boric acid concentration affect the reactivity level in the core.

Whenever the boron concentration of the RCS is changed, plant operation will be such that good mixing is provided in order to ensure that the boron concentration is maintained uniformly throughout the RCS.

The following design bases have been selected to ensure that the uniform RCS boron concentration and temperature will be maintained:

1. Coolant flow is provided by a reactor coolant pump to ensure uniform mixing whenever the boron concentration is changed.

2. The design arrangement of the RCS eliminates deadended sections and other areas of low coolant flow in which nonhomogeneities in coolant temperature or boron concentration could develop.

3. The RCS is designed to operate within the operating parameters, particularly the coolant temperature change limitations.

Before plant cooldown is initiated, the boron concentration in the RCS is increased to the cold shutdown concentration, and the concentration is verified by sampling. Thus during reactor cooldown, no changes are imposed on the boron concentration.

It is therefore concluded that the temperature changes imposed on the RCS during its normal modes of operation do not cause any abnormal or unacceptable reactivity changes.

Also, the design of the RCS is such that the distribution of flow around the system is not subject to the degree of variation which would be required to produce nonhomogeneities in coolant temperature or boron concentration as a result of areas of low coolant flow rate. An exception to this is the pressurizer, but it is of no concern since, as stated in the paragraph above, temperature changes do not cause unacceptable reactivity changes. Operation with 1 reactor coolant pump inoperable is possible under certain conditions and, in this case, there would be backflow in the associated loop, even though the pump itself is prevented from rotating backwards by its anti-rotation device. The backflow through the loop would cause departure from the normal temperature distribution around the loop but would maintain the boron concentration in the loop the same as that in the remainder of the RCS.

The range of coolant temperature variation during normal operation is limited and the associated reactivity change is well within the capability of the control rod group movement.

The RCS provides for heat transfer from the reactor to the steam generators under conditions of forced circulation flow and natural circulation flow. The heat transfer capabilities of the RCS are analyzed in Chapter 15 for various transients.

The heat transfer capability of the steam generators is sufficient to transfer, to the steam and power conversion system, the heat generated during normal operation, and during the initial phase of plant cooldown under natural circulation conditions.

During the second phase of plant cooldown and during cold shutdown and refueling, the heat exchangers of the Residual Heat Removal System are employed. Their capability is discussed in Section 5.5.7.

The pumps of the RCS assure heat transfer by forced circulation flow. Design flow rates are discussed in conjunction with the reactor coolant pump description in Section 5.5.1.

Initial RCS tests are performed to determine the total delivery capability of the reactor coolant pumps. Thus, it is confirmed prior to plant operation that adequate circulation is provided by the RCS.

To assure a heat sink for the reactor under conditions of natural circulation flow, the steam generators are at a higher elevation than the reactor. In the design of the steam generators, consideration is given to provide adequate tube area to ensure that the residual heat removal rate is achieved with natural circulation flow.

5.2.1.2 Design Parameters

The design pressure for the RCS is 2485 psig, except for the pressurizer relief line from the safety valves to the pressurizer relief tank, which is 600 psig, and the pressurizer relief tank, which is 100 psig. The design temperature for the RCS is 650°F, except for the pressurizer and its surge line and relief and safety valve inlet lines which are designed for 680°F, and the pressurizer relief line from the safety valve to the pressurizer relief tank, which is designed for 600°F.

Design parameters for the individual components in the RCS are given in the respective component description subsections of Section 5.5. Component hydrostatic testing is accomplished in accordance with the requirements of the ASME Code, Section III. System hydrostatic testing is discussed in Section 5.2.1.5.

5.2.1.3 Compliance with 10 CFR 50, Section 50.55a

RCS components are designed and fabricated in accordance with the rules of 10 CFR 50, Section 50.55a, "Codes and Standards". The actual addenda of the ASME Code applied in the design of each RCS component is listed in Table 5.2-1.

5.2.1.4 Applicable Code Cases

Regulatory Guides 1.84 and 1.85 are discussed in Appendix 3A.

The ASME Code Cases applicable to the manufacture of the Delta-75 replacement steam generators for the V. C. Summer Nuclear Station are:

N-20-3, applicable to the procurement and manufacture of SB-163 tubing

N-40-1, applicable to eddy current inspection data storage

N-474-1, applicable to Alloy 690 material other than tubing

2142, applicable to UNSNO6052 weld consumables

2143, applicable to UNSW86152 weld consumables

Code cases for which the NRC Staff has provided a disposition are listed in Regulatory Guides 1.84 and 1.85.

Code Case N-60-4 is issued for the reactor internals replacement guide tube support pins and nuts. The limitation on maximum yield strength for the code case included in Regulatory Guide 1.85 is satisfied. RN 01-086

5.2.1.5 Design Transients

The following 5 ASME operating conditions are considered in the design of the RCS.

1. Normal Conditions

   Any condition in the course of startup, operation in the design power range, hot standby and system shutdown, other than upset, emergency, faulted or testing conditions.

2. Upset Conditions (Incidents of Moderate Frequency)

   Any deviation from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system and transients due to loss of load or power. Upset conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of an upset condition shall be included in the design specifications.

3. Emergency Conditions (Infrequent Incidents)

   Those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events shall not cause more than 25 stress cycles having an S_a value greater than that for 10^6 cycles from the applicable fatigue design curves of the ASME Code, Section III.

4. Faulted Conditions (Limiting Faults)

   Those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that consideration of public health and safety is involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

5. Testing Conditions

   Testing conditions are those pressure overload tests including hydrostatic tests, pneumatic tests, and leak tests specified. Other types of tests shall be classified under normal, upset, emergency or faulted conditions.

To provide the necessary high degree of integrity for the equipment in the RCS, the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. To a large extent, the specific transient operating conditions to be considered for equipment fatigue analyses are based upon engineering judgment and experience. The transients selected are representative of operating conditions which prudently should be considered to occur during plant operation and are sufficiently severe or frequent to be of possible significance to component cyclic behavior. The transients selected may be regarded as a conservative representation of transients which, used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of the plant.

The following design conditions are given in the equipment specifications for RCS components.

The design transients and the number of cycles of each that is normally used for fatigue evaluations are shown in Table 5.2-2. In accordance with the ASME Code, faulted conditions are not included in fatigue evaluations.

Loading combination and allowable stresses for ASME Code, Section III, Class 1 components and supports are given in Tables 5.2-3 and 5.2-4.

5.2.1.5.1 Normal Conditions

The following primary system transients are considered normal conditions:

1. Heatup and cooldown at 100°F per hour.

2. Unit loading and unloading at 5% of full power per minute.

3. Step load increase and decrease of 10% of full power.

4. Large step load decrease with steam dump.

5. Steady-state fluctuations.

6. Thermal stratification in the surge line.

Additionally, a new transient was identified when the Model D3 steam generators were replaced with Delta-75 steam generators. The transient is entitled "Feedwater Heaters out of Service" and bounds the entire spectrum of feedwater heater malfunctions (both high pressure and low pressure heaters).

1. Heatup and Cooldown at 100°F per Hour

   The design heatup and cooldown cases are conservatively represented by continuous operations performed at a uniform temperature rate of 100°F per hour. (These operations can take place at lower rates approaching the minimum of 0°F per hour. The expected normal rates are 50°F per hour). For these cases, the heatup occurs from ambient (assumed to be 120°F) to the no-load temperature and pressure condition, and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F per hour will not be attained because of other limitations such as:

   a. Material ductility considerations which establish maximum permissible temperature rates of change, as a function of plant pressure and temperature, which are below the design rate of 100°F per hour.

   b. Slower initial heatup rates when using pump energy only.

   c. Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry and gas adjustments.

2. Unit Loading and Unloading at 5% of Full Power per Minute

The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5% per minute between 15% load and full load. This load swing is the maximum possible, consistent with operation under automatic reactor control. The reactor temperature will vary with load as prescribed by the reactor control system.

3. Step Load Increase and Decrease of 10% of Full Power

The ± 10% step change in load demand is a transient which is assumed to be a change in turbine control valve opening due to disturbances in the electrical network into which the plant output is tied. The reactor control system is designed to restore plant equilibrium without reactor trip following a ± 10% step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15% and 100% full load, the power range for automatic reactor control. In effect, during load change conditions, the reactor control system attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed setpoint at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step decrease in turbine load, the secondary side steam pressure and temperature initially increase, since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the RCS average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With the load decrease, the reactor coolant temperature will ultimately be reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature setpoint change is made as a function of turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash, which reduces the rate of pressure decrease. Subsequently, the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step increase in turbine load, the reverse situation occurs; i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature will be raised to a value above its initial equilibrium value at the beginning of the transient.

4. Large Step Load Decrease With Steam Dump

This transient applies to a step decrease in turbine load from full power of such magnitude, that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature will automatically initiate a secondary side steam dump that will prevent both reactor trip and lifting of steam generator safety valves. Thus, since the Virgil C. Summer Nuclear Station is designed to accept a step decrease of 95% from full power (complete loss of outside load, but retaining the plant auxiliary load of 5%), the steam dump system provides the heat sink to accept approximately 93.6% of the turbine load. The remaining percentage of the total step change is assumed by the reactor control system control rods. If a steam dump system was not provided to cope with this transient, there would be such a strong mismatch between what the turbine is demanding and what the reactor is delivering that a reactor trip and lifting of steam generator safety valves would occur.

5. Steady-State Fluctuations

It is assumed that the reactor coolant temperature and pressure at any point in the system vary around the nominal (steady-state) values. For design purposes, two cases are considered:

a. Initial Fluctuations

These are due to control rod cycling during the first 20 full power months of reactor operation. Temperature is assumed to vary ± 3°F and pressure by ± 25 psi, once during each 2 minute period. The total number of occurrences is limited to 1.5×10^5. These fluctuations are assumed to occur consecutively, and not simultaneously with the random fluctuations.

b. Random Fluctuations

Temperature is assumed to vary by ± 0.5°F and pressure by ± 6 psi, once every 6 minutes. With a 6 minute period, the total number of occurrences during the plant design life does not exceed 3.0×10^6.

6. Thermal Stratification in the Surge Line

On December 20, 1988, the NRC issued Bulletin No. 88-11 regarding pressurizer surge line thermal stratification. The Bulletin requested utilities to establish and implement a program to confirm pressurizer surge line integrity in view of the occurrence of thermal stratification.

Thermal stratification in the pressurizer surge line is the direct result of the difference in densities between the pressurizer water and the generally cooler hot leg water. The lighter pressurizer water tends to float on the cooler heavier hot leg water. The potential for stratification is increased as the difference in temperature between the pressurizer and the hot leg increases and as the insurge or outsurge flow rates decrease.

At power, when the difference in temperature between pressurizer and hot leg is relatively small, the extent and effects of stratification have been observed to be small. However, during certain modes of plant heatup and cooldown, this difference in system temperature could be as large as 320°F, in which case the effects of stratification must be accounted for.

The heatup and cooldown transients for the surge line were developed from monitoring data, historical operation and operator interviews conducted at a large number of plants. The transients were built upon the extensive work done for the Westinghouse Owners Group coupled with plant specific considerations for Virgil C. Summer.

Transients in the surge line are characterized as either insurges, outsurges, or fluctuations. Insurges and outsurges are the more severe transients and result in the greatest change in temperature in the top or bottom of the pipe. An insurge may cool the entire pipe cross section significantly, to very close to the temperature of the RCS hot leg. Conversely, an outsurge can sweep the line and heat the pipe to close to the temperature of the pressurizer.

Fluctuations, as opposed to the insurge-outsurge transients, are caused by relatively insignificant surges and result in variations in the hot-cold interface level. These variations in the interface level do not change the overall global displacement of the pipe and, hence, are analyzed as changes in local stress only.

The redefinition of the thermal fluid transients experienced by the surge line during normal conditions was necessary to reflect the indirectly observed fluid temperature distributions. These redefined thermal fluid transients were developed based on the existing design transient system parameters assumed to exist at the time of the postulated transient and the knowledge gained from the monitoring programs. The redefined thermal fluid transients conservatively account for the thermal stratification phenomena.

7. Feedwater Heaters Out of Service

These transients occur when one or more feedwater heaters are taken out of service. For conservatism, it is assumed that one entire train of high pressure heaters are removed from service, thereby making the transient bounding with regard to the possibility of either a low pressure or high pressure heater train being removed from service. Upon identification of the transient, power should be

reduced to 80% of nominal, and the plant allowed to stabilize. The affected heater train must be taken out of service.

5.2.1.5.2 Upset Conditions

The following primary system transients are considered upset conditions:

1. Loss of load (without immediate reactor trip).
2. Loss of power.
3. Partial loss of flow.
4. Reactor trip from full power.
5. Inadvertent auxiliary spray.
6. Operating basis earthquake.
7. Excessive feedwater flow.

1. Loss of Load (Without Immediate Reactor Trip)

   This transient applies to a step decrease in turbine load from full power (turbine trip) without immediately initiating a reactor trip and represents the most severe pressure transient on the RCS under upset conditions. The reactor eventually trips as a consequence of a high pressurizer level trip initiated by the Reactor Protection System (RPS). Since redundant means of tripping the reactor are provided as part of the RPS, transients of this nature are not expected, but are included to ensure a conservative design.

2. Loss of Power

   This transient applies to a blackout situation involving the loss of all outside electrical power to the station, followed by reactor and turbine trips. Under these circumstances, the reactor coolant pumps are de-energized and, following coastdown of the reactor coolant pumps, natural circulation builds up in the system to some equilibrium value. This condition permits removal of core residual heat through the steam generators which at this time are receiving feedwater from the Emergency Feedwater System. Steam is removed for reactor cooldown through atmospheric relief valves provided for this purpose.

3. Partial Loss of Flow

   This transient applies to a partial loss of flow from full power, in which a reactor coolant pump is tripped out of service as the result of a loss of power to that pump. The consequences of such an accident are a reactor and turbine trip on low

reactor coolant flow, followed by automatic opening of the steam dump system and flow reversal in the affected loop. The flow reversal causes reactor coolant, at cold leg temperature, to pass through the steam generator and be cooled still further. This cooler water then flows through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizable reduction in the hot leg coolant temperature of the affected loop.

4. Reactor Trip From Full Power

A reactor trip from full power may occur from a variety of causes resulting in temperature and pressure transients in the RCS and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of steam remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the RPS causes the control rods to move into the core.

5. Inadvertent Auxiliary Spray

The inadvertent pressurizer auxiliary spray transient will occur if the auxiliary spray valve is opened inadvertently during normal operation of the plant. This will introduce cold water into the pressurizer with a very sharp pressure decrease as a result.

6. Operating Basis Earthquake

The mechanical stresses resulting from the Operating Basis Earthquake (OBE) are considered on a component basis. Fatigue analysis, where required by the codes, is performed by the supplier as part of the stress analysis report. The earthquake loads are a part of the mechanical loading conditions specified in the equipment specifications. The origin of their determination is separate and distinct from those transients resulting from fluid pressure and temperature. They are, however, considered in the design analysis. The number of occurrences for fatigue evaluation is given in Section 3.7.3.1.

7. Excessive Feedwater Flow

The pressure and temperature variations associated with this transient are considered in connection with analyzing the primary and secondary sides of the steam generator, the Reactor Coolant System, and the pressurizer.

This transient is conservatively defined as an umbrella case to cover the occurrence of several events of the same general nature. These include:

- Inadvertent opening of a feedwater control valve;

- Turbine overspeed (110%) with an open feedwater control valve;

- Small steam break with an open feedwater control valve.

The excessive feedwater flow transient results from inadvertent opening of a feedwater control valve when the plant is at hot shutdown and the steam generator is in the no-load condition. The Feedwater, Condensate, and Heater Drain Systems are in operation. The stem of a feedwater control valve has been assumed to fail, with the valve immediately reaching the full-open position. The feedwater flow to the affected loop is assumed to start from essentially zero flow to the value determined by the system resistance and the developed head of all operating feedwater pumps, with no main feedwater flow. Steam flow is assumed to remain at zero, and the temperature of the feedwater entering the steam generator is conservatively assumed to be 32°F. Feedwater flow is isolated on a reactor coolant low T_{avg} signal; a low pressurizer pressure signal actuates the Safety Injection System. Emergency feedwater flow, initiated by the safety injection signal, is assumed to continue with all pumps discharging into the affected steam generator. It is also assumed, for conservatism in the secondary side analysis, that emergency feedwater flows to the steam generators not affected by the malfunctioned valve, in the "unfailed loops." Plant conditions stabilize at the values reached in 600 seconds, at which time emergency feedwater flow is terminated. The plant is then either taken to cold shutdown, or returned to the no-load condition at a normal heatup rate with the Auxiliary Feedwater System under manual control.

For design purposes, this transient is assumed to occur 30 times during the life of the plant.

5.2.1.5.3 Emergency Conditions

No transient is classified as an emergency condition.

5.2.1.5.4 Faulted Conditions

The following primary system transients are considered faulted conditions:

1. Reactor coolant pipe break (large loss of coolant accident).

2. Large steam line break. RN 01-113

3. Steam generator tube rupture.

4. Safe shutdown earthquake. RN 01-113

1. Reactor Coolant Pipe Break (Large Loss of Coolant Accident)

   Following rupture of a reactor coolant pipe resulting in a large loss of coolant, the primary system pressure decreases causing the primary system temperature to decrease. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal will still be at or near the operating temperature by the end of blowdown. It is conservatively assumed that the ECCS is actuated to introduce water at a minimum temperature of 32°F into the RCS. The safety injection signal will also result in reactor and turbine trips.

2. Large Steam Line Break

   This transient is based on the complete severance of the largest steam line. The following conservative assumptions were made:

   a. The reactor is initially in a hot, zero-power condition.

   b. The steam line break results in immediate reactor trip and ECCS actuation.

   c. A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient.

   d. The ECCS operates at design capacity and repressurizes the RCS within a relatively short time.

   The above conditions result in the most severe temperature and pressure variations which the primary system will encounter during a steam break accident.

3. Steam Generator Tube Rupture

   This accident postulates the double ended rupture of a steam generator tube, resulting in a decrease in pressurizer level and reactor coolant pressure. Reactor trip will occur due to the resulting safety injection signal. In addition, safety injection actuation automatically isolates the feedwater lines by closing the feedwater isolation and control valves. When this accident occurs, some of the reactor coolant blows down into the affected steam generator causing the shell side level to rise. The primary system pressure is reduced below the secondary safety valve setting. Subsequent recovery procedures call for isolation of the steam line leading from the affected steam generator. This accident will result in a transient which is no more severe than that associated with a reactor trip from full power. Therefore it requires no special treatment insofar as fatigue evaluation is concerned, and no specific number of occurrences is postulated.

4. Safe Shutdown Earthquake

The mechanical dynamic or static equivalent loads due to the vibratory motion of the Safe Shutdown Earthquake (SSE) are considered on a component basis.

5.2.1.5.5 Test Conditions

The following primary system transients under test conditions are discussed:

1. Turbine roll test.
2. Primary side hydrostatic test. RN 01-113
3. Secondary side hydrostatic test.
4. Primary side leakage test.

1. Turbine Roll Test

This transient is imposed upon the plant during the hot functional test period for turbine cycle checkout. Reactor coolant pump power will be used to heat the reactor coolant to operating temperature (no-load conditions) and the steam generated will be used to perform a turbine roll test. However, the plant cooldown during this test will exceed the 100°F per hour design rate.

This transient occurs before plant startup and the number of cycles is therefore independent of other operating transients.

2. Primary Side Hydrostatic Test

The pressure tests include both shop and field hydrostatic tests which occur as a result of component or system testing. This hydrotest is performed at a water temperature which is compatible with reactor vessel material ductility requirements and a test pressure of 3107 psig (1.25 times design pressure). In this test, the RCS is pressurized to 3107 psig coincident with steam generator secondary side pressure of 0 psig. The number of cycles is independent of other operating transients.

3. Secondary Side Hydrostatic Test

The secondary side of the steam generator is pressurized to 1.25 design pressure with a minimum water temperature of 120°F coincident with the primary side at 0 psig.

These tests may be performed either prior to plant startup, or subsequently following shutdown for major repairs or both. The number of cycles is therefore independent of other operating transients.

4. Primary Side Leakage Test

   Subsequent to each time the primary system has been opened, a leakage test will be performed. During this test, the primary system pressure is, for design purposes, raised to 2500 psia, with the system temperature above the minimum temperature imposed by reactor vessel material ductility requirements, while the system is checked for leaks. In actual practice, the primary system is pressurized to less than 2500 psia as measured at the pressurizer, to prevent the pressurizer safety valves from lifting during the test.

   During this leakage test, the secondary side of the steam generator must be pressurized so that the pressure differential across the tubesheet does not exceed 1600 psi. This is accomplished with the steam, feedwater, and blowdown lines closed off.

5.2.1.6 Identification of Active Pumps and Valves

Valves in the RCPB are tabulated in Section 3.9.2. These valves are identified as being either active or inactive.

Active valves are those whose operability through a mechanical motion is relied upon to perform a safety function during the transients or events considered in each operating condition category. Inactive valves have no required motion, they must only retain their structural integrity.

There are no active pumps in the RCPB. The reactor coolant pumps are classified as inactive.

The design methods and procedures which are used to show that active valves listed in Section 3.9.2 will operate during a faulted condition are described in Section 3.9.2.4.

5.2.1.7 Design of Active Pumps and Valves

The methods that will be used to assure that pumps required to function and valves required to open or close during (or following) a specified plant condition are described in Section 3.9.2.4.

5.2.1.8 Inadvertent Operation of Valves

Those valves which are used in the isolation of the RCPB during normal plant operation, and are not relied on to function after an accident, are redundant valves and their inadvertent operation will not increase the severity of any transient.

5.2.1.9 Stress and Pressure Limits

Inactive components are designed in accordance with the ASME Code, Section III. The codes and stress limits used for Class 1 components are summarized in Section 5.2.1.10.

5.2.1.10 Stress Analysis for Structural Adequacy

5.2.1.10.1 Loading Conditions

The structural stress analyses performed on the reactor coolant system consider the loadings specified as shown in Table 5.2-3. These loads result from thermal expansion, pressure, weight, OBE, SSE, design basis loss of coolant accident, and plant operational thermal and pressure transients.

5.2.1.10.2 Analysis of the Reactor Coolant Loop and Supports

The loads used in the analysis of the reactor coolant loop piping are described in detail below:

1. Pressure

   Pressure loading is identified as either membrane design pressure or general operating pressure, depending upon its application. The membrane design pressure is used in connection with the longitudinal pressure stress and minimum wall thickness calculations in accordance with the ASME Code.

   The term operating pressure is used in connection with determination of the system deflections and support forces. The steady-state operating hydraulic forces based on the system initial pressure are applied as general operating pressure loads to the reactor coolant loop model at changes in direction or flow area.

2. Weight

   A dead weight analysis is performed to meet Code requirements by applying a 1.0 g load downward on the complete piping system. The piping is assigned a distributed mass or weight as a function of its properties. This method provides a distributed loading to the piping system as a function of the weight of the pipe and contained fluid during normal operating conditions.

3. Seismic

   The forcing functions for the coupled building/loop non-linear time history analysis are derived from dynamic response analyses of the reactor building with soil springs subjected to seismic ground motion. Input is in the form of time history ground acceleration that is applied at the anchor point of the soil springs. RN 00-051

For the OBE and SSE seismic analyses, 2 and 4% critical damping, respectively, are used in the reactor coolant loop/supports system analysis. When combining the results of the SSE seismic analysis with the loss of coolant accident analysis results, 5% critical damping is used for the SSE. RN 00-051

4. Loss of Coolant Accident

Blowdown loads are developed in the broken and unbroken reactor coolant loops as a result of transient flow and pressure fluctuations following a postulated pipe break in one of the reactor coolant loops branch lines. Structural consideration of dynamic effects of postulated pipe breaks requires postulation of a finite number of break locations. Postulated pipe break locations are given in Section 3.6.2.

Broken loop time history dynamic analysis is performed for these postulated break cases. Hydraulic models are used to generate time-dependent hydraulic forcing functions used in the analysis of the reactor coolant loop for each break case. For further description of the hydraulic forcing functions, refer to Section 3.6.2.2. RN 99-110

5. Transients

The Code requires satisfaction of certain requirements relative to operating transient conditions. Operating transients are tabulated in Section 5.2.1.5.

The vertical thermal growth of the reactor pressure vessel nozzle centerlines is considered in the thermal analysis to account for equipment nozzle displacement as an external movement.

The hot moduli of elasticity E, the coefficient of thermal expansion at the metal temperature a, the external movements transmitted to the piping due to vessel growth, and the temperature rise above the ambient temperature ΔT, define the required input data to perform the flexibility analysis for thermal expansion.

To provide the necessary high degree of integrity for the RCS, the transient conditions selected for fatigue evaluation are based on conservative estimates of the magnitude and anticipated frequency of occurrence of the temperature and pressure transients resulting from various plant operation conditions.

5.2.1.10.3 Reactor Coolant Loop Models and Methods

The analytical methods used in obtaining the solution consists of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the time-history method for seismic dynamic analysis, and time history integration method for the loss of coolant accident dynamic analysis. RN 00-051

The integrated reactor coolant loops/supports system model is the basic system model used to compute loadings on components, component supports, and piping. The system model includes the stiffness and mass characteristics of the reactor coolant loop piping and components, the stiffness of supports, the stiffnesses of auxiliary line piping which affect the system and the stiffness of piping restraints. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

1. Static

   The reactor coolant loop/supports system model, constructed for the WESTDYN computer program, is represented by an ordered set of data which numerically describes the physical system. Figure 5.2-1 shows an isometric line schematic of this mathematical model. The steam generator and reactor coolant pump vertical and lateral support members are described in Section 5.5.14.

   The spatial geometric description of the reactor coolant loop model is based upon the reactor coolant loop piping layout and equipment drawings. The node point coordinates and incremental lengths of the members are determined from these drawings. Geometrical properties of the piping and elbows along with the modulus of elasticity E, the coefficient of thermal expansion α, the average temperature change from ambient temperature ΔT, and the weight per unit length are specified for each element. The primary equipment supports are represented by stiffness matrices which define restraint characteristics of the supports. Due to the symmetry of the static loadings, the reactor pressure vessel centerline is represented by a fixed boundary in the system mathematical model. The vertical thermal growth of the reactor vessel nozzle centerline is considered in the construction of the model.

   The model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained. These loads are incorporated into the overall load vector.

   After all the sections have been defined in this matter, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points is determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system.

The static solutions for deadweight, thermal, and general pressure loading conditions are obtained by using the WESTDYN computer program. The derivation of the hydraulic loads for the loss of coolant accident analysis of the loop is covered in Section 3.6.4.

2. Seismic

The model used in the static analysis is modified for the dynamic analysis by including the mass characteristics of the piping and equipment. All of the piping loops coupled with the reactor building with soil springs, equipment, and equipment support system are included in the system model. The effect of the equipment motion on the reactor coolant loop/supports system is obtained by modeling the mass and the stiffness characteristics of the equipment in the overall system model. RN 00-051

The steam generator is typically represented by 4 discrete masses. The lower mass is located at the intersection of the centerlines of the inlet and outlet nozzles of the steam generator. The second mass is located at the steam generator upper support elevation, the third mass is located at the feedwater nozzle elevation, and the fourth mass is at the top of the steam generator. RN 00-051

The reactor coolant pump is typically represented by a two discrete mass model. The lower mass is located at the intersection of the centerlines of the pump suction and discharge nozzles.

The reactor vessel and core internals are represented by approximately 14 discrete masses. The masses are lumped at various locations along the length of the vessel and along the length of the representation of the core internals. RN 00-051

The component upper and lower lateral supports are inactive during plant heatup, cooldown and normal plant operating conditions. However, these restraints become active when the plant is at power and under the rapid motions of the reactor coolant loop components that occur from the dynamic loadings and are represented by stiffness matrices and/or individual tension or compression spring members in the dynamic model. The analyses are performed at the full power condition.

The seismic analysis consists of applying an acceleration time history to a coupled reactor building, reactor coolant loop piping, equipment, and equipment support system.

The time history seismic analyses were performed using the proprietary Westinghouse computer code WECAN. RN 00-051

The reduced modal analysis method and modal superposition method are used in the time history seismic analyses. The reduced modal analysis option is used to determine the natural frequencies and mode shapes for a linear, undamped structure.

The modal superposition method gives a time history solution for the response of an arbitrary structure subjected to known nodal forces or ground acceleration time histories. The structure may include linear and non-linear elements.

The seismic responses of the reactor coolant loop support system are computed by performing a time history analysis for each of the three shock directions. The results of the three shock directions are combined by: RN 00-051

$$R_T = [\Sigma_{i=1}^{i=3} R_i^2]^{1/2}$$

where:

R_T = total combined response where i corresponds to each of three orthogonal shock direction combined responses (two horizontal and one vertical shock).

3. Loss of Coolant Accident

The mathematical model used in the static analyses is modified for the loss of coolant accident analysis to represent the severance of the reactor coolant loop piping at the postulated break location. Modifications include addition of the mass characteristic of the piping and equipment. To obtain the proper dynamic solution, a mass, located at the break is included in the mathematical model. The natural frequencies and eigenvectors are determined from this pipe break model.

The time-history hydraulic forces at the node points are combined to obtain the forces and acting at the corresponding structural lumped-mass node points.

The dynamic structural solution for the full power loss of coolant accident is obtained by using a modified-predictor-corrector- integration technique and normal mode theory.

When elements of the system can be represented as single acting members (tension or compression members), they are considered as nonlinear elements, which are represented mathematically by the combination of a gap, a spring, and a viscous damper. The force in this nonlinear element is treated as an externally applied force in the overall normal mode solution. Multiple nonlinear elements can be applied at the same node, if necessary.

The time-history solution is performed in program WESTDYN. The input to this subprogram consists of the natural frequencies, normal modes, applied forces and

nonlinear elements. The natural frequencies and normal modes for the modified reactor coolant loop dynamic model are determined with the WESTDYN program. To properly simulate the release of the strain energy in the pipe, the internal forces in the system at the postulated break location due to the initial steady-state hydraulic forces are determined. The release of the strain energy is accounted for by applying the negative of these internal forces as a step function loading. The initial conditions are equal to zero because the solution is only for the transient problem (the dynamic response of the system from the static equilibrium position). The time-history displacement solution of all dynamic degrees of freedom is obtained using subprogram WESTDYN and employing 4% critical damping.

The loss of coolant accident displacements of the reactor vessel are applied in time history form as input to the dynamic analysis of the reactor coolant loop. The loss of coolant accident analysis of the reactor vessel includes all the forces acting on the vessel including internals reactions, cavity pressure loads, and loop mechanical loads. The reactor vessel analysis is described in Section 5.2.1.10.6.

The time-history displacement response of the loop is used in computing support loads and in performing stress evaluation of the reactor coolant loop piping.

The support loads [F] are computed by multiplying the support stiffness matrix [K] and the displacement vector [δ] at the support point. The support loads are used in the evaluation of the supports.

The time-history displacements are used to determine the internal forces, deflections, and stresses at each end of the piping elements. For this calculation the displacements are treated as imposed deflections on the reactor coolant loop masses. The results of this solution are used in the piping stress evaluation.

4. Transients

Operating transients in a nuclear power plant cause thermal and/or pressure fluctuations in the reactor coolant fluid. The thermal transients cause time-varying temperature distributions across the pipe wall. These temperature distributions resulting in pipe wall stresses may be further subdivided in accordance with the Code into 3 parts, a uniform, a linear, and a nonlinear portion. The uniform portion results in general expansion loads. The linear portion causes a bending moment across the wall and the nonlinear portion causes a skin stress.

The transients as defined in Section 5.2.1.5 are used to define the fluctuations in plant parameters. A one-dimensional finite difference heat conduction program is used to solve the thermal transient problem. The pipe is represented by at least 50 elements through the thickness of the pipe. The convective heat transfer coefficient employed in this program represents the time varying heat transfer due to free and forced convection. The outer surface is assumed to be adiabatic while the inner surface boundary experiences the temperature of the coolant fluid.

Fluctuations in the temperature of the coolant fluid produce a temperature distribution through the pipe wall thickness which varies with time. An arbitrary temperature distribution across the wall is shown in Figure 5.2-2.

The average through-wall temperature, T_A, is calculated by integrating the temperature distribution across the wall. The integration is performed for all time steps so that T_A is determined as a function of time.

$$T_A(t) = \frac{1}{H}\int_0^H T(X,t)dX$$

The range of temperature between the largest and smallest value of T_A is used in the flexibility analysis to generate the moment loadings caused by the associated temperature changes.

$$M = E\alpha\int_0^H \left(X - \frac{H}{2}\right) T(X,t)dX$$

The equivalent thermal moment produced by the linear thermal gradient as shown in Figure 5.2-3 about the mid-wall thickness is equal to:

$$M_L = E\alpha_1 \frac{\Delta T_1}{2} H^2$$

Equating M_L and M, the solution for ΔT_1 as a function of time is:

$$\Delta T_1(t) = \frac{12}{H^2}\int_0^H \left(X - \frac{H}{2}\right) T(X,t)dX$$

The maximum nonlinear thermal gradient, ΔT_2, will occur on the inside surface and can be determined as the difference between the actual metal temperature on this surface and half of the average linear thermal gradient plus the average temperature.

$$\Delta T_2(t) = \left|T(0,t) - T_A(t)\right| - \frac{\left|\Delta T_1(t)\right|}{2}$$

5. Load Set Generation

A load set is defined as a set of pressure loads, moment loads, and through-wall thermal effects at a given location and time in each transient. The method of load set generation is based on Reference [2]. The through-wall thermal effects are functions of time and can be subdivided into four parts:

a. Average temperature (T_A) is the average temperature through-wall of the pipe which contributes to general expansion loads.

b. Radial linear thermal gradient which contributes to the through-wall bending moment (ΔT_1).

c. Radial nonlinear thermal gradient (ΔT_2) which contributes to a peak stress associated with shearing of the surface.

d. Discontinuity temperature (T_A - T_B) represents the difference in average temperature at the cross sections on each side of a discontinuity.

Each transient is described by at least one load sets representing the maximum and minimum stress state during each transient. The construction of the load sets is accomplished by combining the following to yield the maximum (minimum) stress state during each transient.

a. ΔT_1

b. ΔT_2

c. $\alpha_A T_A - \alpha_B T_B$

d. Moment loads due to T_A

e. Pressure loads

This procedure produces at least twice as many load sets as transients for each point.

As a result of the normal mode spectral technique employed in the seismic analysis the load components cannot be given signed values. Eight (8) load sets are used to represent all possible sign permutations of the seismic moments at each point, thus ensuring the most conservative combination of seismic loads are used in the stress evaluation.

For all possible load set combinations, the primary-plus-secondary and peak stress intensities, fatigue reduction factors (K_e) and cumulative usage factors, U, are calculated. The WESTDYN-7 program is used to perform this analysis in accordance with the ASME Code, Section III, Subsection NB-3650. Alternatively, detailed finite element stress analyses may be used to determine primary-plus-secondary and peak stress intensities, for the load set combinations. Since it is impossible to predict the order of occurrence of the transients over the plant life, it is assumed that the transients can occur in any sequence. This is a very conservative assumption. RN 04-019

The combination of load sets yielding the highest alternating stress intensity range is used to calculate the incremental usage factor. The next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles $<10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

5.2.1.10.3.1 Reactor Coolant Loop Analysis Results for Steam Generator Replacement

With the implementation of Leak-Before-Break (LBB) methodology to the V. C. Summer reactor coolant loops, the large RCL breaks have been eliminated from the LOCA analysis inputs. The only set of LOCA inputs re-analyzed to support steam generator replacement and their associated revised operating conditions were the large auxiliary lines, RHR line/primary coolant loop connection, accumulator line/primary coolant loop connection, and pressurizer surge line/primary coolant loop connection. Documentation of the reactor coolant loop analysis as affected by steam generator replacement is furnished in WCAP-9119, Revision 3. The conclusions of the report indicate that the impact of steam generator replacement will not result in reactor coolant loop piping stresses and fatigue usage that exceeds the ASME Code requirements for normal, upset, emergency, and test conditions. The structural integrity and safety-related design requirements of the RCL will be maintained during all conditions of the design specification. RN 00-051

5.2.1.10.4 Primary Component Supports Models and Methods

The static and dynamic structural analyses employ the matrix method and normal mode theory for the solution of lumped-parameter, multi-mass structural models. The equipment support structure models are dual-purpose since they are required 1) to quantitatively represent the elastic restraints which the supports impose upon the loop, and 2) to evaluate the individual support member stresses due to the forces imposed upon the supports by the loop.

Models for the STRUDL or WECAN computer program are constructed for the steam generator lower, steam generator upper lateral, reactor coolant pump lower, and reactor vessel supports. Structure geometry, topology and member properties are used in the modeling. RN 00-051

A description of the supports is found in Section 5.5.14. Detailed models are developed using beam elements and plate elements, where applicable.

The computer programs are used with these models to obtain support stiffness matrices and member influence coefficients for the steam generator and reactor coolant pump supports. Unit force along and unit moment about each coordinate axis are applied to the models at the equipment vertical centerline joint. Stiffness analyses are performed for each unit load for each model.

Joint displacements for applied unit loads are formulated into flexibility matrices. These are inverted to obtain support stiffness matrices which are included in the reactor coolant loop model.

Loads acting on the supports obtained from the reactor coolant loop analysis, support structure member properties, and influence coefficients at each end of each member are input into the WESAN program.

For each support case used, the following is performed:

1. Combine the various types of support plane loads to obtain operating condition loads (normal, upset, or faulted).

2. Multiply member influence coefficients by operating condition loads to obtain all member internal forces and moments.

3. Solve appropriate stress or interaction equations for the specified operating condition. Maximum normal stress, shear stress, and combined load interaction equation values are printed as a ratio of maximum actual values divided by limiting values.

Tables 5.2-16 through 5.2-18 present maximum stresses in each member of the steam generator and reactor coolant pump support structures expressed as a percentage of maximum permissible values for all operating condition loadings. Figures 5.2-10 through 5.2-12 illustrate the support models and show the member locations.

The reactor vessel support structures were analyzed for all loading conditions using a finite element model with the WECAN computer program. Vertical and horizontal forces delivered to the support structures from the reactor vessel shoes were applied to the structure. The maximum stresses in the reactor vessel supports for all loading conditions are presented in Table 5.2-20, expressed as a percentage of the maximum permissible values.

In addition, member compressive axial loads for all Class 1 primary equipment supports do not exceed 0.67 times the critical buckling strength for the faulted condition.

5.2.1.10.5 Analysis of Primary Components

Equipment which serves as part of the pressure boundary in the reactor coolant loop include the steam generators, the reactor coolant pumps, the pressurizer, and the reactor vessel. This equipment is Safety Class 1 and the pressure boundary meets the requirements of the ASME Code, Section III, Subsection NB. This equipment is evaluated for the loading combinations outlined in Table 5.2-3. The equipment is analyzed for 1) the normal loads of dead-weight, pressure and thermal, 2) mechanical transients of OBE, SSE, and pipe ruptures, and 3) pressure and temperature transients outlined in Section 5.2.1.5.

The results of the reactor coolant loop analysis are used to determine the loads acting on the nozzles and the support/component interface locations. These loads are supplied for all loading conditions on an "umbrella" load basis. That is, on the basis of previous plant analyses, a set of loads are determined which should be larger than those seen in any single plant analysis. The umbrella loads represent a conservative means of allowing detailed component analysis prior to the completion of the system analysis. Upon completion of the system analysis, conformance is demonstrated between the actual plant loads and the loads used in the analyses of the components. Any deviations where the actual load is larger than the umbrella load will be handled by individualized analysis.

Seismic analyses are performed individually for the reactor coolant pumps, the pressurizer, and the steam generators. Detailed and complex dynamic models are used for the dynamic analyses. The response spectra corresponding to the building elevation at the highest component/building attachment elevation is used for the component analysis. Seismic analyses for the steam generators are performed using 2% damping for the OBE and 4% damping for the SSE. The analysis of the reactor coolant pump for determination of loads on the motor, main flange, and pump internals is performed using the damping for bolted steel structures, that is, 4% for the OBE and 7% for the SSE (2% for OBE and 4% for SSE is used in the system analysis). This damping is applicable to the reactor coolant pump since the main flange, motor stand, and motor are all bolted assemblies (see Section 5.5). The reactor pressure vessel is qualified by static stress analysis based on loads that have been derived from dynamic analysis.

Reactor coolant pressure boundary components are further qualified to ensure against unstable crack growth under faulted conditions by performing detailed fracture analyses of the critical areas of this boundary. Actuation of the Emergency Core Cooling System produces relatively high thermal stresses in the system. Regions of the pressure boundary which come into contact with Emergency Core Cooling System water are given primary consideration. These regions include the reactor vessel beltline region, the reactor vessel inlet nozzles, and the safety injection nozzles in the piping system.

Two (2) methods of analysis are used to evaluate thermal effects in the regions of interest. The first method is Linear Elastic Fracture Mechanics (LEFM). The LEFM approach to the design against failure is basically a stress intensity consideration in which criteria are established for fracture instability in the presence of a crack. Consequently, a basic assumption employed in LEFM is that a crack or crack-like defect exists in the structure. The essence of the approach is to relate the stress field developed in the vicinity of the crack tip to the applied stress on the structure, the material properties, and the size of defect necessary to cause failure.

The elastic stress field at the crack-tip in any cracked body can be described by a single parameter designated as the stress intensity factor, K. The magnitude of the stress intensity factor K is a function of the geometry of the body containing the crack, the size and location of the crack, and the magnitude and distribution of the stress.

The criterion for failure in the presence of a crack is that failure will occur whenever the stress intensity factor exceeds some critical value. For the opening mode of loading (stresses perpendicular to the major plane of the crack) the stress intensity factor is designated as K_I and the critical stress intensity factor is designated K_{IC}. Commonly called the fracture toughness, K_{IC} is an inherent material property which is a function of temperature and strain rate. Any combination of applied load, structural configuration, crack geometry and size which yields a stress intensity factor K_{IC} for the material will result in crack instability.

The criterion of the applicability of LEFM is based on plasticity considerations at the postulated crack tip. Strict applicability (as defined by ASTM) of LEFM to large structures where plane strain conditions prevail requires that the plastic zone developed at the tip of the crack does not exceed 2.25% of the crack depth. In the present analysis, the plastic zone at the tip of the postulated crack can reach 20% of the crack depth. However, LEFM has been successfully used quite often to provide conservative brittle fracture prevention evaluations, even in cases where strict applicability of the theory is not permitted due to excessive plasticity. Recently, experimental results from Heavy Section Steel Technology (HSST) Program intermediate pressure vessel tests, have shown that LEFM can be applied conservatively as long as the pressure component of the stress does not exceed the yield strength of the material. The addition of the thermal stresses, calculated elastically, which result in total stresses in excess of the yield strength does not affect the conservatism of the results, provided that these thermal stresses are included in the evaluation of the stress intensity factors. Therefore, for faulted condition analyses, LEFM is considered applicable for the evaluation of the vessel inlet nozzle and beltline region.

In addition, it has been well established that the crack propagation of existing flaws in a structure subjected to cyclic loading can be defined in terms of fracture mechanics parameters. Thus, the principles of LEFM are also applicable to fatigue growth of a postulated flaw at the vessel inlet nozzle and beltline region.

For the safety injection and charging line nozzles, which are fabricated from 304 stainless steel, LEFM is not applicable because of extreme ductility of the material. For these nozzles, the thermal effects are evaluated using the principles of Miner's hypothesis of linear cumulative damage in conjunction with fatigue data from constant stress or strain fatigue tests. The cumulative usage fatigue defined as the sum of the ratios of the number of cycles of each transient (n) to the allowable number of cycles for the stress range associated with the transient (N) must not exceed 1.0.

An example of a faulted condition evaluation carried out according to the procedure discussed above is given in Reference [3]. This report discusses the evaluation procedure in detail as applied to a severe faulted condition (a postulated loss of coolant accident).

Valves in the RCS are designated ASME Class 1 and are designed and analyzed to the limits outlined in Table 5.2-4. Valves in sample lines connected to the RCS are not considered to be Safety Class 1 nor ASME Class 1. This is because the nozzles where the lines connect to the primary system piping are orificed to a 3/8 inch hole. This hole restricts the flow such that loss through a severance of one of these lines can be made up by normal charging flow.

To support steam generator replacement, the primary components (i.e., the reactor vessel, reactor internals, reactor coolant pumps, and pressurizer) were reevaluated with the Delta-75 steam generators modeled in the system, and with a NSSS power of 2912 MWt. Revised Reactor Coolant System design transients were compared with the original design basis transients. Where the revised transients indicated a more severe loading, the original evaluation was revisited. The results of the evaluations indicate that in most cases, the conservatively determined results of the original analyses remain bounding for these new conditions. As with the original stress report performed by Chicago Bridge and Iron, Inc., the maximum range of primary plus secondary stress intensity exceeds the $3S_m$ limit defined by the ASME Code for the outlet nozzles and support pads, inlet nozzles and support pads, and closure head flange. The stress intensities calculated by Chicago Bridge and Iron were not affected by the steam generator replacement. CBI performed a simple elastic-plastic analysis according to Paragraph NB-3228.3 of the ASME Code to justify these values. These results are not affected. The fatigue usages for all components which had stress intensities in excess of the Code limits were all less than the Code limit of 1.0. Some of the pressurizer components were calculated to experience a slight increase in fatigue usage, however the cumulative fatigue usage factors for all pressurizer components evaluated remain less than 1.0. In all cases the acceptance limits specified by the appropriate section of the ASME Code Edition applicable for that component were not exceeded. Details regarding the analysis of the reactor vessel and reactor vessel internals are furnished in References [14] and [15].

5.2.1.10.6 Reactor Vessel Support LOCA Loads

5.2.1.10.6.1 Introduction

As indicated in Section 3.6, Postulated breaks in the Reactor Coolant Loop piping, except for branch line connections, have been eliminated. Reactor Coolant loop piping branch nozzle (i.e., accumulator connection, pressurizer surge line, residual heat removal, etc.) breaks are postulated. RN 99-020

5.2.1.10.6.2 Interface Information

Gilbert Associates, Inc. is responsible for reactor containment design and analysis. Stiffness of the primary shield wall beneath the reactor vessel supports and asymmetric cavity pressurization loading was provided by Gilbert to Westinghouse. RN 99-020

All other input information was developed within Westinghouse. These items are as follows: reactor internals properties, loop mechanical loads and loop stiffness, internal hydraulic pressure transients, and reactor support stiffnesses. These inputs allowed formulation of the mathematical models and performance of the analyses, as will be described.

5.2.1.10.6.3 Loading Conditions

As indicated in Section 3.6, postulated breaks in the reactor coolant loop piping, except for branch line connections, have been eliminated. Reactor coolant loop piping branch nozzle (i.e., accumulator connection, pressurizer surge line, residual heat removal, etc.) breaks are postulated.

The reactor internals hydraulic pressure transients were calculated including the assumption that the structural motion is coupled with the pressure transients. This phenomena has been referred to as hydroelastic coupling or fluid-structure interaction. The hydraulic analysis considers the fluid-structure interaction of the core barrel by accounting for the deflections of constraining boundaries which are represented by masses and springs. The dynamic response of the core barrel in its beam bending mode responding to blowdown forces compensates for internal pressure variation by increasing the volume of the more highly pressurized regions. The analytical methods used to develop the reactor internals hydraulics are described in WCAP-8708 [12].

5.2.1.10.6.4 Reactor Vessel and Internals Modeling

The reactor vessel and internals general assembly is shown in Figure 3.9-1. The reactor vessel is restrained by 2 mechanisms: (1) the 3 attached reactor coolant loops with the steam generator and reactor coolant pump primary supports and (2) six reactor vessel supports, one beneath each reactor vessel nozzle. The reactor vessel supports are described in Section 5.5.14 and are shown in Figures 5.5-7, 5.2-4 and 5.2-5. The support shoe provides restraint in the horizontal directions and for downward reactor vessel motion.

The mathematical model of the reactor pressure vessel is a 3-dimensional nonlinear finite element model which represents the dynamic characteristics of the reactor vessel and its internals in 6 geometric degrees of freedom. The model was developed using the WECAN computer code. The model consists of 3 concentric structural submodels connected by nonlinear impact elements and stiffness matrices. The first submodel represents the reactor vessel shell and the associated support system. The second submodel represents the reactor core barrel, neutron panels, lower support plate, tie plates, and secondary core support components. This submodel is physically located inside the first, and is connected to it by a stiffness matrix at the internals support ledge. Core barrel to vessel shell impact is represented by nonlinear elements at the core barrel flange, core barrel outlet nozzle, and lower radial support locations. The third and innermost submodel represents the upper support plate, guide tubes, support columns, upper and lower core plates, and fuel. The third submodel is connected to the first and

second by stiffness matrices and nonlinear elements. The first, second, and third submodels are shown in Figures 5.2-6a, 5.2-6b, and 5.2-6c respectively.

The hydrodynamic mass matrices are a function of the properties of 2 cylinders with a fluid in the cylindrical annulus, specifically; inside and outside of the annulus, density of the fluid and length of the cylinders. Vertical segmentation of the reactor core barrel allows inclusion of radii variations along the reactor core barrel height and approximates the effects of reactor core barrel beam deformation. These mass matrices were inserted between selected nodes on the core barrel and reactor vessel shell submodels as shown in Figure 5.2.7.

The structure is divided into a finite number of members or elements. The inertia and stiffness matrices, as well as the force array, are first calculated for each element in the local coordinates. Employing appropriate transformation, the element global matrices and arrays are then computed. Finally, the global element matrices and arrays are assembled into the global structural matrices and arrays, and used for dynamic solution of the differential equation of motion for the structure.

5.2.1.10.6.5 Analytical Methods

The time history effects of the cavity pressurization loads, internals loads and loop mechanical loads are combined and applied simultaneously to the applicable nodes of the mathematical model of the reactor vessel and internals. The analysis is performed by numerically integrating the differential equations of motion to obtain the transient response. The output of the analysis includes, among other things, the displacements of the reactor vessel and the loads in the reactor vessel supports. The loads from the postulated pipe break on the vessel supports are combined with other applicable faulted condition loads and subsequently used to calculate the stresses in the supports. Also, the reactor coolant loop is analyzed by applying the reactor vessel displacements to the reactor coolant loop model. The resulting loads and stresses in the piping, components and supports are then combined with those from the loop dynamic blowdown analysis and the adequacy of the system is verified. Thus, the effect of vessel displacements upon loop response and the effect of loop blowdown upon vessel displacements are both evaluated.

5.2.1.10.6.6 Results of the Analysis

As previously discussed, the application of LBB methodology at V. C. Summer excludes the large RCL breaks from requiring analysis. The most limiting breaks considered for the dynamic analysis of V. C. Summer are the accumulator line break (cold leg) and pressurizer surge line break (hot leg).

LOCA loads applied to the V. C. Summer reactor pressure vessel system consist of reactor internal hydraulic loads (vertical and horizontal), and reactor coolant loop mechanical loads. All the loads are calculated individually and combined in a time history manner.

Reactor coolant loop mechanical loads are applied to the RPV (Reactor Pressure Vessel) nozzles by the primary coolant loop piping, and are given in Table 5.2-13.

The severity of a postulated break in a reactor vessel is related to 2 factors: the distance from the vessel to the break and the break opening area. Due to the interaction between the internals structure and the decompression wave, cold leg breaks result in larger reactor internal hydraulic forces. The RPV system LOCA analysis results consist of time history displacements and impact forces. The accumulator line break and the surge line break displacements are given in Table 5.2-14. Also, the reactor vessel/internals interface loads for these 2 breaks are given in Table 5.2-15. The results given in Tables 5.2-14 and 5.2-15 were compared with the original results from the RPV inlet nozzle break (large break). This comparison showed that the system LOCA response for the accumulator line break and the surge line break at uprated plant conditions with the Delta-75 steam generators are bounded by the main line loop breaks, eliminated by the application of leak-before-break methodology. Consequently, there is no adverse impact on the structural integrity of the V. C. Summer reactor internals.

5.2.1.10.7 Stress Criteria for Class 1 Components and Component Supports

1. Components

   All Class 1 components are designed and analyzed for the design, normal, and upset conditions to the rules and requirements of the ASME Code, Section III. The design analysis or test methods and associated stress or load allowable limits used in evaluation of faulted conditions are those that are defined in Appendix F of the ASME Code with supplementary options outlined below:

   a. Elastic System Analysis and Component Inelastic Analysis.

      This is an acceptable method of evaluation for faulted conditions if the rules provided below are met for component supports, and if primary stress limits for components are taken as greater of 0.70 S_u or S_y + 1/3 (S_u - S_y) for membrane stress and greater of 0.70 S_{ut} or S_y + 1/3 (S_{ut} - S_y) for membrane-plus-bending stress, where material properties are taken at appropriate temperature.

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

   b. Elastic/Inelastic System Analysis and Component/Test Load Method

      The test load method given in F-1370(d) is an acceptable method of qualifying components in lieu of satisfying the stress/load limits established for the component analysis.

2. Component Supports

   All Class 1 component supports are designed and analyzed, as discussed in Section 5.2.1.10.4, for the design, normal, upset, and faulted conditions. Allowable stresses for the Class 1 component supports are provided in Table 5.2-5.

Loading combinations and allowable stresses for ASME Code, Section III, Class 1 components and component supports are given in Tables 5.2-3, 5.2-4 and 5.2-5. For faulted condition evaluations, the effects of the SSE and loss of coolant accident are combined using the square root of the sum of the squares method. Justification for this method of load combination has been submitted to the NRC as WCAP-9279 and WCAP-9283.

5.2.1.10.8 Computer Program Descriptions

The following computer programs have been used in dynamic and static analyses to determine mechanical loads, stresses, and deformations of Seismic Category I components and equipment. These are described and verified in Reference [4].

1. WESTDYN -static and dynamic analysis of redundant piping systems.

2. STHRUST - hydraulic loads on loop components from blowdown information.

3. WESAN - reactor coolant loop equipment support structures analysis and evaluation.

4. WECAN - finite element structural analysis

5.2.1.11 Analysis Methods for Faulted Conditions

The methods used for the evaluation of the faulted conditions are contained in Section 5.2.1.10.

5.2.1.12 Protection Against Environmental Factors

A discussion of the protection provided for the principal components of the RCS against environmental factors is found in Section 3.11.

5.2.1.13 Compliance with Code Requirements

A brief description of the analyses and methods used to assure compliance with the applicable codes is provided in Section 5.2.1.10.

5.2.1.14 Stress Analysis for Emergency and Faulted Conditions Loadings

The stress analysis used for emergency and faulted condition loadings are discussed in Section 5.2.1.10.

5.2.1.15 Stress Levels in Category I Systems

Summary results of the typical analyses of the Category I systems are included in the latest revisions of Westinghouse Reports WCAP-9119, Vol. II and Vol. I; WCAP-9803, Vol. IV, and WCAP-9802, Vol. III.

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5.2.1.16 Analytical Methods for Stresses in Pumps and Valves

The pressure boundary portions of the Class 1 pump in the RCS (the reactor coolant pumps) are designed and analyzed according to the requirements of NB-3400.

The pressure boundary portions of Class 1 valves in the RCS are designed and analyzed according to the requirements of NB-3500. These valves are identified in Section 3.9.2.

5.2.1.17 Analytical Methods for Evaluation of Pump Speed and Bearing Integrity

The reactor coolant pump shaft is designed so that its operating speed is below the first shaft critical speed as discussed in Section 5.5.1.3.6.

Reactor coolant pump bearing integrity is discussed in Section 5.5.1.3.4.

5.2.1.18 Operation of Active Valves Under Transient Loadings

The tests and analyses that are performed on active valves to assure operation under transient loadings are described in Section 3.9.2.4.

5.2.1.19 Field Run Piping

There is no field run safety class piping. All safety class piping down to 1 inch is depicted on and fabricated to plan and section drawings and any applicable analyses are performed accordingly. Approved design criteria are used for pipe hanger design.

5.2.2 OVERPRESSURIZATION PROTECTION

5.2.2.1 Location of Pressure Relief Devices

Pressure relief devices for the RCS comprise the 3 pressurizer safety valves and 3 power operated relief valves shown in Figure 5.1-1, Sheet 2; these discharge to the pressurizer relief tank through a common header. Other relief valves in the reactor building that discharge to the pressurizer relief tank are itemized in Table 5.2-6.

5.2.2.2 Mounting of Pressure Relief Devices

The NSSS supplier provides protection pads for the pressurizer. The architect engineer designs the supports for the safety relief valves and the attached piping in accordance with the installation guidelines and suggested physical layout criteria provided by the NSSS supplier. The architect engineer determines the reactions on the valves and attached piping to limit the piping reaction loads to acceptable values.

5.2.2.3 Report on Overpressure Protection

The pressurizer is designed to accommodate pressure increases (as well as decreases) caused by load transients. The spray system condenses steam to prevent the pressurizer pressure from reaching the setpoint of the power operated relief valves during a step reduction in power level equivalent to 10% of full rated load.

The spray nozzle is located in the top head of the pressurizer. Spray is initiated when the pressure controlled spray demand signal is above a given setpoint. The spray rate increases proportionally with increasing compensated error signal until it reaches a maximum value. The compensated error signal is the output of PID (Proportional plus Integral plus Derivative) controller, the input to which is an error signal based on the difference between actual pressure and a reference pressure.

The pressurizer is equipped with power operated relief valves which limit system pressure for a large power mismatch, and thus prevent actuation of the fixed high pressure reactor trip. The relief valves are operated automatically or by remote manual control. The operation of these valves also limits the undesirable opening of the spring-loaded safety valves. Remotely operated stop valves are provided to isolate the power operated relief valves if excessive leakage occurs. The relief valves are designed to limit the pressurizer pressure to a value below the high pressure trip setpoint for all design transients up to and including the design percentage step load decrease with steam dump but without reactor trip.

Output signals from the pressurizer pressure protection channels are isolated and used for pressure control. These are used to control pressurizer spray and heaters and power operated relief valves. Pressurizer pressure is sensed by fast response pressure transmitters with a time response of better than 0.2 seconds.

In the event of a complete loss of heat sink, e.g., no steam flow, protection of the RCS against overpressure is afforded by pressurizer and steam generator safety valves along with any of the following reactor trip functions:

1. Reactor trip on turbine trip (if the turbine is tripped at above 50% power); RN 98-129

2. High pressurizer pressure reactor trip;

3. Overtemperature ΔT reactor trip;

4. Low feedwater flow reactor trip or;

5. Low-low steam generator water level reactor trip.

The ASME Code pressure limit is 110% of the 2485 psig design pressure. This limit is not exceeded as discussed in Reference [5]. The report describes in detail the pressure relief devices, location, reliability, and sizing. Transient analysis data is provided for the worst cases that require safety valve actuation as well as those cases which do not.

A detailed functional description of the process equipment associated with the high pressure trip is provided in Reference [6].

The upper limit of overpressure protection is based upon the positive surge of the reactor coolant produced as a result of turbine trip under full load, i.e., a 100% load mismatch assuming that the core continues to produce full power with no steam dump. The self-actuated safety valves are sized on the basis of steam flow from the pressurizer to accommodate this surge at a setpoint of 2500 psia and a total accumulation of 3%.

The actual installed capacity of the safety valves is always greater than the capacity calculated from the sizing analysis and is indicated so by the ratio of safety valve flow to peak surge rate being greater than 1.0. Note that no credit is taken for the relief capability provided by the power operated relief valves during this surge.

The RCS design and operating pressure together with the safety, power relief and pressurizer spray valve setpoints and the protection system setpoint pressures are listed in Table 5.2-7.

System component whose design pressure and temperature are less than the RCS design limits are provided with overpressure protection devices and redundant isolation means. System discharge from overpressure protection devices is collected in the pressurizer relief tank.

As specified in NUREG 0660, item II.K.3 Table C.3 item 3, SCE&G will report safety/relief valve failures and challenges during startup testing and plant operation.

5.2.2.4 ECCS Check Valve Testing

Check valves on the discharge side of low and high head safety injection systems, and the safety injection accumulator subsystems that are classified as ASME Code, Category AC are listed in Table 5.2-7a. RN 98-109 99-091

The ECCS check valve system, shown by Figure 5.2-8, provides the means for leak testing individual check valves, classified Category AC, on the discharge side of the low head safety injection systems and safety injection accumulator subsystems independent of other valves. Testing of these valves will be performed during plant startup following each refueling shutdown. Reactor coolant pressure will be at least 1000 psig. RN 99-091

ECR 50650 added a back pressure regulator set at approximately 400 psig to allow controlled leakage to the PRT. By installing a back pressure regulator around XVT00037-SI, a method of preventing pressure building up in an isolated system is provided. This is done to reduce the potential for leakage into and pressurization of interfacing systems. RN 13-020

Acceptance criteria for leakage are in accordance with Technical Specification 3.4.6.2. Test results are documented in accordance with the ASME Code of Record, prescribed under 10CFR50.55a. These check valves are normally tested during plant startup following a refueling shutdown. Should a valve be found to leak excessively, it will be repaired and retested prior to a return to normal operation. RN 99-091 04-012

5.2.2.5 RCS Pressure Control During Low Temperature Operation RN 99-091

Administrative procedures are developed to aid the operator in controlling RCS pressure during low temperature operation.

Cold overpressurization protection will be provided by the RHR suction relief valves during low RCS temperature operation. Analysis has shown that 1 RHR suction relief valve is sufficient to prevent violation of 10CFR50, Appendix G limits due to anticipated mass and heat input transients.

5.2.2.5.1 System Operation

The RHR system is normally lined up to the RCS with the reactor shut down, primary coolant temperature ≤350°F and pressure ≤450 psig. The RHR suction relief valves are set at 450 psig and provide the necessary overpressurization protection during cold plant operations. Annunciators are provided in the control room to alert the operator if reactor coolant temperature is < 300°F and any of the RHR suction isolation valves are not open.

5.2.2.5.2 Evaluation of Low Temperature Overpressure Transients

Pressure Transient Analyses

ASME Section III, Appendix G, establishes guidelines and limits for RCS pressure primarily for low temperature conditions (≤350°F). The relief system discussed in 5.2.2.5 satisfies these conditions as discussed in the following paragraphs.

Transient analyses were performed to insure that a RHR relief setpoint of 450 psig is sufficient. This setpoint maintains reactor coolant system pressure within acceptable limits following all credible overpressurization incidents occurring in the plant during low temperature, water solid operation.

The mass input transient analysis was performed assuming the most severe event involving a single centrifugal charging pump. Specifically, a loss of air incident is

postulated, whereby the flow control valve on the charging line fails open and, simultaneously, the flow control valve on the letdown line fails closed.

The heat input mechanism considered for analysis involved a RCS pump startup in 1 loop with a water solid condition and temperature asymmetry in the reactor coolant system, whereby the steam generators were at a higher temperature than the remainder of the system. A 50°F mismatch was assumed to exist between the RCS (100°F) and the secondary side of the steam generators. (At lower temperatures, the mass input case is the limiting transient condition).

Both analyses took into account the single failure criteria and therefore, the operation of 1 RHR suction relief valve was assumed to be available for pressure relief. Consistent with WCAP-14040-NP-A, Revision 2[7], the design credits the fact that overpressure events mostly occur during isothermal conditions in the RCS and therefore use the steady state Appendix G limit in judging the adequacy of the RHR relief valve capacity and lift setting. Calculated peak pressures for both the design basis mass and heat addition transients remain substantially less than the allowable limits, which are reflected in the heatup and cooldown limit curves for normal operation within the Technical Specifications. Thus, low temperature overpressure transients do not constitute impairment to vessel integrity or plant safety. RN 00-001

<u>OBE Evaluation</u>

A fluid systems evaluation has been performed to analyze the potential for overpressure transients following an OBE. The basis of the evaluation assumes the plant air system is inoperable since it is not seismically qualified. The results of the evaluation follow and demonstrate that overpressure transients following an OBE are not a concern.

1. A loss of plant air during the first part of plant cooldown (i.e., decay heat removal via SG, normal CVCS letdown and just prior to placing the RHRS on line at 350°F) would cause the low pressure letdown isolation valves to fail close and the charging flow control valve to fail open. These conditions would create a net mass addition to the RCS causing the pressure to increase. However, the pressure increase would be acceptable since the pressurizer safety valves would limit system pressure (2485 psig) within allowable values (Technical Specification Figures 3/4 4-26 and 3/4 4-27).

2. A loss of plant air during the second part of plant cooldown (i.e., decay heat removal via the RHRS and temperature less than 350°F) would cause the low pressure letdown valve to fail close and charging flow control valve to fail open similar to that discussed above. These conditions would create a net mass addition to the system which would be relieved by the RHR relief valves which are set at 450 psig and thus maintain the pressure within allowable values.

The flow paths of the common relief headers from the pressurizer safety and RHRS relief valves to the pressurizer relief tank are assured following an OBE by design of the headers for seismic loads.

For the various modes described above, the pressurizer safety and RHRS relief valves provide pressure relief for the postulated transients following an OBE and thus maintain the primary system within the allowable pressure/temperature limits.

5.2.2.5.3 Administrative Procedures

Although the system described in Section 5.2.2.5.1 is installed to maintain RCS pressure within allowable limits, administrative procedures are recommended for minimizing the potential for any transient that could actuate the overpressure relief system. The following discussion highlights these procedural controls, listed in hierarchy of their function in preventing RCS cold overpressurization transients.

Of primary importance is the basic method of operation of the plant. Normal plant operating procedures will maximize the use of a pressurizer cushion (steam/N_2 bubble) during periods of low pressure, low temperature operation. This cushion will dampen the plants' response to potential transient generating inputs, providing easier pressure control with the slower response rates.

An adequate cushion eliminates some potential transients such as reactor coolant pump induced heat input and slows the rate of pressure rise for others. In conjunction with the previously discussed alarms, this provides reasonable assurance that most potential transients can be terminated by operator action before the overpressure relief system actuates.

However, for those modes of operation when water solid operation may still be possible, procedures will further highlight precautions that minimize the potential for developing an overpressurization transient. The following specific recommendations will be made:

1. Do not isolate the residual heat removal inlet lines from the reactor coolant loop unless the charging pumps are stopped. This precaution is to assure there is a relief path from the reactor coolant loop to the residual heat removal suction line relief valves when the RCS is at low pressure (less than 500 psi) and is water solid.

2. Whenever the plant is water solid and the reactor coolant pressure is being maintained by the low pressure letdown pressure control valve (PCV-145), low pressure letdown from the Residual Heat Removal System (via HCV-142) must be in operation. During this mode of operation, all 3 letdown orifices must also remain open. RN 05-016

3. If all reactor coolant pumps have stopped for more than 5 minutes during plant heatup, and the reactor coolant temperature is greater than the charging and seal injection water temperature, do not attempt to restart a pump unless a steam

bubble is formed in the pressurizer. This precaution will minimize the pressure transient when the pump is started and the cold water previously injected by the charging pumps is circulated through the warmer reactor coolant components. The steam bubble will accommodate the resultant expansion as the cold water is rapidly warmed.

4. If all reactor coolant pumps are stopped and the reactor coolant system is being cooled down by the residual heat exchangers, a non-uniform temperature distribution may occur in the reactor coolant loops. Do not attempt to restart a reactor coolant pump unless a steam bubble is formed in the pressurizer.

5. During plant cooldown, all steam generators should be connected to the steam header to assure a uniform cooldown of the reactor coolant loops.

6. At least 1 reactor coolant pump must remain in service until the reactor coolant temperature is reduced to 160°F.

These special precautions backup the normal operational mode of maximizing periods of steam bubble operation so that cold overpressure transient prevention is continued during periods of transitional operations.

The specific plant configurations of ECCS testing and alignment to prevent developing cold overpressurization transients during these limited periods of plant operation, given as follows: 98-01

1. To preclude inadvertent ECCS actuation during heatup and cooldown, procedures require blocking the pressurizer pressure safety injection signal when pressurizer pressure has decreased below 1985 psig and the steamline safety injection signal when T_{avg} decreases below 552°F. 98-01

2. During further cooldown, closure and power lockout of the accumulator isolation valves and power lockout of the non-operating charging pumps will be performed with the RCS pressure maintained between 900 and 950 psig, providing additional backup to step 1 above. 98-01

3. The recommended procedure for periodic ECCS pump performance testing will be to test the pumps during normal power operation or at hot shutdown conditions. This precludes any potential for developing a cold overpressurization transient.

   Should cold shutdown testing of the pumps be desired, it will be recommended that the test be done when the vessel is open to atmosphere again precluding overpressurization potential.

   If cold shutdown testing with the vessel closed is necessary, the procedures will require charging pump discharge valve closure and RHRS alignment to both

isolate potential ECCS pump input and to provide backup benefit of the RHRS relief valves.

4. "S" signal circuitry testing, if done during cold shutdown, will also require RHRS alignment and non-operating charging pump power lockout to preclude developing cold overpressurization transients.

The above procedural recommendations covering normal operations with a steam bubble, transitional operations where potentially water solid, followed by specific testing operations provide in-depth cold overpressure prevention, augmenting the installed overpressure relief system.

5.2.3 GENERAL MATERIAL CONSIDERATIONS

5.2.3.1 Material Specifications

Material specifications used for the principal pressure retaining applications in each component comprising the RCPB are listed in Table 5.2-8 for ASME Class 1 primary components and Table 5.2-9 for ASME Class 1 and 2 auxiliary components. The materials are procured in accordance with the material specification requirements and include the special requirements of the ASME Code, Section III, plus Addenda and Code Cases as are applicable and appropriate to meet Appendix B of 10CFR50 in the Federal Register, Vol. 35, No. 125. In some cases, Table 5.2-9 may not be totally inclusive of the material specifications used in the listed applications. However, the listed specifications are representative of those materials utilized. All of the materials used are procured in accordance with ASME Code requirements.

The welding materials used for joining the ferritic base materials of the RCPB, conform to or are equivalent to ASME Material Specifications SFA 5.1, 5.2, 5.5, 5.17, 5.18 and 5.20. They are tested and qualified to the requirements of ASME Section III rules. In addition the ferritic materials of the reactor vessel beltline are restricted to the following maximum limits of copper, phosphorous and vanadium to reduce sensitivity to irradiation embrittlement in service:

Element	Base Metal (%)	As Deposited Weld Metal (%)
Copper	0.10 (Ladle) 0.12 (Check)	0.10
Phosphorous	0.012 (Ladle) 0.017 (Check)	0.015
Vanadium	0.05 (Check)	0.05 (as residual)

Regulatory Guides 1.34, 1.43, 1.50 and 1.66 are discussed in Appendix 3A.

The welding materials used for joining the austenitic stainless steel base materials of the RCPB conform to ASME Material Specifications SFA 5.4 and 5.9. They are tested and qualified to the requirements of ASME Section III rules.

The welding materials used for joining nickel-chromium-iron alloy in similar base material combination and in dissimilar ferritic or austenitic base material combination conform to ASME Material Specifications SFA 5.11 and 5.14. They are tested and qualified to the requirements of ASME Section III rules and are used only in procedures which have been qualified to these same rules.

5.2.3.2 Compatibility With Reactor Coolant

All of the ferritic low alloy and carbon steels which are used in principal pressure retaining applications are provided with corrosion resistant cladding on all surfaces that are exposed to the reactor coolant. This cladding material has a chemical analysis which is at least equivalent to the corrosion resistance of Types 304 and 316 austenitic stainless steel alloys or nickel-chromium-iron alloy, martensitic stainless steel and precipitation hardened stainless steel. Ferritic low alloy and carbon steel nozzles are safe ended with either stainless steel wrought materials, stainless steel weld metal analysis A-7, or nickel-chromium iron alloy weld metal F-Number 43. The latter buttering material requires further safe ending with austenitic stainless steel base material after completion of the post weld heat treatment when the nozzle is larger than a 4 inch nominal inside diameter and/or the wall thickness is greater than 0.531 inches. The reactor vessel main coolant nozzles are safe ended with nickel-chromium-iron alloy weld metal F-Number 43, and will necessitate dissimilar metal field welds of nickel-chromium-iron alloy to austenitic stainless steel.

The cladding on ferritic type base materials receives a post weld heat treatment.

All of the austenitic stainless steel and nickel-chromium-iron alloy base materials with primary pressure retaining applications are used in the solution anneal heat treat condition. These heat treatments are as required by the material specifications.

During subsequent fabrication, these materials are not heated above 800°F other than locally by welding operations. The solution annealed surge line material is subsequently formed by hot bending followed by a resolution annealing heat treatment.

All other austenitic stainless steel piping in contact with reactor coolant is required to be bent at temperatures above 1200°F. If the bending is performed below 1950°F, piping is solution annealed after bending. All stainless steel piping is cooled rapidly between temperatures of 1500°F and 800°F after bending or annealing.

Degreasing is also required prior to subjecting piping to elevated temperatures before, during or after bending to prevent carbonization. Austenitic stainless steel material is subsequently tested in accordance with Practice E and the accompanying test Practice A of ASTM A 262.

5.2.3.3 Compatibility With External Insulation and Environmental Atmosphere

In general, all of the materials listed in Tables 5.2-8 and 5.2-9 which are used in principal pressure retaining applications and which are subject to elevated temperature during system operation are in contact with thermal insulation that covers their outer surfaces.

The thermal insulation used on the RCPB is either a metallic reflective stainless steel type or a mass type which is encapsulated in stainless steel.

In the event of coolant leakage, the ferritic materials will show increased general corrosion rates. Where minor leakage is anticipated from service experience, such as valve packing, pump seals, etc., only materials which are compatible with the coolant are used. These are as shown in Tables 5.2-8 and 5.2-9. Ferritic materials exposed to coolant leakage can be readily observed as part of the inservice visual and/or nondestructive inspection program to assure the integrity of the component for subsequent service.

5.2.3.4 Chemistry of Reactor Coolant

The RCS chemistry specifications are given in Table 5.2-10.

The RCS water chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the specifications.

The Chemical and Volume Control System provides a means for adding chemicals to the RCS which control the pH of the coolant during initial startup and subsequent operation, scavenge oxygen from the coolant during startup and control the oxygen level of the coolant due to radiolysis during all power operations subsequent to startup and to chemically degasify and oxygenate the RCS during shutdowns for refueling operations. The oxygen content and pH limits for power operations are shown in Table 5.2-10. RN 05-035

The pH control chemical employed is Lithium-7 hydroxide. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/inconel systems. In addition, lithium is produced in solution from the neutron irradiation of the dissolved boron in the coolant. The Lithium-7 hydroxide is introduced into the RCS via the charging flow. The solution is prepared in the laboratory and poured into the chemical mixing tank. Reactor makeup water is then used to flush the solution to the suction header of the charging pumps. The concentration of Lithium-7 hydroxide in the RCS is maintained in the range specified for pH control. If the concentration exceeds this range, the cation bed demineralizer is employed in the letdown line in series operation with the mixed bed demineralizer. Since the amount of lithium to be removed is small and its buildup can be readily calculated and determined by analysis, the flow through the cation bed demineralizer is not required to be full letdown flow.

During reactor startup from the cold condition, hydrazine is employed as an oxygen scavenging agent. The hydrazine solution is introduced into the RCS in the same manner as described above for the pH control agent.

Dissolved hydrogen is employed to control and scavenge oxygen produced due to radiolysis of water in the core region. Sufficient partial pressure of hydrogen is maintained in the volume control tank such that the specified equilibrium concentration of hydrogen is maintained in the reactor coolant. A self-contained pressure control valve maintains a minimum pressure in the vapor space of the volume control tank. This can be adjusted to provide the correct equilibrium hydrogen concentration.

During reactor shutdown for subsequent refueling operations or to open the RCS, hydrogen peroxide is utilized to chemically degasify the reactor coolant and also may be utilized to oxygenate to RCS causing a release of corrosion products from the reactor vessel for subsequent clean up using the CVCS system. RN 05-035

Components with stainless steel sensitized in the manner expected during component fabrication and installation will operate satisfactorily under normal plant chemistry conditions in PWR systems because chlorides, fluorides and oxygen are controlled to very low levels.

5.2.4 FRACTURE TOUGHNESS

5.2.4.1 Compliance With Code Requirements

Assurance of adequate fracture toughness of ferritic materials in the RCPB (ASME Section III Class I Components) is provided by compliance with the requirements for fracture toughness testing included in NB 2300 of Section III of the ASME Code, Summer 1972 Addenda, and 10 CFR 50, Appendix G, to the extent possible. A summary of the fracture toughness data for the reactor pressure vessel material is given in Tables 5.2-11 and 5.2-21, and Figures 5.2-16 through 5.2-23.

The fracture toughness of the reactor vessel beltline region has been evaluated according to the requirements of the Pressurized Thermal Shock (PTS) rule [20]. Using the PTS rule and surveillance capsule data [21], the evaluation documented in WCAP-16306-NP, Revision 0 [22], shows a maximum calculated RT_{PTS} value for the most limiting plate, A9154-1, of 152°F at 32 EFPY and 159°F at 56 EFPY. The limiting RT_{PTS} values are well below the screening criteria of 270°F specified in the PTS rule. RN 00-001 04-042

Ferritic bolting materials used in the reactor coolant pressure boundary are fabricated in accordance with the following addenda of the ASME Code Section III, all of which predate the issuance of 10 CFR 50 Appendix G:

Reactor Vessel (closure head bolts)	1971 Edition, thru Summer 1971 Addenda
Steam Generator (manway bolts)	1971 Edition, thru Summer 1971 Addenda
Pressurizer (manway bolts)	1971 Edition, thru Summer 1971 Addenda
Reactor Coolant Pump (main flange and seal housing bolts)	1971 Edition, thru Summer 1972 Addenda

These materials comply with the fracture toughness requirements of the applicable code. These requirements are compared to the requirements of 10 CFR Part 50 Appendix G below.

	Test Temperature	ft. lbs	MLE
1971 Edition, Summer 1971 Addendum	-	35	-
1971 Edition, Summer 1972 Addendum	≤ 40°F	-	25
10 CFR 50 Appendix G, 1980	Lower of preload or service temp. (usually ≈ 50°F)	45	25

The fracture toughness data available for these materials are shown in Table 5.2-22.

5.2.4.2 Acceptable Fracture Energy Levels

Initial upper shelf fracture energy levels for materials of the reactor vessel beltline region (including welds) meet the minimum 75 foot-pound shelf energy requirement of 10 CFR 50, Appendix G.

5.2.4.3 Operating Limitations During Startup and Shutdown

Startup and shutdown operating limitations are based on the properties of the core region materials of the reactor pressure vessel. These limitations are reflected within the Technical Specifications in the form of heatup and cooldown limit curves which show allowable combination of pressure and temperature for specific temperature change rates. The heatup and cooldown limit curves for normal operation are derived in WCAP-16305-NP, Revision 0 [23] using the methodology from WCAP-14040-NP-A, Revision 4 [7]. RN 00-001 RN 04-042

As outlined in WCAP-16305-NP, Revision 0, heatup and cooldown limit curves are calculated using the unirradiated RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} (ART) of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (transverse to the major rolling direction for late material) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in the reactor vessel steels. Regulatory Guide 1.99, Revision 2[24] is used to predict radiation embrittlement and for the calculation of Adjusted Reference Temperature (ART) values (IRT_{NDT} + ΔRT_{NDT} + margin for uncertainties) at the ¼ T and ¾ T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The most limiting ART values are used in the generation of heatup and cooldown limit curves for normal operation. RN 00-001

As required by Appendix G to 10 CFR Part 50, the Technical Specification heatup limit curves also include temperature limits for inservice leak tests and hydrotests which are used to define additional pressure-temperature limits for core operation (i.e., criticality limit curves) to provide additional margin during actual power production.

5.2.4.3.1 Temperature Difference Between the Pressurizer and Reactor Coolant Loop

Thermal stratification in the pressurizer surge line is a direct result of the difference in temperatures between the pressurizer water and the generally cooler Reactor Coolant System (RCS) water during plant startup and shutdown. The hotter, lighter pressurizer water tends to float on the cooler, heavier RCS water. The potential for thermal stratification is directly proportional to the difference in temperature (system ΔT) between the pressurizer and the Reactor Coolant System. The potential for

stratification is increased as the system ΔT increases. At power, when the system ΔT is relatively small, the extent and effects of thermal stratification have been observed to be negligible. However, during certain modes of plant startup and shutdown, the system ΔT could be as large as 320°F, in which case the effects of thermal stratification are significant.

As required by the NRC Bulletin 88-11, the effects of thermal stratification of the surge line were included in a structural and fatigue analysis of the piping. The analysis demonstrated that the pipe stress and fatigue factors conform the ASME Code limits for all anticipated thermal stratification conditions up to a system ΔT limit of 320°F.

In order to remain within the design basis parameters of this analysis, a system ΔT limit of 320°F is necessary during plant startup and shutdown.

5.2.4.4 Compliance with Reactor Vessel Material Surveillance Program Requirements

Refer to Section 5.4.

5.2.4.5 Reactor Vessel Annealing

Refer to Section 5.4.

5.2.5 AUSTENITIC STAINLESS STEEL

The unstabilized austenitic stainless steel material specifications used for the RCPB, systems required for reactor shutdown and systems required for emergency core cooling are listed in Tables 5.2-8 and 5.2-9.

The unstabilized austenitic stainless steel material for the reactor vessel internals which are required for emergency core cooling for any mode of normal operation or under postulated accident conditions and for core structural load bearing members are listed in Table 5.2-12.

The above tabulated materials are procured in accordance with the material specification requirements and include the special requirements of the ASME Code, Section III, plus Addenda and Code Cases as are applicable to meet Appendix B of 10 CFR 50 in the Federal Register, Vol. 35, No. 125.

Regulatory Guide 1.44 is discussed in Appendix 3A.

5.2.5.1 Cleaning and Contamination Protection Procedures

It is required that austenitic stainless steel materials used in the fabrication, installation and testing of nuclear steam supply components and systems be handled, protected, stored and cleaned according to recognized and accepted methods which are designed to minimize contamination which could lead to stress corrosion cracking. The rules

covering these controls are stipulated in process specifications. As applicable, these process specifications supplement the equipment specifications and purchase order requirements of every individual austenitic stainless steel component or system procured for the NSSS, regardless of the ASME Code Classification. The site cleaning procedures recommended in Regulatory Guide 1.37 are discussed in Appendix 3A.

To assure that manufacturers and installers adhere to the requirements of these specifications, surveillance of operations is conducted either in residence at the manufacturer's plant and the construction site or, when residency is not practical, during periodic engineering and quality assurance visitations and audits at these locations. Any deviation from these rules requires corrective action.

5.2.5.2 Solution Heat Treatment Requirements

The austenitic stainless steels listed in Tables 5.2-8, 5.2-9 and 5.2-12 are utilized in the final heat treated condition required by the respective ASME Code Section II material specification for the particular type or grade of alloy.

5.2.5.3 Material Inspection Program

Austenitic stainless steel materials of product forms with simple shapes need not be corrosion tested provided that the solution heat treatment is followed by water quenching. Simple shapes are defined as all plates, sheets, bars, pipe and tubes, as well as forgings, fittings, and other shaped products which do not have inaccessible cavities or chambers that would preclude rapid cooling when water quenched. When testing is required, the tests are performed in accordance with ASTM A 262-70, Practice A or E, as amended by Westinghouse Process Specification 84201 MW.

5.2.5.4 Unstabilized Austenitic Stainless Steels

The unstabilized austenitic stainless steels used in the RCPB and components are listed in Tables 5.2-8 and 5.2-9.

The materials are used in the as-welded condition as discussed in Section 5.2.5.2. The control of the water chemistry is stipulated in Section 5.2.3.4.

5.2.5.5 Prevention of Intergranular Attack of Unstabilized Austenitic Stainless Steels

Unstabilized austenitic stainless steels are subject to intergranular attack (IGA) provided that 3 conditions are present simultaneously. These are:

1. An aggressive environment, e.g., an acidic aqueous medium containing chlorides or oxygen.

2. A sensitized steel.

3. A high temperature.

If any one of the 3 conditions described above is not present, intergranular attack will not occur. Since high temperatures cannot be avoided in all components in the NSSS, Westinghouse relies on the elimination of conditions 1 and 2 to prevent intergranular attack on wrought stainless steel components.

The water chemistry in the RCS of a Westinghouse Pressurized Water Reactor (PWR) is rigorously controlled to prevent the intrusion of aggressive species. In particular, the maximum permissible oxygen and chloride concentrations are 0.10 ppm and 0.15 ppm, respectively. Reference [8] describes the precautions taken to prevent the intrusion of chlorides into the system during fabrication, shipping, and storage. The use of hydrogen overpressure precludes the presence of oxygen during operation. The effectiveness of these controls has been demonstrated by both laboratory tests and operating experience. The long time exposure of severely sensitized stainless in early plants to PWR coolant environments has not resulted in any sign of intergranular attack. Reference [8] describes the laboratory experimental findings and the Westinghouse operating experience. The additional years of operations since the issuing of Reference [8] have provided further confirmation of the earlier conclusions. Severely sensitized stainless steels do not undergo any intergranular attack in Westinghouse PWR coolant environments.

In spite of the fact there never has been any evidence that PWR coolant water attacks sensitized stainless steels, Westinghouse considers it good metallurgical practice to avoid the use of sensitized stainless steels in the NSSS components. Accordingly, measures are taken to prohibit the purchase of sensitized stainless steels and to prevent sensitization during component fabrication. Wrought austenitic stainless steel stock used for components that are part of the RCPB, systems required for reactor shutdown, systems required for emergency core cooling and reactor vessel internals that are relied upon to permit adequate core cooling for normal operation or under postulated accident conditions is utilized in 1 of the following conditions:

1. Solution annealed and water quenched.

2. Solution annealed and cooled through the sensitization temperature range within less than approximately 5 minutes.

It is generally accepted that these practices will prevent sensitization. Westinghouse has verified this by performing corrosion tests (ASTM 393) on as-received wrought material.

Westinghouse recognizes that the heat affected zones of welded components must, of necessity, be heated into the sensitization temperature range, 800 to 1500°F. However, severe sensitization, i.e., continuous grain boundary precipitates of chromium carbide, with adjacent chromium depletion, can still be avoided by control of welding parameters and welding processes.

The heat input [(1)] and associated cooling rate through the carbide precipitation range are of primary importance. Westinghouse has demonstrated this by corrosion testing a number of weldments.

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(1) Heat input is calculated according to the formula:

$$H = \frac{(E)(I)(60)}{S}$$

Where: H = joules/in; E = volts; I = Amperes; and S = Travel Speed in inches/minute.

Of 25 production and qualification weldments tested, representing all major welding processes, and a variety of components, and incorporating base metal thicknesses from 0.10 to 4.0 inches, only portions of 2 were severely sensitized. Of these, one involved a heat input of 120,000 joules, and the other involved a heavy socket weld in relatively thin walled material. In both cases, sensitization was caused primarily by high heat inputs relative to the section thickness. However, in only the socket weld did the sensitized condition exist at the surface, where the material is exposed to the environment. The component has been redesigned to eliminate this weld.

Westinghouse controls the heat input in all austenitic pressure boundary weldments by:

1. Prohibiting the use of block welding.

2. Limiting the maximum interpass temperature to 350°F.

3. Exercising approval rights on all welding procedures.

To further assure that these controls are effective in preventing sensitization, Westinghouse per WCAP-8678 (Reference [16]) conducted additional intergranular corrosion tests of qualification mock-ups of primary pressure boundary and core internal component welds, including the following: RN 98-175

1. Reactor vessel safe ends.

2. Pressurizer safe ends.

3. Surge line and reactor coolant pump nozzles.

4. Control rod drive mechanisms head adapters.

5. Control rod drive mechanisms seal welds.

6. Control rod extensions.

7. Lower instrumentation penetration tubes.

To summarize, Westinghouse has a 4 point program designed to prevent intergranular attack of austenitic stainless steel components.

1. Control of primary water chemistry to ensure a benign environment.

2. Utilization of materials in the final heat treated condition and the prohibition of subsequent heat treatments in the 800 to 1500°F temperature range.

3. Control of welding processes and procedures to avoid heat affected zone sensitization.

4. Confirmation that the welding procedures used for the manufacture of components in the primary pressure boundary and of reactor internals do not result in the sensitization of heat affected zones.

Both operating experience and laboratory experiments in primary water have conclusively demonstrated that this program is 100% effective in preventing intergranular attack in Westinghouse NSSS's utilizing unstabilized austenitic stainless steel.

5.2.5.6 Retesting Unstabilized Austenitic Stainless Steels Exposed to Sensitization Temperatures

Unstabilized austenitic stainless steels are not normally exposed to the sensitization range of 800 to 1500°F during fabrication into components except as described in Section 5.2.3.2. If, during the course of fabrication, the steel is inadvertently exposed to the sensitization temperature range, 800 to 1500°F, the material may be tested in accordance with ASME A393 or A262 as amended by Westinghouse Process Specification 84201 MW to verify that it is not susceptible to intergranular attack, except that testing is not required for:

1. Cast metal or weld metal with a ferrite content of 5% or more.

2. Material with a carbon content of 0.03% or less that is subjected to temperatures in the range of 800 to 1500°F for less than 1 hour.

3. Material exposed to special processing provided the processing is properly controlled to develop a uniform product and provided that adequate documentation exists of service experience and/or test data to demonstrate that the processing will not result in increased susceptibility to intergranular stress corrosion.

If it is not verified that such material is not susceptible to intergranular attack, the material will be resolution annealed and water quenched or rejected.

5.2.5.7 Control of Delta Ferrite in Austenitic Stainless Steel Welding

Regulatory Guide 1.31, describes a method for implementing General Design Criterion 1 of Appendix A to 10 CFR 50 and Appendix B to 10 CFR 50 with regard to control of welding austenitic stainless steel components and systems. The Interim Regulatory Position of this guide, March 1974, describes an alternative method of control. The following paragraphs describe the methods used by Westinghouse and the verification of these methods for austenitic stainless steel welding on this application. Additional discussion on the recommendations of Regulatory Guide 1.31 is given in Appendix 3A.

The welding of austenitic stainless steel is controlled to mitigate the occurrence of microfissuring or hot cracking in the weld. Although published data and experience have not confirmed that fissuring is detrimental to the quality of the weld, it is recognized that such fissuring is undesirable in a general sense. Also, it has been well documented in the technical literature that the presence of delta ferrite is one of the mechanisms for reducing the susceptibility of stainless steel welds to hot cracking. However, there is insufficient data to specify a minimum delta ferrite level below which the material will be prone to hot cracking. It is assumed that such a minimum lies somewhere between 0 and 3% delta ferrite.

The scope of these controls discussed herein encompasses welding processes used to join stainless steel parts in components designed, fabricated or stamped in accordance with the ASME Code, Section III Class 1, 2, and core support components. Delta ferrite control is appropriate for the above welding requirements except where no filler metal is used or for other reasons such control is not applicable. These exceptions include electron beam welding, autogenous gas shielded tungsten arc welding, explosive welding, and welding using fully austenitic welding materials.

The fabrication and installation specifications require welding procedure and welder qualification in accordance with Section III, and include the delta ferrite determinations for the austenitic stainless steel welding materials that are used for welding qualification testing and for production processing. Specifically, the undiluted weld deposits of the "starting" welding materials are required to contain a minimum of 5% delta ferrite(1) as determined by chemical analysis and calculation using the appropriate weld metal constitution diagrams in Section III. When new welding procedure qualification tests are evaluated for these applications, including repair welding of raw materials, they are performed in accordance with the requirements of Section III and Section IX.

The results of the destructive and nondestructive tests are reported in the procedure qualification record in addition to the information required by Section III.

(1) The equivalent ferrite number may be substituted for percent delta ferrite.

The "starting" welding materials used for fabrication and installation welds of austenitic stainless steel materials and components meet the requirements of Section III. The austenitic stainless steel welding material conforms to ASME weld metal analysis A-7, Type 308 for all applications except Type 308L weld metal analysis may be substituted for consumable inserts when used for weld root closures. Bare weld filler metal, including consumable inserts, used in inert gas welding processes conform to ASME SFA-5.9, and are procured to contain not less than 5% delta ferrite according to Section III. Weld filler metal materials used in flux shielded welding processes conform to ASME SFA-5.4 or SFA-5.9 and are procured in a wire-flux combination to be capable of providing not less than 5% delta ferrite in the deposit according to Section III. Welding materials are tested using the welding energy inputs to be employed in production welding.

Combinations of approved heats and lots of "starting" welding materials are used for all welding processes. The welding quality assurance program includes identification and control of welding material by lots and heats as appropriate. All of the weld processing is monitored according to approved inspection programs which include review of "starting" materials, qualification records and welding parameters. Welding systems are also subject to quality assurance audit including calibration of gages and instruments; identification of "starting" and completed materials; welder and procedure qualifications; availability and use of approved welding and heat treating procedures; and documentary evidence of compliance with materials, welding parameters and inspection requirements. Fabrication and installation welds are inspected using nondestructive examination methods according to Section III rules.

To further assure the reliability of these controls, Westinghouse has completed a verification program, which is described in Reference [9]. The program has been approved as a valid approach to verify the Westinghouse hypothesis and is considered an acceptable alternative to production weld testing for conformance with the NRC Interim Position on Regulatory Guide 1.31. The Regulatory Staff's acceptance letter and topical report evaluation were received on December 30, 1974. The results of the verification program, which do not support the original hypothesis, are summarized in Reference [10].

It should be noted that the criteria discussed above concerning delta ferrite determinations were incorporated in the Virgil C. Summer Nuclear Station components that were involved in the Westinghouse verification program; these components cannot necessarily be identified. Those components not involved in the verification program were fabricated in accordance with applicable ASME Code requirements, which do not include delta ferrite determinations. Therefore, the delta ferrite determinations performed on the Virgil C. Summer Nuclear Station components are in addition to the applicable ASME Code requirements.

5.2.6 PUMP FLYWHEELS

The integrity of the reactor coolant pump flywheel is assured on the basis of the following design and quality assurance procedures.

5.2.6.1 Design Basis

The calculated stresses at operating speed are based on stresses due to centrifugal forces. The stress resulting from the interference fit of the flywheel on the shaft is less than 2000 psi at zero speed, but this stress becomes zero at approximately 600 rpm because of radial expansion of the hub. The primary coolant pumps run at approximately 1190 rpm and may operate briefly at overspeeds up to 109% (1295 rpm) during loss of outside load. For conservatism, however, 125% of operating speed was selected as the design speed for the primary coolant pumps. The flywheels are given a preoperational test of 125% of the maximum synchronous speed of the motor.

5.2.6.2 Fabrication and Inspection

The flywheel consists of 2 thick plates bolted together. The flywheel material is produced by a process that minimizes flaws in the material and improves its fracture toughness properties, such as vacuum degassing, vacuum-melting, or electroslag remelting. Each plate is fabricated from A533, Grade B, Class 1 steel. Supplier certification reports are available for all plates and demonstrate the acceptability of the flywheel material on the basis of the requirements of Regulatory Guide 1.14.

No welding, including repair welding, is performed on the flywheels. The tensile (from material certifications) and Charpy-V-notch data for the flywheel discs are tabulated in Tables 5.2-23 and 5.2-24.

Flywheel blanks are flame-cut from the A533, Grade B, Class 1 plates with at least 1/2 inch of stock left on the outer and bore radii for machining to final dimensions. The finished machined bores, keyways, and drilled holes are subjected to magnetic particle or liquid penetrant examinations. The finished flywheels are subjected to 100% volumetric ultrasonic inspection per Paragraphs NB-2532.1 and NB-2532.2 of the ASME Code, Section III.

The reactor coolant pump motors are designed such that the flywheel is available by removing the cover to provide access to allow an inservice inspection program in accordance with requirements of Section XI of the ASME Code and the recommendations of Regulatory Guide 1.14 (see Appendix 3A).

5.2.6.3 Acceptance Criteria and Compliance with Regulatory Guide 1.14

The reactor coolant pump motor flywheel conforms to the following material acceptance criteria.

1. The nil-ductility transition temperature (NDTT) of the flywheel material is obtained by 2 drop weight tests (DWT) which exhibit "no-break" performance at 20°F in accordance with ASTM E-208. The above drop weight tests demonstrate that the NDTT of the flywheel material is no higher than 10°F.

2. A minimum of 3 Charpy V-notch specimens from each plate is tested at ambient (70°F) temperature in accordance with the specification ASTM E-23. The Charpy V-notch (C_v) energy in both the parallel and normal orientation with respect to the rolling direction of the flywheel material is at least 50 foot pounds at 70°F to demonstrate that RT_{NDT} is no higher than 10°F. A lower bound K_{Ic} versus temperature curve from tests of SA-533 Grade B, Class 1 and SA-508 Class 2 steels is provided in ASME Boiler and Pressure Vessel Code, Section XI, Appendix A, Figure A-4200-1. The temperature scale of this figure is referenced to RT_{NDT}, the reference nil-ductility temperature. The fracture toughness K_{Ic} given in Figure A-4200-1 is based on lower bound data from static fracture toughness tests. Reference of this curve to the guaranteed RT_{NDT} of +10°F gives a minimum static fracture toughness of 150 ksi $in^{1/2}$ at 90°F. The expected normal operating temperature of the Virgil C. Summer Nuclear Station reactor coolant pump motor flywheels is in excess of 110°F. Therefore, the minimum static fracture toughness requirement of 150 ksi $in^{1/2}$ is exceeded. A lower bound K_{Id} reference curve (see Figure 5.2-3) has been constructed from dynamic fracture toughness data generated in A533, Grade B, Class 1 steel [11]. All data points are plotted on the temperature scale relative to the NDTT. The construction of the lower bound curve below which no single test point falls, combined with the use of dynamic data when flywheel loading is essentially static, together represent a large degree of conservatism. Reference of this curve to the guaranteed NDTT of +10°F indicates that, at the predicted flywheel operating temperature of 110°F, the minimum fracture toughness is in dynamic stress intensity factor must be at least 100 KSI-$in^{1/2}$. This conforms to the Regulatory Guide 1.14 requirement that the dynamic stress intensity factor must be at least 100 KSI-$in^{1/2}$.

Precautionary measures taken to preclude missile formation from primary coolant pump components assure that the pumps will not produce missiles under any anticipated accident condition. Each component of the primary pump motors has been analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy case.

Thus, it is concluded that flywheel plate materials are suitable for use and can meet Regulatory Guide 1.14 acceptance criteria on the bases of suppliers certification data.

5.2.7 REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS

Leakage detection systems are required to detect, and to the extent practical, identify the source of reactor coolant leakage. The airborne radiation monitoring system, described in Sections 11.4 and 12.2.4, provides input which is sensitive to leakage detection when activity is present in reactor coolant. Portions of this system, in addition to quantitative leak detection systems located in the reactor building, enable the operator to adequately locate and analyze the reactor coolant pressure boundary leakage.

5.2.7.1 Leakage Detection Method

Leakage detection system instrumentation sensitivity is 1 gpm or less. Response time of less than 1 hour is consistent with the requirements of General Design Criterion 30. Sufficient range overlap and multiple instruments ensure shorter overall response time for leakage detection. Reactor building sump level instrumentation is Seismic Category I. The various methods of leakage detection employed are discussed in Sections 5.2.7.1.1 through 5.2.7.1.4 and 5.2.7.8.

5.2.7.1.1 Activity Detection

Consistent with the recommendations of Regulatory Guide 1.45 (see Appendix 3A), radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS leakage but have recognized limitations on achieving the recommended response time of 1 hour or less when there are few fuel cladding defects and reactor coolant activity levels are low.

The reactor building air sample line radiation monitor (RM-A2) is the most sensitive to reactor coolant leakage. Alarms are provided to alert the operator to the onset of or changes in reactor coolant leakage. The monitor is typically set to provide the most sensitive response without causing an excessive number of spurious alarms in order to provide an early indication of the onset of or changes in reactor coolant leakage, and an audible alarm is provided in the control room when the monitor's setpoint is exceeded. Section 12.2.4 discusses airborne radiation monitoring instrumentation, sensitivities and setpoints. The radiation monitor (RM-A2) satisfies the seismic requirements of Regulatory Guide 1.45.

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Leakage from the reactor coolant pressure boundary into the component cooling water system is detected by process radiation monitors (RM-L2A and RM-L2B). These radiation monitors are described in Section 11.4.

5.2.7.1.2 Detection by Temperature, Pressure or Drain Flows

Reactor building temperature and pressure monitors provide indirect indication of gross reactor coolant pressure boundary leakage. However, the most sensitive leakage detection system, other than activity detection, is the reactor building cooling unit condensate drain flow alarm. A flow switch is located in each of the common condensate drain headers from the reactor building cooling units (see Figures 5.2-9 and 9.3-5). Each flow switch is set to actuate an alarm in the control room, should a flow rate exceeding 0.5 gpm occur. The response time for this system to indicate a 1 gpm reactor coolant leak is approximately 15 minutes. The reactor building temperature and pressure monitors and the flow switches satisfy the seismic requirements of Regulatory Guide 1.45.

5.2.7.1.3 Leak Detection Sumps

The Reactor Building Sump is used to collect and quantify leakage originating in the Reactor Building. Detection of leakage is accomplished by obtaining a flow rate into the sump from comparison of level changes over a specified time period. Whenever the measured flow rate into the sump from unidentified sources exceeds 1 gallon per minute, a "greater than 1 gpm" alarm is actuated. To ensure reliability, the level monitor for the RB sump is Seismic Category I. This detection method satisfies the requirements of Regulatory Guide 1.45 as one of the diverse methods for determination of reactor coolant leakage employed in the design of Virgil C. Summer Nuclear Station. RN 08-017

The Incore Instrument Sump, although not credited in the VCSNS SER nor in Technical Specifications, provides another means to collect and quantify unidentified leakage originating in the Reactor Building.

The CRDM condensate drain line flow transmitter installed under ECR 50677 monitors leakage in the range of 0.5 GPM to 1.0 GPM. The new flow transmitter provides enhanced RCS leak detection but is not credited in the VCSNS SER nor in the Technical Specifications. RN 13-021

5.2.7.1.4 Indirect Detection

Low pressurizer level indication or a discrepancy between net letdown and makeup flow, if unexplained, are good indications of reactor coolant leakage.

5.2.7.2 Indication in Control Room

Display instrumentation for the variables providing indirect indication for the leakage detection systems has audible alarms and measured variable recording devices in the control room. Instrumentation is of sufficient redundancy and quantity to assure that leakage is monitored. Regulatory Guide 1.45 is discussed in Appendix 3A.

5.2.7.3 Limits for Reactor Coolant Leakage

Reactor coolant pressure boundary leakage is defined as leakage through a nonisolable fault in a reactor coolant system component body, pipe wall or vessel wall. Unidentified leakage limitations are discussed in the Technical Specifications. Unidentified leakage is expected to be less than 1 gpm. Identified leakage less than 10 gpm is acceptable provided that it does not interfere with detection of unidentified leakage by the leakage detection systems.

5.2.7.4 Unidentified Leakage

Unidentified leakage is any reactor coolant system leakage which is not Identified or Controlled leakage (see Section 5.2.7.5).

5.2.7.5 Maximum Allowable Total Leakage

The maximum allowable total leakage includes controlled leakage and identified leakage.

Controlled leakage is defined as seal water flow to reactor coolant pump seals. Operation is limited if this flow exceeds 33 gpm with the supply modulating valve fully open and reactor coolant system pressure at 2235 psig, nominal. Flow instrumentation is provided for the purpose of monitoring this leakage. 98-01

Identified leakage includes the following:

1. Leakage into closed systems, such as flange seals and valve leakoffs to collection tanks, normal letdown, sampling of auxiliary systems and pressurizer relief valve discharge to pressurizer relief tank. These leakage paths contain temperature elements to indicate leakage flow, as well as high level tank, high flow rate and radiation monitor alarms in auxiliary systems.

2. Other leakages not related to the reactor coolant pressure boundary which do not interfere with leakage detection system operation.

3. Reactor coolant leakage through a steam generator (not a steam generator tube rupture) to the secondary system. This leakage is limited to less than a total of 150 gal/day from each steam generator. This leakage is monitored by radiation monitors RM-L3 and RM-A9, as well as by periodic sampling discussed in Section 11.4. RN 03-050

5.2.7.6 Differentiation between Identified and Unidentified Leaks

The leakage detection system provides for differentiation between identified and unidentified leakage since identified leakage flows into closed containers or does not interfere with detection system operation. The leakage detection system provides both qualitative and quantitative information for use by the operator in determining the location and magnitude of leakage.

5.2.7.7 Sensitivity and Operability Tests

Proper functioning of leak detection instrumentation is assured by testing and calibration procedures. Sensitivity and calibration of radiation monitoring equipment are discussed in Section 11.4 and 12.2.4.

5.2.7.8 Intersystem Leakage

Reliable methods of intersystem leakage detection for systems connected to the reactor coolant system are described below:

1. Safety Injection System

Safety injection system cold leg connection leakage will be detected by lifting of residual heat removal relief valves 8864A or 8864B. 98-01

Safety injection system hot leg connection leakage will be detected by lifting of residual heat removal relief valve 8865.

Accumulator connection leakage will be detected by:

a. Redundant high accumulator level indication and alarm, and/or

b. Redundant high accumulator pressure indication and alarm.

High head cold leg injection and high head hot leg recirculation line leakage is not of concern since these are normally exposed to charging flow. 98-01

2. Residual Heat Removal System

Residual heat removal suction connection leakage will be detected by lifting of residual heat removal suction relief valves 8708A or 8708B accompanied by increased pressurizer relief tank level and temperature indication and alarm.

Residual heat removal discharge connection leakage will be detected by lifting of residual heat removal relief valves 8864A or 8864B. 98-01

3. Waste Processing System

Cold leg leakage into the waste processing system will be collected in the reactor coolant drain tank and detected by:

a. An increased load on the collection system, and/or

b. Lifting of reactor coolant drain tank relief valve 7169.

Leakage from the reactor vessel flange leakoff connection will be detected by high temperature indication and alarm from the leakage piping.

4. Pressurizer Relief Tank

Reactor coolant system leakage to the pressurizer relief tank will be detected by:

a. High temperature indication and alarm in the pressurizer safety valve and power operated relief valve discharge lines, and/or

b. High temperature or high level indication and alarms in the pressurizer relief tank, and

c. The acoustical type leak monitoring system and pressurizer safety valve alarm, and RN 14-037

d. PORV position indication.

5. Chemical and Volume Control System

Since the only chemical and volume control system to reactor coolant system connections are utilized for reactor coolant system letdown and charging, intersystem leakage detection is not relevant.

6. Valve Leakage

As discussed in Section 9.3.3.2, Item 10, and illustrated by Figures 9.3-13 and 9.3-13a, valve packing box leakage from various systems is monitored by means of level switches.

7. Main Steam System

Steam generator tube leakage will be detected by:

a. Use of steam generator blowdown and sampling (see Sections 9.3.2 and 10.4.8) which includes use of radiation monitors, RM-L3 and RM-L10 (see Section 11.4.2) and/or

b. Main plant vent exhaust radiation monitor, RM-A3 (see Section 11.4.2) and/or

c. Turbine room sump radiation monitor, RM-L8 (see Section 11.4) and/or RN 03-050

d. Condenser exhaust monitor, RM-A9 (see Section 11.4).

8. Component Cooling Water System

As discussed in Section 11.4.2, radiation monitors, RM-L2A and/or RM-L2B, will detect intersystem leakage into the component cooling water system.

Residual heat removal heat exchanger leakage will be detected by high temperature indication and alarm.

Reactor coolant drain tank heat exchanger leakage will be detected by:

a. High temperature indication and alarm and/or

b. High flow indication.

Reactor coolant pump thermal barrier leakage will be detected by:

a. High temperature indication and alarm and/or

b. High flow indication and alarm.

Reactor coolant pump upper and lower bearing leakage will be detected by:

a. High temperature indication and alarm and/or

b. High flow indication.

Letdown heat exchanger leakage will be detected by:

a. High temperature indication and alarm and/or

b. High flow indication.

Seal water heat exchanger leakage will be detected by:

a. High temperature indication and alarm and/or

b. High flow indication.

Residual heat removal pump leakage will be detected by:

a. High temperature indication and alarm and/or

b. High flow indication.

5.2.8 INSERVICE INSPECTION PROGRAM

Refer to Section 5.7.

5.2.9 REFERENCES

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15. WCAP-13777, "Addendum to Final (Summary) Stress Report 157" PWR Vessel Virgil C. Summer Nuclear Station Revision 2 (Uprating/Steam Generator Replacement Evaluation)," June 1993.

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18. WCAP-9802, "Structure Analysis of the Reactor Coolant Loop/Support for the V. C. Summer Station Volume III, Reactor Coolant Loop Branch Nozzle Analysis." RNs 98-176 00-051

19. WCAP-9803, "ASME Section III Class 1 Piping Stress Analysis for the V. C. Summer Station Volume IV, Piping Analysis of the Class 1 Auxiliary Piping Systems." RNs 98-176 00-051

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TABLE 5.2-1

APPLICABLE CODE ADDENDA FOR RCS COMPONENTS

Reactor Vessel	ASME III, 1971 Ed.
Steam Generator (∆75)	ASME III, 1971 Ed. through Summer ‘71 (Design & Analysis) ASME III, 1986 Ed. (Construction)
Pressurizer	ASME III, 1971 Ed. through Summer ‘71
CRDM Housing	
Full Length	ASME III, 1971 Ed. through Winter ‘72
Part Length	ASME III, 1971 Ed through Winter ‘72
CRDM Head Adapter	ASME III, 1971 Ed. through Summer ‘71
Reactor Coolant Pump	ASME III, 1971 Ed. through Summer ‘72
Reactor Coolant Pressure Boundary Pipe:	
(1) 1 1/2" CVCS Seal Injection Line	ASME III, 1974 Ed through Summer ‘76
(2) 1" and smaller Class 1 Piping	ASME III, 1974 Ed through Summer ‘75
(3) 29" I.D. Hot Leg Pipe, 31" I.D. Cross Over Leg Pipe, 27.5" I.D. Cold Leg Pipe	ASME III, 1971 Ed. through Winter '71
(4) Class I Pipe Fatigue Qualification	ASME III, 1977 Ed. through Summer ‘79
(5) All other piping	ASME III, 1971 Ed. through Summer ‘73
Valves	
Pressurizer Safety	ASME III, 1971 Ed. through Winter ‘72
Other	ASME III, 1971 Ed. through Summer ‘71

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TABLE 5.2-2

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

Normal Conditions	Occurrences
1. Heatup and cooldown at 100°F/hr (pressurizer cooldown 200°F/hr)	200 (each)
2. Unit loading and unloading at 5% of full power/min	13,200 (each)
3. Step load increase and decrease of 10% of full power	2,000 (each)
4. Large step load decrease	200
5. Steady-state fluctuations	
a. Initial Fluctuations	1.5×10^5
b. Random Fluctuations	3.0×10^6
6. Feedwater Heater Out of Service	40
Upset Conditions	
1. Loss of load, (without immediate reactor trip)	80
2. Loss of power	40
3. Partial loss of flow	80
4. Reactor trip from full power	400
5. Inadvertent auxiliary spray	10
6. Operating basis earthquake (20 earthquakes of 20 cycles each)	400
7. Excessive Feedwater Flow	30
Faulted Conditions[1]	
1. Reactor coolant branch line pipe break	1
2. Large steam line break	1
3. Steam generator tube rupture	(included in 4 above, reactor trip from full power)
4. Safe Shutdown Earthquake	1

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TABLE 5.2-2 (Continued)

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

Test Conditions	Occurrences
1. Turbine roll test	10
2. Primary side hydrostatic test	5
3. Secondary side hydrostatic test	5
4. Primary side leakage test	50

[1] In accordance with the ASME Boiler and Pressure Vessel Code, Section III, faulted conditions are not included in fatigue evaluation.

TABLE 5.2-3

LOADING COMBINATIONS FOR ASME CLASS 1
COMPONENTS AND SUPPORTS

Condition Classification	Loading Combination
Design	Design Pressure, Design Temperature, Deadweight, Operating Basis Earthquake
Normal	Normal Condition Transients, Deadweight
Upset	Upset Condition Transients, Deadweight, Operating Basis Earthquake
Faulted	Faulted Condition Transients, Deadweight, Safe Shutdown Earthquake or (Safe Shutdown Earthquake and Pipe Rupture Loads)

TABLE 5.2-4

ALLOWABLE STRESSES FOR ASME SECTION III CLASS 1 COMPONENTS[1]

Operating Condition Classification	Vessels/Tanks	Piping	Pumps	Valves	Component Supports
Normal	ASME Section III	ASME Section III	ASME Section III	ASME Section III	See Section 5.2.1.10.7
Upset	ASME Section III	ASME Section III	ASME Section III	ASME Section III	See Section 5.2.1.10.7
Faulted	ASME Section III	ASME Section III	ASME Section III	[2]	See Section 5.2.1.10.7
	See Section 5.2.1.11	See Section 5.2.1.11	See Section 5.2.1.11		
			(No active Class 1 pump used)		

P_e, P_m, P_b, Q_t, C_p, S_n, & S_m as defined by Section III ASME Code

(1) A test of the components may be performed in lieu of analysis.
(2) Class 1 Valve Faulted Condition Criteria

TABLE 5.2-4 (Continued)

ALLOWABLE STRESSES FOR ASME SECTION III CLASS 1 COMPONENTS[(1)]

Active	Inactive
a) Calculated P_m from paragraph NB3545.1 with Internal Pressure P_s = 1.25P_s $P_m \leq 1.5S_m$	a) Calculated Pm from paragraph NB3545.1 with Internal Pressure P_s = 1.50P_s $P_m \leq 2.4S_m$ or 0.7 Su
b) Calculated Sn from paragraph NB345.2 with C_p = 1.5 P_s = 1.25Ps $Q_t^2 = 0$ Ped = 1.3X value of Ped from equations of NB3545.2 (b)(1) $S_n \leq 3S_m$	b) Calculate Sn from paragraph NB3545.2 with C_p = 1.5 P_s = 1.50Ps $Q_t^2 = 0$ Ped = 1.3X value of Ped from equations of NB3545.2 (b)(1) $S_n \leq 3\ S_m$

(1) A test of the components may be performed in lieu of analysis.

TABLE 5.2-5

ALLOWABLE STRESSES FOR PRIMARY EQUIPMENT SUPPORTS

Loading Conditions	Stress Limits
Design/Normal	AISC, Seventh Edition[(1)], Part 1, Allowable Stresses
Upset	AISC, Seventh Edition, Part 1, Allowable Stresses
Faulted	Stresses ≤ yield strength of material. Local yielding is permitted but limited so that the structural integrity of the system is maintained. As an alternative to the above, 80 percent of L_T[(2)] may be used.

---

(1) Specification for the Design, Fabrication and Erection of Structural Steel for Buildings.

(2) L_T = The limits established for the analysis need not be satisfied if it can be shown from the test of a prototype or model that the specified loads (dynamic or static equivalent) do not exceed 80 percent of L_T, where L_T is the ultimate load or load combination used in the test. In using this method, account is taken of the size effect and dimensional tolerances (similitude relationships) which may exist between the actual component and the tested models to assure that the loads obtained from the test are a conservative representation of the load carrying capability of the actual component under postulated loading for faulted conditions.

TABLE 5.2-6

RELIEF VALVE DISCHARGE TO THE PRESSURIZER RELIEF TANK

Reactor Coolant System

3	Pressurizer Safety Valves	Figure 5.1-1, Sheet 2
3	Pressurizer Power Operated Relief Valves	Figure 5.1-1, Sheet 2
1	Reactor Vessel Head Vent System Letdown Line	Figure 5.5-13

Emergency Core Cooling System

3	Low Head Safety Injection/Residual Heat Removal Pump Discharge Lines	Figure 6.3-1, Sheet 2

Residual Heat Removal System

2	Residual Heat Removal Pump Suction Line from the Reactor Coolant System Hot Legs	Figure 5.5-4

Chemical and Volume Control System

2	Charging Pump Suction	Figure 9.3-16, Sheet 2
1	Seal Water Return Line	Figure 9.3-16, Sheet 1
1	Letdown Line	Figure 9.3-16, Sheet 1

TABLE 5.2-7

REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS

	PSIG	
Hydrostatic Test Pressure	3107	
Design Pressure	2485	
Safety Valves (Begin to Open)	2485	
High Pressure Reactor Trip	2380	RN 98-174
Power Relief Valves	2335 (1) (2)	
High Pressure Alarm	2310 (2)	
Pressurizer Spray Valves (Full Open)	2310 (2)	
Pressurizer Spray Valves (Begin to Open)	2260 (2)	
Proportional Heaters (Begin to Operate)	2250 (2)	
Operating Pressure	2235	
Proportional Heater (Full Operation)	2220 (2)	
Backup Heaters On	2210 (2)	
Low Pressure Alarm	2210 (2)	
Pressurizer Relief Valve Interlock	1985	RN 98-174
Low Pressure Reactor Trip	1870	

(1) At 2335 psig, a pressure signal initiates actuation (opening) of these valves. Remote manual control is also provided.

(2) Setpoint is affected by integral portion of controller.

TABLE 5.2-7a

CHECK VALVES CLASSIFIED AS CATEGORY AC

Valve Tag Number	Valve Tag Number	Valve Tag Number
8998A	8992A	8973B
8998B	8992B	8973C
8998C	8992C	8974A
8997A	8990A	8974B
8997B	8990B	8956A
8997C	8990C	8956B
8995A	8988A	8956C
8995B	8988B	
8995C	8948A	
8993A	8948B	
8993B	8948C	
8993C	8973A	

RN 99-091

TABLE 5.2-8

REACTOR COOLANT PRESSURE BOUNDARY
MATERIALS CLASS 1 PRIMARY COMPONENTS

CLASS 1 PRIMARY COMPONENTS

Reactor Vessel Components

Shell & Head Plates (other than core region)	SA533 Gr A, B, or C, Class 1 or 2 (Vacuum treated)	
Shell Plates (core region) (Vacuum treated)	SA533 Gr A or B, Class 1	
Shell, Flange & Nozzle Forgings Nozzle Safe Ends	SA508 Class 2 and 3 Weld Filler Metal Alloy 600 except "A" Hot Leg which is Alloy 690	RN 04-031
CRDM and/or ECCS Appurtenances Upper Head	SB166 or 167 and SA182 Type F304	
Instrumentation Tube Appurtenances - Lower Head	SB166 or 167	RN 01-113
Closure Studs, Inserts and Adaptors	SA-540 Gr B-24 Class 3 (as modified by Code Case 1605)	RN 08-030 14-003 15-021
Closure Nuts, Washers	SA-540 Gr B-23/24 Class 3	
Core Support Pads	SB166 with Carbon less than 0.10%	
Monitor Tubes & Vent Pipe	SA312 or 376 Type 304 or 316 or SB167	
Vessel Supports, Seal Ledge & Heat Lifting Lugs	SA516 Gr 70 Quenched & Tempered or SA533 Gr A, B or C, Class1 or 2. (Vessel supports may be of weld metal buildup of equivalent strength)	
Cladding & Buttering	Stainless steel weld Metal Analysis A-7 and Ni-Cr-Fe Weld Metal F-Number 43 (Alloy 600 except Buttering on "A" Hot Leg which is Alloy 690)	RN 04-031

Steam Generator Components

Shell	SA508 Class 3a	
Pressure Forgings (including nozzles and tubesheet)	SA508 Class 3a	
Nozzle Safe Ends	Forged Stainless Steel	RN 01-113
Channel Head	SA508 Class 3a	
Tubes	SB163 Ni-Cr-Fe Thermally treated, procured to Code Case N-20-3	

TABLE 5.2-8 (Continued)

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS
CLASS 1 PRIMARY COMPONENTS

Steam Generator Components (Cont'd)	
Cladding	Austenitic Stainless
Channel Head Surfaces	Steel 309L/308L
Tubesheet Primary Side Surfaces	Ni-Cr-Fe (N06082/W86182)
Closure Studs	SA193 Gr B-7
Closure Nuts	SA194 Gr 7
Pressurizer Components	
Pressure Plates	SA533 Gr A, B or C, Class 1 or 2
Pressure Forgings	SA508 Class 2 or 3
Nozzle Safe Ends	SA182 or 376 Type 316 or 316L and Ni-Cr-Fe Weld Metal F-Number 43
Cladding and Buttering	Stainless Steel Weld Metal Analysis A-7 and Ni-Cr-Fe Weld Metal F-Number 43
Closure Bolting	SA193 Gr B-7
Pressurizer Safety Valve Forgings	SA182 Type F316
Reactor Coolant Pump	
Pressure Forgings	SA182 F304, F316, F347 or F348
Pressure Casting	SA351 Gr CF8, CF8A, or CF8M
Tube & pipe	SA213, SA376 or SA312- Seamless Type 304 or 316
Pressure Plates	SA240 Type 304 or 316
Bar Material	SA479 Type 304 or 316
Closure Bolting	SA193, SA320, SA540, SA453, Gr 660
Flywheel	SA533 Gr B, Class 1

RN 01-113

TABLE 5.2-8 (Continued)

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS
CLASS 1 PRIMARY COMPONENTS

Reactor Coolant Piping	
Reactor Coolant Pipe	SA376 Gr 304N
Reactor Coolant Fittings	SA351 Gr CF8A
Branch Nozzles	SA182 Code Case 1423-2 Gr 304N
Surge Line	SA376 Gr 304, 316 or F304N
Auxiliary Piping 1/2" through 12" and wall schedules 40S through 80S (ahead of second isolation valve)	ANSI B36.19
All other Auxiliary Piping (ahead of second isolation valve)	ANSI B36.10
Socket Weld Fittings	ANSI B16.11
Piping Flanges	ANSI B16.5
Full Length Control Rod Drive Mechanism	
Latch Housing	SA182 Gr F304 or SA351 Gr CF8
Rod Travel Housing	SA182 Gr F304 or SA336 Gr F8
Cap	SA479 Type 304
Welding Materials	SFA5.4 and 5.9 Type 308 or 308L
Part Length Control Rod Drive Mechanism	
Latch Housing	SA182 Gr F304 or SA351 Gr CF8
Rod Travel Housing	SA182 Gr F304 or SA336 Gr F8
Closure Components	SA479 Type 304 and SA453 Gr 660
Welding Materials	SFA5.4 and 5.9 Type 308 and 308L

TABLE 5.2-9

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS
CLASS 1 AND 2 AUXILIARY COMPONENTS

Valves	
Bodies	SA182 Type F316 or SA351 Gr CF8 or CF8M
Bonnets	SA182 Type F316 or SA351 Gr CF8 or CF8M
Discs	SA182 Type F316 or SA564 Gr 630 Cond 1100°F Heat Treatment or SA351 Gr CF8 or CF8M
Pressure Retaining Bolting	SA453 Gr 660
Pressure Retaining Nuts	SA453 Gr 660 or SA194 Gr 6
Auxiliary Heat Exchangers	
Heads	SA240 Type 304
Nozzle Necks	SA182 Gr F304
Tubes	SA213 TP 304
Tubesheets	SA182 Gr F304
Shells	SA240 and SA312 Type 304
Auxiliary Pressure Vessels, Tanks, Filters, etc,	
Shells & Heads	SA240 Type 304 or SA264 consisting of SA537 CL1 with SA240 Type 304 L Cladding
Flanges & Nozzles	SA182 Gr F304 and SA350 Gr LF2, with SA240 Type 304 L Cladding
Piping	SA312 and SA240 TP304 or TP316 Seamless
Pipe Fittings	SA403 WP304 Seamless
Closure Bolting & Nuts	SA193 Gr B7 and SA194 Gr 2H

TABLE 5.2-9 (Continued)

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS
CLASS 1 AND 2 AUXILIARY COMPONENTS

Auxiliary Pumps

Pump Casing & Heads	SA351 Gr CF8 or CF8M, SA182 Gr F304 or F316
Flanges & Nozzles	SA182 Gr F304 or F316, SA403 Gr WP316L Seamless
Piping	SA312 TP304 or TP316 Seamless
Stuffing or Packing Box Cover	SA351 Gr CF8 or CF8M, SA240 TP304 or TP316
Pipe Fittings	SA403 Gr WP316L Seamless
Closure Bolting & Nuts	SA193 Gr B6, B7, or B8M and SA194 Gr 2H or Gr 8M, SA193 Gr B6, B7 or B8M; SA453 Gr 660; and Nut 8, SA194 Gr 2H, Gr 8M, and Gr 6

TABLE 5.2-10

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Electrical Conductivity	Affected by the concentration of boric acid and alkali present. Expected range is $<$ 1 to 40 μMhos/cm at 25°C.	RN 05-025
Solution pH	Affected by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration) at 25°C.	RN 05-025
Oxygen, ppm, maximum	Oxygen concentration of the reactor coolant is maintained below 0.1 ppm for plant operation above 250°F. Hydrazine may be used to chemically scavenge oxygen during heatup.	
Chloride, ppm, maximum	0.15	
Fluoride, ppm, maximum	0.15	
Hydrogen, cc (STP)/kg H_2O Reactor Power Level above 1 MWt, excluding decay heat during subcritical operation.	25-50	
Total Suspended Solids ppm, maximum	1.0	
pH Control Agent (Li^7OH)	Lithium concentration is coordinated with the boron concentration.	
Boric Acid, ppm B	Variable from 0 to approximately 4000	

TABLE 5.2-11

REACTOR VESSEL UNIRRADIATED TOUGHNESS DATA (TRANSVERSE) RN 00-001

Component	Heat Number	Material Type	Cu %	P %	NDTT (°F)	Min. 50 Ft-Lb 35 Mil Temp. (°F)	RT_{NDT} (°F)	Avg. Upper Shelf (ft-lb)
Closure Head	A9231-1	A533-B-Class 1	-	-	-20	40	-20	106
Head Flange	5297-V-1	SA508 Class 2	-	-	10	<60	10	129
Vessel Flange	5301-V-1	SA508 Class 2	-	-	0	<60	0	172
Inlet Nozzle	436B-1	SA508 Class 2	-	-	-20	<40	-20	130
Inlet Nozzle	436B-2	SA508 Class 2	-	-	0	<60	0	114.5
Inlet Nozzle	436B-3	SA508 Class 2	-	-	-20	<40	-20	135
Outlet Nozzle	437B-1	SA508 Class 2	-	-	-10	<50	-10	146
Outlet Nozzle	437B-2	SA508 Class 2	-	-	-10	<50	-10	165
Outlet Nozzle	437B-3	SA508 Class 2	-	-	0	<50	0	150
Nozzle Shell	C9955-2	A533-B-Class 1	0.13	0.010	-20	78	18	100.5
Nozzle Shell	C0123-2	A533-B-Class 1	0.12	0.009	-30	86	26	91
Inter. Shell	A9154-1	A533-B-Class 1	0.10	0.009	-20	90	30	80.5
Inter. Shell	A9153-2	A533-B-Class 1	0.09	0.006	-20	40	-20	106.5
Lower Shell	C9923-2	A533-B-Class 1	0.08	0.005	-10	70	10	91.5
Lower Shell	C9923-1	A533-B-Class 1	0.08	0.005	-30	70	10	106
Trans. Ring	A9249-1	A533-B-Class 1	-	-	-40	23	-37	107
Bottom Head	A9231-2	A533-B-Class 1	-	-	-10	42	-10	134
Core Region Weld	-	-	0.05	0.013	-50	16	-44	84 RN 00-001
Weld HAZ	-	-	-	-	-70	-37	-70	130

TABLE 5.2-12

REACTOR VESSEL INTERNALS FOR EMERGENCY CORE COOLING

Forgings	SA182 Type F304	
Plates	SA240 Type 304	
Pipes	SA312 Type 304 Seamless or SA376 Type 304	
Tubes	SA213 Type 304	
Bars	SA479 Type 304	
Castings	SA351 Gr CF8 or CF8A	
Bolting	Westinghouse PD Spec. 70041 EA Westinghouse PD Spec. 15106 DA	
Nuts	SA193 Gr B8	
Locking Devices	SA479 Type 304	
Weld Buttering	Stainless Steel Weld Metal Analysis A-7	
Guide tube support pins and nuts	SA-193 Gr B&M Type 316 per Code Case N-60-4	RN 01-086

TABLE 5.2-13

REACTOR COOLANT MECHANICAL LOADS

LOAD COMPONENT	ACCUMULATOR LINE BREAK (REF. 37)	PRESSURIZER SURGE LINE BREAK (REF. 37)
Axial Load (Fx)(Kips)	1652.6	1656.6
Vertical Load (Fy)(Kips)	-48.0	28.0
Lateral Load (Fz) (Kips)	-5.0	2.0
Axial Moment (Mx) (Ft-Kips)	310.3	121.7
Vertical Moment (My) (Ft-Kips)	83.8	4.4
Lateral Moment (Mz) (Ft-Kips)	-329.2	350.3

TABLE 5.2-14

MAXIMUM RPV DISPLACEMENTS (Node #3)

LOAD COMPONENT	ACCUMULATOR LINE BREAK	PRESSURIZER SURGE LINE BREAK
UX (in)		
Max.	+ .0022	+ .02825
Min.	- .0046	- .0267
UY (in)		
Max.	+ .0048	- .113
Min.	- .0143	- .00401
UZ (in)		
Max.	+ .0520	0.0
Min.	2.1E-5	- .04

TABLE 5.2-15

SUMMARY OF REACTOR VESSEL/INTERNALS INTERFACE LOADS

INTERFACE/COMPONENT	ACCUMULATOR LINE BREAK (LBS)	PRESSURIZER SURGE LINE BREAK (LBS)
Vessel/Barrel Flange (Hor.)	0.87E6	0.17E6
Vessel/Barrel Flange (Vert.)	2.9E6	2.37E6
Core Barrel Outlet Nozzle (Hor.)	0.4E6	0.21E6
Vessel/Upper Support Flange (Hor.)	1.5E5	-
Vessel Head/Upper Support Flange (Vert.)	1.1E6	.61E6
Lower Radial Key (Hor.)	3.20E6	2.41E6

TABLE 5.2-16

STEAM GENERATOR LOWER SUPPORT MEMBERS STRESSES

Member	Member Stress (% of allowable /loading condition) Normal	Upset	Faulted	
13	-			
14	-			
15	-	83 (1)	97 (1) (2)	
16	-			
17	-			
18	-			
20	-			
22	-	81 (1)	99 (1) (2)	
24	-			
26	-			
43				
44	27	65	93	RN 00-051
45				
46				

(1) Includes the effects of pipe supports and platform attachments.
(2) Conservatively includes faulted conditions thermal loads added to SRSS (SSE, LOCA).

RN 95-029 01-113

TABLE 5.2-17

STEAM GENERATOR UPPER SUPPORT MEMBER STRESSES

Member	Member Stress (% of allowable/loading condition) Normal	Upset	Faulted
(26,28) Snubbers *	-	-	-
70	-	25.9	25
71	-	41.1	41
72	-	37	35
36-69	-	54.5	78

RN 00-051

* Snubbers have been eliminated.

TABLE 5.2-18

REACTOR COOLANT PUMP SUPPORT MEMBER STRESSES

Member	Member Stress (% of allowable/loading condition) Normal	Upset	Faulted
4	-	58	81
5	-	-	-
6	-	43	75
7, 8, 9	16	89	75

TABLE 5.2-21

REACTOR VESSEL BELTLINE MATERIAL FRACTURE TOUGHNESS DATA

Beltline Material	Plate No.	Cu (%)	P (%)	RT_{NDT}* (°F)	Upper Shelf Energy (ft-lb)	32/56 EFPY Fluence **** (10^{19} N/cm^2)	32/56 EFPY ΔRT_{NDT}** (°F)
Nozzle Shell	C-9955-2	.13	.010	18	103,106.5,93	--	--
Nozzle Shell	C0123-2	.12	.009	26	97,87,89	--	--
Inter. Shell	A9154-1	.10	.009	30	82.5,76.5,83	3.92/6.80	87.9/94.8
Inter. Shell	A9153-2	.09	.006	-20	101,108,111.5	3.92/6.80	78.4/84.6
Lower Shell	C9923-1	.08	.005	10	104,114,100	3.92/6.80	69.0/74.4
Lower Shell	C9923-2	.08	.005	10	93,94,88	3.92/6.80	69.0/74.4
Core Region Welds	***	.05	.013	-44	79,87,87	1.36/2.35	45.8/51.9
Weld HAZ	A9154-1	-	-	-70	137,135,118	1.36/2.35	45.8/51.9

RN 04-042

* Determined from drop weight tests and full Charpy curve tests normal to the major working direction. Value reported is the higher of the NDTT temperature or the temperature 60°F below the 50 ft-lb/35 mil lateral expansion temperature.

** Based on WCAP-16298-NP, Revision 0, using methods identified in Regulatory Guide 1.99 Revision 2. Most limiting material is plate A9154-1. Test specimens from this plate are included in the reactor vessel surveillance program. The values were determined using results of capsules U, V, X, W, and Z test reports.

*** All core region welds were made with RACO INMM wire (heat no. 4P4784) and Linde 124 flux (lot no. 3930). The same heat of weld wire and lot of flux was also used to fabricate the surveillance program weldment. A post weld heat treatment at 1100°F-1175°F for 12 hours was applied to all the weld test material.

**** Calculated fast neutron fluence (E > 1.0 MeV) values at the inner surface of the reactor vessel (per WCAP-16298-NP, Revision 0). These values were projected using results of the Capsule Z radiation analysis.

TABLE 5.2-22

BOLTING MATERIAL FRACTURE TOUGHNESS DATA

Component	Test Number	Test Temp (°F)	Material	Charpy V-Notch (Ft-LB)	Lateral Expansion (Mils)
RV Closure Heat Studs	Heat 80751 Bar 66	10	A540-B24	50-51-52	31-30-32
Heat Studs	Bar 66-1	10	A540-B24	55-54-56	35-30-35
Heat Studs	Bar 70	10	A540-B24	57-56-57	36-33-35
Heat Studs	Bar 70-1	10	A540-B24	52-51-50	28-28-29
Heat Studs	Bar 75	10	A540-B24	48-51-48	32-33-28
Heat Studs	Bar 75-1	10	A540-B24	51-52-53	29-32-32
Heat Studs	Bar 81	10	A540-B24	50-47-51	28-26-27
Heat Studs	Bar 81-1	10	A540-B24	54-55-54	32-31-30
Steam Generator Closure Studs *	See Note	50°F Max	SA193 GrB7	45	25
Steam Generator Closure Studs *	See Note	50°F Max	SA194 GrB7	45	25
Pressurizer Manway Bolts	To 4560	40	SA193 GrB7	89-86.5-86.5	59-60-59
Reactor coolant heat pump seal housing bolts	8082109	10	SA193 GrB23	72-74.5-72.5	50-51-49

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* Specific test data available in Steam Generator Final Data Package for each Steam Generator. Data shown is per Design Specification 411A78, Rev.3

TABLE 5.2-22 (Continued)

BOLTING MATERIAL FRACTURE TOUGHNESS DATA

Component	Test Number	Test Temp (°F)	Material	Charpy V-Notch (ft-lb)	Lateral Expansion (Mils)
Reactor Coolant Pump Main Flange Bolts	heat 123123 lot 1	35	SA540 1324c14	73,74,75 79,81,81 68,74,71 83,85,81	45,47,45 46,48,49 40,45,41 51,44,43
Reactor Coolant Pump Main Flange Bolts	heat 123123 lot 3	35		77,76,76 75,70,70 58,58,60	51,47,49 44,41,41 38,37,37
Reactor Coolant Pump Main Flange Bolts	heat 123123 lot 7	35		62,63,62 70,68,66	39,37,37 42,41,39
Reactor Coolant Pump Main Flange Bolts	heat 123123 lot 8	35		63,63,62 64,69,65	33,37,34 38,42,39
Reactor Coolant Pump Main Flange Bolts	heat 123123 lot 9	35		62,62,63 65,64,64	35,37,36 38,38,38

TABLE 5.2-23

FLYWHEEL DISC TENSILE DATA

Motor Unit	Melt Number	Yield Strength (psi)	Tensile Strength (psi)
01	B9109	70,000	98,500
		68,700	84,000
	D1985	68,700	90,900
		72,700	92,400
02	D1985	68,700	90,900
		72,700	92,400
	D1985	68,700	90,900
		72,700	92,400
03	B9694	66,700	87,900
		66,500	85,400
	D1295	66,600	84,000
		72,900	90,900

RN 01-113

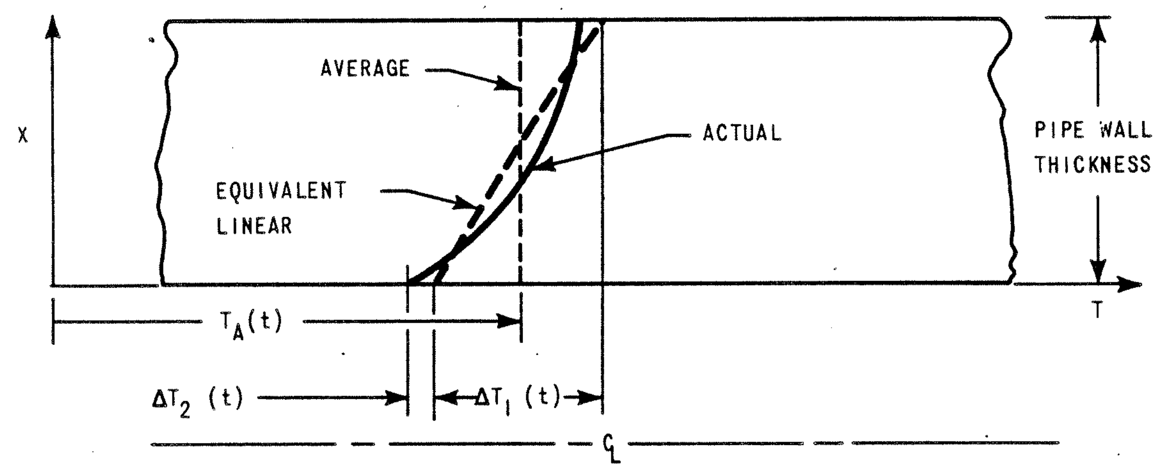
TABLE 5.2-24

FLYWHEEL DISC CHARPY V-NOTCH DATA

Motor Unit	Melt Number	Lateral Expansion (in)	Charpy V-Notch (ft-lb)	Orientation (longitudinal (L) or transverse (T))	Test Temperature (°F)	T_{NDT}* (°F)	RT_{NDT} (°F)
01	B9109	0.090, 0.091, 0.092	120, 110, 118	(T)	70	10	10
		0.099, 0.098, 0.099	155, 148, 163	(L)	70	10	10
	D1985	0.053, 0.058, 0.059	70, 66, 69	(T)	30	10	10
		0.068, 0.061, 0.063	83, 88, 79	(L)	30	10	10
02	D1985	0.053, 0.058, 0.059	70, 66, 69	(T)	30	10	10
		0.068, 0.061, 0.063	83, 88, 79	(L)	30	10	10
	D1985	0.053, 0.058, 0.059	70, 66, 69	(T)	30	10	10
		0.068, 0.061, 0.063	83, 88, 79	(L)	30	10	10
03	D9694	0.078, 0.083, 0.082	100, 106, 98	(T)	70	10	10
		0.099, 0.099, 0.098	198, 200, 199	(L)	70	10	10
	D1295	0.058, 0.057, 0.064	62, 68, 70	(T)	70	10	10
		0.096, 0.095, 0.098	148, 150, 158	(L)	70	10	10

RN 01-113

* Based on two drop weight tests exhibiting no break performance @ 20°F.

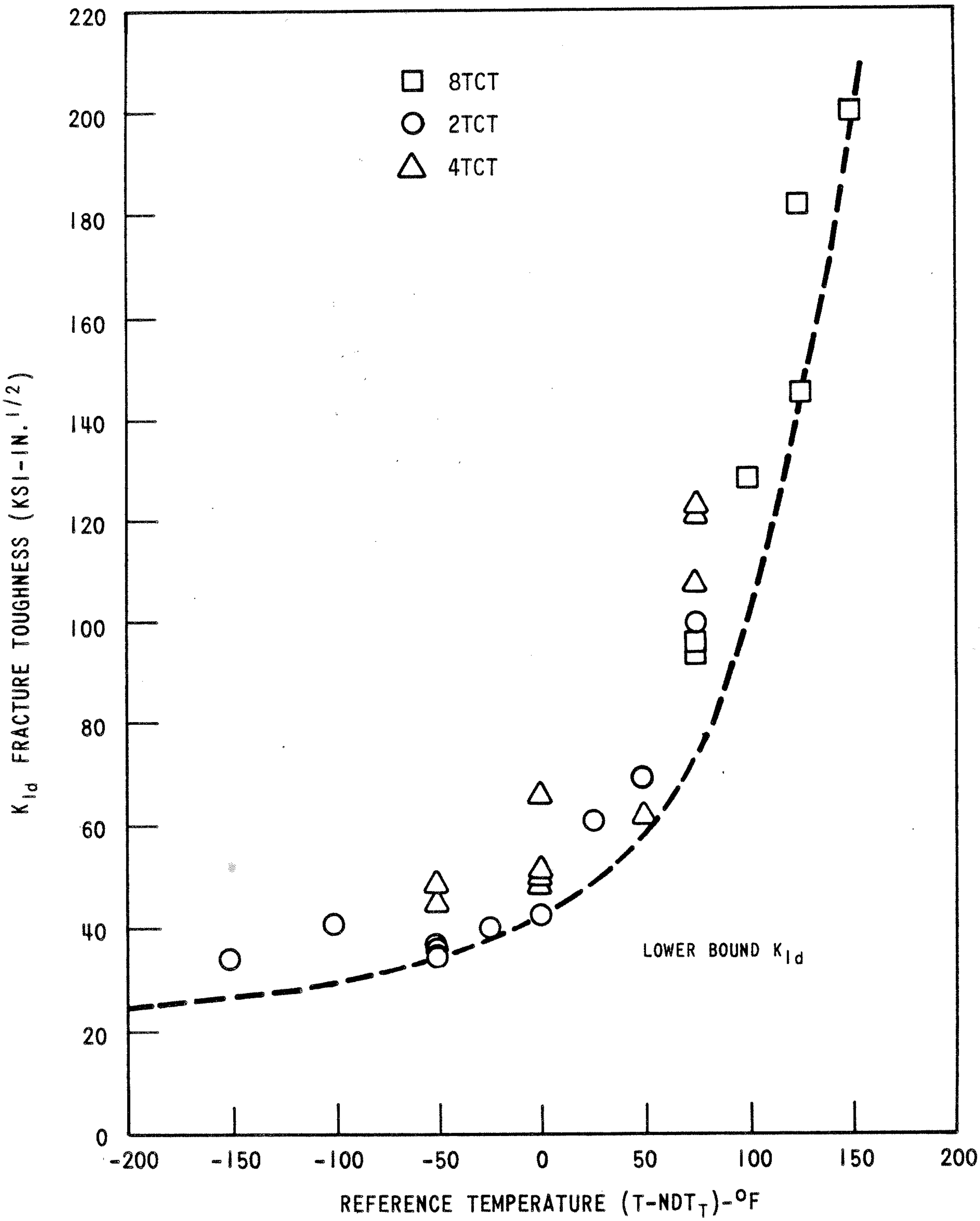


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

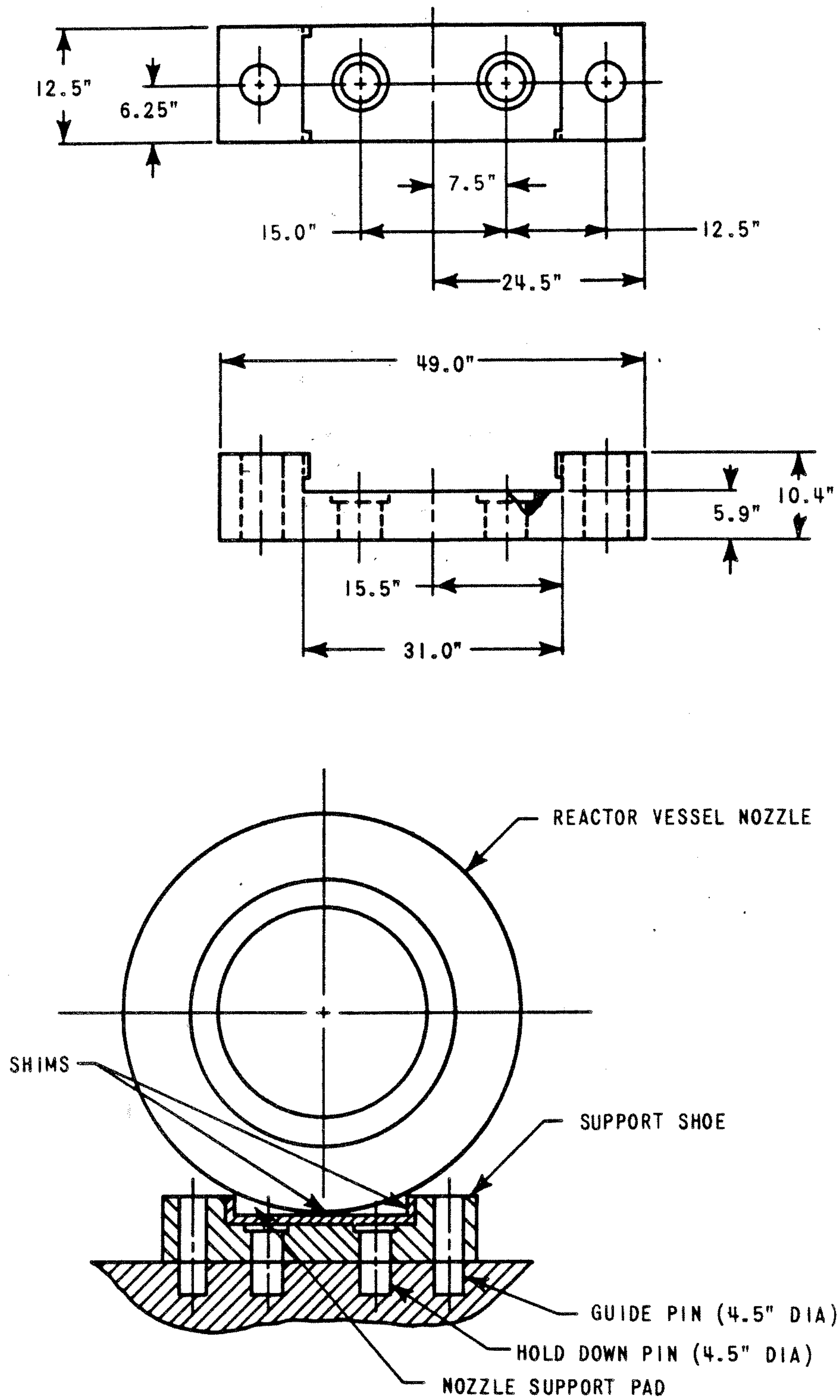
Through - Wall Thermal Gradients

Figure 5.2-2



Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
K_{Id} Lower Bound Fracture Toughness A533V (Reference WCAP - 7623) Grade B Class 1 Figure 5.2-3

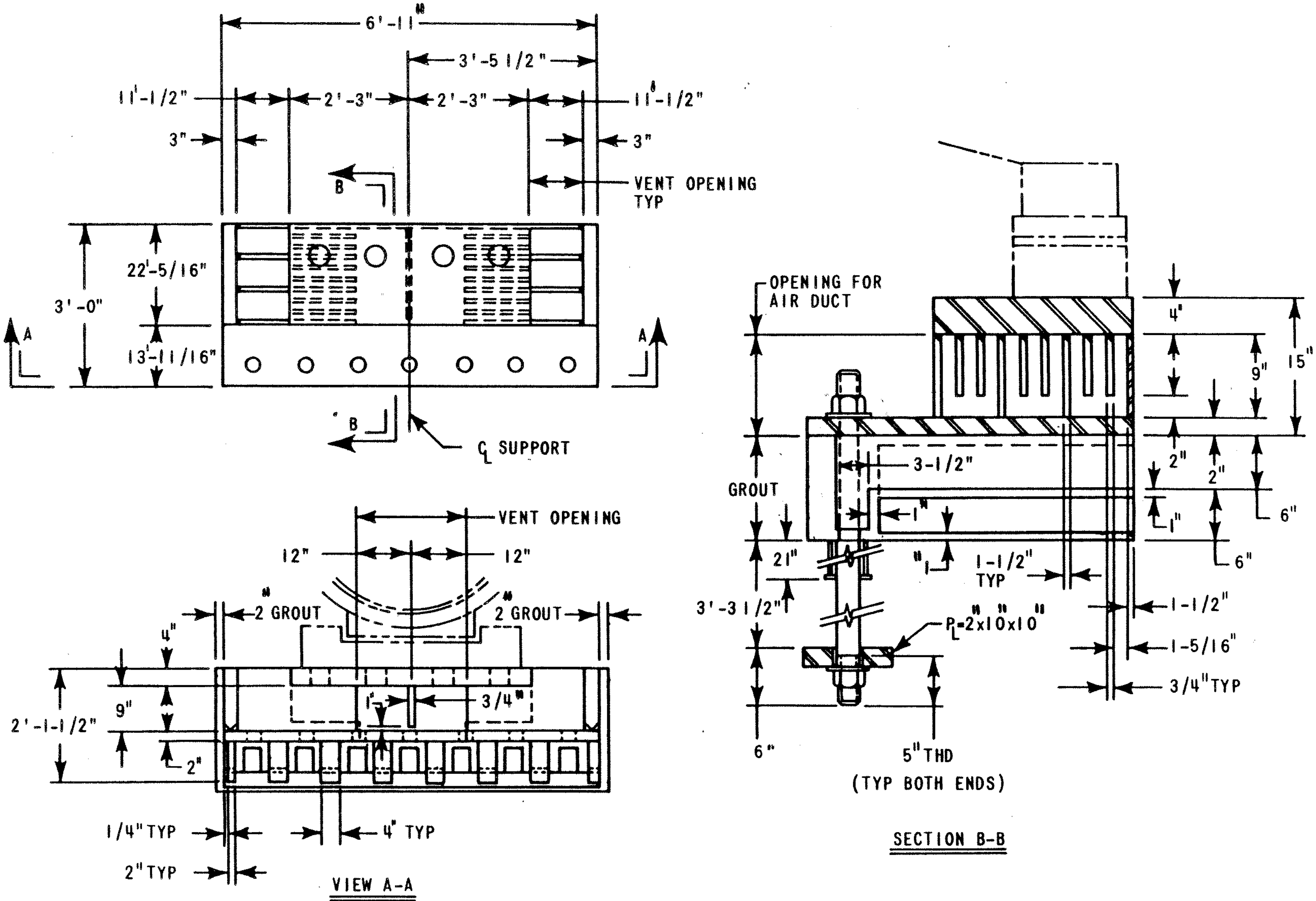


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Vessel Support
Shoe

Figure 5.2-4



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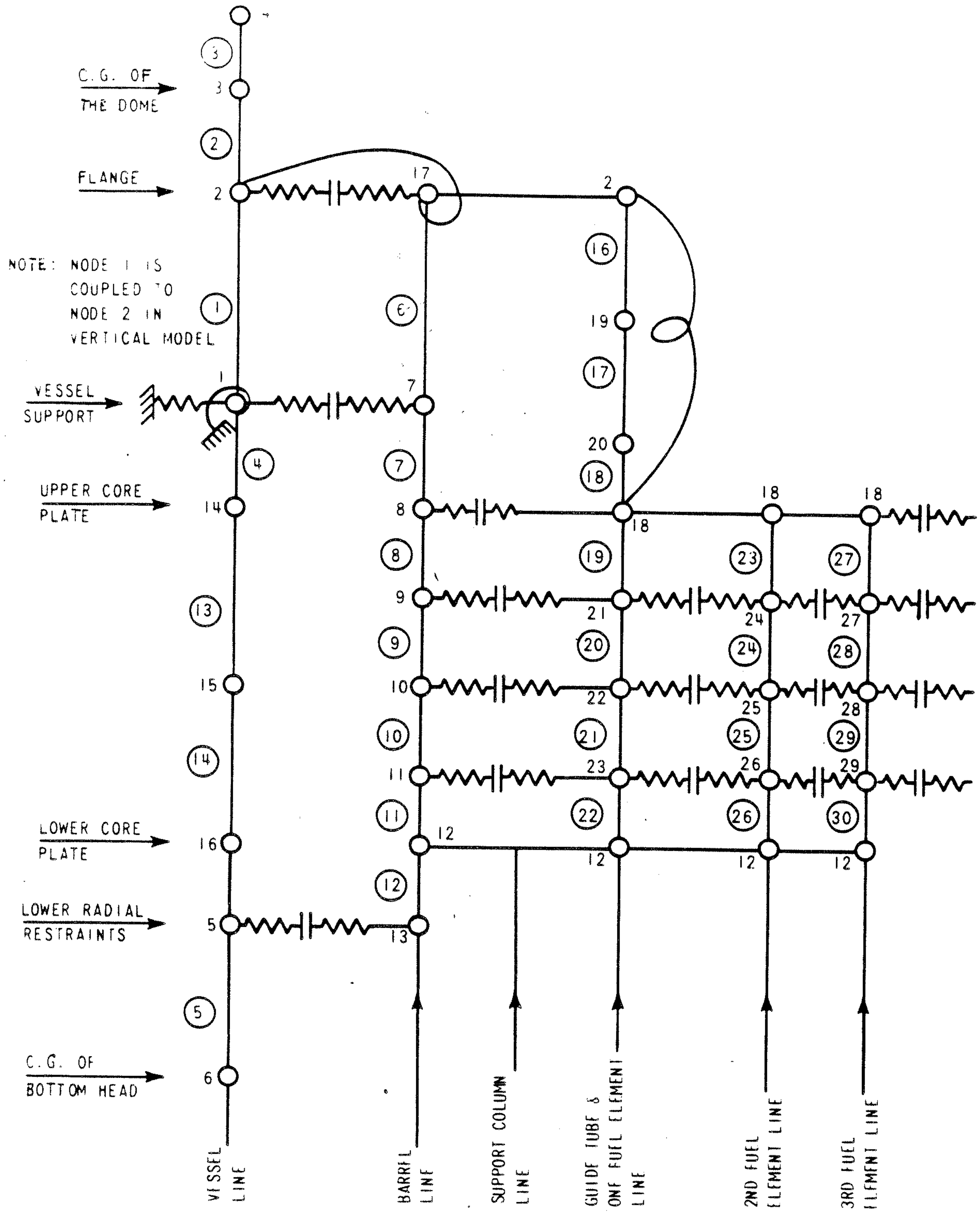
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Vessel Support Box

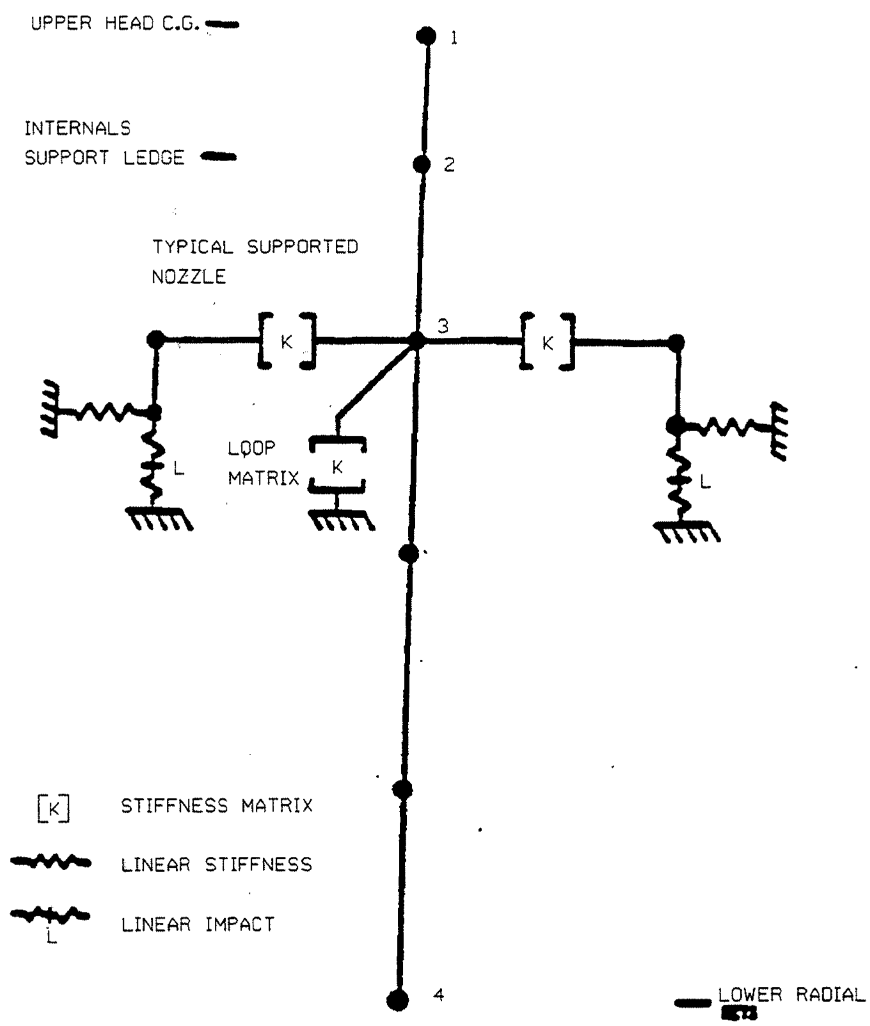
Figure 5.2-5



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Mathematical Model for Horizontal Response

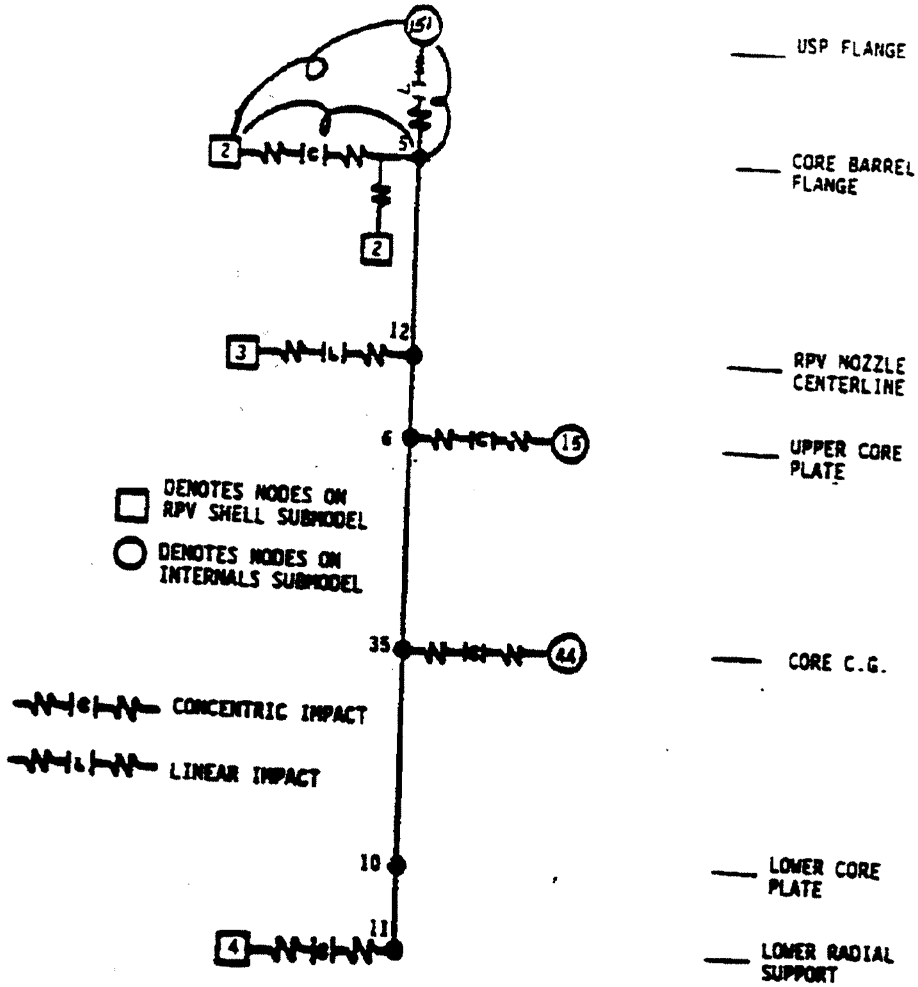
Figure 5.2-6



Reactor Vessel and Typical Supported Nozzle

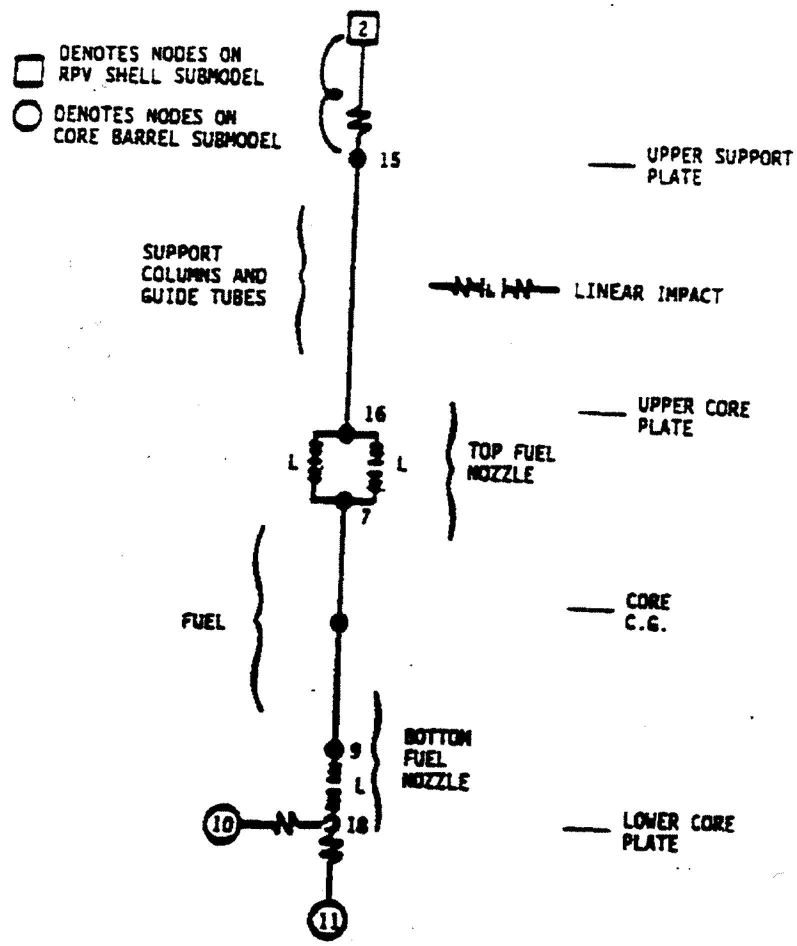
SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Reactor Vessel Shell Finite Element 1st Submodel Figure 5.2-6 (a)

AMENDMENT 96-02
JULY 1996



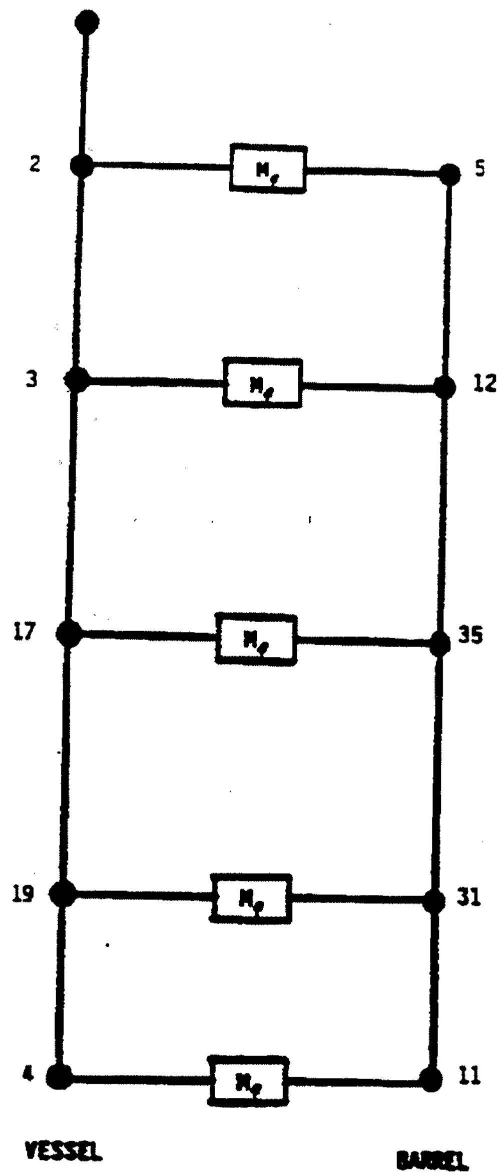
Core Barrel Submodel

SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Reactor Core Barrel Finite Element 2nd Submodel Figure 5.2-6 (b)



Internals Submodel

SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Core Support and Fuel Finite Element 3rd Submodel Figure 5.2-6 (c)



Hydrodynamic Masses in Vessel/Barrel Downcomer Annulus

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

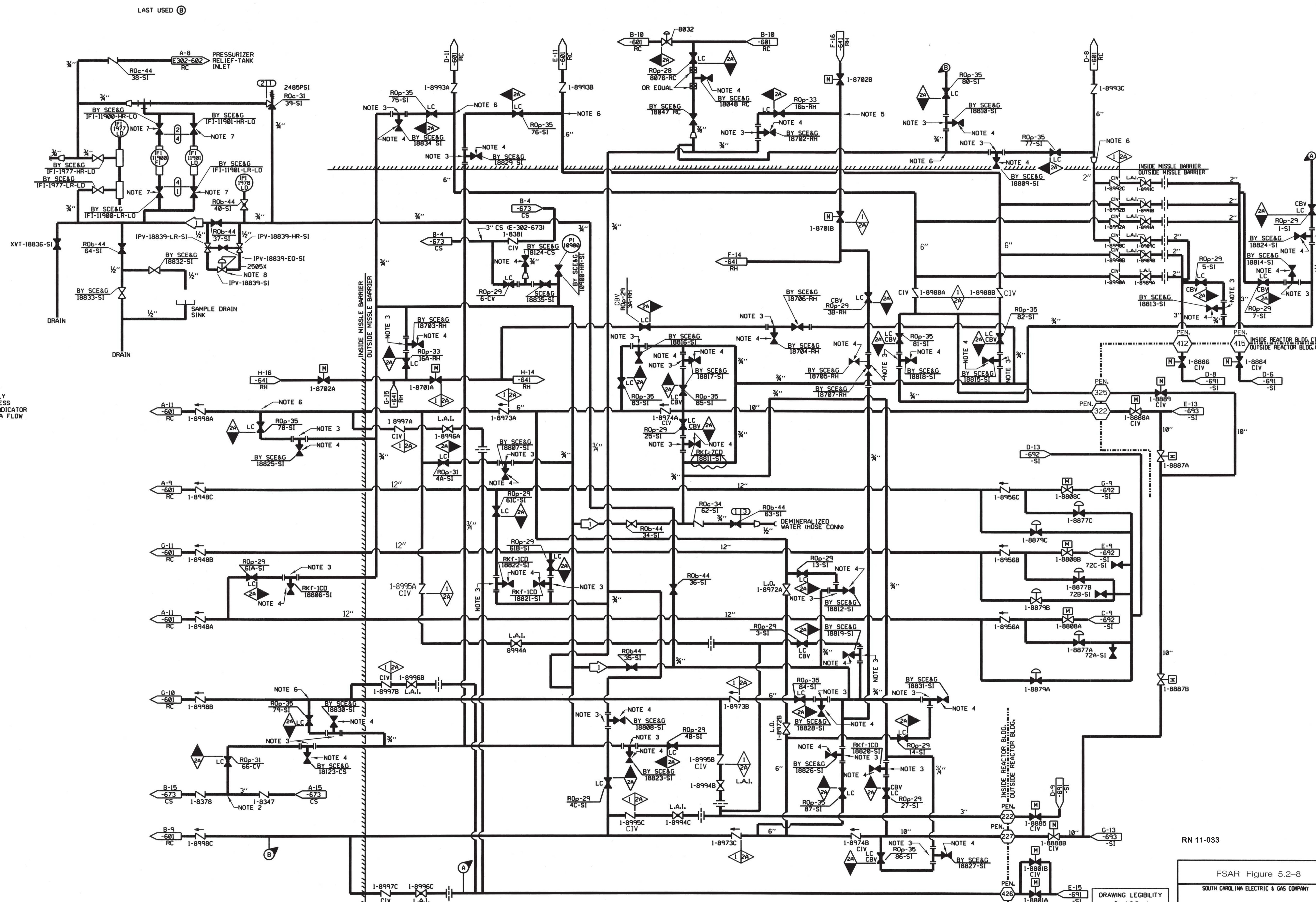
Vessel - Core Barrel
Finite Element Interface
Model

Figure 5.2-7

SYSTEM DATA					
⇨	LB/HR	PSIG	°F	BY	REMARKS
1		1000	120	ECD	
1		240	120	DVR	FLOW THRU FI11900-11901

NOTES:

1. ALL PIPING TO BE NON-NUCLEAR SAFETY CLASS, LINE SPEC 2501X
2. 1.050 ODX 375 ID FLOW RESTRICTOR IS REQUIRED TO ALLOW TRANSITION FROM SAFETY CLASS 1 TO SAFETY CLASS 2A
3. SPOOL PIECE REQUIRED SPOOL PIECE REMAINS IN EXCEPT WHEN LINE IS TO DRAINED UP STREAM VALVES ARE LOCKED CLOSED DURING NORMAL OPERATION
4. BLEED VALVE FOR SYSTEM FILL WITH DEMINERALIZED WATER PRIOR TO TESTING NUPRO P SERIES OR EQUAL
5. 2.380 OD X 375" ID FLOW RESTRICTOR IS REQUIRED TO ALLOW TRANSITION FROM SAFETY CLASS 1 TO SAFETY CLASS 2A
6. ¾ SW CONN EQUV TO 6000# A.S.A. HALF CPLG DRILLED FOR 375 ID FLOW RESTRICTOR IS REQUIRED TO ALLOW TRANSITION FROM SAFETY CLASS 1 TO SAFETY CLASS 2A
7. UPSTREAM & DOWNSTREAM VALVES TO IFI-11900 & 11901 ARE TO BE CLOSED AT THE START OF TESTING PROCEDURE FOR USING THESE INDICATORS IS
   1 PROVIDE A BLEED THRU IFI 1977 FLOW INDICATOR OR DRAIN VALVE 64 SI IN ORDER TO REDUCE PRESSURE TO LESS THAN 240 PSIG ON PI-1978
   2 OPEN DOWNSTREAM VALVE (S) (IFI-11900-HR-LD OR IFI-11901-HR-LD)
   3 OPEN UPSTREAM VALVE (5) (IFI-11900 LRR LD OR IFI 11901 LR-LD)
   4 CLOSE BLEED PATH AFTER FLOW IS READ. UPSTREAM VALVE (S) IS TO BE CLOSED FIRST & DOWNSTREAM VALVE (S) CLOSED LAST IFI-11900 LOW RANGE FLOW INDICATOR IS ONLY TO BE USED WHEN IFI-1977 READS A FLOW LESS THAN 14GPM IFI-11901 LO LOW RANGE FLOW INDICATOR IS ONLY TO BE USED WHEN IFI-11900 READS A FLOW LESS THAN 0 6 GPM
8. LINE SPEC FOR IPV-18839-SI TUBING IS 2505X.



RN 11-033

HYDROTEST TEMP. 60° F

P#	NORMAL PSIG	NORMAL °F	UPSET PSIG	UPSET °F	UPSET DURATION	HYDRO	BY	CHKD	REMARKS
4	240	200	240	200			DVR		
3	100	100	110	150	<1%	165	DJK		
2	3	120	350	435	<1%	525	DJK		
1	1000	335	2485	650	<1%	3730	PLL		

DESIGN DATA

DRAWING LEGIBILITY
CLASS 1
SCE&G CAD ENHANCED

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
16	4/8/98	JMR	REVISED PER CGSS-98-0138	MGR	ROB
15	11/13/97	AVN	REVISED PER CGSS-97-0579	LEK	MGR
20	11/04/2011	JTS	REVISED PER ECR-71545	MGR	GFK
19	10/17/2011	JTS	REVISED PER ECR-71545	MGR	GFK
18	05-13-08	TGB	REVISED PER ECR-50650	MGR	RAT
17	7/19/05	TGB	CADD ENHANCED PER ECR-50239	RHM	MGR

ESSENTIAL

FSAR Figure 5.2–8

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

PIPING SYSTEM FLOW DIAGRAM

ECCS CHECK VALVE TESTING

SCE&G A SCANA COMPANY

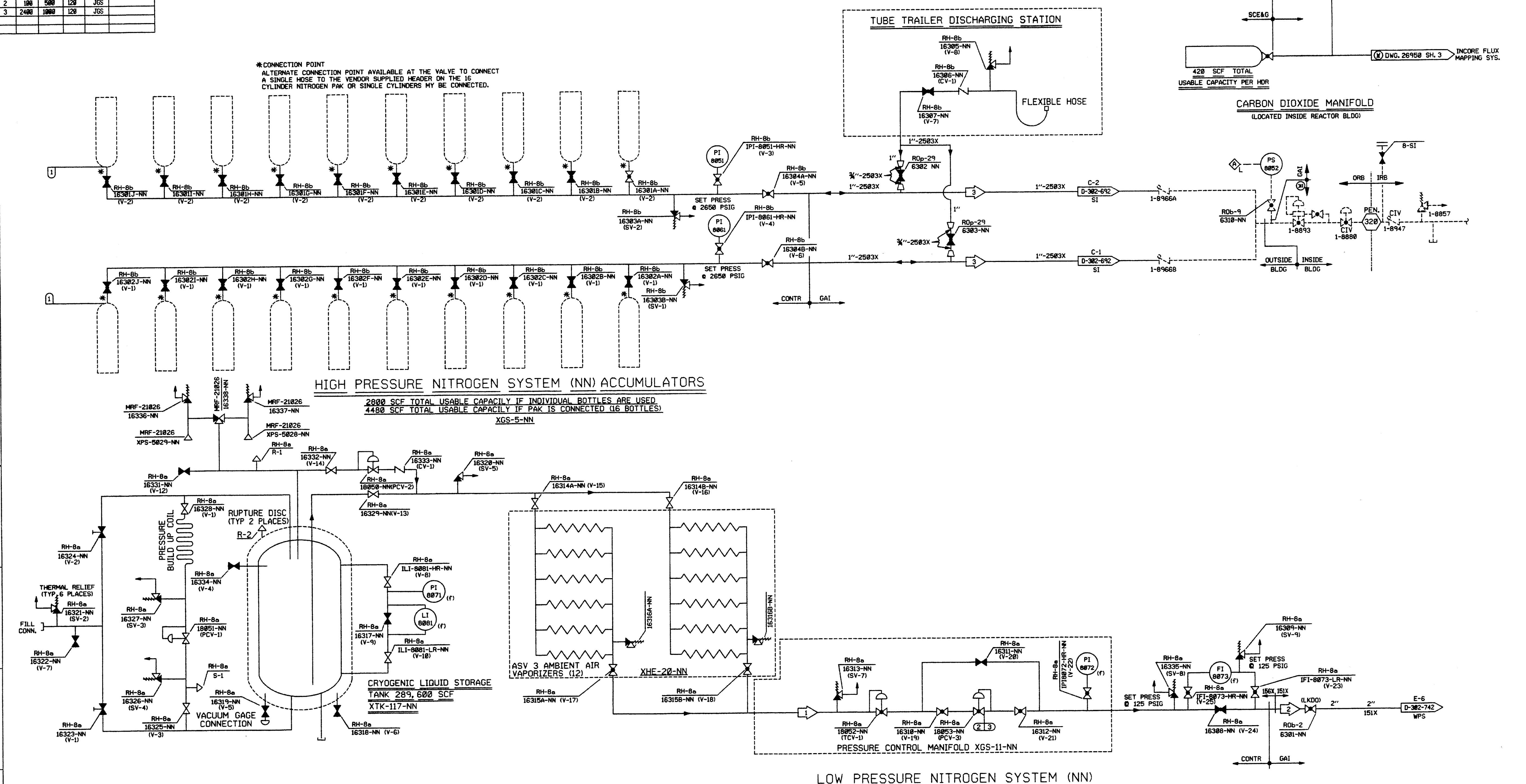
DESIGN ENGINEERING

V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. SRM	2. MGR	3. MRH

SCALE NO SCALE | DRAWING NUMBER D-302-812 | SHT. NUMBER | REV 20

NO	PSIG	SCFM	°F	BY	REMARKS
SYSTEM DATA					
1	230	500	120	JGS	
2	100	500	120	JGS	
3	2400	1000	120	JGS	



NOTES
1. ALL VALVE NUMBERS OTHER THAN THE 6300 NUMBERS ARE BY AIR PRODUCTS AND CHEMICALS INC.
2. SYSTEM SHALL RECEIVE AN INITIAL SERVICE LEAK TEST PER ANSI B3110 AT DESIGN PRESSURES INDICATED.

* SEE NOTE 2

#	PSIG	°F	PSIG	°F	DURA.	PNEU *	BY	CHKD	REMARKS
	NORMAL		UPSET						
3	110	120	125	120	<1%	110	JGS		
2	230	120	245	120	<1%	230	JGS		
1	2400	120	2650	120	<1%	2450	PLL		

DESIGN DATA

1/4" = 1'-0" 0' 5' 10' 15' 20'

ESSENTIAL

RN 09-006
May 2009

DRAWING LEGIBILITY
CLASS 1
SCE&G CAD ENHANCED

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
22	2/4/09	JTS	CADD ENHANCED PER ECR-50239	MGR	DDJ
21	3/17/04	DDJ	REVISED PER ECR-50492	MGR	RAT
20	11/12/97	AVN	REVISED PER CGSS-97-0579	LEK	MGR
19	01/15/96	AVN	REVISED PER MRF-22640	MGR	JEW
18	12-06-93	SRM	REVISED PER CGSS-93-2789	MGR	RBS
17	02-16-93	SRM	REVISED PER CGSS-29584	RHM	LEK

FSAR Figure 5.2–8A

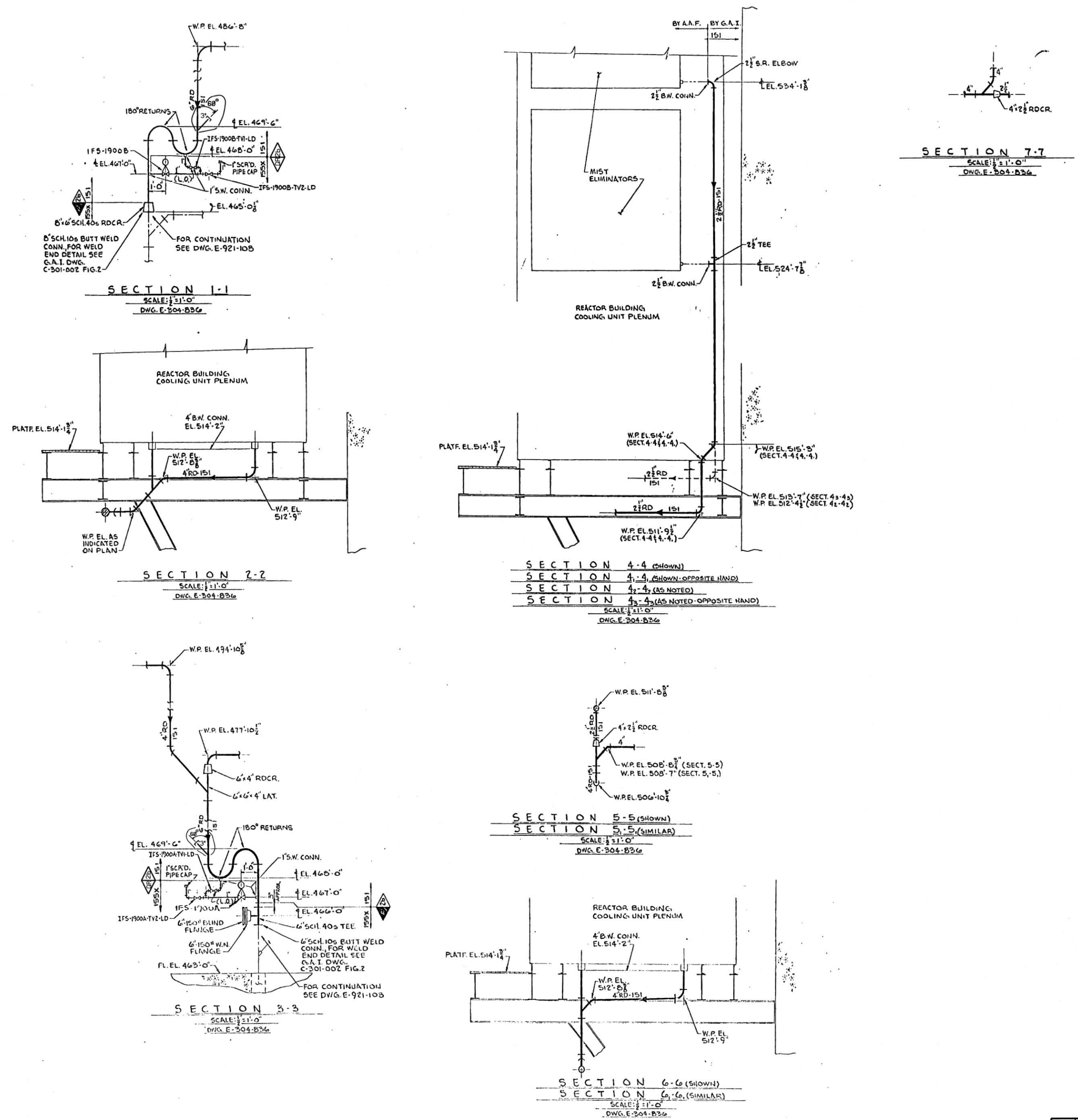
SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION

PIPING SYSTEM FLOW DIAGRAM
NSSS CARBON DIOXIDE
AND NITROGEN SUPPLY

SCE&G A SCANA COMPANY
DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. SRM	2. RHM	3. LEK

SCALE NO SCALE
DRAWING NUMBER D-302-717
SHT. NUMBER
REV 22



THIS IS A NUCLEAR SAFETY RELATED DOCUMENT. NO DEVIATION SHALL BE INITIATED OR PERFORMED WITHOUT PRIOR DOCUMENTATION AND WRITTEN APPROVAL

RN 11-019

FSAR Figure 5.2–9

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

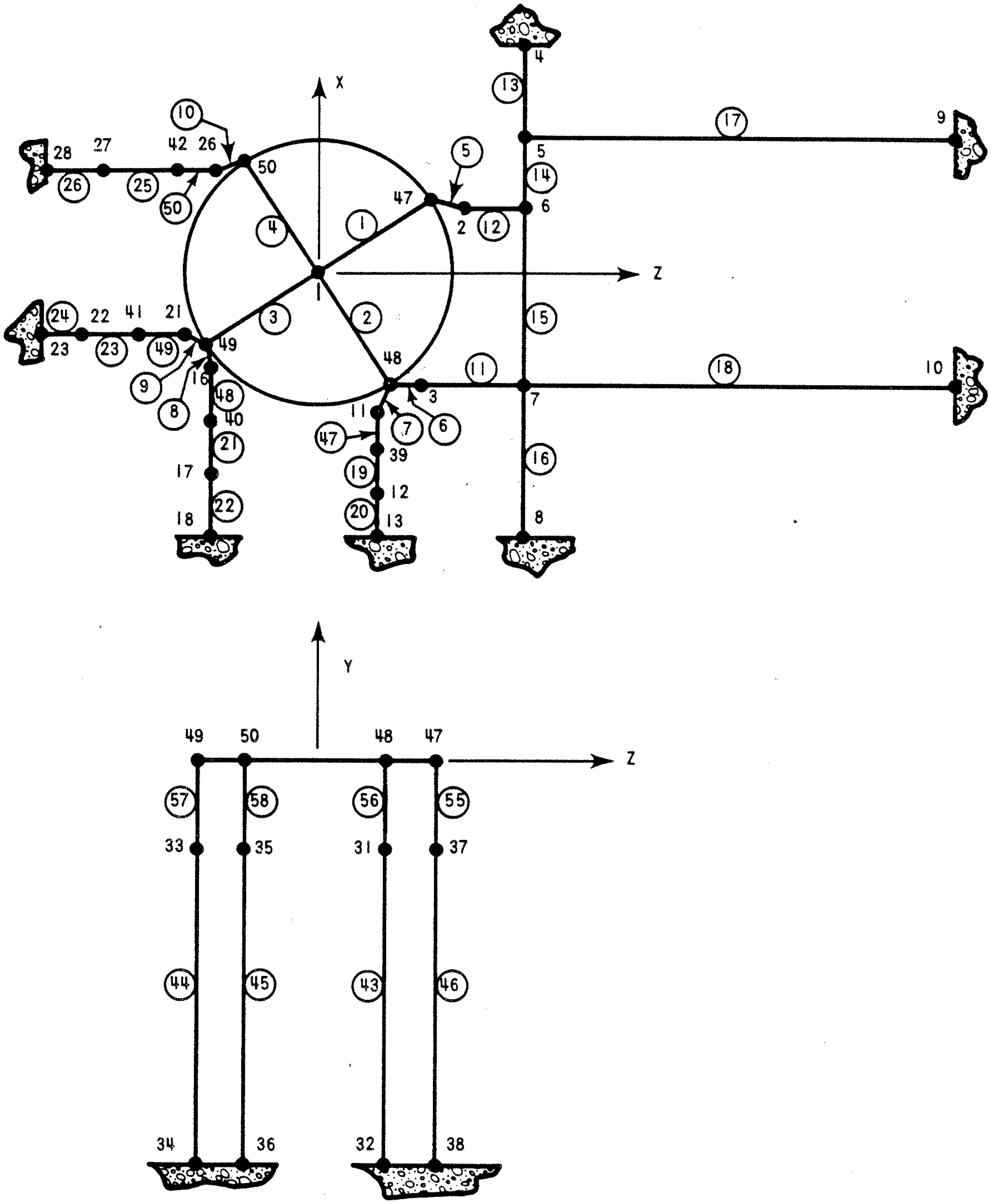
PIPING SYSTEM
REACTOR BUILDING
COOLING UNIT DRAIN

DESIGN ENGINEERING
SCE&G A SCANA COMPANY — V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. JMH	2. JHR	3. RLJ

SCALE: NO SCALE — DRAWING NUMBER: E-304-837 — SHT. NUMBER — REV: 9

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
9	7/11/11	RHM	REVISED PER ECR-50756	MGR	RS
8	9/26/01	DDJ	REVISED PER ECR-70028	MGR	DDJ
7	09/30/93	JMH	REVISED PER GCSS-93-2031	JHR	RLJ

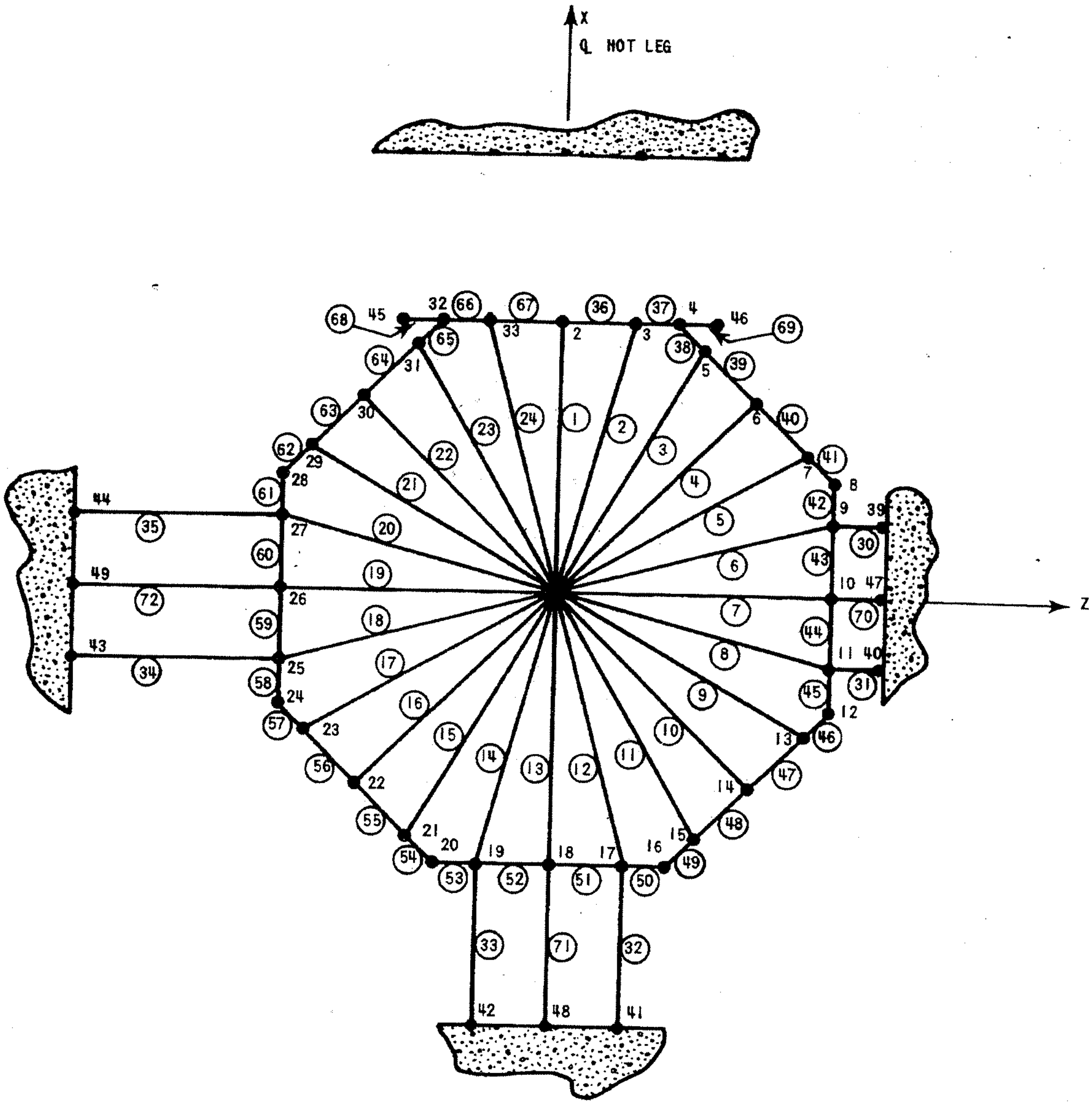


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Steam Generator Lower Support Model

Figure 5.2-10



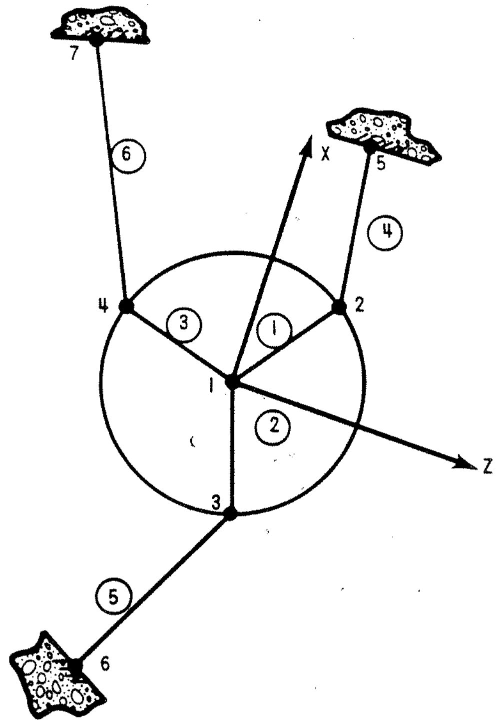
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Steam Generator Upper

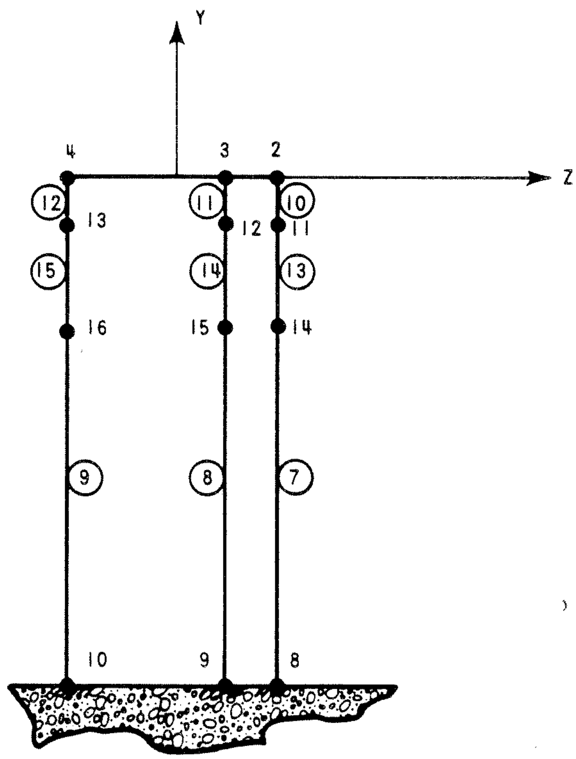
Support Model

Figure 5.2-11

PLAN VIEW



ELEVATION VIEW



Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

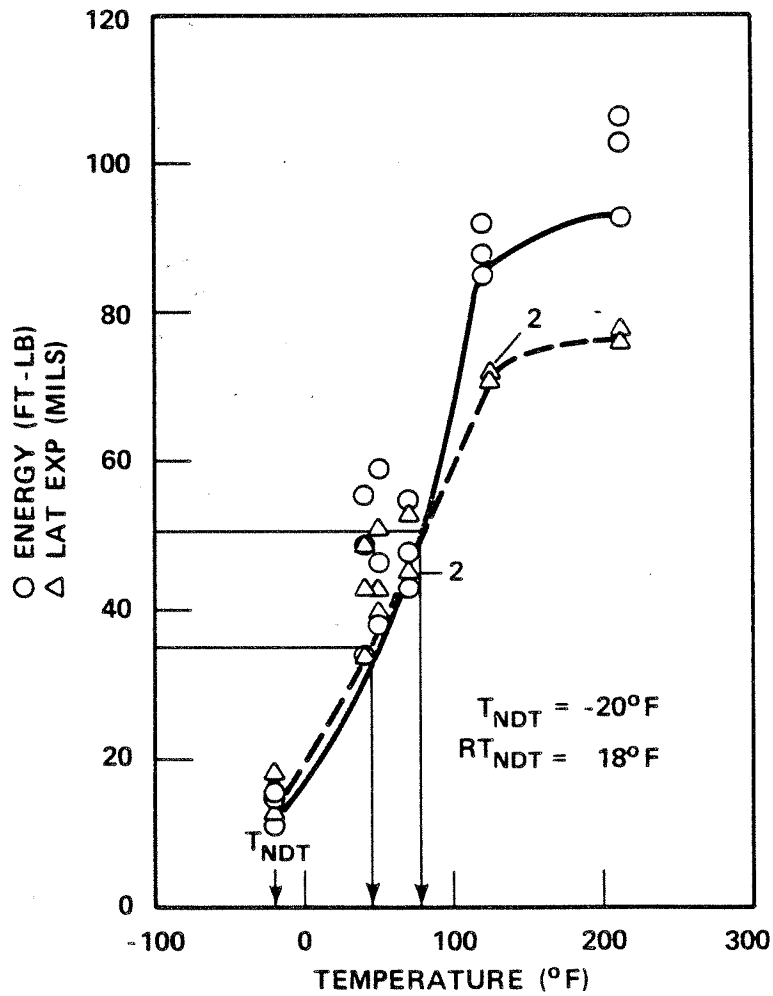
Reactor Coolant Pump Support Model

Figure 5.2-12

FIGURES 5.2-13 and 5.2-14
Deleted per RN 00-001

02-01

Figure 5.2-15 Deleted By Amendment 1, August 1985

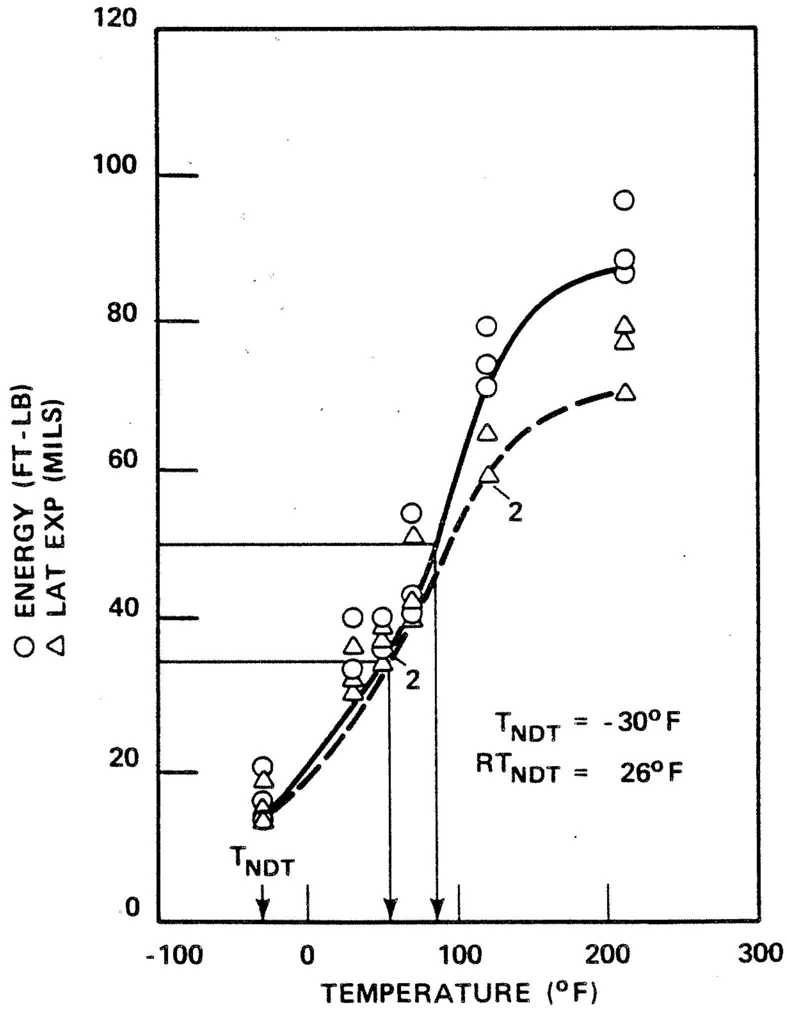


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Charpy V - Notch Curve for Nozzle Shell Plate C9955 - 2

Figure 5.2-16

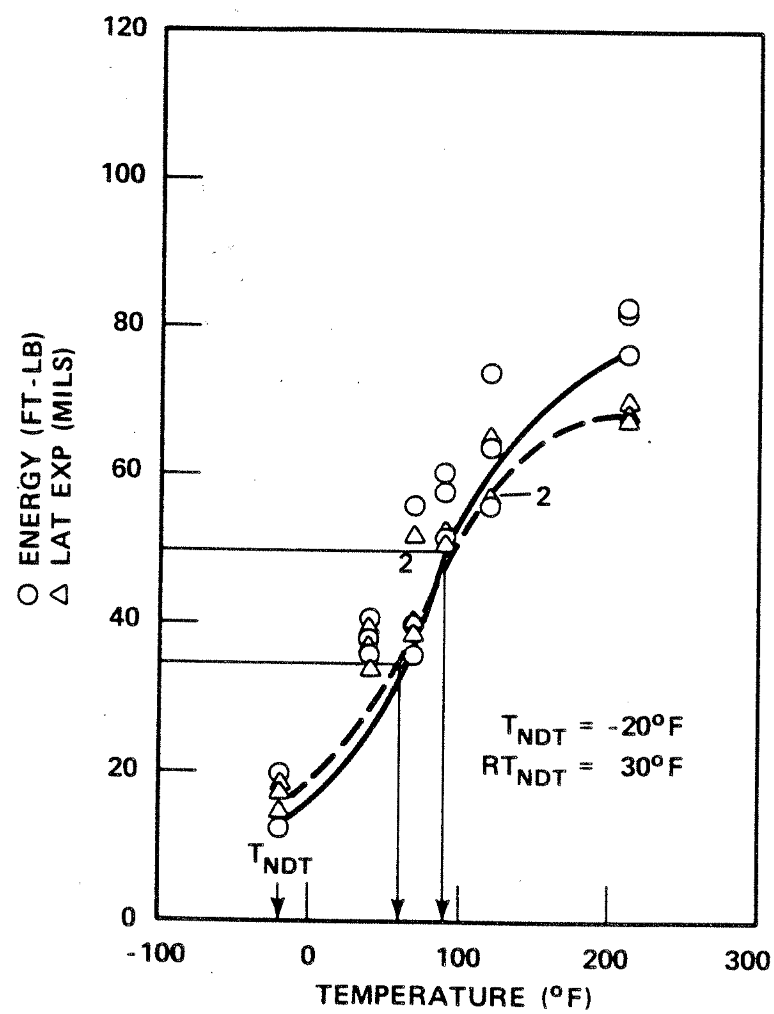


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Charpy V - Notch Curve for Nozzle Shell Plate C0123 - 2

Figure 5.2-17

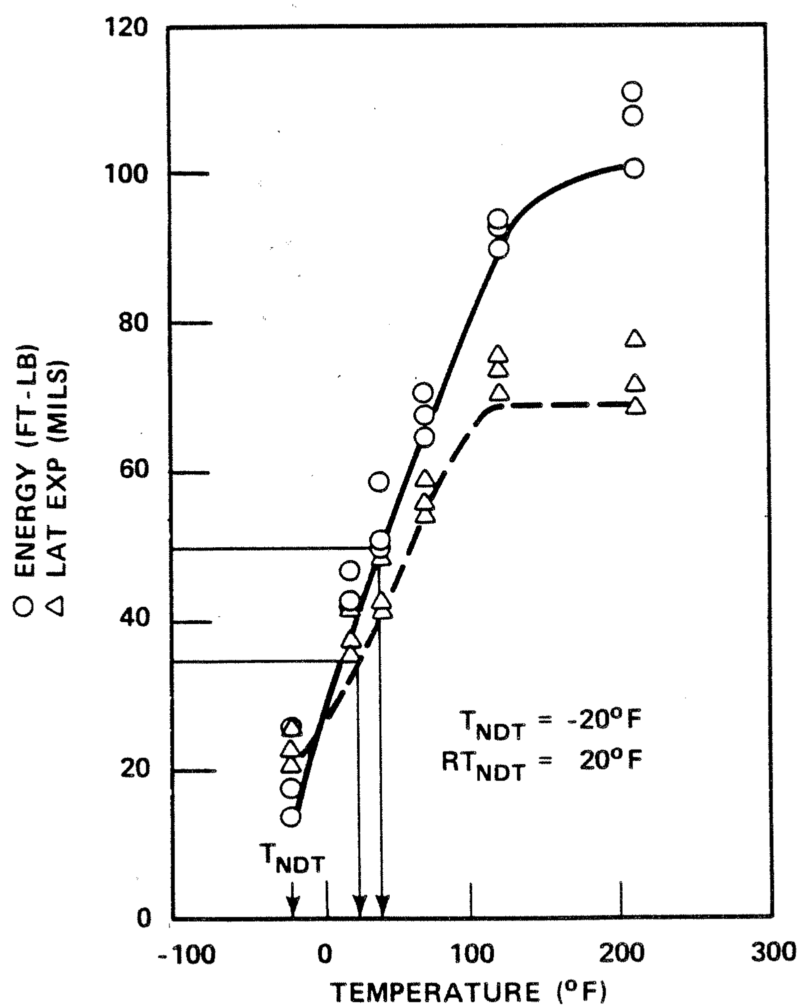


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Charpy V - Notch Curve for Intermediate Shell Plate A9154 - 1

Figure 5.2-18

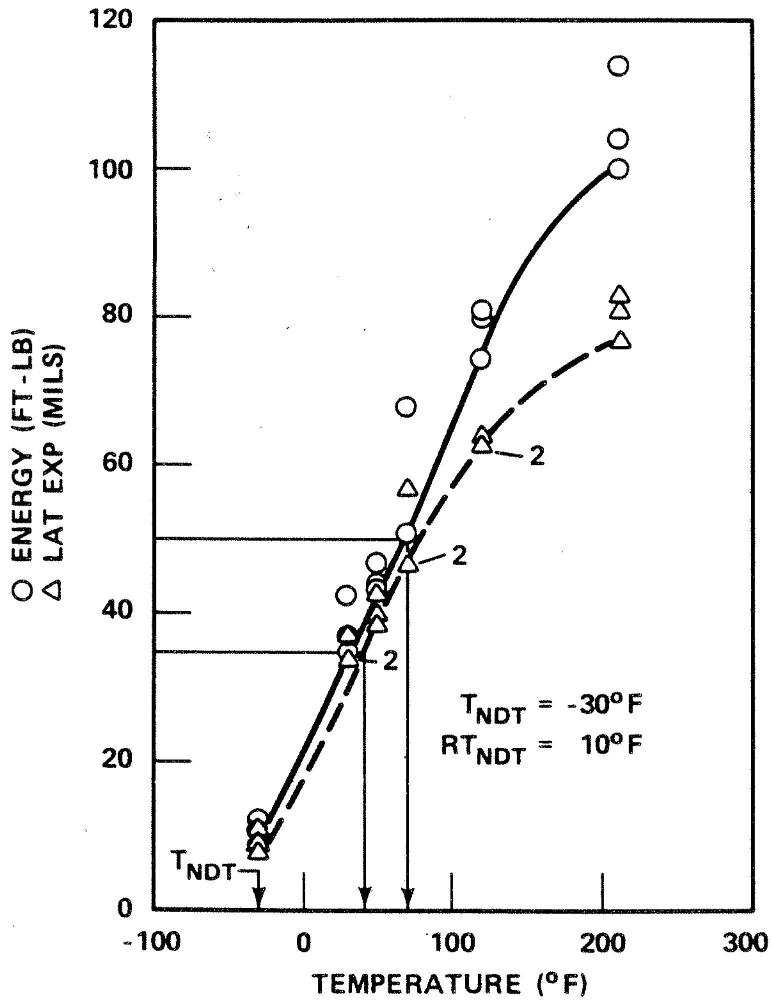


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Charpy V - Notch Curve for Intermediate Shell Plate A9153 - 2

Figure 5.2-19

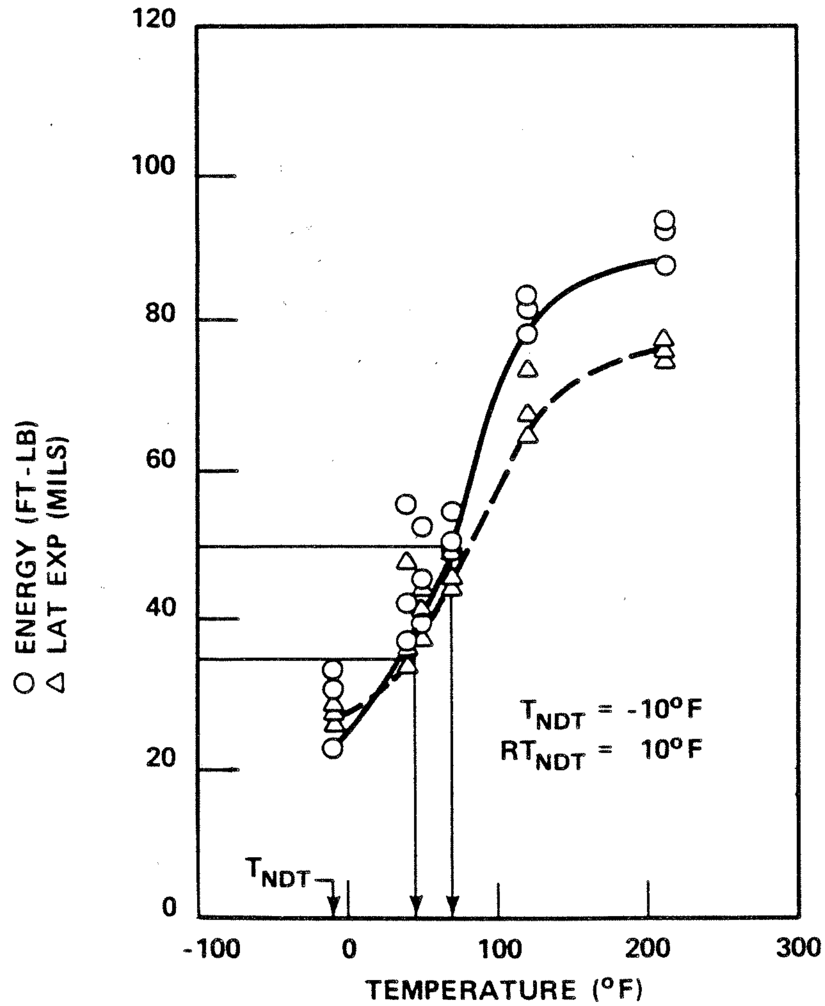


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Charpy V - Notch Curve for Lower Shell Plate C9923 - 1

Figure 5.2-20

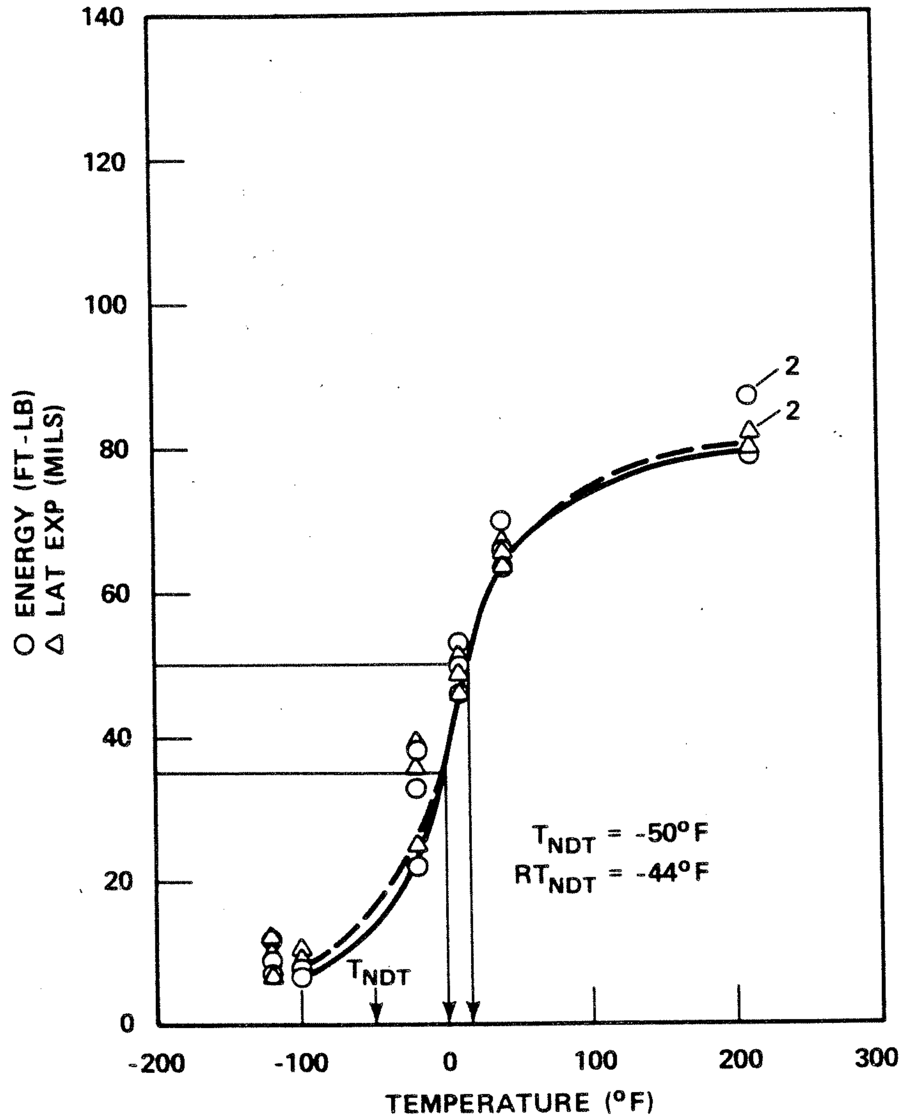


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Charpy V - Notch Curve for Lower Shell Plate C9923-2

Figure 5.2-21

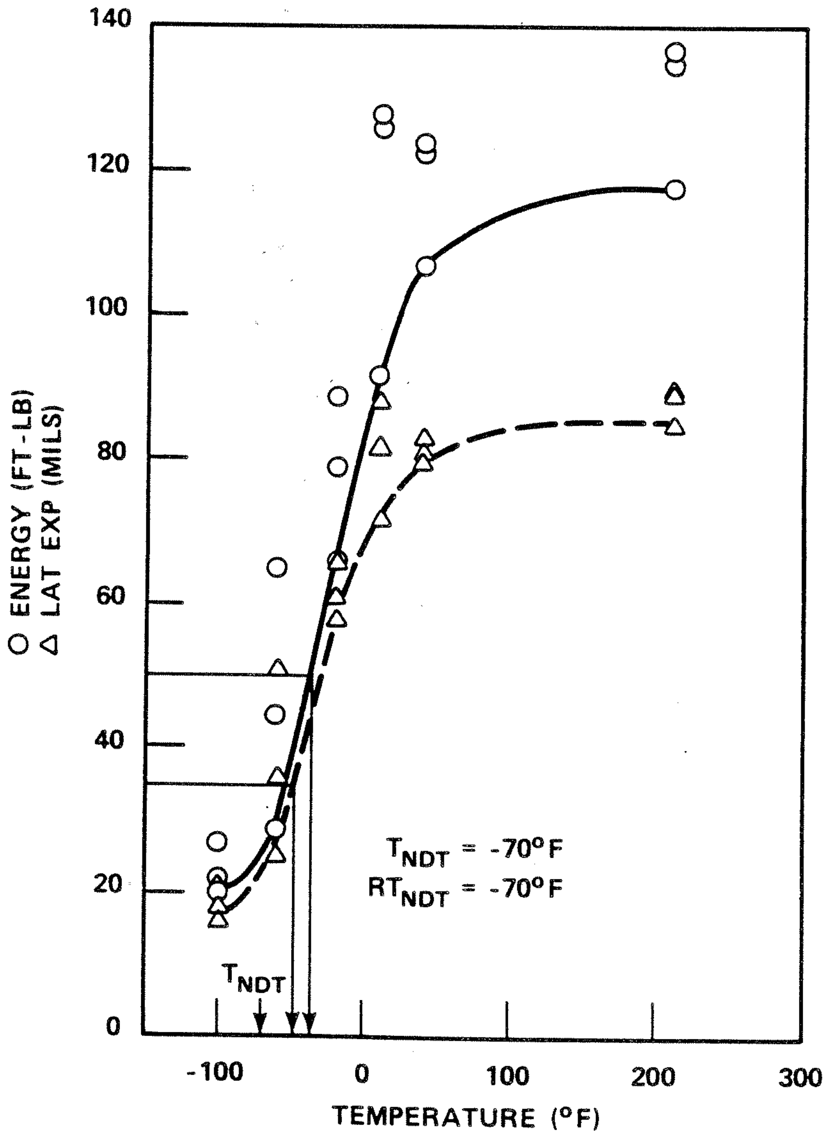


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Charpy V - Notch Curve for Core Beltline Region Weld

Figure 5.2-22



Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Charpy V - Notch Curve for Core Beltline Region Haz Metal

Figure 5.2-23

5.3 THERMAL HYDRAULIC SYSTEM DESIGN

5.3.1 ANALYTICAL METHODS AND DATA

The thermal and hydraulic design bases of the Reactor Coolant System (RCS) is summarized in Table 5.1-1 and described further in Sections 4.3 and 4.4 in terms of core heat generation rates, departure from nucleate boiling ratio (DNBR), analytical models, peaking factors, and other relevant aspects of the reactor.

5.3.2 OPERATING RESTRICTIONS ON PUMPS

The minimum net positive suction head (NPSH) and minimum seal injection flowrate must be established before operating the reactor coolant pumps. With the minimum 6 gpm seal injection flowrate established, the operator will have to verify that the system pressure satisfies NPSH requirements and that the number 1 seal bypass flow satisfies seal flow requirements.

5.3.3 POWER-FLOW OPERATING MAP (BWR)

Not applicable to pressurized water reactors.

5.3.4 TEMPERATURE-POWER OPERATING MAP

The NSSS is designed to operate with a full power average coolant temperature ranging from 572°F to 587.4°F. The relationship between RCS temperature and power is shown in Figure 5.3-1 for thermal design conditions with a Tavg of 587.4°F at a core power of 2900 MWt.

The effects of reduced core flow due to inoperative pumps is discussed in Sections 5.5.1, 15.2.5, and 15.3.4.

5.3.5 LOAD FOLLOWING CHARACTERISTICS

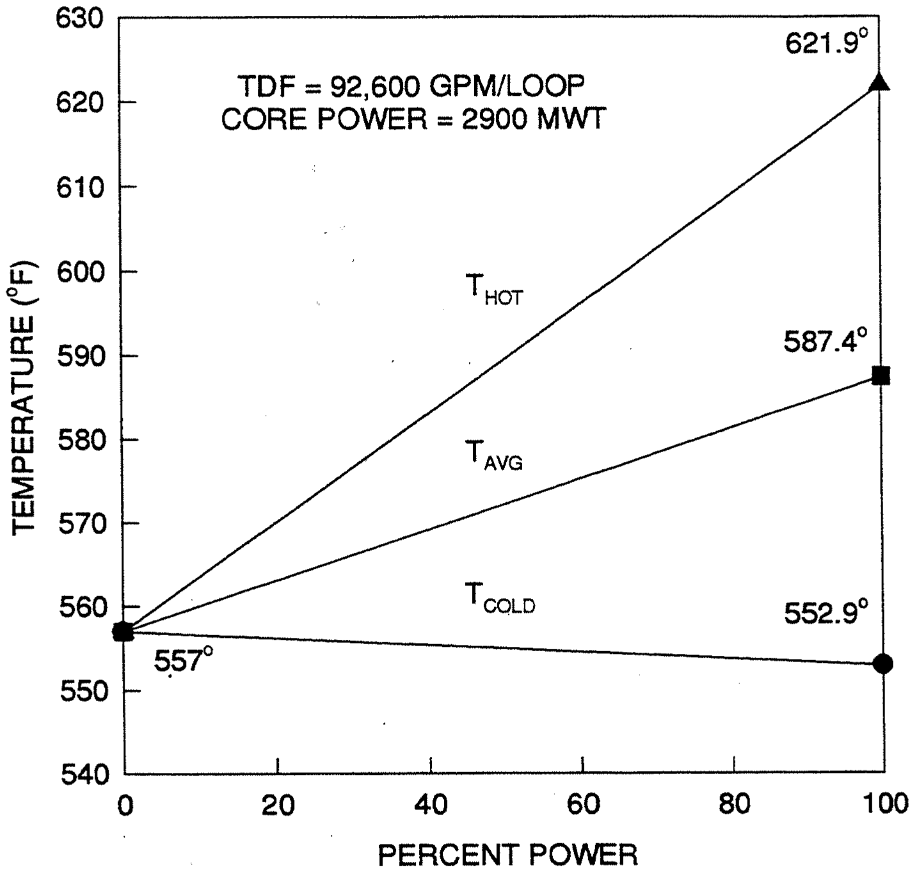
The RCS is designed on the basis of steady-state operation at full power heat load. The reactor coolant pumps utilize single speed motors as described in Section 5.5 and the reactor power is controlled to maintain average coolant temperature at a value which is a linear function of load, as described in Section 7.7. Operation with 1 pump out of service requires adjustment only in reactor trip system setpoints as discussed in Section 7.2.

5.3.6 TRANSIENT EFFECTS

Transient effects on the RCS are evaluated in Chapter 15.

5.3.7 THERMAL AND HYDRAULIC CHARACTERISTICS SUMMARY TABLE

The thermal and hydraulic characteristics are given in Tables 4.3-1, 4.4-1, and 5.1-1. 00-01



SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Reactor Coolant System Temperature - Percent Power Map Figure 5.3-1

5.4 REACTOR VESSEL AND APPURTENANCES

Section 5.4 has been divided into four principal subsections, viz., design bases, description, evaluation and tests and inspections for the reactor vessel and its appurtenances consistent with the requirements of the introductory paragraph of Section 5.4 of the "Standard Format and Content of Safety Analyses Reports for Nuclear Power Plants, Revision 1." The following specific information required by the guide is cross referenced below.

	Guide Reference	FSAR Section(s)
5.4.1	Protection of Closure Studs	5.4.2.2
5.4.2	Special Processes for Fabrication and Inspection	5.2.5, 5.4.2.1, 5.4.4
5.4.3	Features for Improved Reliability	5.4.1, 5.4.2.1
5.4.4	Quality Assurance Surveillance	5.4.2, 5.4.4
5.4.5	Materials and Inspections	5.2.3, 5.2.4, 5.4.4
5.4.6	Reactor Vessel Design Data	Table 5.4-1
5.4.7	Reactor Vessel Schematic (BWR)	Not Applicable

5.4.1 DESIGN BASES

5.4.1.1 Codes and Specifications

The vessel is Safety Class 1. Design and fabrication of the reactor vessel is carried out in strict accordance with ASME Section III, Class 1.

Material specifications are in accordance with the ASME Code requirements and are given in Section 5.2.3.

5.4.1.2 Design Transients

Cyclic loads are introduced by normal power changes, reactor trip, startup and shutdown operations. These design base cycles are selected for fatigue evaluation and constitute a conservative design envelope for the projected plant life. Vessel analysis results in a usage factor that is less than 1.

With regard to the thermal and pressure transients involved in the loss of coolant accident, the reactor vessel is analyzed to confirm that the delivery of cold emergency core cooling water to the vessel following a loss of coolant accident does not cause a loss of integrity of the vessel.

The results of the analysis required by the design specifications prove that the vessel is in compliance with the fatigue and stress limits of Section III of the ASME Boiler and Pressure Vessel Code. The loadings and transients specified for the analysis are based on the most severe conditions expected during service. The heatup and cooldown rates imposed by plant operating limits are 50°F per hour for normal operations and 100°F per hour under abnormal or emergency conditions. These rates are reflected in the vessel design specifications.

A control rod housing failure does not cause propagation of failure to adjacent housings or to any other part of the Reactor Coolant System (RCS) boundary.

Design transients are discussed in detail in Section 5.2.1.5.

5.4.1.3 Protection Against Non-Ductile Failure

Protection against non-ductile failure is discussed in Section 5.2.4.

5.4.1.4 Inspection

The internal surface of the reactor vessel is capable of inspection periodically using visual and/or nondestructive techniques over the accessible areas. During refueling, the vessel cladding is capable of being inspected in certain small areas between the closure flange and the primary coolant inlet nozzles, and, if deemed necessary, the core barrel is capable of being removed, making the entire inside vessel surface accessible.

The closure head is examined visually during each refueling. Optical devices permit a selective inspection of the cladding, control rod drive mechanism nozzles, and the gasket seating surface. The knuckle transition piece, which is the area of highest stress of the closure head, is accessible on the outer surface for visual inspection, dye penetrant or magnetic particle, and ultrasonic testing. The closure studs can be inspected periodically using visual, magnetic particle and/or ultrasonic techniques.

5.4.2 DESCRIPTION

The reactor vessel fabricated by Chicago Bridge and Iron, Inc., is cylindrical, with a welded hemispherical bottom head and a removable, bolted, flanged, and gasketed, hemispherical upper head. The reactor vessel flange and head are sealed by two hollow metallic O-rings. Seal leakage is detected by means of two leakoff communications: one between the inner and outer ring and one outside the outer O-ring. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The reactor vessel closure head contains head adaptors. These head adaptors are tubular members, attached by partial penetration welds to the underside of the closure head. The upper end of these adaptors contain acme threads for the assembly of control rod drive mechanisms or instrumentation adaptors. During RF-20 and RF-21, repairs were performed to mitigate Primary Water Stress Corrosion Cracking (PWSCC) in the CRDM J-groove welds in the Reactor Vessel Upper Head enclosure (per ECR-50846 and ECR-50846D). The repair

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consisted of an Alloy 690 (weld material 52M) weld overlay to cover the Alloy 600 weld material at the CRDM nozzle J-groove welds at nozzle penetration #9, 15, 19, 22, 31, 37, 43, 51 and 52. The weld repair method prescribed by Westinghouse WCAP-15987-P, Revision 2-P-A, “Technical Basis for the Embedded Process for Repair of Reactor Vessel Head Penetrations,” was adopted by V. C. Summer during RF-20 and gained further approval by the NRC on April 30, 2014, under Relief Request RR-4-05, “Alternative Weld Repair for Reactor Vessel Upper Head Penetrations.” The repair covered the entire surface of Alloy 600 weld material exposed to primary water as well as the outer diameter of the penetration tubing below the weld. CRDM housing caps were installed to CRDM housing at penetration #1 and 18 to catch any potential thermal sleeve remnant due to thermal sleeve degradation. The Part Length CRDM drive rods at penetration #1, 18 and 19 were cut due to the weld repair and the thermal sleeve degradation. Canopy Seal Clamp Assemblies have been installed at the lower canopy seal weld on some CRDMs. These clamps greatly reduce the likelihood of any non-pressure boundary RCS leakage emanating from the lower canopy seal weld. The seal
arrangement at the upper end of the instrumentation adaptors consists of a welded flexible canopy seal. Inlet and outlet nozzles are located symmetrically around the vessel. Outlet nozzles are arranged on the vessel to facilitate optimum layout of the RCS equipment. The inlet nozzles are tapered from the coolant loop vessel interfaces to the vessel inside wall to reduce loop pressure drop.

RN 12-042 14-024 15-023

RN 08-029

The bottom head of the vessel contains penetration nozzles for connection and entry of the nuclear incore instrumentation. Each nozzle consists of a tubular member made of Inconel. Each tube is attached to the inside of the bottom head by a partial penetration weld.

Internal surfaces of the vessel which are in contact with primary coolant are weld overlay with 0.125 inch minimum of stainless steel. The exterior of the reactor vessel is insulated with canned stainless steel reflective sheets. The insulation is a minimum of three inches thick and contoured to enclose the top, sides, and bottom of the vessel. All the insulation modules are removable but the access to vessel side insulation is limited by the surrounding concrete.

The reactor vessel nozzle insulation consists of a noncrushable nonmetalic material clad with stainless steel.

5.4.2.1 <u>Fabrication Processes</u>

1. The use of severely sensitized stainless steel as a pressure boundary material has been prohibited and has been eliminated by both a select choice of material and by programming the method of assembly.

   Refer to Appendix 3A for a discussion of Regulatory Guide 1.44.

2. Minimum preheat requirements have been established for pressure boundary welds using low alloy weld material. Special preheat requirements have been

added for stainless steel cladding of low stressed areas. Controlled limitations are placed on preheat requirements by Westinghouse. The purpose of placing limitations on preheat requirements is the addition of precautionary measures to decrease the probabilities of weld cracking by decreasing temperature gradients, lower susceptibility to brittle transformation, prevention of hydrogen embrittlement and reduction in peak hardness.

Refer to Appendix 3A for a discussion of Regulatory Guide 1.50.

3. The control rod drive mechanism head adaptor threads and surfaces of the guide studs are chrome plated to prevent possible galling of the mated parts.

4. At all locations in the reactor vessel where stainless steel and Inconel are joined, the final joining beads are Inconel weld metal in order to prevent cracking.

5. Core region shells fabricated of plate material have longitudinal welds which are angularly located away from the peak neutron exposure experienced in the vessel, where possible.

6. The location of full penetration weld seams in the upper closure head and vessel bottom head are restricted to areas that permit accessibility during inservice inspection.

7. The stainless steel clad surfaces are sampled to assure that composition and delta ferrite requirements are met.

8. The procedure qualification for cladding low alloy steel (SA508 Class 2) requires a special evaluation to assure freedom from underclad cracking.

Refer to Appendix 3A for a discussion of Regulatory Guide 1.43.

Principal design parameters of the reactor vessel are given in Table 5.4-1.

5.4.2.2 Protection of Closure Studs

Procedures require the closure studs, nuts, and washers to be removed from the reactor cavity in preparation for refueling. These items may be placed in storage racks and stored at convenient locations on the reactor building operating deck prior to removal of the reactor closure head. Alternatively, these components may remain attached to the closure head and moved with the head, as it is transported to the storage pedestal. RN 02-024

The reactor closure studs are removed from the refueling cavity prior to flooding and are never exposed to the borated refueling cavity water. The stud holes in the reactor vessel flange are sealed with special plugs before removing the reactor closure, thus preventing leakage of the borated refueling water into the stud holes.

5.4.3 EVALUATION

5.4.3.1 Steady-State Stresses

Evaluation of steady-state stresses is discussed in Section 5.2.1.10.

5.4.3.2 Fatigue Analysis Based on Transient Stresses

Fatigue analysis of transient stresses is discussed in Sections 5.2.1.5 and 5.2.1.10.

5.4.3.3 Thermal Stresses Due to Gamma Heating

The stresses due to gamma heating in the vessel wall are also calculated by the vessel vendor and combined with the other design stresses.

They are compared with the code allowable limit for mechanical plus thermal stress intensities to verify that they are acceptable. The gamma stresses are low and thus have a negligible effect on the stress intensity in the vessel.

5.4.3.4 Thermal Stresses Due to Loss of Coolant Accident

Fracture mechanics evaluation of the reactor vessel due to thermal stresses following a loss of coolant accident are discussed in Section 5.2.1.10.

5.4.3.5 Heatup and Cooldown

Heatup and cooldown requirements for the reactor vessel material are discussed in Section 5.2.4 and the Technical Specifications. RN 99-136

5.4.3.6 Irradiation Surveillance Programs

10CFR50, Appendix H, requires that neutron dosimetry be present to monitor the reactor vessel and measure damage associated with fast neutron exposure (neutron fluence). The originally installed internal surveillance capsules are not sufficient to provide this monitoring capability throughout the VCSNS extended license period.

ECR 50571 changed the methodology of the Irradiation Surveillance Program at the Virgil C. Summer Nuclear Station. During Refueling Outage 15, the last in-vessel surveillance specimen was removed from its in-vessel location and new dosimetry capsules were installed exterior to the reactor vessel (ex-vessel) for the measurement of neutron exposure. RN 05-017

Presented below are both a description of the new ex-vessel dosimetry program as well as a description of the former in-vessel program, used for the plant’s first 14 operating cycles.

5.4.3.6.1 Ex-Vessel Irradiation Surveillance Program

5.4.3.6.1.1 Summary

An Ex-Vessel Neutron Dosimetry system is installed to monitor neutron fluence in the reactor vessel and vessel support structure. Passive neutron sensors are positioned in the reactor cavity annulus, between the reactor vessel and the biological shield wall. These sensor capsules are connected to, and supported by, stainless steel bead chains, which are suspended from a support bar installed just below the reactor vessel inlet and outlet nozzles. Capsules are then raised and lowered into position (from beneath the Reactor Vessel) to periodically remove the sensors for evaluation. This design ensures a correct and repeatable axial and azimuthal placement of the dosimetry relative to well-known reactor features, which is essential for a long-term monitoring program.

5.4.3.6.1.2 Background

The Code of Federal Regulations, Title 10, Part 50, Appendix H, requires that neutron dosimetry be present to monitor the reactor vessel throughout plant life and that material specimens be used to measure damage associated with the end-of-life fast neutron exposure of the reactor vessel. The Ex-Vessel Neutron Dosimetry Program is designed to provide a verification of fast neutron exposure distributions within the reactor vessel wall and to establish a mechanism to enable long term monitoring of those portions of the reactor vessel and vessel support structure that could experience significant radiation induced increases in reference nil ductility transition temperature (RT_{NDT}) over the service lifetime of the plant. RN 05-017

When used in conjunction with data from internal surveillance capsules and the results of neutron transport calculations, the reactor cavity neutron measurements allow the projection of embrittlement gradients through the reactor vessel wall with a minimum degree of uncertainty. Minimizing the uncertainty in the neutron exposure projections will help to assure that the reactor can be operated in the least restrictive mode possible with respect to:

- 10CFR50 Appendix G pressure/temperature limit curves for normal heat up and cool down of the reactor coolant system,
- Emergency Response Guideline (ERG) pressure/temperature limit curves, and
- Pressurized thermal shock (PTS) RT_{NDT} screening criteria.

The installed ex-vessel surveillance program satisfies the regulatory guidance provided in the Generic Aging Lessons Learned (GALL) Report (NUREG-1801), Section XI.M31 Reactor Vessel Surveillance; which states in part:

A plant with a surveillance program containing capsules with projected fast neutron fluence exceeding a 60-year fluence at the end of 40 years, i.e., a lead factor greater

than 1.5, should remove the capsules when they reach the 60-year exposure. One of these capsules should be tested to meet the requirements of ASTM E185 and the remaining capsules should be placed in storage without material testing. Subsequently, an alternative dosimetry program will need to be instituted to monitor reactor vessel neutron exposure through the license renewal period.

5.4.3.6.1.3 Sensor Sets

To achieve its goals, the Ex-Vessel Neutron Dosimetry Program employs advanced sensor sets which consist of both encapsulated dosimeters and gradient chains.

Encapsulated dosimeters include radiometric monitors and are employed at discrete locations within the reactor cavity to measure axial variations in neutron spectrum over the core height, particularly near the top of the fuel where back scattering of neutrons from primary loop nozzles and reactor vessel support structures can produce significant differences.

Radiometric Monitors include cadmium-shielded foils of the following metals: copper, titanium, iron, nickel, niobium, and cobalt-aluminum. Cadmium shielded fast fission reactions include U-238 and Np-237 in vanadium encapsulated oxide detectors. Bare iron and cobalt monitors are also included.

Stainless steel gradient chains are used in conjunction with the encapsulated dosimeters to complete the mapping of the neutron environment between the discrete locations chosen for spectrum determinations. At each of the azimuthal locations selected for spectrum measurements, stainless steel gradient chains extend over the full height of the active fuel. RN 05-017

Gradient Chains are stainless steel bead chains which connect and support the dosimeter capsules containing the radiometric monitors. These segmented chains provide iron, nickel, and cobalt reactions that are used to complete the determination of the axial and azimuthal gradients. The high purity iron, nickel, and cobalt-aluminum foils contained in the multiple foil sensor sets provide a direct correlation with the measured reaction rates from these gradient chains. These cross-comparisons permit the use of the gradient measurements to derive neutron flux distributions in the reactor cavity with a high level of confidence.

The design of the dosimetry support system and the gradient chains/chain stops ensure correct and repeatable axial and azimuthal placement of the dosimetry relative to well-known reactor features.

5.4.3.6.1.4 Physical Installation

The Ex-Vessel Neutron Dosimeters are located in the annular space between the reactor vessel reflective insulation and the primary biological shield wall. The dosimetry loops are suspended from a stainless steel support bar that is installed just below the reactor vessel nozzles (approximately eight inches above the top of the active fuel).

Located between the A-Loop inlet and B-Loop outlet nozzles, the gradient chain loops are aligned at 270, 285, 300, and 315 degrees (reactor coordinates). The bar is supported from above by two 1/8-inch stainless steel support chains. The support chains are, in turn, attached to a spring hook attachment plates that are field-welded to the carbon steel reactor cavity liner plate.

These sensor sets consists of a small (1.0 x 0.5 x 3.875 inch-long) aluminum dosimeter capsule connected to and supported by stainless steel bead chain. Clamps are installed at the bottom of each chain loop, which are in turn attached to the grating platform beneath the reactor vessel, to prevent relative movement of the capsule during plant operation.

5.4.3.6.1.5 Supporting Analysis

At the conclusion of the first irradiation cycle the multiple foil sensor sets will be removed from the reactor cavity for evaluation. At the same time replacement dosimetry will be installed in the reactor cavity to provide monitoring for subsequent irradiation cycles.

The evaluation of the dosimetry sets includes radiochemical analysis of each RM sensor. All laboratory procedures conform to the latest version of ASTM dosimetry standards and practices established for light-water reactor applications. Equivalent full-power reaction rates will be computed from the measured specific activities based on the irradiation history. The chemical composition of the stainless steel gradient chain is determined by taking the ratio of the Mn-54, Co-58, and Co-60 specific activities as measured in the pure metal foils and as measured in the bead chain segments adjacent to the dosimetry capsules containing the foils. RN 05-017

Plant specific discrete ordinates transport theory calculations will be performed to determine the neutron and gamma ray environment within the reactor geometry. The specific methods applied are consistent with those described in WCAP-15557, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology", S. L. Anderson, August 2000 and meet the requirements of Regulatory Guide RG-1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence", March 2001.

Analysis from the prior in-vessel surveillance capsules (U withdrawn at End of Cycle (EOC) 1, V withdrawn at EOC 3, X withdrawn at EOC 5, W withdrawn at EOC 10, Z withdrawn at EOC 14, and Y withdrawn at EOC 15) provides a consistent set of calculations and measurements detailing the integrated neutron exposure of the Virgil C. Summer reactor vessel from plant startup. The analysis of the ex-vessel neutron dosimetry removed at the end of Cycle 16 will build upon that work. This will provide the greatest accuracy in determining the fast neutron exposure rate distribution incident on and through the reactor vessel wall and will provide a sound basis for planning future evaluations of ex-vessel neutron dosimetry.

5.4.3.6.2 In-Vessel Irradiation Surveillance Program

5.4.3.6.2.1 Summary

In the In-Vessel Irradiation Surveillance Program, the evaluation of the radiation damage was based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile and 1/2 T (thickness) compact tension (CT) fracture mechanics test specimens. The program was directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach. The program conformed with ASTM-E-185 "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," and 10 CFR 50, Appendix H. RN 05-017

5.4.3.6.2.2 Specimen Capsules

The reactor In-Vessel Surveillance Program used six specimen capsules. The capsules were located in guide baskets welded to the outside of the neutron shield pads and were positioned directly opposite the center portion of the core. The capsules were removed when the vessel head was removed and could be replaced when the internals are removed.

Each surveillance capsule contained the following materials:

a) Transverse orientation specimen from the limiting core region intermediate shell course plate A9154-1.

b) Longitudinal orientation specimen from the limiting core region intermediate shell course plate A9154-1.

c) Specimen of weldment joining sections of materials from intermediate shell plate A9154-1 and adjoining lower shell course plate C9923-2. The weldment is fabricated from the same heat of weld wire (Heat Number 4P4784) and Linde 124 flux (Lot Number 3930) as used to fabricate the core region girth and vertical weld seams.

d) Heat affected zone of plate A9154-1.

Each specimen was composed of the same material as the plate or weldment represented by the specimen. RN 05-017 The copper and phosphorous content and fracture toughness data of each beltline plate and the core region weldments are provided in Table 5.2-21. The originally installed location of each capsule in the reactor vessel is shown on Figure 5.4-1. RN 05-017

The six capsules contained 54 tensile specimens, 360 Charpy V-notch specimens (which include weld metal and weld heat affected zone material), and 72 CT specimens. RN 05-017 Archive material sufficient for two additional capsules will be retained.

The test specimen types, and their orientation, for each surveillance material were as follows: RN 05-017

Surveillance Material	Quantity of Test Specimen Types: Charpy	Tensile	1/2 - CT
Plate A9154-1 (transverse orientation)	15	3	4
Plate A9154-1 (longitudinal orientation)	15	3	4
Weld metal	15	3	4
Heat Affected Zone of Plate A9154-1	15	-	-

Dosimeters, including Ni, Cu, Fe, Co-Al, Cd shielded Co-Al, Cd shielded Np-237, and Cd shielded U-238, were placed in filler blocks drilled to contain them. The dosimeters permit evaluation of the flux seen by the specimens and the vessel wall. In addition, thermal monitors made of low melting point alloys were included to monitor the maximum temperature of the specimens. The specimens were enclosed in a tight fitting stainless steel sheath to prevent corrosion and ensure good thermal conductivity. The complete capsule was helium leak tested. As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01 percent was made for surveillance material and as deposited weld metal. RN 05-017

Each of the six capsules contained the following specimens: RN 05-017

Material	Number of Charpys	Number of Tensiles	Number of CT's
Limiting Base Material [1]	15	3	4
Limiting Base Material [2]	15	3	4
Weld Metal [3]	15	3	4
Heat Affected Zone	15	-	-

(1) Specimens oriented in the major rolling or working direction.

(2) Specimens oriented normal to the major rolling or working direction.

(3) Weld metal to be selected per ASTM-E-185.

5.4.3.6.2.3 Removal Schedule RN 05-017

The schedule for removal of specimen capsules from the reactor vessel was as follows:

Capsule	Location (deg)	Capsule Lead Factor	Removal time [(a)]	Calculated Fluence (n/cm^2) [(b)]
U	343	3.32	1.12 (EOC 1)	6.542×10^{18}
V	107	3.72	2.93 (EOC 3)	1.538×10^{19}
X	287	3.84	5.03 (EOC 5)	2.543×10^{19}
W	110	3.40	10.78 (EOC 10)	4.664×10^{19}
Z	340	3.40	16.67 (EOC 14)	6.712×10^{19}
Y	290	3.40	17.28 (EOC 15)	(c)

RN 00-001

RN 05-017

(a) Effective full power years from plant startup. End of Cycle (EOC) value given in parenthesis. Note that core thermal power was uprated from 2775 to 2900 MW_{th} starting with operating cycle 10.

(b) Plant specific evaluation per WCAP-16298-NP, Revision 0[2] using Capsule Z dosimetry data. RN 00-001 04-042

(c) Placed in storage for possible future use. RN 05-017

5.4.3.6.3 Measurement of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples RN 05-017

Measurement of Integrated Fast Neutron Flux and Fluence at the Irradiation Samples, and a description of how they were derived, are included in WCAP-16298-NP [2]. RN 04-042

5.4.3.6.4 Calculation of Integrated Fast Neutron (E 1.0 MeV) Flux at the Irradiation Samples RN 05-017

Calculations of Integrated Fast Neutron Flux and Fluence at the Irradiation Samples, and a description of the methodology used, are included in WCAP-16298-NP [2]. RN 04-042

5.4.3.7 Capability for Annealing the Reactor Vessel

There are no special design features which would prohibit the insitu annealing of the vessel. In the unlikely case that an annealing operation is required to restore the properties of the vessel material opposite the reactor core because of neutron irradiation damage, a metal temperature greater than 650°F would be applied for a maximum period of 168 hours. Various modes of heating may be used depending on the temperature.

The reactor vessel materials surveillance program is adequate to accommodate the annealing of the reactor vessel. Sufficient specimens are available to evaluate the effects of the annealing treatment.

5.4.4 TESTS AND INSPECTIONS

The reactor vessel quality assurance program is given in Table 5.4-2.

Inservice inspection is in accordance with the ASME Code, Section XI.

5.4.4.1 Ultrasonic Examinations

During fabrication, angle beam inspection of 100 percent of plate material is performed to detect discontinuities that may be undetected by longitudinal wave examination, in addition to the design code straight beam ultrasonic test.

5.4.4.2 Penetrant Examinations

The partial penetration welds for the control rod drive mechanism head adaptors and the bottom instrumentation tubes are inspected by dye penetrant after the root pass in addition to code requirements. Core support block attachment welds are inspected by dye penetrant after the first layer of weld metal and after each 1/2 inch of weld metal.

5.4.4.3 Magnetic Particle Examination

The magnetic particle examination requirements below are in addition to the magnetic particle examination requirements of Section III of the ASME Code.

All magnetic particle examinations of materials and welds shall be performed in accordance with the following:

1. Prior to the final post weld heat treatment (only by the Prod, Coil or Direct Contact Method).

2. After the final post weld heat treatment (only by the Yoke Method).

The following surfaces and welds shall be examined by magnetic particle methods. The acceptance standards shall be in accordance with Section III of the ASME Code.

5.4.4.3.1 Surface Examinations

1. Magnetic particle examine all exterior vessel and heat surfaces after the hydrostatic test.

2. Magnetic particle examine all inside diameter surfaces of carbon and low alloy steel products that have their properties enhanced by accelerated cooling. This inspection to be performed after forming and machining (if performed) and prior to cladding.

5.4.4.3.2 Weld Examination

Magnetic particle examine weld metal buildup for vessel supports, closure head lifting lugs and refueling seal ledge to the reactor vessel after the first layer and each 1/2 inch of weld metal is deposited. All pressure boundary welds are examined after back chipping or back grinding operations.

5.4.4.4 Inservice Inspection

The full penetration welds in the following areas of the installed irradiated reactor vessel are available for visual and/or nondestructive inspection.

1. Vessel shell (from the inside surface).

2. Primary coolant nozzles (from the inside surface).

3. Closure head (from the inside and outside surfaces). Bottom head (from the outside surface).

4. Closure studs, nuts and washers.

5. Field welds between the reactor vessel, nozzles and the main coolant piping.

6. Vessel flange seal surface.

The design considerations which have been incorporated into the system design to permit the above inspections are as follows:

1. All reactor internals are completely removable. The tools and storage space required to permit these inspections are provided.

2. The closure head is stored dry on the reactor operating deck during refueling to facilitate direct visual inspection.

3. All reactor vessel studs, nuts and washers can be removed to dry storage during refueling.

4. Removable plugs are provided in the primary shield. The insulation covering the nozzle welds may be removed.

5. Access holes are provided in the lower internals barrel flange to allow remote access to the reactor vessel internal surfaces between the flange and the nozzles without removal of the internals.

The reactor vessel presents access problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps have been incorporated into the design and manufacturing procedures in preparation for the periodic nondestructive tests which are required by the ASME inservice inspection code. These are:

1. Shop ultrasonic examinations are performed on all internally clad surfaces to acceptance and repair standards to assure an adequate cladding bond to allow later ultrasonic testing of the base metal from inside surface. The size of cladding bonding defect allowed is 1/4 inch by 3/4 inch, with the greater direction parallel to the weld in the region bounded by 2T (T = wall thickness) on both sides of each full penetration pressure boundary weld. Unbonded areas exceeding 0.442 square inches (3/4 inch diameter) in all other regions are rejected.

2. The design of the reactor vessel shell is a clean, uncluttered cylindrical surface to permit future positioning of the test equipment without obstruction.

3. The weld deposited clad surface on both sides of the welds to be inspected are specifically prepared to assure meaningful ultrasonic examinations.

4. After the shop hydrostatic testing, selected areas of the reactor vessel are ultrasonically tested and mapped to facilitate the inservice inspection program.

The vessel design and construction enables inspection and compliance with the ASME Code, Section XI, 1974 Edition.

5.4.5 REFERENCE

1. Soltesz, R. G., et al, "Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation, Volume 5 - Two-Dimensional Discrete Ordinates Technique," WANL-PR-(LL)-034, August, 1970.

2. C. M. Burton, J. Conermann, E. T. Hayes, "Analysis of Capsule Z from The South Carolina Electric & Gas Company V. C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-16298-NP, Revision 0, August 2004. RN 04-042

3. Bamford, W. H., et al, “Technical Basis for the Embedded Flaw Process for Repair of Reactor Vessel Head Penetrations,” WCAP-15987-P Revision 2-P-A (Proprietary), December 2003.

4. Letter from Chief Robert J. Pascarelli (NRC) to Mr. Thomas D. Gatlin, “Virgil C. Summer Nuclear Station, Unit 1 – Alternative Request Weld Repair for Reactor Vessel Upper Head Penetrations (TAC NO. MF3546),” April 30, 2014.

RN 15-023

TABLE 5.4-1

REACTOR VESSEL DESIGN PARAMETERS

Design/Operating Pressure, psig	2485/2317
Design Temperature, °F	650
Overall Height of Vessel and Closure Head, ft-in (bottom head outside diameter to top of control rod mechanism adapter)	41
Thickness of Insulation, in (minimum)	3
Number of Reactor Closure Head Studs	58
Diameter of Reactor Closure Head/Studs, in (minimum shank)	5-13/16
Inside Diameter of Flange, in	149-15/16
Outside Diameter of Flange, in	184
Inside Diameter at Shell, in	157
Inlet Nozzle Inside Diameter, in	27-1/2
Outlet Nozzle Inside Diameter, in	29
Clad Thickness, in (minimum)	1/8
Lower Head Thickness, in (minimum)	4-7/8
Vessel Belt-Line Thickness, in (minimum)	7-3/4
Closure Head Thickness, in	5-3/4

TABLE 5.4-2

REACTOR VESSEL QUALITY ASSURANCE PROGRAM

Forgings	RT [1]	UT [1]	PT [1]	MT [1]
1. Flanges		Yes		Yes
2. Studs, Nuts		Yes		Yes
3. Head Adapters		Yes	Yes	
4. Head Adapter Tube		Yes	Yes	
5. Instrumentation Tube		Yes	Yes	
6. Main Nozzles		Yes		Yes
7. Nozzle Safe Ends (If forging is employed)		Yes	Yes	
Plates		Yes		Yes
Weldments				
1. Main Seam	Yes	Yes		Yes
2. CRD Head Adapter Connection			Yes	
3. Instrumentation Tube Connection			Yes	
4. Main Nozzle	Yes	Yes		Yes
5. Cladding		Yes	Yes	
6. Nozzle Safe Ends (If forging)	Yes	Yes	Yes	
7. Nozzle Safe Ends (If weld deposit)	Yes	Yes	Yes	
8. Head Adapter Forging to Head Adapter Tube	Yes		Yes	
9. All Ferritic Welds Accessible after Hydrotest		Yes		Yes
10. All Non-Ferritic Welds Accessible after Hydrotest		Yes	Yes	
11. Seal Ledge				Yes
12. Head Lift Lug				Yes
13. Core Pad Welds			Yes	

[1] RT - Radiographic, UT - Ultrasonic, PT - Dye Penetrant, MT - Magnetic Particle

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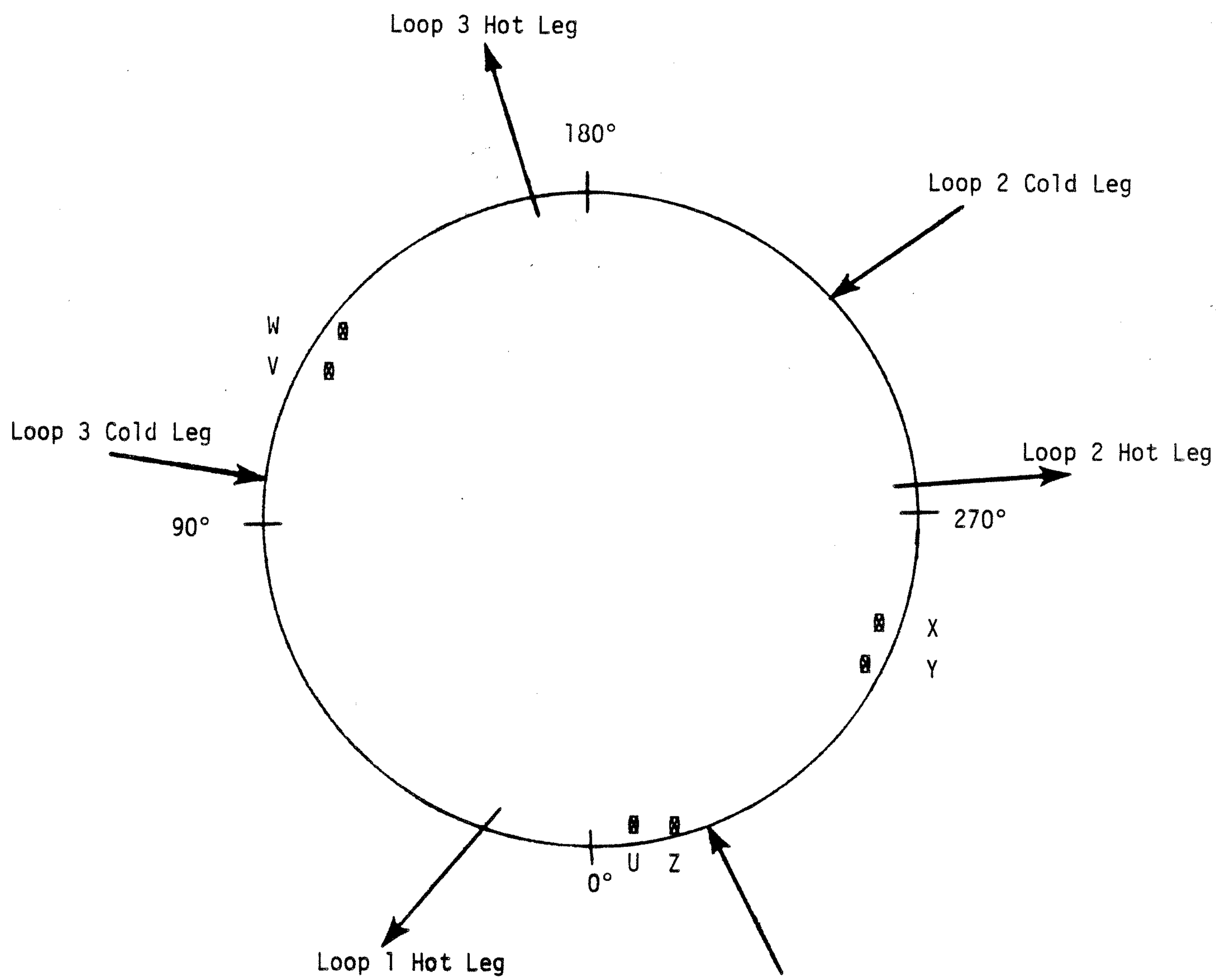

Specimen	L
U	343°
V	107°
W	110°
X	287°
Y	290°
Z	340°

Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Surveillance Capsule Locations

Figure 5.4-1

5.5 COMPONENT AND SUBSYSTEM DESIGN

5.5.1 REACTOR COOLANT PUMPS

5.5.1.1 Design Bases

The reactor coolant pump ensures an adequate core cooling flowrate and hence sufficient heat transfer, to maintain a departure from nucleate boiling ratio (DNBR) greater than 1.30 within the parameters of operation. The required net positive suction head is by conservative pump design always less than that available by system design and operation.

Sufficient pump rotation inertia is provided by a flywheel, in conjunction with the impeller and motor assembly, to provide adequate flow during coastdown. This flow, following an assumed loss of pump power, provides the core with adequate cooling.

The reactor coolant pump motor is tested, without mechanical damage, at overspeeds up to and including 125% of normal speed. The integrity of the flywheel during a LOCA is demonstrated in Reference [1]. Reference [1] is undergoing generic review by the NRC.

The reactor coolant pump is shown in Figure 5.5-1. The reactor coolant pump design parameters are given in Table 5.5-1.

Code and material requirements are provided in Section 5.2.

5.5.1.2 Design Description

The reactor coolant pump is a vertical, single stage, centrifugal, shaft seal pump designed to pump large volumes of reactor coolant at high temperatures and pressures.

The pump consists of 3 areas from bottom to top. They are the hydraulics, shaft seals, and the motor.

1. The hydraulic section consists of an impeller, diffuser, casing, thermal barrier heat exchanger, lower radial bearing, bolting ring, motor stand, and pump shaft.

2. The shaft seal section consists of 3 devices. They are the number 1, controlled leakage, film riding face seal and the number 2 and number 3 rubbing face seals. These seals are contained within the main flange and seal housing.

3. The motor section consists of a vertical solid shaft, squirrel cage induction type motor, an oil lubricated double tilting pad Kingsbury type thrust bearing, 2 oil lubricated radial bearings, and a flywheel.

Attached to the bottom of the pump shaft is the impeller. The reactor coolant is drawn up through the impeller, discharged through passages in the diffuser, and out through the discharge nozzle in the side of the casing. Above the impeller is a thermal barrier heat exchanger which limits heat transfer between hot system water and seal injection water.

High pressure seal injection water is introduced through the thermal barrier wall. A portion of this water flows up around the bearing through the seals; the remainder flows down through the thermal barrier where it acts as a buffer to prevent system water from entering the radial bearing and seal section of the unit. The thermal barrier heat exchanger provides a means of cooling system water to an acceptable level in the event that seal injection flow is lost. The water lubricated journal type pump bearing, mounted above the thermal barrier heat exchanger, has a self-aligning spherical seat.

The reactor coolant pump motor bearings are of conventional design. The radial bearings are the segmented pad type, and the thrust bearings are tilting pad Kingsbury bearings. All are oil lubricated. The lower radial bearing and the thrust bearings are submerged in oil, and the upper radial bearing is oil fed from an impeller integral with the thrust runner.

The motor is an air-cooled, Class F thermalastic epoxy insulated, squirrel cage induction motor. The rotor and stator are of standard construction and are cooled by air. Six (6) resistance temperature detectors are located throughout the stator to sense the winding temperature. The top of the motor consists of a flywheel and an anti-reverse rotation device.

Each of the reactor coolant pumps is equipped for continuous monitoring of reactor coolant pump shaft and frame vibration levels. Shaft vibration is measured by 2 relative shaft probes mounted on top of the pump seal housing. The probes, 1 in line with the pump discharge and the other perpendicular to the pump discharge, are mounted in the same horizontal plane near the pump shaft. Frame vibration is measured by 2 velocity velomitors located 90 degrees apart in the same horizontal plane and mounted at the top of the motor support stand. RN 13-001 Proximeters and converters convert the probe signals to linear output which is displayed on monitor meters in the control room. The monitor meters automatically indicate the highest output from the relative probes and velomitors; RN 13-001 manual selection allows monitoring of individual probes.

Indicator lights display caution and danger limits of vibration, which are adjustable over the full range of the motor scale.

All parts of the pump in contact with the reactor coolant are austenitic stainless steel except for special parts, such as seals and bearings. Component cooling water is supplied to the 2 oil coolers on the pump motor and to the pump thermal barrier heat exchanger.

In addition, the reactor coolant pumps receive pump shaft seal cooling water from the Chemical and Volume Control System (CVCS). The component cooling water supply to the thermal barrier is provided as a backup heat sink to the CVCS cooling water flow to the reactor coolant pump seals. A failure of the CCWS supply to the thermal barrier would not jeopardize operation of the reactor coolant pump or cause loss of reactor coolant. The individual CCWS return lines from the upper and lower bearing oil coolers, as well as the return line from the thermal barrier, are equipped with seismically qualified flow indicators powered from the vital bus (associated). These flow indicators actuate alarms on the main control board. The temperature indicator and alarms on the CCWS return lines are located on the main control board. The reactor coolant pump motor bearing temperatures are supplied as input to the process computer. A high temperature causes the computer to alarm and identify the out-of-limit temperatures. The CCWS isolation valve monitor lights, which would indicate valve closure, are located on the main control board.

The operator action necessary to trip the reactor and stop the reactor coolant pumps is not complicated and is a direct logical result of the event symptoms alarmed and indicated. An operator response time of 10 minutes is conservative and appropriate for this event during normal operation.

Actual tests performed by Westinghouse Electric Corporation on reactor coolant pumps of the type provided for the Virgil C. Summer Nuclear Station show that interruption of the CCWS flow for 10 minutes or less will not result in damage to the pump.

The pump shaft, seal housing, thermal barrier, main flange, and motor stand can be removed from the casing without disturbing the reactor coolant piping. The flywheel is available for inspection by removing the cover.

The performance characteristic, shown in Figure 5.5-2, is common to all of the fixed speed mixed flow pumps and the "knee" at about 45% design flow introduces no operational restrictions, since the pumps operate at full speed.

5.5.1.3 Design Evaluation

5.5.1.3.1 Pump Performance

The reactor coolant pumps are sized to deliver flow at rates which equal or exceed the required flowrates. Initial Reactor Coolant System (RCS) tests confirm the total delivery capability. Thus, assurance of adequate forced circulation coolant flow is provided prior to initial plant operation.

The Reactor Trip System ensures that pump operation is within the assumptions used for loss of coolant flow analyses, which also assures that adequate core cooling is provided to permit an orderly reduction in power if flow from a reactor coolant pump is lost during operation.

An extensive test program has been conducted for several years to develop the controlled leakage shaft seal for pressurized water reactor applications. Long term tests were conducted on less than full scale prototype seals as well as on full size seals. Operating plants continue to demonstrate the satisfactory performance of the controlled leakage shaft seal pump design.

The support of the stationary member of the number 1 seal ("seal ring") is such as to allow large deflections, both axial and tilting, while still maintaining its controlled gap relative to the seal runner. Even if all the graphite were removed from the pump bearing, the shaft could not deflect far enough to cause opening of the controlled leakage gap. The "spring-rate" of the hydraulic forces associated with the maintenance of the gap is high enough to ensure that the ring follows the runner under very rapid shaft deflections.

Testing of pumps with the number 1 seal entirely removed (full reactor pressure on the number 2 seal) shows that relatively small leakage rates would be maintained for long periods of time; even if the number 1 seal fails entirely, the number 2 seal would maintain these small leakage rates. The plant operator is warned of number 1 seal damage by the increase in number 1 seal leakoff. Following warning of excessive seal leakage conditions, the plant operator should close the number 1 seal leakoff line and secure the pump, as specified in the instruction manual. It may be concluded that gross leakage from the pump does not occur if the proper operator action is taken subsequent to warning of excessive seal leakage conditions.

The effect of loss of offsite power on the pump itself is to cause a temporary stoppage in the supply of injection flow to the pump seals and also of the cooling water for seal and bearing cooling. The emergency diesel generators are started automatically due to loss of offsite power so that seal injection flow is automatically restored.

5.5.1.3.2 Coastdown Capability

It is important to reactor operation that the reactor coolant continues to flow for a short time after a reactor trip. In order to provide this flow in a station blackout condition, each reactor coolant pump is provided with a flywheel. Thus, the rotating inertia of the pump, motor, and flywheel is employed during the coastdown period to continue the reactor coolant flow. The coastdown flow transients are provided in the figures in Section 15.3.

The pump/motor system is designed for Seismic Category 1. The integrity of the bearings is described in Section 5.5.1.3.4. Hence, it is concluded that the coastdown capability of the pumps is maintained even under the most adverse case of a loss of offsite power coincident with the safe shutdown earthquake. Core flow transients and figures are provided in Section 15.2. RN 99-110

5.5.1.3.3 Flywheel Integrity

Demonstration of integrity of the reactor coolant pump flywheel is discussed in Section 5.2. Additional discussion is contained in Reference [1].

5.5.1.3.4 Bearing Integrity

The design requirements for the reactor coolant pump bearings are primarily aimed at ensuring a long life with negligible wear, so as to give accurate alignment and smooth operation over long periods of time. The surface bearing stresses are held at a very low value and even under the most severe seismic transients do not begin to approach loads which cannot be adequately carried for short periods of time.

Because there are no established criteria for short time stress-related failures in such bearings, it is not possible to make a meaningful quantification of such parameters as margins to failure, safety factors, etc. A qualitative analysis of the bearing design, embodying such considerations, gives assurance of the adequacy of the bearing to operate without failure.

Low oil levels in the lube oil sumps signal an alarm in the control room and require shutting down of the pump. Each motor bearing contains embedded temperature detectors, and so initiation of failure, separate from loss of oil, is indicated and alarmed in the control room as a high bearing temperature. This, again, requires pump shutdown. If these indications are ignored and the bearing proceeded to failure, the low melting point of Babbitt metal on the pad surfaces ensures that sudden seizure of the shaft will not occur. In this event the motor continues to operate, as it has sufficient reserve capacity to drive the pump under such conditions. However, the high torque required to drive the pump will require high current which will lead to the motor being shutdown by the electrical protection systems.

5.5.1.3.5 Locked Rotor

It may be hypothesized that the pump impeller might severely rub on a stationary member and then seize. Analysis has shown that under such conditions, assuming instantaneous seizure of the impeller, the pump shaft fails in torsion just below the coupling to the motor, disengaging the flywheel and motor from the shaft. This constitutes a loss of coolant flow in the loop. Following such a postulated seizure, the motor continues to run without any overspeed, and the flywheel maintains its integrity, as it is still supported on a shaft with 2 bearings. Flow transients are provided in the figures in Section 15.4 for the assumed locked rotor.

There are no other credible sources of shaft seizure other than impeller rubs. Any seizure of the pump bearing is precluded by graphite in the bearing. Any seizure in the seals results in a shearing of the anti-rotation pin in the seal ring. The motor has adequate power to continue pump operation even after the above occurrences. Indications of pump malfunction in these conditions are initially by high temperature signals from the bearing water temperature detector and excessive number 1 seal

leakoff indications respectively. Following these signals, pump vibration levels are checked. Excessive vibration indicates mechanical trouble and the pump is shutdown for investigation.

5.5.1.3.6 Critical Speed

The reactor coolant pump shaft is designed so that its operating speed is below its first critical speed. This shaft design, even under the most severe postulated transient, gives low values of actual stress.

5.5.1.3.7 Missile Generation

Each component of the pump is analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy casing. Further discussion and analysis of missile generation are contained in Reference [1].

5.5.1.3.8 Pump Cavitation

The minimum net positive suction head required by the reactor coolant pump at running speed is approximately a 192 foot head (approximately 85 psi). In order for the controlled leakage seal to operate correctly it is necessary to have a minimum differential pressure of approximately 200 psi across the seal. This corresponds to a primary loop pressure at which the net positive suction head requirement is exceeded and no limitation on pump operation occurs from this source. At this pressure the net positive suction head requirement is exceeded and the pump can be successfully operated.

5.5.1.3.9 Pump Overspeed Considerations

For turbine trips actuated by either the Reactor Trip System or the Turbine Protection System, the generator and reactor coolant pumps are maintained connected to the external network for at least 30 seconds to prevent any pump overspeed condition (see Section 8.2).

An electrical fault requiring immediate trip of the generator (with resulting turbine trip) could result in an overspeed condition. However, the turbine control system and the turbine intercept valves limit the overspeed to less than 120%. In case a generator trip de-energizes the pump buses, the reactor coolant pump motors will be transferred to offsite power within 6 to 10 cycles. Further discussion of pump overspeed considerations is contained in Reference [1]. RN 11-015

5.5.1.3.10 Anti-Reverse Rotation Device

Each of the reactor coolant pumps is provided with an anti-reverse rotation device in the motor. This anti-reverse mechanism consists of 5 pawls mounted on the outside diameter of the flywheel, a serrated ratchet plate mounted on the motor frame, a spring return for the ratchet plate, and 2 shock absorbers.

After the motor has slowed and come to a stop, the dropped pawls engage the ratchet plate and, as the motor tends to rotate in the opposite direction, the ratchet plate also rotates until it is stopped by the shock absorbers. The rotor remains in this position until the motor is energized again. When the motor is started, the ratchet plate is returned to its original position by the spring return.

As the motor begins to rotate, the pawls drag over the ratchet plate. When the motor reaches sufficient speed, the pawls are bounded into an elevated position and are held in that position by friction resulting from centrifugal forces acting upon the pawls. Considerable shop testing and plant experience with the design of these pawls have shown high reliability of operation.

5.5.1.3.11 Shaft Seal Leakage

Leakage along the reactor coolant pump shaft is controlled by 3 shaft seals arranged in series such that reactor coolant leakage to the containment is essentially 0. Charging flow is directed to each reactor coolant pump via a $\leq$ 5 micron seal water injection filter. RN 98-128 The flow is injected into the reactor coolant pump through a pipe in the thermal barrier flange and is directed down to a point between the pump shaft bearing and the thermal barrier cooling coils. Here the flow enters the shaft annulus; a portion flows down past the thermal barrier cooling cavity, labyrinth seals and into the RCS; the remainder flows up the pump shaft annulus cooling the lower shaft bearing. This flow provides a back pressure on the number 1 seal and a controlled flow through the seal. Above the seal most of the flow leaves the pump via the number 1 seal leakoff line. Minor flow passes through the number 2 seal and leakoff line and the number 3 seal and leakoff line. This arrangement assures essentially 0 leakage of reactor coolant or trapped gases from the pump.

5.5.1.3.12 Seal Discharge Piping

Discharge pressure from the number 1 seal is reduced to that of the volume control tank. Water from each pump number 1 seal is piped to a common manifold, through the seal water return filter, and through the seal water heat exchanger where the temperature is reduced to that of the volume control tank. The number 2 and number 3 leakoff lines dump number 2 and number 3 seal leakage to the reactor coolant drain tank.

5.5.1.3.13 Loss of Component Cooling Water or Seal Injection

Component cooling water is provided to the reactor coolant pump thermal barrier heat exchanger, as well as to the upper and lower motor bearing oil coolers. In addition, seal injection flow is supplied to the pumps from the chemical and volume control system. These cooling supplies are discussed in the following paragraph and are shown schematically in Figure 5.5-16. Detailed flow diagrams of the Chemical and Volume Control System and the Component Cooling Water System are shown in FSAR Figures 9.3-16 and 9.2-5, respectively.

Consequences of the loss of either of these supplies are also discussed in the following paragraphs.

Seal injection flow, at a slightly higher pressure and at a lower temperature than the Reactor Coolant System, enters the pump through a pipe connection on the thermal barrier flange (see Figure 5.5-17) and is directed to a point between the pump radial bearing and the thermal barrier heat exchanger. Here the flow splits with a portion flowing down through the thermal barrier labyrinth (where it acts as a buffer to prevent reactor coolant from entering the radial bearing and seal section of the pump) and into the Reactor Coolant System. The remainder of the seal injection water flows up through the pump radial bearing and the shaft seals and is discharged via the seal leakoffs.

The pump shaft seal section consists of 3 seals in series, which are contained within a seal housing. The seal arrangement is shown in Figure 5.5-18. The No. 1 seal, the primary seal of the pump, is a controlled-leakage, film-riding seal. Most of the seal injection flow (directed to the seals) is discharged through the No. 1 seal leakoff, which is piped to the volume control tank. Minor leakage passes through the No. 2 and No. 3 seals, which are rubbing-face type seals. The No. 2 and No. 3 seal leakoffs are directed to the reactor coolant drain tank. This arrangement minimizes leakage of water and vapor into the containment.

The thermal barrier is a welded assembly consisting of a flanged cylindrical shell, a series of concentric stainless steel cans, a heat exchanger coil assembly, and 2 flanged water connections.

Component cooling water enters the thermal barrier through a flanged connection on the thermal barrier flange (see Figure 5.5-17). The cooling water flows through the inside of the coiled stainless steel tubing in the heat exchanger and exits through another flanged connection on the thermal barrier flange. During normal operation, the thermal barrier limits the heat transfer from the reactor coolant to the pump internals.

The upper bearing assembly contains an oil-cooled pivoted-pad radial guide bearing (upper guide bearing), as well as a double acting oil-cooled Kingsbury-type thrust bearing (see Figure 5.5-19). The thrust bearing shoes are positioned above and below a common runner to accommodate thrust in both directions. The shoes are mounted on equalizing pads, which distribute the thrust load equally to all the shoes.

The oil is circulated through an external oil-to-water shell and tube heat exchanger (oil cooler) to which component cooling water is supplied.

The lower guide bearing is a pivoted-pad radial bearing, similar to the upper guide bearing.

The entire lower guide bearing assembly is located in the lower oil reservoir, which contains an integral oil-to-water coil type heat exchanger (see Figure 5.5-19).

Component cooling water is supplied to this heat exchanger.

Should a loss of seal injection to the RCPs occur, the pump radial bearing and seals are lubricated by reactor coolant flowing up through the pump. Under these conditions, the CCWS continues to provide flow to the thermal barrier heat exchanger and the heat exchanger, functioning in its backup capacity, cools the reactor coolant before it enters the pump radial bearing and the shaft seal area. The loss of seal injection flow may result in a temperature increase in the pump bearing area, a temperature increase in the seal area, and a resultant increase in the number one seal leak rate; however, pump operation can be continued (for up to 24 hours), provided these parameters remain within the allowable limits.

Should a loss of CCW to the RCPs occur, the Chemical and Volume Control System continues to provide seal injection flow to the RCPs; the seal injection flow is sufficient to prevent damage to the seals with a loss of thermal barrier cooling. Consequently, the RCP can continue to run following a loss of thermal barrier cooling provided that pump seal temperatures remain within allowable limits. However, the loss of CCW to the motor bearing oil coolers will result in an increase in oil temperature and a corresponding rise in motor bearing metal temperature. It has been demonstrated by testing, that the reactor coolant pumps will incur no damage as a result of a CCW flow interruption of 10 minutes.

Two (2) safety related transmitters are provided to redundantly monitor component cooling water flow to the upper and lower reactor coolant pump bearings. Two (2) additional safety related transmitters are provided to redundantly monitor component cooling water flow to the reactor coolant pump thermal barriers. These transmitters provide flow indication and actuate low flow alarms in the control room.

A discussion of the loss of seal injection is provided above. This discussion justifies the use of non-safety grade instrumentation for seal injection flow, since loss of seal injection is not limiting in terms of continued pump operation and does not require immediate operator action.

As discussed above, a loss of CCW to the motor bearing oil coolers will result in an increase in oil temperature and a corresponding rise in motor bearing temperature. Since the loss of CCW to the thermal barrier does not, in itself, affect operation of the RCP, a simultaneous loss of CCW to the thermal barrier and to the motor bearing oil coolers is no worse than a loss of CCW only to the motor bearing oil coolers. Westinghouse contends that the loss of CCW to the RCPs will not result in an instantaneous seizure of a single pump and, further, that instantaneous seizure of 2 pumps simultaneously is not a credible ultimate consequence.

Instead, it is Westinghouse's technical opinion that a more realistic ultimate consequence will be an abbreviated coastdown. If a limiting condition of the babbitt metal is considered, an increasing coefficient of friction, as well as an increasing retarding torque is expected. However, in view of the large rotational inertia of the pump/motor assembly, Westinghouse maintains that an instantaneous seizure will not result.

Because an initial seizure is not expected, it is not possible to define a precise point in time at which a sequential seizure would be anticipated. Therefore, for the purpose of defining the time expected between sequential seizures, the following discussion will be presented in terms of sequential occurrences of reaching a "high" bearing temperature. The upper thrust bearing exhibits the limiting temperature; therefore, an upper thrust bearing temperature of 240°F has been chosen arbitrarily as the "high" temperature. It should be noted that the use of this value does not imply pump seizure at this temperature.

Variables affecting the steady state operating temperature of the bearings include the following:

a. Surface finish of the bearing and runner.

b. Bearing (and oil pumping mechanism) clearances.

c. Inlet temperature of water to heat exchanger (oil cooler).

d. Condition of oil-to-water heat exchanger (oil cooler) (i.e., extent of fouling).

e. Condition of oil.

f. Amount of oil in oil pot.

g. Oil temperature.

These variables would be expected to interact concurrently in a manner which individualizes the performance of the bearings during actual steady state plant operation.

In order to quantify the resultant variation in performance, Westinghouse has collected data from an operating plant. This data demonstrates that the upper thrust bearings operate at different steady state temperatures (i.e., 128°F, 132°F, 135°F, and 145°F). It should be noted that this data was collected from a 4 loop plant; the following evaluation based on this data is applicable to a 3 loop plant (i.e., Virgil C. Summer).

Using these actual steady state operating values (A-128°F, B-132°F, C-135°F, D-145°F) and assuming a conservative 5°F/minute linear heatup rate after a loss of CCW, sequential occurrences of reaching the high bearing temperature could be expected at the time intervals tabulated below. (See Figures 5.5-20 and 5.5-21.)

Sequential Motors	Δ Operating Temperature(°F)	Time Interval(minutes)
A and B	4	0.8
B and C	3	0.6
C and D	10	2.0
A and C	7	1.4
B and D	13	2.6
A and D	17	3.4

To summarize, 2 bearings sequentially reaching a temperature of 240°F could be expected at a minimum time interval of 0.6 minutes and at a maximum time interval of 3.4 minutes.

Westinghouse has obtained motor bearing heatup data. These test data show actual values of bearing temperatures following a loss of CCW. The test data presented in Figure 5.5-22 will be examined relative to the above discussion. The test runs, which were performed at different times using different motors, demonstrate similar heatup rates; this fact supports the assumption of identical linear heatup rates made in the previous discussion. In addition, the average heatup rates evidenced in the test data are less than 3.3°F/minute, which substantiates the use of 5°F/minute as a conservative value. The actual test data, although limited, is supportive of the assumptions posed in defining the time intervals tabulated above.

In conclusion, Westinghouse contends that a single or multiple pump seizures as the result of a loss of CCW to the RCPs is not a credible event. However, in our judgment and based on the above discussion, 2 RCP motor upper thrust bearings could sequentially reach a "high" bearing temperature of 240°F at a minimum time interval of 0.6 minutes (or approximately 40 seconds).

Section 15.4.4 of the Virgil C. Summer FSAR presents the analysis of a single RCP locked rotor. It should be pointed out that the Section 15 analysis assumes an instantaneous seizure of a reactor coolant pump rotor on a non-mechanistic basis. Westinghouse contends that a postulated mechanistic instantaneous seizure of a pump rotor due to a loss of CCW to the RCP will not occur and is not a credible event.

However, in response to the NRC request, the results of a second non-mechanistic instantaneous seizure occurring at 40 seconds after a first non-mechanistic instantaneous seizure have been evaluated. Although a Section 15 approach was utilized to evaluate this situation, Westinghouse does not recognize a postulated mechanistic instantaneous locked rotor as a credible consequence of the loss of CCW to the RCPs.

Assuming that a second pump seizure occurs 40 seconds after a first pump seizure, no noticeable change is seen in the Reactor Coolant System pressure and the clad temperature transients. Furthermore, even if the time interval between the sequential seizures is reduced to 10 seconds, no noticeable change is seen in the Reactor Coolant System pressure and the clad temperature transients.

The hypothetical seizure of 1 RCP results in a low flow reactor trip approximately 1 second after the initiation of the event. As a result of the fast reactor trip and the consequential decrease in core heat flux, the Reactor Coolant System pressure and the clad temperature reach the peak values at about 2.5 seconds and then start to decrease.

Because the core has been shut down, at 40 seconds - or even 10 seconds - after a pump seizure, the Reactor Coolant System pressure and the clad temperature transients have decreased to a point at which a second pump seizure results in no noticeable change in the transients.

Operating procedures are provided for a loss of component cooling water and seal injection to the reactor coolant pumps and/or motors. Included in these operating procedures is the provision to trip the reactor if component cooling water flow, as indicated by the instrumentation discussed previously, is lost to the reactor coolant pump motors and cannot be restored within 10 minutes. The reactor coolant pumps will also be tripped following the reactor trip.

This section provides a description of testing performed and the test results which constitute the basis of reactor coolant pump operation for 10 minutes without CCW with no resultant damage. Two (2) RCP motors have been tested with interrupted CCW flow; these tests were conducted at the Westinghouse Electro Mechanical Division. In both cases, the reactor coolant pumps were operated to achieve "hot" (2230 psia, 552°F) equilibrium conditions. After the bearing temperatures stabilized, the cooling water flow to the upper and lower motor bearing oil coolers was terminated and bearing (upper thrust, lower thrust, upper guide, and lower guide) temperatures were monitored. A bearing metal temperature of 185°F was established as the maximum test temperature. When that temperature was reached, the cooling water flow was restored.

In both tests, the upper thrust bearing exhibited the limiting temperatures. Figure 5.5-21 shows the upper thrust bearing temperature versus time. In both cases, 185°F was reached in approximately 10 minutes.

The maximum test temperature of 185°F is also the suggested alarm setpoint temperature and the suggested trip temperature is 195°F. It should be noted that the melting point of the babbitt bearing metal exceeds 400°F.

The information presented above constitutes the basis of the RCP qualification for 10 minute operation without CCW with no resultant damage.

5.5.1.4 Tests and Inspections

The reactor coolant pumps can be inspected in accordance with the ASME Code, Section XI, for Inservice Inspection of Nuclear Reactor Coolant Systems.

Any full penetration welds in the pressure boundary are prepared with a smooth surface transition between weld metal and parent metal for radiographic inspection. However, the pump casing is cast in one piece, eliminating welds in the casing.

Support feet are cast integral with the casing to eliminate a weld region.

The design enables disassembly and removal of the pump internals for usual access to the internal surface of the pump casing.

The reactor coolant pump quality assurance program is given in Table 5.5-2.

5.5.2 STEAM GENERATORS

5.5.2.1 Design Bases

The replacement steam generators installed in the V. C. Summer plant are Westinghouse Delta-75, feedring type steam generators. To facilitate replacement in the plant, the geometric characteristics of the Delta-75 steam generators are identical to the original D3 steam generators. Additionally, the Delta-75 steam generators must provide an equal or better level of performance than the original steam generators. The Delta-75 steam generators are expected to supply steam at about 1.3% higher than the original D3 steam generators at a plant thermal rating of 2787 MWt. At the Engineered Safeguards Power Rating of 2912 MWt, the Delta-75 steam generators are expected to deliver steam at about 2.0% above that projected for the original D3 steam generators.

The steam generators are designed and analyzed according to the requirements of Section III of the ASME Code, 1971 Edition through Summer 1971 Addenda, and constructed according to the 1986 Edition. To minimize the impact on steam generator supports and seismic considerations, the total weight of each empty Delta-75 steam generator is ≤ 360 tons.

Steam generator design data is given in Table 5.3-3. The design sustains transient conditions given in Section 5.2.1. Although the required secondary side ASME classification is Class 2, Class 1 requirements are applied for all pressure retaining portions of the steam generator. Assurance of adequate fracture toughness of all pressure boundary materials, is, therefore, as described in Section 5.2.4 and complies with Article NB-2300 of Section III of the ASME Code. Rupture of a steam generator tube is discussed in Section 15.4.3.

The internal moisture separation equipment is designed to ensure that moisture carryover does not exceed 0.10% by weight under the following conditions:

1. Steady-state operation up to 100% of full load steam flow, with water at the normal operating level.

2. Loading or unloading at a rate of 5% of full load power steam flow per minute in the range from 15% to 100% of full load steam flow.

3. A step load change of 10% of full power in the range from 15% to 100% full load steam flow.

The water chemistry in the reactor side is selected to provide the necessary boron content for reactivity control and to minimize corrosion of RCS surfaces. The water chemistry of the steam side and its effectiveness in corrosion control is discussed in Section 10.3.5.

5.5.2.2 Design Description.

The original VCSNS steam generators were Model D3 type, which employed the use of an integral preheater at the feedwater inlet. The original steam generators were replaced during the Refuel 8 outage, approximately September, 1994, with Westinghouse Delta-75 steam generators. The Delta-75 steam generator has the same profile dimensions as the original D3 steam generator. Like the D3 steam generator, the Delta-75 is a vertical shell and U-tube evaporator with integral moisture separation equipment. The Delta-75 steam generator is shown in Figure 5.5-3. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tubesheet. Manways are provided for access to both sides of the divided head. The feedwater inlet on the original D3 steam generators was located several feet above the top surface of the tubesheet, on the cold leg or return side of the U-tube bundle. The incoming feedwater was forced through a back and forth path within the preheater by several staggered tube support plates. This path allowed the feedwater temperature to be rapidly heated almost to saturation temperature before entering the boiler section. In the Delta-75, the feedwater inlet is located at approximately 2/3 of the steam generator height and is distributed equally around the circumference of the steam generator shell. Feedwater enters the tube bundle by flowing downward between the steam generator

external shell and inner wrapper barrel. An open area at the bottom of the wrapper barrel permits the feedwater to enter the tube bundle. The simplified design of the Delta-75 steam generator supports a more simplified steam generator operating procedure and eliminates some of the low power operating restrictions applied to the D3 steam generators. The total steam generator heat transfer surface area is increased in the Delta-75 to account for the thermal performance characteristics of the preheater. Steam is generated and flows upward through the moisture separators and through the flow restrictor outlet nozzle at the top of the steam drum. The Delta-75 utilizes high efficiency centrifugal steam separators, which remove most of the entrained water. Chevron dryers are employed to increase the steam quality to a minimum of 99.90% (0.10% moisture).

The steam generator channel head and tubesheet are protected from the primary water by applying an autogenous weld deposited stainless steel cladding to the primary surfaces of the channel head and Inconel to the tubesheet. The cladding surface is 02-01 machined to a smooth condition and electropolished thereby reducing the collection of radioactive contamination inside the steam generators during refueling and maintenance periods.

Steam generator materials of construction are listed in Table 5.2-8. All materials are selected and fabricated in accordance with the requirements of the ASME Code, Section III, and Code Case N-20-3. The steam generator tubing material is Alloy 690. Alloy 690 material represents the state of the art technology for heat transfer tubing. The tubing receives a heat treatment process after forming which results in a grain boundary structure which has been shown by test to be exceptionally resistive to primary water stress corrosion cracking (PWSCC), which initiates tube degradation in the expansion region within the tubesheet area and has resulted in the majority of tube plugging in the original D3 steam generators at V. C. Summer. The tube expansion process used for the replacement steam generator is not expected to result in similar strain levels or patterns as observed in the original D3 steam generators. Alloy 690 tubing material is also exceptionally resistive to outer diameter stress corrosion cracking (ODSCC), which has affected the secondary side of the tubes in operating plants at the tube support plate intersections.

Ferritic material in the primary side of the steam generators includes the following: channel head casting, tubesheet, primary nozzles, manway covers, and manway studs and nuts. Fracture toughness data for the steam generator bolting materials is shown in Table 5.2-22. Fracture toughness data for the channel head, tubesheet, and associated weldments will be consistent with ASME Code. As-built data generally shows the test data to far exceed the ASME Code minimum values. The materials of construction and general design of the steam generator shell of the Delta-75 have resulted in a reduction in the number of pressure boundary welds required for assembly of the shell.

5.5.2.3 Design Evaluation

5.5.2.3.1 Forced Convection

The effective heat transfer coefficient is determined by the physical characteristics of the Delta-75 steam generator and the fluid conditions in the primary and secondary systems for the nominal 100% design case. It includes a conservative allowance for fouling and uncertainty. A designed heat transfer area is provided to permit the achievability of the full-design heat-removal rate. Although margin for tube fouling is available, operating experience to date has not indicated that steam generator thermal performance decreases over a long term period. Adequate tube surface area is selected to ensure that the full design heat removal rate is achieved.

5.5.2.3.2 Natural Circulation Flow

The steam generators, which provide a heat sink, are at a higher elevation than the reactor core, which is the heat source. Thus natural circulation is assured for the removal of decay heat.

5.5.2.3.3 Tube and Tubesheet Stress Analyses

Tube and tubesheet stress analyses of the steam generator, which are discussed in Section 5.2, confirm that the steam generator tubesheet will withstand the loading caused by a loss of reactor coolant (LOCA). Routine inservice inspections of the steam generator tubing will assess tubing integrity and adequacy for continued operation based upon the 40% depth of indication limit specified by Section XI of ASME Code. Tube degradation existing at this level will maintain integrity consistent with the recommendations of Regulatory Guide 1.121 during the transient events identified in Section 5.2.

5.5.2.3.4 Corrosion

The Delta-75 steam generator utilizes thermally treated Alloy 690 tube material. This material is used in the replacement steam generators at North Anna Unit 1 and Indian Point Unit 3. Thermally treated Alloy 690 tubing is also used in a number of plants, which include the Turkey Point replacements, Surry replacements, D. C. Cook Unit 2 replacements, and Prairie Island replacements, Wolf Creek, Seabrook and Millstone Unit 3.

To date, the only form of tube degradation which has been identified in Model F-type units is fretting wear between the tube and steam generator anti-vibration bars (AVB). Only a small number of tubes (expected to be less than 1% total over the life of the steam generators) would be subject to this phenomena. The AVB design in the Delta-75 steam generators maintain smaller, more tightly controlled tube-to-AVB gaps, increased tube-to-AVB contact area, and utilize more AVB's than do the Model F steam generators. These factors should greatly lower the potential for tube fretting at AVB intersections in the Delta-75 steam generators.

5.5.2.3.5 Compatibility of Steam Generator Tubing with Primary and Secondary Coolant

As previously discussed, the Alloy 690 tubing installed in the Delta-75 steam generators at V. C. Summer have been shown by test (and operating experience) to be exceptionally resistant to both PWSCC and ODSCC. The Delta-75 Alloy 690 tubing specification is in accordance with the requirements of ASME Code Case N-20-3, ASME Code Section III, ASME Code Section II Specification SB-163, and meets the EPRI Guidelines for Procurement of Alloy 690 steam generator tubing, NP-6743-L, February 1991.

The steam generator tube expansion process employed in the tubesheet region results in reduced residual stress levels compared to the original D3 steam generators. This reduced residual stress will lower the potential for PWSCC in this region. Additionally, no evidence of PWSCC has been identified in either the Model F type steam generators operating today, or the replacement steam generators manufactured by Westinghouse.

The potential for ODSCC to develop in the Delta-75 steam generators is also greatly reduced. The axial flow paths around the steam generator tubes provided by the trifoil support design significantly reduces crevice area and contaminant hideout potential. The stainless steel tube support plate material does not represent a potential for magnetite generation or general corrosion product buildup. Also, the most recent revision of the EPRI Secondary Water Chemistry Guidelines have incorporated a sodium-chloride molar ratio control philosophy, which will help to maintain an as neutral as possible crevice chemistry within the entire steam generator secondary side.

5.5.2.3.6 Flow Induced Vibration

The potential for tube wall degradation attributable to mechanical or flow induced excitation is exceptionally low in a feedring type steam generator design. Flow induced vibration at the tube support plate intersections has not been observed in Model F type steam generators.

5.5.2.3.7 Tube Denting

Localized steam generator tube diameter reductions were first discovered during the April 1975 steam generator inspection at the Surry Unit 2 plant. This discovery was evidenced by eddy current signals, resembling those produced by scanning dents, and by difficulty in passing the standard 0.700 inch diameter eddy current probe through the tubes at their intersections with the support plates. Subsequent to the initial finding, steam generator inspections at other operating plants revealed essentially identical results.

The phenomena of tube denting is attributed to a localized buildup of corrosion products (due to magnetite generation in the support plate crevice) in the tube-to-tube support plate crevice, which eventually deforms the tube wall as the corrosion products continue to build. The implementation of All Volatile Treatment (AVT) chemistry on the secondary side of the steam generator has minimized the potential of denting. Additionally, the Delta-75 steam generators at the V. C. Summer plant employ stainless steel tube support plates, which should preclude the formation of corrosion products within the crevice. The Delta-75 tube support plates employ a limited contact trifoil hole shape which permits axial flow around the tube, which also minimizes the buildup of corrosion products in this area. The original D3 steam generators utilized carbon steel support plates with drilled tube holes of a diameter approximately 0.013 inch larger in diameter than the tubes, which can be filled with corrosion products.

5.5.2.4 Test and Inspections

The steam generator quality assurance program is given in Table 5.5-4. During manufacture, cleaning is performed on the primary and secondary sides of the steam generator in accordance with written procedures which follow the guidance of Regulatory Guide 1.37 and other industry standards, such as those developed by the American Society of Testing and Materials (ASTM). Radiographic inspections, liquid penetrant inspections, magnetic particle inspections, ultrasonic inspections, and the implementation of the associated acceptance criteria for each are instituted according to the requirements of the ASME Code.

The steam generators are designed to permit inservice inspections in accordance with the requirements of Section XI to the ASME Code, and steam generator tubing eddy current examination consistent with the requirements of V.C. Summer Technical Specifications 3/4.4.5, “Steam Generator Tube Integrity,” and 6.8.4.k, “Steam Generator Program.” RN 99-110 RN 10-016

When required, tests, using a Lithium or radioactive Sodium tracer, are conducted to properly account for moisture carryover in the plants calorimetric and precision RCS flow measurement. RN 99-110

5.5.3 REACTOR COOLANT PIPING

5.5.3.1 Design Bases

The RCS piping is designed and fabricated to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions. Stresses are maintained within the limits of Section III of the ASME Nuclear Power Plant Components Code. Code and material requirements are provided in Section 5.2.

Materials of construction are specified to minimize corrosion/erosion and ensure compatibility with the operating environment.

The piping in the RCS is Safety Class 1 and is designed and fabricated in accordance with the ASME Code, Section III, Class 1 requirements.

Stainless steel pipe conforms to ANSI B36.19 for sizes 1/2 inch through 12 inches and wall thickness Schedules 40S through 80S. Stainless steel pipe outside of the scope of ANSI B36.19 conforms to ANSI B36.10.

The minimum wall thicknesses of the loop pipe and fittings are not less than that calculated using the ASME Code, Section III, Class 1 formula of Paragraph NB-3641.1 (3) with an allowable stress value of 17,550 psi. The pipe wall thickness for the pressurizer surge line is Schedule 160. The minimum pipe bend radius is five nominal pipe diameters; ovality does not exceed 6%.

All butt welds, branch connection nozzle welds, and boss welds are of a full penetration design.

Processing and minimization of sensitization are discussed in Sections 5.2.3 and 5.2.5.

Flanges conform to ANSI B16.5.

Socket weld fittings and socket joints conform to ANSI B16.11.

Inservice inspection is discussed in Section 5.2.8.

5.5.3.2 Design Description

Principal design data for the reactor coolant piping is given in Table 5.5-5.

The piping is seamless forged; the fittings are cast without longitudinal welds and electroslag welds. Piping and fittings comply with the requirements of the ASME Code, Section II, Parts A and C, Section III, and Section IX.

The RCS piping is specified in the smallest sizes consistent with system requirements. This design philosophy results in the reactor inlet and outlet piping diameters given in Table 5.5-5. The line between the steam generator and the pump suction is larger to reduce pressure drop and improve flow conditions to the pump suction.

The reactor coolant piping and fittings which make up the loops are austenitic stainless steel. There is no electroslag welding on these components. All smaller piping which comprise part of the RCS such as the pressurizer surge line, spray and relief line, loop drains, and connecting lines to other systems are also austenitic stainless steel. The nitrogen supply line for the pressurizer relief tank is carbon steel. All joints and connections are welded, except for the pressurizer code safety valves, where flanged joints are used. Thermal sleeves are installed at some points in the system where high thermal stresses could develop due to rapid changes in fluid temperature during normal operational transients. These points include: RN 03-051

1. The pressurizer surge line connection at the pressurizer.

2. Pressurizer spray line connection at the pressurizer.

Piping connections from auxiliary systems are made above the horizontal centerline of the reactor coolant piping, with the exception of:

1. Residual heat removal pump suction lines, which are 45° down from the horizontal centerline. This enables the water level in the RCS to be lowered in the reactor coolant pipe while continuing to operate the Residual Heat Removal System, should this be required for maintenance.

2. Loop drain lines and the connection for temporary level measurement of water in the RCS during refueling and maintenance operation.

3. The differential pressure taps for flow measurement, which are downstream of the steam generators on the first 90° elbow. The tap arrangement is discussed in Section 5.6.

4. The pressurizer surge line, which is attached at the horizontal centerline.

5. The safety injection connections to the hot leg, for which inservice inspection requirements and space limitations dictate location on the horizontal centerline.

Penetrations into the coolant flow path are limited to the following:

1. The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force.

2. The reactor coolant sample system taps protrude into the main stream to obtain a representative sample of the reactor coolant.

3. The resistance temperature detector (RTD) hot leg scoops extend into the reactor coolant at locations 120° apart in the cross sectional plane. In the original design, these scoops collected a representative temperature sample for the RTD manifold. In the current design, they provide a convenient location for the narrow range thermowell mounted RTDs.

4. The wide range temperature detectors and the cold leg fast response temperature detectors are located in resistance temperature detector wells that extend into the reactor coolant pipes.

The RCS piping includes those sections of piping interconnecting the reactor vessel, steam generator, and reactor coolant pump. It also includes the following:

1. Charging line and alternate charging line from the system isolation valve up to the branch connections on the reactor coolant loop.

2. Letdown line and excess letdown line from the branch connections on the reactor coolant loop to the system isolation valve.

3. Pressurizer spray lines from the reactor coolant cold legs to the spray nozzle on the pressurizer vessel.

4. Residual heat removal lines to or from the reactor coolant loops up to the designated check valve or isolation valve.

5. Safety injection lines from the designated check valve to the reactor coolant loops.

6. Accumulator lines from the designated check valve to the reactor coolant loops.

7. Loop fill, loop drain, sample (1), and instrument (1) lines to or from the designated isolation valve to or from the reactor coolant loops.

8. Pressurizer surge line from one reactor coolant loop hot leg to the pressurizer vessel inlet nozzle.

9. Resistance temperature detector scoop element, pressurizer spray scoop, sample connection (1) with scoop, reactor coolant temperature element installation boss, and the temperature element well itself.

10. All branch connection nozzles attached to reactor coolant loops.

11. Pressure relief lines from nozzles on top of the pressurizer vessel up to and through the power operated pressurizer relief valves and pressurizer safety valves.

12. Seal injection water and labyrinth differential pressure lines to or from the reactor coolant pump inside reactor building.

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(1) Liquid filled lines with a 3/8 inch flow restricting orifice qualify as Safety Class 2a; in the event of a break in one of these Safety Class 2a lines, the normal makeup system is capable of providing makeup flow while maintaining pressurizer water level. Instrument impulse lines connected to the pressurizer steam space above the water level have been upgraded to Safety Class 1, since failure of these lines could cause automatic operation of the ECCS systems. RN 07-012

13. Auxiliary spray line from the isolation valve to the pressurizer spray line header.

14. Sample lines[(1)] from pressurizer to the isolation valve.

Details of the materials of construction and codes used in the fabrication of reactor coolant piping and fittings are discussed in Section 5.2.

5.5.3.3 Design Evaluation

Piping load and stress evaluation for normal operating loads, seismic loads, blowdown loads, and combined normal, blowdown, and seismic loads are discussed in Section 5.2.1.

5.5.3.3.1 Material Corrosion/Erosion Evaluation

The water chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the specifications.

The design and construction are in compliance with the ASME Code, Section XI. Pursuant to this, all pressure containing welds out to the second valve that delineates the RCS boundary are available for examination with removable insulation.

Components constructed with stainless steel will operate satisfactorily under normal plant chemistry conditions in pressurized water reactor systems because chlorides, fluorides, and particularly oxygen are controlled to very low levels (see Section 5.2.3).

Periodic analysis of the coolant chemical composition is performed to monitor the adherence of the system to desired reactor coolant water quality listed in Table 5.2-10. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System and process sampling system which are described in Sections 9.3.4 and 9.3.2, respectively.

5.5.3.3.2 Sensitized Stainless Steel

Sensitized stainless steel is discussed in Sections 5.2.3 and 5.2.5.

5.5.3.3.3 Contaminant Control

Contamination of stainless steel and Inconel by copper, low melting temperature alloys, mercury, and lead is prohibited. Only selected lubricants, which are not deleterious to stainless steel, are used.

Prior to application of thermal insulation, the austenitic stainless steel surfaces are cleaned and analyzed to a halogen limit of 0.0015 mg Cl/dm^2 and 0.0015 F/dm^2.

5.5.3.3.4 Alloy 600 Dissimilar Metal Welds

The RCS Piping includes Alloy 600 dissimilar metal welds at the Reactor Vessel Nozzles, Steam Generator Nozzles, Pressurizer Surge Line Nozzle, Pressurizer Safety Valves, Power Operated Relief Valves, and Pressurizer Spray Nozzle.

A thru-wall leak was identified on the Reactor Vessel "A" Loop Hot Leg Nozzle to Pipe Weld during Refuel 12. The weld was completely removed and a pipe spool piece was welded in place. The root cause was determined to be Primary Water Stress Corrosion Cracking (PWSCC). The Reactor Vessel Nozzle to Pipe Weld was made using Alloy 690. Alloy 690 is a material with more resistance to PWSCC. RN 04-031

Due to the high susceptibility to PWSCC in the Reactor Vessel Hot Leg Nozzle to Pipe Welds, mitigative measures were taken during Refuel 13. The Mechanical Stress Improvement Process (MSIP) is a remediation process for pipe-to-pipe and pipe-to-nozzle welds that are susceptible to stress corrosion cracking from welding induced residual stresses. MSIP equipment applies a narrow permanent radial deformation adjacent to a pipe weld that redistributes the residual stresses in the weld region producing compressive stress at the inside diameter of the weld joint. This process removes one of the three factors (material susceptibility, environment, and high residual tensile stresses) required to create PWSCC in RCS components. MSIP was applied to the Reactor Vessel "B" and "C" Loop Hot Leg Nozzle to Pipe Welds.

The Pressurizer Nozzle Alloy 600 dissimilar metal welds were mitigated during Refuel 17 by the application of a full structural weld overlay. This weld overlay was applied to each of the nozzles on the pressurizer utilizing alloy 52M filler metal which is high in chromium and more resistant to PWSCC (consistent with alloy 690). This pre-emptive mitigation provided a full structural overlay to each of the nozzles and provides an ultrasonic NDE inspectable geometry which meets the ASME volumetric inspection requirements. The pre-weld overlay geometry did not meet the volumetric inspection requirements of the ASME XI code. Application of the weld overlay also provides a beneficial compressive stress to the underlying pipe, safe end, nozzle, and welds. RN 08-022

5.5.3.4 <u>Tests and Inspections</u>

The RCS piping quality assurance program is given in Table 5.5-6.

Volumetric examination is performed throughout 100% of the wall volume of each pipe and fitting in accordance with the applicable requirements of Section III of the ASME Code for all pipe 27-1/2 inches and larger. All unacceptable defects are eliminated in accordance with the requirements of the same section of the code.

A liquid penetrant examination is performed on both the entire outside and inside surfaces of each finished fitting in accordance with the criteria of the ASME Code, Section III. Acceptance standards are in accordance with the applicable requirements of the ASME Code, Section III.

The pressurizer surge line conforms to SA-376 Grade 304, 304N, or 316 with supplementary requirements S2 (transverse tension tests) and S6 (ultrasonic test). The S2 requirement applies to each length of pipe. The S6 requirement applies to 100% of the piping wall volume.

The end of pipe sections, branch ends, and fittings are machined back to provide a smooth weld transition adjacent to the weld path. All butt welds are ground smooth to permit inservice inspection in accordance with the ASME Code, Section XI. There is one pipe-to-pipe weld ("A" Hot Leg). RN 04-031

5.5.4 STEAM OUTLET FLOW RESTRICTOR (STEAM GENERATOR)

5.5.4.1 Design Bases

Each steam generator is provided with a flow restrictor having several small diameter venturi-type throats. The flow restrictors are designed to limit steam flowrate consequent to the unlikely event of a steam line rupture; thereby, reducing the cooldown rate of the primary system and limiting stresses of internal steam generator components.

The flow restrictor is designed to minimize unrecovered pressure loss coincident with limiting accident flowrate to an acceptable value.

Although it is not considered to be part of the pressure vessel boundary, the restrictor is constructed of material specified in Section III of the ASME Code and is Seismic Category 1.

5.5.4.2 Design Description

The flow restrictor is an assembly of 7 smaller nozzles installed within the steam outlet nozzle of the steam generator. Upper and lower internal discs support an outer circle of 6 nozzles and 1 central nozzle. The discs and outer cylinder are made of steel plates and pipe and the venturi nozzles are Inconel. The flow restrictor assembly is welded into the main steam generator outlet by a circumferential weld between the rim of the outer cylinder and the inside surface of the outlet nozzle.

5.5.4.3 Design Evaluation

The equivalent throat diameter of the steam generator outlet is 16 inches and the resultant pressure drop through the restrictors at 100% steam flow is approximately 3.4 psig. The steam side weld to the outlet nozzle is in compliance with manufacturing and quality control requirements of the ASME Code, Section III, and is Seismic Category 1.

5.5.4.4 Tests and Inspections

The restrictors are not a part of the steam system boundary. No tests or inspections of the restrictors are anticipated beyond those performed in the fabricator’s shop.

5.5.5 MAIN STEAM LINE ISOLATION SYSTEM

The main steam line isolation system is discussed in Sections 10.3.2 and 10.3.3. The discussion presented in the referenced sections include information on measures taken to reduce potential leakage of radioactivity to the environment in the event of a main steam line break.

5.5.6 REACTOR CORE ISOLATION COOLING SYSTEM

Not applicable to pressurized water reactors.

5.5.7 RESIDUAL HEAT REMOVAL SYSTEM

The Residual Heat Removal System (RHRS) transfers heat from the RCS to the Component Cooling Water System (CCWS) to reduce the temperature of the reactor coolant to the cold shutdown temperature at a controlled rate during the second part of normal plant cooldown and maintains this temperature until the plant is started up again.

During the first phase of cooldown, the temperature of the RCS is reduced by transferring heat from the RCS to the steam and power conversion system through the steam generators.

Parts of the RHRS also serve as parts of the Emergency Core Cooling System (ECCS) during the injection and recirculation phases of a loss of coolant accident (see Section 6.3).

The RHRS is also used to transfer refueling water between the refueling cavity and the refueling water storage tank at the beginning and end of the refueling operations.

5.5.7.1 Design Bases

RHRS design parameters are listed in Table 5.5-7.

The RHRS is placed in operation approximately 4 hours after reactor shutdown when the temperature and pressure of the RCS are approximately 350°F and less than 425 psig, respectively. The RHRS heat exchanger design heat load is based upon transferring 1/2 of the decay heat load at 20 hours after reactor shutdown (for the original RTP of 2775 MWt) with 5600 gpm CCWS flow (CCWS temperature is limited to 120°F maximum and will decrease towards 105°F as the cooldown proceeds) and 3750 gpm RHRS flow. With the current licensed RTP of 2900 MWt each RHRS heat exchanger is capable of transferring 1/2 of the decay heat load at 24 hours after reactor shutdown with the same flowrates and CCWS temperature given above. 02-01

At the original RTP of 2775 MWt, the coupled RHRS/CCWS/SWS system is capable of cooling the RCS from 350°F beginning at 4 hours after reactor shutdown from an extended run at full power to 200°F within 4 hours after the start of the RHRS cooldown 02-01

and to 140°F within 16 hours after the start of the RHRS cooldown. With the current licensed RTP of 2900 MWt the coupled RHRS/CCW/SWS system is still capable of cooling the RCS from 350°F beginning at 4 hours after reactor shutdown from an extended run at full power to 200°F within 4 hours after the start of the RHRS cooldown and to 140°F within 20 hours after the start of the RHRS cooldown. The RHRS heat load during the transient includes the heat from 1 RCP (RCP off after RCS reaches 160°F), core decay heat, and the sensible heat of the RCS metal and fluid; also, RCS cooldown rate is limited to 50°F/hr. 02-01

5.5.7.1.1 Compliance with BTP RSB 5-1

The safe shutdown design basis of the Virgil C. Summer Nuclear Station is hot standby. Under abnormal conditions, the plant is designed to remain in a safe hot standby condition until (a) normal systems can be restored to permit either return to power operation or cooldown to cold shutdown conditions, or (b) sufficient systems capability can be restored (depending on plant condition) to permit cooldown to cold shutdown conditions under abnormal plant conditions. This design basis is considered to constitute a safe design.

BTP RSB 5-1 establishes specific design requirements that address the various system functions that are required to achieve and maintain a safe hot standby and cold shutdown conditions. BTP RSB 5-1 requires plants with construction permits docketed after January 1, 1978, to comply in full with the design requirements of the BTP. Plants with construction permits docketed prior to January 1, 1978, (including Virgil C. Summer Nuclear Station) are required to address the BTP technical requirements and demonstrate partial compliance.

The following is a discussion of the Virgil C. Summer Nuclear Station compliance with the technical requirements of BTP RSB 5-1. This discussion demonstrates that under the postulated condition of BTP RSB 5-1 the Virgil C. Summer Nuclear Station can be maintained in a safe hot standby condition and taken to Residual Heat Removal System (RHRS) initiation conditions within approximately 36 hours, including credit for limited manual actions outside the control room to operate and/or repair a limited number of components that are not safety-grade or single failure proof.

Itemized below are the technical requirements of BTP RSB 5-1 followed by a general discussion of the Virgil C. Summer Nuclear Station compliance. The technical requirements section is then followed by more detailed discussion in sections entitled Cold Shutdown Scenario, Single Failure Evaluation and Natural Circulation.

1. Provide safety-grade steam generator dump valves, operators, air, and power supplies which meet the single failure criterion.

   One (1) safety grade steam generator power operated relief valve is provided for each of the 3 steam generators. Safety grade remote operators and power supplies are not required since hot standby can be achieved and maintained using

the safety grade steam generator safety valves. The steam generator power operated relief valves are provided with handwheels and can be operated locally to permit plant cooldown. See the cold shutdown scenario and single failure evaluation provided below (Part II, Removal of Residual Heat).

2. Provide the capability to cooldown to cold shutdown in a reasonable period of time assuming the most limiting single failure and loss of offsite power or show that manual actions inside or outside containment or return to hot standby until the manual actions or maintenance can be performed to correct the failure provides an acceptable alternative. 02-01

   If a condition occurred requiring cold shutdown of the plant, design features permit the maintenance of a hot standby condition for an indefinite period of time. The plant is capable of being cooled via natural convection and reaching RHRS initiating conditions including the time required to perform any manual actions.

3. Provide the capability to depressurize the Reactor Coolant System with only safety-grade systems assuming a single failure and loss of offsite power or show that manual actions inside or outside containment or remaining at hot standby until manual actions or repairs are complete provides an acceptable alternative.

   The plant can be maintained in a safe hot standby condition while any required manual actions are taken. See the cold shutdown scenario and single failure evaluation provided below (Part IV, Depressurization).

4. Provide the capability for borating with only safety grade systems assuming a single failure and loss of offsite power, or show that manual actions inside or outside containment or remaining at hot standby until manual actions or repairs are completed provides an acceptable alternative.

   The plant can be maintained in a safe hot standby condition while any required manual actions are taken. See the cold shutdown scenario and single failure evaluation provided below (Part III, Boration and Makeup).

5. Provide the system and component design features necessary for the prototype testing of both the mixing of the added borated water and the cooldown under natural circulation conditions with and without a single failure of a steam generator atmospheric dump valve.

The Virgil C. Summer Nuclear Station's ability to address boron mixing and cooldown test in natural circulation was originally to be demonstrated through referencing testing which was to be conducted at the Diablo Canyon nuclear plant. Since that time, other plants have demonstrated the following results.

During natural circulation testing at Farley 2, North Anna II, Salem II and Sequoyah I, the following observations were made:

a. All plants (3 or 4 loop) exhibited very similar RCS behavior indicating similar RCS characteristics.

b. By directly comparing North Anna and Farley natural circulation test results, these plants effectively exhibited identical behavior in terms of: time to stability following the reactor coolant pump trip; core flow distribution; core power distribution; and flow rates as indicated by the loop delta T's.

Results of the boron mixing and cooldown tests at Sequoyah, Salem, and North Anna indicate that:

a. The results of the North Anna test conclusively demonstrated the ability of the RCS to mix boron under natural circulation conditions as was also demonstrated by Salem and Sequoyah.

b. The tests performed at Salem and North Anna were performed at the end of cycle with a decay heat level on the order of 1% of the rated power where the Sequoyah test was done with nuclear heat of approximately 2% of the rated power. These power levels (and flow) are nominally lower than those expected following an emergency condition since the operator would initiate a boration quickly following the reactor trip to assure shutdown margin.

c. The capability of the RCS to perform a natural circulation cooldown at a rate of approximately 50°F/hr was demonstrated by all of these tests. This is at a higher rate than the Westinghouse emergency procedures (25°F/hr).

d. All plants demonstrated that no core temperature distribution anomalies were induced by the addition of boric acid into the RCS or the cooldown process while using natural circulation.

Since the tests at all plants successfully and conclusively demonstrated the capability of the plant to mix boric acid and cool down under natural circulation conditions, there would be no benefit from the Virgil C. Summer Nuclear Station performing this test.

a. Commit to providing specific procedures for cooling down using natural circulation and submit a summary of these procedures.

   Specific procedures for cooling down using natural circulation are provided in the Virgil C. Summer Nuclear Station "EOPs" which includes natural circulation cooldown with boron mixing. A summary of the procedures is provided in the cold shutdown scenario and single failure evaluation provided below. 00-01

b. Provide a seismic Category 1 AFW supply for at least 4 hours at Hot Shutdown plus cooldown to the RHR system cut-in based on the longest time (for only onsite or offsite power and assuming the worst single failure), or show that an adequate alternate Seismic Category 1 source will be available.

   Sufficient emergency feedwater is provided in the Seismic Category 1 condensate storage tank to permit 2 hours of operation at hot standby plus cooldown to RHR initiation conditions. In addition, a long term source of Emergency Feedwater is provided by a connection to the Seismic Category 1 Service Water System. See the cold shutdown scenario and single failure evaluation provided below (Part II, Removal of Residual Heat).

c. Provide for collection and containment of RHR pressure relief or show that adequate alternative methods of disposing of discharge are available.

   The RHR relief valves discharge to the pressurizer relief tank located inside containment.

5.5.7.1.1.2 Cold Shutdown Scenario

The safe shutdown design basis of Virgil C. Summer Nuclear Station is hot standby. The plant can be maintained in a safe hot standby condition while manual actions are taken to permit achievement of cold shutdown conditions following a safe shutdown earthquake with loss of offsite power. Under such conditions the plant is capable of achieving RHR initiation conditions (approximately 350°F and less than 425 psia), including the time required for any manual actions. To achieve and maintain cold shutdown, 4 key functions must be performed. These are: (1) circulation of the reactor coolant, (2) removal of residual heat, (3) boration and makeup, and (4) depressurization. 02-01

1. Circulation of Reactor Coolant

   Circulation of the reactor coolant has 2 stages in a cooldown from hot standby to cold shutdown. The first stage is from hot standby to 350°F. During this stage, circulation of the reactor coolant is provided by natural circulation with the reactor core as the heat source and steam generators as the heat sink. Steam release from the steam generators is initially via the steam generator safety valves and

occurs automatically as a result of turbine and reactor trip. Steam release for cooldown is via the steam generator power operated relief valves. If air and control power are available, these valves will be operated from the main control board. A back-up air compressor can be manually loaded on the diesel generator busses. If these power sources are not available, communications will be established between the control room and auxiliary operators at the valves. These valves can be manually operated by handwheel. The steam generator power operated relief valves are easily accessible for local operation at floor level or by permanently installed platforms in the Seismic Category 1 penetration access area. The ability to operate these valves locally by handwheel was verified in the plant hot functional testing program. As a result of the installed noise reduction features on the valve, noise levels were sufficiently low that plant personnel were able to communicate with the control room and operate the valves with no difficulty. RN 10-014

The status of each steam generator can be monitored using Class 1E instrumentation located in the control room. Separate indication channels for both steam generator pressure and water level are available.

Feedwater to the steam generators is provided from the Emergency Feedwater System which has a 500,000 gallon Seismic Category 1 condensate storage tank (160,054 usable gallons are reserved for emergency feedwater, see Section 9.2.6.1) as the primary source and 2 separate Seismic Category 1 piping sub-systems. The first sub-system is composed of 2 motor driven pumps each powered from a different emergency power train, and the second sub-system incorporates a turbine driven pump which can receive motive steam from either of 2 steam generators. Backup is from the Seismic Category 1 Service Water System. There are additional sources of feedwater backup which can be manually accessed. The operation of the Emergency Feedwater System can be monitored using Class 1E instrumentation located in the control room. 99-01

The second stage of reactor coolant circulation is from 350°F to cold shutdown. During this stage, circulation of the reactor coolant is provided by the residual heat removal pumps.

2. Removal of Residual Heat

Removal of residual heat also has 2 stages in a cooldown from hot standby to cold shutdown. The first stage is from hot standby to 350°F.

During this stage, the steam generators act as the means of heat removal from the Reactor Coolant System. When the operators are ready to begin the cooldown, the steam generator power operated relief valves are opened slightly. As the cooldown proceeds, the operators will occasionally adjust these valves to increase the amount they are open. This allows a reasonable cooldown rate to be maintained. Feedwater makeup to the steam generators is provided from the Emergency Feedwater System. The Emergency Feedwater System has the ability 99-01

to remove decay heat by providing feedwater to all 3 steam generators for extended periods of operation.

The second stage is from 350°F to cold shutdown. During this stage, the Residual Heat Removal (RHR) System is brought into operation. The residual heat removal exchangers in the RHR system act as the means of heat removal from the Reactor Coolant System. In the RHR heat exchanger, the residual heat is transferred to the Component Cooling System which transfers the heat to the Service Water System. The Component Cooling and the Service Water Systems are both designed to Seismic Category 1. The RHR System includes 2 residual heat removal pumps and 2 residual heat removal heat exchangers. Each RHR Pump is powered from different emergency power trains and each RHR heat exchanger is

cooled by a different component cooling loop. If any component in one RHR loop becomes inoperable, cooldown of the plant is not compromised; however, the time for cooldown would be extended.

The operation of the RHR System can be monitored using Class 1E instrumentation in the control room.

3. Boration and Makeup

Boration is accomplished using portions of the Chemical and Volume Control System (CVCS). Boric acid 4 wt. % (approximately 7,000 ppm) from the boric acid tanks is supplied to the suction of the centrifugal charging pumps by the boric acid transfer pumps. The centrifugal charging pumps inject the borated water into the Reactor Coolant System via the normal charging and/or reactor coolant pump seal injection flow paths. The 2 boric acid tanks, 2 boric acid transfer pumps, centrifugal charging pumps, and the associated piping are of Seismic Category 1 design. There is sufficient boric acid capacity to provide for a cold shutdown with the most reactive rod withdrawn. The boric acid transfer pumps are each powered from different emergency power trains. The boric acid tank level can be monitored using Class 1E instrumentation in the control room to verify the operability of the boration portion of the CVCS. RN 99-110

Makeup, in excess of that required for boration, can be provided from the refueling water storage tank (RWST) using centrifugal charging pumps and the same injection flow paths as described for boration. Two (2) motor operated valves, each powered from different emergency power trains and connected in parallel, will transfer the suction of the charging pumps to the RWST. Makeup from the RWST can be monitored using Class 1E instrumentation in the control room.

4. Depressurization

Depressurization is normally accomplished using the reactor coolant pumps and the normal spray lines to the pressurizer.

Depressurization can be accomplished using the pressurizer power operated relief valves after boration and RCS cooldown to 450°F. This method consists of (1) discharging reactor coolant from the pressurizer to the pressurizer relief tank via the pressurizer power operated relief valves, and (2) allowing the pressurizer to cool via ambient heat losses as the Reactor Coolant System is maintained at 350°F via natural circulation. 02-01

5. Maintaining RCS Temperature and Pressure Without Letdown

Letdown is isolated by fail closed air operated valves and is assumed to be unavailable during accident conditions. The cold shutdown scenario of maintaining RCS temperature and pressure without letdown is considered the limiting cold shutdown scenario and is presented to demonstrate cold shutdown capability under abnormal conditions. As additional components are assumed to be available, the cold shutdown scenario is simplified. For example, as the reactor vessel head vent valves or the pressurizer PORVs are assumed to be available, the boration and makeup function or the depressurization function, respectively, is simplified as RCS letdown becomes available.

In performing the cooldown, the operator will integrate the functions of heat removal, boration and makeup, and depressurization so that these functions can be accomplished without letdown from the Reactor Coolant System. Boration, cooldown, and depressurization will be accomplished in a series of short steps arranged to keep Reactor Coolant System temperature and pressure and pressurizer level in the desired relationships. The boration requirements will be evaluated by the operator prior to initiating cooldown and depressurization. Based on initial plant conditions, the operator and/or the contents of the boric acid tanks. Once the plant is cooled to 350°F and depressurized to 425 psia, Residual Heat Removal System operation is initiated and the Reactor Coolant System is taken to cold shutdown conditions.

To demonstrate that boration and depressurization can be done without letdown, a simpler scenario can be used. First the operators integrate the cooldown and boration functions taking advantage of the RCS inventory contraction resulting from the cooldown. Finally, the operators use auxiliary spray from the CVCS to depressurize the plant to RHRS initiating conditions. The SCE&G calculation to demonstrate this capability assumes worst case boration requirements based on core end of life/peak xenon conditions and the following RCS initial conditions following plant trip: 02-01

RCS Temperature	557°F
RCS Pressure	2250 psia
Pressurizer Water Volume	350 ft^3
Pressurizer Steam Volume	1050 ft^3

The cooldown from 557°F to 350°F decreases the volume of water in the RCS by
approximately 1420 cubic feet. This assumes that the pressurizer is not cooled 02-01
and the water level is maintained at the initial condition. Makeup for contraction is
supplied by 4 wt. % boric acid stored in the boric acid tanks at 70°F. A boric acid
tank volume of approximately 1170 cubic feet is required to maintain the reactor
within the technical specification shutdown requirements at 350°F. A boric acid 02-01
tank volume of approximately 1170 cubic feet will expand to approximately 1310
cubic feet as it is heated to the RCS temperature of 350°F. The volume required
for boration requirements at 350°F is less than the contraction volume available at
350°F.

To calculate if depressurization can be accomplished without letdown and without
taking the plant water solid, it was assumed that the pressurizer was at saturated
conditions with 350 cubic feet of water, 1050 cubic feet of steam, and the
pressurizer metal at 653°F (2250 psia). It was further assumed that no additional
water would be removed from the pressurizer by the cooldown contraction. With
these assumptions, including the effect of heat input from the pressurizer metal, it
was determined that spraying in approximately 450 cubic feet of 70°F water would
provide saturated conditions at 425 psia (450°F) with a water volume of 913 cubic 02-01
feet and a steam volume of 487 cubic feet.

Once depressurized to 425 psia, RHRS operation is initiated and cooldown is
continued to cold shutdown conditions. The cooldown from 350°F to 200°F further 02-01
decreases the volume of water in the RCS. This assumes that the pressurizer is
not cooled and the pressurizer water level is maintained at the level resulting from
depressurization. There is no additional boration required in going from Mode 4 02-01
(350°F) to Mode 5 (200°F). This is expected since the shutdown margin
requirements are less restrictive in Mode 5 than in Mode 4.

The results of the calculations described above demonstrate that, based on the
assumed initial conditions, boration and depressurization with 4 wt. % RN
(approximately 7,000 ppm) boric acid can be accomplished without letdown and 99-110
without taking full credit for the available volume created by the cooldown
contraction. Should boration without letdown prove impractical due to any
combination of plant conditions or equipment failures, letdown can be achieved by
discharging RCS inventory via the pressurizer power operated relief valves or the
reactor vessel head vent valves.

6. Instrumentation

   Class 1E instrumentation is available in the control room to monitor the key functions associated with achieving cold shutdown. This instrumentation is discussed in Section 7.5 (Safety Related Display Instrumentation) and includes the following:

   a. RCS wide range temperature

   b. RCS wide range pressure

   c. Pressurizer water level

   d. Steam generator water level (per steam generator)

   e. Steam line pressure (per steam line)

   f. RWST level

   g. Boric acid tank level (per boric acid tank)

   h. Reactor building pressure

   This instrumentation is sufficient to monitor the key functions associated with cold shutdown and to maintain the RCS within the desired pressure, temperature, and inventory relationships. Operation of the auxiliary systems that service the RCS can be monitored by the control room operator, if desired, via remote communication with an operator in the plant.

5.5.7.1.1.3 Single Failure Evaluation

1. Circulation of the Reactor Coolant

   a. From Hot Standby to 350°F (Refer to FSAR Figures 5.1-1, 10.3-1, and 10.3-4) - Three (3) reactor coolant loops and steam generators are provided, any 2 of which can provide sufficient natural circulation flow to provide adequate core cooling. Even with the most limiting single failure (of a steam generator power operated relief valve), 2 of the reactor coolant loops and steam generators remain available.

   b. From 350°F to cold shutdown (Refer to FSAR Tables 5.5-7, 5.5-8, and Figure 5.5-4); 2 RHR pumps are provided, either 1 of which can provide adequate circulation of the reactor coolant.

2. Removal of Residual Heat

a. From Hot Standby to 350°F [Refer to FSAR Figures 10.3-1, 10.3-4, 10.4-16, 9.2-2 (Sheets 1 through 4), and 9.2-3]. RN 09-002

(1) Steam generator power operated relief valves - Three (3) are provided (1 per steam generator), any 2 of which are sufficient for residual heat removal. In the event of a single failure, 2 power operated relief valves remain available.

(2) Emergency feedwater pumps - Two (2) motor driven and 1 steam driven emergency feedwater pumps are provided. In the event of a single failure, 2 pumps remain available, either of which can provide sufficient feedwater flow.

(3) Flow control valves - Air operated, fail open valves are provided. In the event of a single failure of 1 flow control valve (which effects flow to 1 steam generator from either a motor driven pump or the steam driven pump), emergency feed flow can still be provided to all 3 steam generators from the other pumps.

(4) Backup source - A backup source of emergency feedwater can be provided to the suction of the emergency feedwater pumps from either train of the Seismic Category 1 Service Water System.

b. From 350°F to 200°F [Refer to FSAR Table 5.5-15 and Figure 5.5-4, and Figures 9.2-1, 9.2-2 (Sheets 1 through 4), 9.2-3, and 9.2-4]. RN 09-002

(1) RHR Suction Isolation Valves 8701A and 8702A (RHR Pump 1) and 8701B and 8702B (RHR Pump 2) - The 2 valves in each RHR subsystem are each powered from different emergency power trains. Failure of either power train could prevent initiation of RHR cooling in the normal manner from the control room. In the event of such a failure, the affected valve can be de-energized and opened with its handwheel. Another method of operating these valves is by the use of alternate temporary power. Any other single failure can be tolerated as it would only affect one of the RHR subsystems, and adequate cooling can be provided by the redundant subsystem.

(2) RHR pumps 1 and 2 - Each pump is powered from a different emergency power train. In the event of a single failure, either pump can provide sufficient RHR flow.

(3) RHR heat exchangers 1 and 2 - If either heat exchanger is unavailable for any reason, the remaining heat exchanger is unavailable for any reason, the remaining heat exchanger can provide sufficient heat removal capability.

(4) RHR Flow Control Valves HCV603A and B - If either of these normally open fail open valves closes spuriously, sufficient RHR cooling can be provided by the unaffected RHR train.

(5) RHR/SIS Cold Leg Isolation Valves 8888A and B - If either of these normally open, motor operated valves, which are powered from different emergency power trains, closes spuriously, sufficient RHR cooling can be provided by the unaffected RHR train. The affected valve can be de-energized and opened with its handwheel.

(6) Component Cooling Water System - Two (2) redundant subsystems are provided for safety related loads. Either subsystem can provide sufficient heat removal via one of the RHR heat exchangers.

(7) Service Water System - Two (2) redundant subsystems are provided for safety related loads. Either subsystem can provide sufficient heat removal via 1 of the CCW heat exchangers.

3. Boration and Makeup (Refer to FSAR Figures 5.1-1, 6.3-1, and 9.3-16)

a. Boric Acid Tanks 1 and 2 - Two (2) boric acid tanks are provided. Each tank contains sufficient 4% (approximately 7,000 ppm) boric acid to borate the Reactor Coolant System for cold shutdown. RN 99-110

b. Boric Acid Transfer Pumps 1 and 2 - Each pump is powered from a different emergency power train. In the event of a single failure, either pump can provide sufficient boric acid flow.

c. Isolation Valve 8104 - If valve 8104, which is supplied from emergency power and is normally closed, cannot be opened due to power train or operator failure, it can be opened locally with its handwheel. If valve 8104 cannot be opened with its handwheel, an alternate flow path is available via a) air operated, fail open valve FCV-113A and normally closed manual valve 8439, or b) gravity feed through normally closed manual valves 8329 and 8331.

d. Refueling Water Storage Tank Isolation Valves LCV-115B and LCV-115D - Each valve is powered from a different emergency power train, only one of these normally closed motor operated valves needs to be opened to provide a makeup flow path from the RWST to the centrifugal charging pumps.

e. Centrifugal Charging Pumps A, B, and C - Pumps A and B are powered from a different emergency power train. In the event of a single failure, any 1 pump can provide sufficient boration or makeup flow.

f. Charging Pump Suction Isolation Valves 8130A, B and 8131A, B - If 1 of these normally open motor operated valves--each of which is powered from a different emergency power train--closes spuriously, operator action can be used to de-energize the valve operator and reopen the valve with its handwheel.

g. Normal Charging Flow Control Valve FCV-122 - This normally open valve fails open on loss of air or power. If FCV-122 closes spuriously, the charging pumps can operate on their miniflow circuits until operator action can open bypass valve 8403.

h. Normal Charging Isolation Valves 8107 and 8108 - If either of these normally open, motor operated valves--each of which is powered from a different emergency power train -- closes spuriously, operator action can be used to de-energize the valve operator and reopen the valve with its handwheel.

i. Normal Charging Isolation Valve 8146 - If the normally open valve closes spuriously, alternate charging valve 8147, which fails open, can be used. RN 11-039

j. Charging Pump Discharge Isolation Valves 8132 A, B, and 8133 A, B - If 1 of these normally open motor operated valves--each of which is powered from a different emergency power train--closes spuriously, operator action can be used to de-energize the valve operator and reopen the valve with its handwheel. Power to valves 8133A, B is normally locked out to prevent spurious operation. Reference FSAR Section 6.3.2.20. 98-01

k. Reactor Coolant Pump Seal Injection Isolation Valve 8105 - If this normally open motor operated valve closes spuriously, operator action can be used to de-energize the valve operator and reopen the valve with its handwheel (also see item p.). RN 11-027

l. Reactor Coolant Pump Seal Injection Flow Control Valve HCV-186 - This normally open valve fails open on loss of air or power. If HCV-186 closes spuriously, the charging pumps can operate on their miniflow circuits until operator action can open bypass valve 8389 (also see item p.). RN 11-027

m. Reactor Coolant Pump Seal Injection Valves 8102 A, B, and C - If any of these normally open motor operated valves closes spuriously, operator action can be used to de-energize the valve operator and reopen the valve with its handwheel.

n. Cold Leg Injection Isolation Valves 8801 A and B - Each valve is powered from a different emergency power train; only 1 of these normally closed motor operated valves needs to be opened to provide a makeup path and source for boration.

o. Reactor Vessel Head Vent Valves 8095 A, B, and 8096 A, B - These valves fail as-is on loss of power. They are arranged with 2 valves in series in each of 2 parallel paths. The isolation valves in series in each flow path are powered from separate emergency power supplies. One (1) normally closed isolation valve and 1 normally open valve are located in each flow path. This valving and power supply arrangement ensures that 1 path from the reactor vessel head can be opened assuming a single failure. One (1) path is sufficient to permit letdown from the Reactor Coolant System to augment boration and makeup operations.

p. If total RCP seal injection flow falls below 10 gpm, the Alternate Seal Injection (ASI) System (part of CVCS) is designed to automatically supply a total of 20 gpm RWST water downstream of HCV-186 and XVT-8105. The ASI system is an ASME Code Class 2 system (pressure boundary); however, the power source for the ASI pump (XPP0230) is from an NNS power supply. RN 11-027

4. Depressurization (Refer to FSAR Figure 9.3-16)

a. Pressurizer Power Operated Relief Valves PCV-444B and PCV-445A - These normally closed valves fail closed on loss of air or power. However, the valves are redundant, are powered by separate emergency electrical power supplies, and have backup seismic category air supply accumulators. The operability of either valve is sufficient to permit depressurization.

b. Auxiliary Spray Valve 8145 - This normally closed valve fails closed on loss of air or power. If 8145 is stuck closed as a result of a single failure, the redundant Seismic Category 1 pressurizer PORVs can be used to depressurize the RCS as described in the alternate method for depressurization.

5. Instrumentation

Sufficient instrumentation is provided to monitor from the control room the key function associated with cold shutdown. All necessary indications are redundant. Thus, in the event of a single failure, the operator can make comparisons between duplicate information channels or between functionally related channels in order to identify the particular malfunction. Refer to FSAR Section 7.5 (Safety Related Display Instrumentation) for applicable details.

6. Qualification

The equipment discussed in the cold shutdown scenario is safety grade with the following exceptions. These are few in number and of such a nature that local manual actions and/or equipment repair could be performed while the plant is maintained in the hot standby condition while preparations are made to go to cold shutdown. These manual actions would not prevent the plant from achieving residual heat removal system initiations within 36 hours.

a. Steam generator power operated relief valves - These air operated valves are provided with safety grade remote operators; however, remote control provisions are control grade. Hot standby can be achieved and maintained using the safety grade steam generator safety valves. The steam generator power operated relief valves are provided with handwheels which can be operated locally to permit plant cooldown.

b. Charging line isolation valves (8146, 8147), Charging line auxiliary spray valve (8145) - Seal injection line hand control valve (HCV-186). Charging line flow control valve (FCV-122).

   These air operated valves are not provided with safety grade remote operators, air supplies, or power supplies. Their failure consequences were discussed in the Parts 3 and 4 of the single failure evaluation.

c. Residual Heat Removal System Flow Control Valve HCV-603 A/B - This air operated valve is not provided with a safety grade remote operator, air supply, or power supply. Its failure consequence was discussed in Part 2 of the single failure evaluation.

d. Pressurizer relief tank - This tank is a non-nuclear safety class and non-Seismic Category 1 tank. Its failure does not affect the ability of the Virgil C. Summer Nuclear Station to achieve cold shutdown.

e. Pressurizer power operated relief valves - These air operated valves are not supplied with safety grade remote operators; however, they are powered from separate vital DC electrical power supplies and 2 of them have Seismic Category 1 air supply accumulators. These valves are discussed in Part 4 of the cold shutdown scenario. RN 98-020

5.5.7.1.1.4 Natural Circulation

The natural circulation capabilities of the Virgil C. Summer Nuclear Station are compared with 3 and 4 loop test results in Section 5.5.7.1.1, Item 5.

5.5.7.2 System Description

The RHRS, as shown in Figure 5.5-4 consists of 2 residual heat exchangers, 2 residual heat removal pumps, and the associated piping, valves, and instrumentation necessary for operational control. The inlet lines to the RHRS are connected to the hot legs of 2 reactor coolant loops, while the return lines are connected to each of the cold legs and to 2 hot legs of the reactor coolant loops. These return lines are also the ECCS low head cold leg injection/recirculation lines and the hot leg recirculation lines (see Figure 6.3-1). 99-01

The RHRS suction lines are isolated from the RCS by 2 motor operated valves in series and a relief valve, all located inside the Reactor Building. Each discharge line is isolated from the RCS by 2 check valves located inside the Reactor Building and by a motor operated valve located outside the Reactor Building. (The check valves and the motor operated valve on each discharge line are not shown as part of the RHRS; these valves are shown as part of the ECCS, see Figure 6.3-1.)

During RHRS operation, reactor coolant flows from the RCS to the residual heat removal pumps, through the tube side of the residual heat exchangers, and back to the RCS. The heat is transferred to the component cooling water circulating through the shell side of the residual heat exchangers.

Coincident with operation of the RHRS, a portion of the reactor coolant flow may be diverted from downstream of the residual heat exchangers to the Chemical and Volume Control System (CVCS) low pressure letdown line for cleanup and/or pressure control. By regulating the diverted flowrate and the charging flow, the RCS pressure may be controlled. Pressure regulation is necessary to maintain the pressure range dictated by the fracture prevention criteria requirements of the reactor vessel and by the number 1 seal differential pressure and net positive suction head requirements of the reactor coolant pumps.

The RCS cooldown rate is manually controlled by regulating the reactor coolant flow through the tube side of the residual heat exchangers. A line containing a flow control valve bypasses each residual heat exchanger and is used to maintain a constant return flow to the RCS. Instrumentation is provided to monitor system pressure, temperature and total flow.

The RHRS is also used for filling the refueling cavity before refueling. After refueling operations, water is pumped back to the refueling water storage tank until the water level is brought down to the flange of the reactor vessel. The remainder of the water is removed via a drain connection at the bottom of the refueling cavity.

When the RHRS is in operation, the water chemistry is the same as that of the reactor coolant. Provision is made for the process sampling system to extract samples from the flow of reactor coolant downstream of the residual heat exchangers. A local sampling point is also provided on each residual heat removal train between the pump and heat exchanger.

The RHRS functions in conjunction with the high head portion of the ECCS to provide injection of borated water from the refueling water storage tank into the RCS cold legs during the injection phase following a loss of coolant accident.

In its capacity as the low head portion of the ECCS, the RHRS provides long term recirculation capability for core cooling following the injection phase of the loss of coolant accident. This function is accomplished by aligning the RHRS to take fluid from the reactor building sump, cool it by circulation through the residual heat exchangers, and supply it to the core directly as well as via the centrifugal charging pumps.

The use of the RHRS as part of the ECCS is more completely described in Section 6.3.

5.5.7.2.1 Component Description

The materials used to fabricate RHRS components are in accordance with the applicable code requirements. All parts of components in contact with borated water are fabricated or clad with austenitic stainless steel or equivalent corrosion resistant material.

Portions of the system that are designed to carry ECCS fluid outside containment during the recirculation phase of ECCS operation are designed to minimize leakage to the atmosphere as described in Section 6.3.2.11.3. RN 04-005

Component codes and classifications are given in Section 3.2 and component parameters are listed in Table 5.5-8.

5.5.7.2.1.1 Residual Heat Removal Pumps

Two (2) pumps are installed in the RHRS. The pumps are sized to deliver reactor coolant flow through the residual heat exchangers to meet the plant cooldown requirements. The use of 2 separate residual heat removal trains assures that cooling capacity is adequately maintained should 1 pump become inoperative.

The residual heat removal pumps are protected from overheating and loss of suction flow by minimum flow bypass lines that assure flow to the pump suction. A valve located in each minimum flow line is regulated by a signal from the flow switch located in each heat exchanger discharge header. The control valves open when RHR discharge flow to the RCS reaches the minimum flow setpoint, which is set to prevent RHR pump flow from approaching the minimum design flow of 500 gpm. The control valves close when flow reaches the close setpoint at nearly the maximum range of the flow switch, or approximately 1500 gpm. RN 03-047

A pressure sensor in each pump discharge header provides a signal for an indicator in the control room. A high pressure alarm is also actuated by the pressure sensor. The 2 pumps are vertical, centrifugal units with mechanical seals on the shafts. All pump surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

5.5.7.2.1.2 Residual Heat Exchangers

Two (2) residual heat exchangers are installed in the system. The heat exchanger design is based on heat load and temperature differences between reactor coolant and component cooling water existing at 20 hours after reactor shutdown from an extended full power run at 2775 MWt when the temperature difference between the 2 systems is small. For the current licensed RTP of 2900 MWt, the corresponding decay heat level occurs at 24 hours after reactor shutdown.

The installation of 2 heat exchangers in separate and independent residual heat removal trains assures that the heat removal capacity of the system is adequately maintained if 1 train becomes inoperative.

The residual heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell. The tubes are welded to the tubesheet to prevent leakage of reactor coolant.

5.5.7.2.1.3 Residual Heat Removal System Valves

Valves that perform a modulating function are equipped with 2 sets of packings and an intermediate leakoff connection that discharges to the drain header.

Manual and motor operated valves have backseats to facilitate repacking and to limit stem leakage when the valves are open. Leakage connections are provided where required by valve size and fluid conditions.

5.5.7.2.2 System Operation

5.5.7.2.2.1 Reactor Startup

Generally, while at cold shutdown condition, decay heat from the reactor core is being removed by the RHRS. The number of pumps and heat exchangers in service depends upon the heat load at the time.

At initiation of the plant startup, the RCS is completely filled, and the pressurizer heaters are energized. The RHRS is operating and is connected to the CVCS via the low pressure letdown line to control reactor coolant pressure. During this time, the RHRS acts as an alternate letdown path. The manual valves downstream of the residual heat exchangers leading to the letdown line of the CVCS are opened. The control valve in the line from the RHRS to the letdown line of the CVCS is then manually adjusted in the control room to permit letdown flow. Failure of any of the valves in the line from the RHRS to the CVCS has no safety implications, either during startup or cooldown.

After the reactor coolant pumps are started, the residual heat removal pumps are stopped but pressure control via the RHRS and the low pressure letdown line is continued until the pressurizer steam bubble is formed. Indication of steam bubble formation is provided in the control room by the damping out of the RCS pressure fluctuations, and by pressurizer level indication. The RHRS is then isolated from the RCS and the system pressure is controlled by normal letdown and the pressurizer spray and pressurizer heaters.

5.5.7.2.2.2 Power Generation and Hot Standby Operation

During power generation and hot standby operation, the RHRS is not in service but is aligned for operation as part of the ECCS.

5.5.7.2.2.3 Reactor Cooldown

Reactor cooldown is defined as the operation which brings the reactor from no-load temperature and pressure to cold shutdown conditions.

The initial phase of reactor cooldown is accomplished by transferring heat from the RCS to the steam and power conversion system through the use of the steam generators.

When the reactor coolant temperature and pressure are reduced to approximately 350°F and less than 425 psig, approximately 4 hours after reactor shutdown, the 02-01 second phase of cooldown starts with the RHRS being placed in operation. It should be noted that given the decay heat models and the RCP heat assumed, the RHRS system, under worst case design conditions, cannot begin to cooldown the RCS until approximately 6 hours after an extended full power run at the current licensed RTP of 2900 MWt (although, it can be placed into service in parallel with the steam dump system at approximately 4 hours after reactor shutdown). Plant procedures do verify that heat removal occurs via RHRS before fully transferring from the steam dump system.

Startup of the RHRS includes a warmup period during which time reactor coolant flow through the heat exchangers is limited to minimize thermal shock. The rate of heat removal from the reactor coolant is manually controlled by regulating the coolant flow through the residual heat exchangers. By adjusting the control valves downstream of the residual heat exchangers the mixed mean temperature of the return flows is controlled. Coincident with the manual adjustment, each heat exchanger bypass valve is regulated to give the required total flow.

The reactor cooldown rate is limited by RCS equipment cooling rates based on allowable stress limits, as well as the operating temperature limits of the CCWS. As the reactor coolant temperature decreases, the reactor coolant flow through the residual heat exchangers is increased by adjusting the control valve in each heat exchanger's tube side outlet line.

Assuming that only 1 heat exchanger and pump are in service and that the heat exchanger is supplied with component cooling water at design flow and at a temperature not exceeding 120°F, the coupled RHRS/CCWS/SWS system is capable of reducing the RCS temperature from 350°F at 6 hours after shutdown from an extended full power run at the current licensed RTP of 2900 MWt to 200°F within 30 hours from the beginning of the cooldown. The RHRS heat load during the transient includes the heat from one RCP, core decay heat, and the sensible heat of the RCS metal and fluid.

As cooldown continues, the pressurizer is filled with water and the RCS is operated in the water solid condition.

At this stage, pressure control is accomplished by regulating the charging flowrate and the rate of letdown from the RHRS to the CVCS.

After the reactor coolant pressure is reduced and the temperature is 140°F or lower, the RCS may be opened for refueling or maintenance once the remaining shutdown operations are considered complete.

5.5.7.2.2.4 Refueling

Both residual heat removal pumps are utilized during refueling to pump borated water from the refueling water storage tank to the refueling cavity. During this operation, the isolation valves in the inlet lines of the RHRS are closed, and the isolation valves from the refueling water storage tank are opened.

Station operating procedures include instructions for draining the Reactor Coolant System for vessel head removal prior to refueling operations. The instruction includes the installation of clear hose between 1 of the Reactor Coolant System loop drains and the pressurizer relief line vent to provide level indication after the pressurizer level indication is off scale. The procedure also states that after the Reactor Coolant System level has reached 4 to 12 inches below the reactor vessel flange, draining operations are secured. These actions insure that air will not be introduced into the RHR System via the Reactor Coolant System during refueling operations.

The reactor vessel head is lifted. The refueling water is then pumped into the reactor vessel through the normal RHRS return lines and into the refueling cavity through the open reactor vessel. After the water level reaches the normal refueling level, the refueling water storage tank supply valves are closed, the inlet isolation valves are opened, and residual heat removal is resumed. RN 14-018

During refueling, the RHRS is maintained in service with the number of pumps and heat exchangers in operation as required by the heat load.

Following refueling, the residual heat removal pumps are used to drain the refueling cavity to the top of the reactor vessel flange by pumping water from the RCS to the refueling water storage tank.

5.5.7.3 Design Evaluation

5.5.7.3.1 System Availability and Reliability

General Design Criterion 34 requires that a system to remove residual heat be provided. The safety function of this system is to transfer fission product decay heat and other residual heat from the core at a rate sufficient to prevent fuel or pressure boundary design limits from being exceeded. Safety grade systems are provided in the plant design to perform this safety function. The NSSS safety grade systems which perform this function, for all plant conditions except a LOCA, are: the RCS and steam generators, which operate in conjunction with the Emergency Feedwater System, and the steam generator safety valves; and the RHRS which operates in conjunction with the reactor plant Component Cooling Water System and the Service Water System. The BOP safety grade systems which perform this function, for all plant conditions except LOCA, are: the Emergency Feedwater System, the steam generator safety valves, which operate in conjunction with the Reactor Coolant System and the steam generators; and the reactor plant Component Cooling Water and Service Water Systems, which operate in conjunction with the RHRS. For LOCA conditions, the safety grade system which performs the function of removing residual heat from the reactor core is the ECCS, which operates in conjunction with the reactor plant Component Cooling Water System and the Service Water System.

The Emergency Feedwater System, along with the steam generator safety valves provides a completely separate, independent, and diverse means of performing the safety function of removing residual heat, which is normally performed by the RHRS when RCS temperature is less than 350°F. The Emergency Feedwater System is capable of performing this function for an extended period of time following plant shutdown.

The RHRS is provided with 2 residual heat removal pumps, and 2 residual heat removal heat exchangers arranged in 2 separate, independent flow paths. To assure reliability, each residual heat removal pump is connected to a different vital bus. Each residual heat removal train is isolated from the RCS on the suction side by 2 motor operated valves in series. Each motor operated valve receives power via a separate motor control center and the 2 valves in series in each train receive their power from a different vital bus. Each suction isolation valve is also independently and diversely interlocked to 1 of 2 RCS wide range pressure instruments to prevent opening the valves when the RCS pressure is above 425 psig. Independence is accomplished by aligning each RCS wide range pressure transmitter to a different vital bus. Diversity is accomplished through the use of 2 RCS wide range pressure instruments which employ different pressure sensing principals. Also, at strategically identified local high points, the RHRS is provided with indicating air-traps to ensure the RHRS remains void free (reference NRC GL 2008-01). RN 09-003

RHRS operation for normal conditions is accomplished from the control room. The redundancy in the RHRS design provides the system with the capability to maintain its cooling function even with major single failures, such as failure of an RHR pump, valve, or heat exchanger since the redundant train can be used for continued heat removal. See Table 5.5-15 for a single active failure analysis of the RHRS.

Although such major system failures are within the system design basis, there are other less significant failures which can prevent opening of the RHR suction isolation valves from the control room. Since these failures are of a minor nature, improbable to occur, and easily corrected outside the control room, with ample time to do so, they have been realistically excluded from the engineering design basis. Such failures are not likely to occur during the limited time period in which they can have any effect (i.e., when opening the suction isolation valves to initiate RHR operation); however, even if they should occur, they have no adverse safety impact and can be readily corrected. In such a situation, the emergency feedwater system and steam generator safety valves can be used to perform the safety function of removing residual heat and can be used to continue the plant cooldown below 350°F, until the RHR system is made available. 98-01

One (1) failure which can prevent opening the RHR suction isolation valves from the control room is a failure of an electrical power train. Such a failure is extremely unlikely to occur during the few minutes per year operating time, during which it can have any consequence. If such an unlikely event should occur, several alternatives are available. The most realistic approach would be to obtain restoration of offsite power, which can be expected to occur in less than 1/2 hour. Other alternatives are to restore the emergency diesel generator to operation, to bring in an alternate power source, or to open the affected valves with their manual handwheels.

The only impact of either of the above types of failure is some delay in initiating RHR operation, while action is taken to open the RHR suction isolation valves. This delay has no adverse safety impact because of the capability of the emergency feedwater system and steam generator safety relief valves to continue to remove residual heat.

The safe shutdown design basis of the Virgil C. Summer Nuclear Station is hot standby, as it is for all other Westinghouse designed pressurized water reactors. Hot standby is a safe and stable plant condition which can be maintained for an extended period of time following any Condition II, III, or IV event. In the hot standby condition, residual heat removal, in compliance with GDC 34, is provided by the Emergency Feedwater System in conjunction with the steam generator safety valves. Cross connections from the Service Water System to the Emergency Feedwater System provide a long term (i.e., greater than 7 days) source of emergency feedwater.

In view of the above described capability, the Residual Heat Removal (RHR) System is not required to fully comply with the single failure requirement of GDC 34. For most major single failures, the RHR System does fully comply with the single failure criterion. The only situation where full compliance is not provided is for opening the RHR suction isolation valves (to initiate RHR cooling) in the event of the failure of 1 of the safeguards power trains. In the event of such a failure, sufficient time is available to either compensate for the single failure by manual actions inside (i.e., by opening a suction valve with its manual handwheel) or outside (by providing an alternate temporary power supply to a suction valve) the containment or to restore the failed safeguards power train to operation.

See Section 5.5.7.1.1 for compliance with BTP RSB 5-1.

The maximum cooldown rate which can result if both RHR flow control valves and both RHR bypass valves all simultaneously fail in such a manner as to permit maximum flow through the RHR heat exchangers (a low probability event considering the few hours a year when it could cause any effect) depends on several factors including the RHR flow rate, the Component Cooling Water System flow rates and temperatures, and other heat loads on the Component Cooling Water System. One (1) of the key factors is the RCS temperature, since the heat removal rate depends on the temperature differential between the RHR (RCS) flow and the component cooling water flow in the RHR heat exchanger. Typically, it is impossible to maintain a cooldown rate even as high as the design rate of 50°F/hr when the RCS temperature is less than 250°F, even with the maximum flow through the RHR heat exchangers.

Even if maximum flow through the RHR heat exchangers was experienced at the instant of initiating RHR operation and no operator action was taken, it is unlikely that the cooldown would exceed 100°F in the first hour. The cool down rate in the subsequent hours would be much less than 100°F/hr. The maximum possible cooldown rate from 350°F to 250°F would not exceed 200°F/hr. Calculations have been done which show that, from a stress standpoint, a cooldown rate greater than 200°F/hr is acceptable for such a hypothetical cooldown from 350°F to 250°F even though, as discussed above, the actual maximum rate of cooldown at or below 250°F is not expected to exceed 50°F/hr.

Although such a hypothetical cooldown event is acceptable assuming no operator action, it should be noted that the operator can significantly limit the maximum possible cooldown rate by merely stopping one of the RHR pumps.

5.5.7.3.2 Leakage Provisions and Activity Release

In the event of a loss of coolant accident or during normal recirculation mode, radioactive fluid may be recirculated through part of the RHRS exterior to the reactor building. If the residual heat removal pump seal should fail, the water would spill out on a floor in a shielded compartment. Each residual heat removal pump compartment contains a sump which is equipped with an alarm system set to indicate leakage of

greater than 45 gpm. The alarm system is annunciated in the control room so the operator can determine the location of the leak. Redundant sump pumps are capable of handling leakage flows up to 50 gpm each. In the case of failure of both sump pumps, each residual heat removal pump room has a water tight volume of at least 200 ft^3 to accommodate a 50 gpm leak for at least 30 minutes. This provides protection against pump flooding in the event of a 50 gpm leak. Any leakage, whether handled by the residual heat removal pump room sumps or the Auxiliary Building sump pumps, is directed to holdup tanks in the liquid waste processing system. Residual heat removal piping and pumps in each cubicle can be remotely isolated by motor operated valves so they can be drained and flushed prior to being repaired.

The maximum discharge rate from a moderate energy pipe crack in the RHR system is approximately 714 gpm. 02-01

Since such a leak rate would have no effect on core cooling until it resulted in emptying the RCS loops, the time available to the operator to take action is at least 73 minutes. 02-01

The operator would be alerted to the leak by decreasing pressurizer level and the pressurizer low level alarm. Depending on the location of the crack, the operator could also be alerted by RHR pump compartment sump level alarms. The only action necessary to recover from the event is to close the suction isolation valve(s) of the affected RHR train and to initiate cooling on the other RHR train, if it is not already operating. Depending on the amount of water lost before isolating the leak, the operator may be required to makeup to the RCS.

5.5.7.3.3 Overpressurization Protection

Each inlet line to the RHRS is equipped with a pressure relief valve which protects the system from inadvertent overpressurization during plant cooldown or startup. Each valve has a relief flow capacity of 900 gpm at a set pressure of 450 psig. Analyses have been conducted to confirm the capability of the RHRS relief valve to prevent overpressurization of the RHRS. All credible events were examined for their potential to overpressurize the RHRS. These events included normal operating conditions, infrequent transients, and abnormal occurrences. The analyses confirmed that 1 relief valve has the capability to keep the RHRS maximum pressure within 10CFR50 Appendix G limits.

Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve to relieve the maximum possible back-leakage through the valves separating the RHRS from the RCS. Each valve has a relief flow capacity of 20 gpm at a set pressure of 600 psig. These relief valves are located in the ECCS (see Figure 6.3-1).

The fluid discharged by the suction side relief valves is collected in the pressurizer relief tank. The fluid discharged by the discharge side relief valves is collected in the recycle holdup tanks of the boron recycle system.

5.5.7.3.4 Prevention of Exposure of the RHRS to Normal RCS Operating Pressure

The design of the RHRS includes 2 motor-operated gate isolation valves in series on each inlet line between the high pressure RCS and the lower pressure RHRS. They are closed during normal operation and are only opened for residual heat removal and RCS cold overpressure protection during a plant cooldown after the RCS pressure is reduced to less than 425 psig and the RCS temperature is reduced to approximately 350°F. 02-01
During a plant startup, the inlet isolation valves are shut after drawing a bubble in the pressurizer and prior to increasing RCS pressure above 425 psig. Power to the isolation valves is manually locked out during normal operation.

The 2 inlet isolation valves in each residual heat removal subsystem are independently and diversely interlocked with pressure signals to prevent their being opened whenever the RCS pressure is greater than approximately 425 psig. The autoclosure interlock of these valves has been removed as per WCAP-11835. An alarm has been added to alert the operator if the valves are not closed when the RCS pressure increases above the alarm setpoint of 520 psig.

The use of 2 independently powered motor-operated valves in each of the 2 inlet lines, along with an independent and diverse pressure interlock to prevent them from being opened, and a RCS pressure high with RHR suction valves not closed alarm assures a 02-01
design which meets applicable single failure criteria. Not only more than one single failure, but also different failure mechanisms, must be postulated to defeat the function of preventing possible exposure of the RHR system to normal RCS operating pressure. This protective interlock design, in combination with alarm, administrative controls, and plant operating procedures, provide the means for accomplishing the protective function. For further information on the instrumentation and control features, see Section 7.6.2.

The RHR inlet isolation valves are provided with red-green position indicator lights on the main control board and a ESF monitor light for proper valve position. ESF monitor light is independent and diverse from valve position indication.

Isolation of the low pressure RHRS from the high pressure RCS is provided on the discharge side by 2 check valves in series. These check valves are located in the ECCS and their testing is described in Section 5.2.2.4.

5.5.7.3.5 Shared Function

The safety function performed by the RHRS is not compromised by its normal function which is normal plant cooldown. The valves associated with the RHRS are normally aligned to allow immediate use of this system in its engineered safety features mode of operation. The system has been designed in such a manner that 2 redundant flow circuits are available, assuring the availability of at least 1 train for safety purposes.

The normal plant cooldown function of the RHRS is accomplished through a suction line arrangement which is independent of any safety function. The cold leg cooldown return lines are arranged in parallel redundant circuits and are utilized also as the low head injection lines to the RCS. Utilization of the same return circuits for cold leg cool down lends assurance to the proper functioning of these lines for engineered safety features purposes.

5.5.7.3.6 Radiological Consideration

The highest radiation levels experienced by the RHRS are those which would result from a loss of coolant accident. Following a loss of coolant accident, the RHRS is used as part of the ECCS. During the recirculation phase of emergency core cooling, the RHRS is designed to operate for up to a year pumping water from the Reactor Building sump, cooling it, and returning it to the Reactor Building to cool the core.

Since, except for some valves and piping, the RHRS is located outside the reactor building, most of the system is not subjected to the high levels of radioactivity in the Reactor Building post accident environment.

The operation of the RHRS does not involve a radiation hazard for the operators since the system is controlled remotely from the control room. If maintenance of the system is necessary, the portion of the system requiring maintenance is isolated by remotely operated valves and/or manual valves with stem extensions which allow operation of the valves from a shielded location. The isolated piping is drained and flushed before maintenance is performed.

5.5.7.4 Tests and Inspections

Periodic visual inspections and preventive maintenance are conducted during plant operation according to normal industrial practice.

The instrumentation channels for the residual heat removal pump flow instrumentation devices are calibrated during each refueling operation if a check indicates that recalibration is necessary.

Due to the role the RHRS has in sharing components with the ECCS, the residual heat removal pumps are tested as a part of the ECCS testing program (see Section 6.3.4). Preoperational testing is discussed in Chapter 14.

5.5.8 REACTOR COOLANT CLEANUP SYSTEM

The Chemical and Volume Control System provides reactor coolant cleanup and is discussed in Section 9.3.4. The radiological considerations are discussed in Chapter 11.

5.5.9 MAIN STEAM LINE AND FEEDWATER PIPING

Main steam line and feedwater piping are discussed in Sections 10.3 and 10.4.7, respectively.

5.5.10 PRESSURIZER

5.5.10.1 Design Bases

The general configuration of the pressurizer is shown in Figure 5.5-6. The design data of the pressurizer are given in Table 5.5-9. Codes and material requirements are provided in Section 5.2.

The pressurizer provides a point in the RCS where liquid and vapor can be maintained in equilibrium under saturated conditions for pressure and control purposes.

Stress analysis of the pressurizer components were performed at the plant conditions associated with steam generator replacement. While a slight increase in the fatigue usage for several of the pressurizer components was noted, the acceptance limits of Section III of the ASME Code were not exceeded. Therefore, the pressurizer components will not be overstressed or subject to fatigue usage during the remainder of the projected plant lifetime.

5.5.10.1.1 Pressurizer Surge Line

The surge line is sized to limit the pressure drop between the RCS and the safety valves with maximum allowable discharge flow from the safety valves. Overpressure of the RCS does not exceed 110% of the design pressure.

The surge line and thermal sleeve at the pressurizer end are designed to withstand the thermal stresses resulting from volume surges of relatively hotter or colder water which may occur during operation. RN 03-051

The pressurizer surge line nozzle diameter is given in Table 5.5-9 and the pressurizer surge line dimensions are shown in Figure 5.1-1, Sheet 2.

5.5.10.1.2 Pressurizer

The volume of the pressurizer is equal to or greater than the minimum volume of steam, water, or total of the 2 which satisfies the following requirements:

1. The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes.

2. The water volume is sufficient to prevent the heaters from being uncovered during a 10% step load increase to full power.

3. The steam volume is large enough to accommodate the surge resulting from 50% reduction of full load with automatic reactor control and 40% steam dump without the water level reaching the high level reactor trip point.

4. The steam volume is large enough to prevent water relief through the safety valves following a loss of load with the high water level initiating a reactor trip, without reactor control or steam dump.

5. The pressurizer will not empty following reactor trip and turbine trip.

6. The emergency core cooling signal is not activated during reactor trip and turbine trip.

5.5.10.2 Design Description

5.5.10.2.1 Pressurizer Surge Line

The pressurizer surge line connects the pressurizer to 1 reactor hot leg. The line enables continuous coolant volume pressure adjustments between the RCS and the pressurizer.

5.5.10.2.2 Pressurizer

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads constructed of carbon steel, with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant. A stainless steel liner or tube is used in lieu of cladding in some nozzles.

The high strength welds joining the component parts of the pressurizer are the 4 girth welds joining the upper head, 3 shell barrels, and the lower head, and the 3 longitudinal welds joining the 3 sections of the shell barrels. The various weld and flux combinations used in these welds are listed in Table 5.5-18 along with the appropriate fracture toughness test information.

The surge line nozzle and removable electric heaters are installed in the bottom head. The heaters are removable for maintenance or replacement. A thermal sleeve is provided to minimize stresses in the surge line nozzle. A screen at the surge line nozzle and baffles in the lower section of the pressurizer prevent an insurge of cold water from flowing directly to the steam/water interface and assist mixing.

Spray line nozzles, relief, and safety valve connections are located in the top head of the vessel. Spray flow is modulated by automatically controlled air operated valves. The spray valves can also be operated manually in the control room.

A small continuous spray flow is provided through a manual bypass valve around the power operated spray valves to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excessive cooling of the spray piping.

During an outsurge from the pressurizer, flashing of water to steam, and generating of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. During an insurge from the RCS, the spray system, which is fed from 2 cold legs, condenses steam in the vessel to prevent the pressurizer pressure from reaching the setpoint of the power operated relief valves for normal design transients. Heaters are energized on high water level during insurge to heat the subcooled surge water that enters the pressurizer from the reactor coolant loop.

SA533 Grade A Class 2 material was used in the Virgil C. Summer pressurizer. (SA533 Grade A Class 2 material was not used in primary side (RCPB) pressure retaining applications of the Virgil C. Summer steam generators.) The actual fracture toughness data for the SA533 Grade A Class 2 material in the pressurizer are tabulated in Table 5.5-17.

Material specifications are provided in Table 5.2-8 for the pressurizer, pressurizer relief tank, and the surge line. Design transients for the components of the RCS are discussed in Section 5.2.1.5. Additional details on the pressurizer design cycle analysis are given in Section 5.5.10.3.5.

The pressurizer nozzle weld overlays as noted in section 5.5.3.3.4 identified that the overlay weld filler material used was alloy 52M. This high chromium weld filler metal is compatible with the carbon steel, stainless steel and existing Inconel materials in the underlying components. This weld metal was applied utilizing a temperbead method to eliminate the need for pre and post weld heat treatment. RN 08-022

5.5.10.2.2.1 Pressurizer Support

The skirt type support is attached to the lower head and extends for a full 360 degrees around the vessel. The lower part of the skirt terminates in a bolting flange with bolt holes for securing the vessel to its foundation. The skirt type support is provided with ventilation holes around its upper perimeter to assure free convection of ambient air past the heater plus connector ends for cooling.

5.5.10.2.2.2 Pressurizer Instrumentation

Refer to Chapter 7 and Section 5.6 for details of the instrumentation associated with pressurizer pressure, level, and temperature.

5.5.10.2.2.3 Spray Line Temperatures

Temperatures in the spray lines from 2 loops are measured and indicated. Alarms from these signals are actuated by low spray water temperature. Alarm conditions indicate insufficient flow in the spray lines.

5.5.10.2.2.4 Safety and Relief Valve Leakage Monitoring

Monitoring of the status of the pressurizer safety and relief valves is by the following methods:

a. Temperature detectors in the piping downstream of the valves. (An increase in discharge line temperature is an indication of leakage through the valve.)

b. Acoustical type monitors to detect safety valve leakage.

c. Pressure/temperature/level of the pressurizer relief tank.

d. Valve limit switches on the pressurizer power operated relief valves which indicate valve open/closed position.

All 4 methods for detecting leakage through the pressurizer safety and relief valves are monitored in the control room.

5.5.10.3 <u>Design Evaluation</u>

5.5.10.3.1 System Pressure

Whenever a steam bubble is present within the pressurizer, RCS pressure is maintained by the pressurizer. A safety limit has been set to ensure that the RCS pressure does not exceed the maximum transient value allowed under the ASME Code, Section III, and thereby assure continued integrity of the RCS components.

Evaluation of plant conditions of operation which follow indicate that this safety limit is not reached.

During startup and shutdown, the rate of temperature change is controlled by the operator. Heatup rate is controlled by pump energy and by the pressurizer electrical heating capacity. This heatup rate takes into account the continuous spray flow provided to the pressurizer. When the reactor core is shutdown, the heaters are de-energized.

When the pressurizer is filled with water (i.e., during initial system heatup) and near the end of the second phase of plant cooldown, RCS pressure is maintained by the letdown flowrate via the Residual Heat Removal System.

5.5.10.3.2 Pressurizer Performance

The normal operating water volume at full load conditions is 60% of the free internal vessel volume. Under part load conditions, the water volume in the vessel is reduced for proportional reductions in plant load to 25% of free vessel volume at 0 power level. The various plant operating transients are analyzed and the design pressure is not exceeded with the pressurizer design parameters as given in Table 5.5-9.

5.5.10.3.3 Pressure Setpoints

The RCS design and operating pressure together with the safety, power relief and pressurizer spray valves setpoints, and the protection system setpoint pressures are listed in Table 5.2-7. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics.

5.5.10.3.4 Pressurizer Spray

Two (2) separate, automatically controlled spray valves with remote manual overrides are used to initiate pressurizer spray. In parallel with each spray valve is a manual throttle valve which permits a small continuous flow through both spray lines to reduce thermal stresses and thermal shock when the spray valves open and to help maintain uniform water chemistry and temperature in the pressurizer. Temperature sensors with low alarms are provided in each spray line to alert the operator to insufficient bypass flow. The layout of the common spray line piping to the pressurizer forms a water seal which prevents the steam buildup back to the control valves. The spray rate is selected to prevent the pressurizer pressure from reaching the operating setpoint of the power relief valves during a step reduction in power level of 10% of full load.

The pressurizer spray lines and valves are large enough to provide adequate spray using as the driving force the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg. The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force. The spray valves and spray line connections are arranged so that the spray will operate when 1 reactor coolant pump is not operating. The line may also be used to assist in equalizing the boron concentration between the reactor coolant loops and the pressurizer.

A flow path from the Chemical and Volume Control System to the pressurizer spray line is also provided. This additional facility provides auxiliary spray to the vapor space of the pressurizer during cooldown if the reactor coolant pumps are not operating. The thermal sleeves on the pressurizer spray connection and the spray piping are designed to withstand the thermal stresses resulting from the introduction of cold spray water.

5.5.10.3.5 Pressurizer Design Analysis

The occurrences for pressurizer design cycle analysis are defined as follows:

1. The temperature in the pressurizer vessel is always, for design purposes, assumed to equal saturation temperature for the existing RCS pressure, except in the pressurizer steam space subsequent to a pressure increase. In this case, the temperature of the steam space will exceed the saturation temperature since an isentropic compression of the steam is assumed.

The only exception of the above occurs when the pressurizer is filled solid during plant startup and cooldown.

2. The temperature shock on the spray nozzle is assumed to equal the temperature of the nozzle minus the cold leg temperature. The temperature shock on the surge nozzle is assumed to equal the pressurizer water space temperature minus the hot leg temperature.

3. Pressurizer spray is assumed to be initiated instantaneously to its design value as soon as the RCS pressure increases 40 psi above the nominal operating pressure. Spray is assumed to be terminated as soon as the RCS pressure falls 40 psi below normal operating pressure.

4. Unless otherwise noted, pressurizer spray is assumed to be initiated once per occurrence of each transient condition. The pressurizer surge nozzle is also assumed to be subject to 1 temperature transient per transient condition, unless otherwise noted.

5. At the end of each transient, except the faulted conditions, the RCS is assumed to return to a load condition consistent with the plant heatup transient.

6. Temperature changes occurring as a result of pressurizer spray are assumed to be instantaneous. Temperature changes occurring on the surge nozzle are also assumed to be instantaneous.

7. Whenever spray is initiated in the pressurizer, the pressurizer water level is assumed to be at the no load level.

5.5.10.4 Tests and Inspections

The pressurizer is designed and constructed in accordance with the ASME Code, Section III.

To implement the requirements of the ASME Code, Section XI, the following welds are designed and constructed to present a smooth transition surface between the parent metal and the weld metal. The path is ground smooth for ultrasonic inspection.

1. Support skirt to the pressurizer lower head.

2. Surge nozzle to the lower head.

3. Nozzles to the safety, relief, and spray lines.

4. Nozzle to safe end attachment welds.

5. All girth and longitudinal full penetration welds.

6. Manway attachment welds.

The liner within the safe end nozzle region extends beyond the weld region to maintain a uniform geometry for ultrasonic inspection.

The Pressurizer nozzles for the surge, spray, safety and relief lines had full structural weld overlays applied during Refuel 17. These weld overlays were designed to ensure that a smooth surface was applied such that the ultrasonic inspection volumetric requirements are maintained. The result of the weld overlays resulted in 100% ultrasonic inspection volumetric coverage of the respective nozzles and underlying welds and components. A Performance Demonstration Initiative (PDI) linear phased array inspection technique was used to perform the post weld overlay inspections. RN 08-022

Peripheral support rings are furnished for the removable insulation modules.

The pressurizer quality assurance program is given in Table 5.5-10.

5.5.11 PRESSURIZER RELIEF TANK

5.5.11.1 Design Bases

Design data for the pressurizer relief tank are given in Table 5.5-11. Codes and materials of the tank are given in Section 5.2.

The tank design is based on the requirement to absorb a discharge of pressurizer steam equal to 110% of the volume above the full power pressurizer water level setpoint. The tank is not designed to accept a continuous discharge from the pressurizer.

The volume of water in the tank is capable of absorbing the heat from the assumed discharge, assuming an initial temperature of 120°F and increasing to a final temperature of 200°F. If the temperature in the tank rises above 120°F during plant operation, the tank is cooled by spraying in cool water and draining out the warm mixture to the waste processing system.

5.5.11.2 Design Description

The pressurizer relief tank condenses and cools the discharge from the pressurizer safety and relief valves. Discharge from smaller relief valves located inside the containment is also piped to the relief tank. In addition, in the event that the reactor vessel head vent system is used, the pressurizer relief tank receives the discharge from the Reactor Coolant System. An itemized list identifying the discharges to the pressurizer relief tank is provided in Table 5.2-6. The tank normally contains water and a predominantly nitrogen atmosphere; however, provision is made to permit the gas in the tank to be periodically analyzed to monitor the concentration of hydrogen and/or oxygen.

By means of its connection to the waste processing system, the pressurizer relief tank provides a means for removing any noncondensable gases from the RCS which might collect in the pressurizer vessel.

Steam is discharged through a sparger pipe under the water level. This arrangement provides for condensing and cooling the steam by mixing it with water that is near ambient temperature. The tank is also equipped with an internal spray and a drain which are used to cool the tank following a discharge. A flanged nozzle is provided on the tank for the pressurizer discharge line connection to the sparger pipe.

5.5.11.2.1 Pressurizer Relief Tank Pressure

The pressurizer relief tank pressure transmitter provides an indication of pressure relief tank pressure. An alarm is provided to indicate high tank pressure.

5.5.11.2.2 Pressurizer Relief Tank Level

The pressurizer relief tank level transmitter supplies a signal for an indicator with high and low level alarms.

5.5.11.2.3 Pressurizer Relief Tank Water Temperature

The temperature of the water in the pressurizer relief tank is indicated, and an alarm actuated by high temperature informs the operator that cooling to the tank contents is required.

5.5.11.3 Design Evaluation

The volume of water in the tank is capable of absorbing a discharge of 110% of the pressurizer steam volume above the full power water level setpoint. Water temperature in the tank is maintained at the nominal containment temperature.

The rupture disks on the relief tank have a relief capacity equal to or greater than the combined capacity of the pressurizer safety valves. The tank design pressure is twice the calculated pressure resulting from the maximum design safety valve discharge described above. The tank and rupture disc holders are also designed for full vacuum to prevent tank collapse if the contents cool following a discharge without nitrogen being added.

The discharge piping from the safety and relief valves to the relief tank is sufficiently large to prevent backpressure of the safety valves from exceeding 20% of the setpoint pressure at full flow.

5.5.12 VALVES

5.5.12.1 Design Bases

As noted in Section 5.2, all valves out to and including the second valve normally closed or capable of automatic or remote closure, larger than 3/4 inch, are Safety Class 1, and ASME III, Code Class 1 valves. All 3/4 inch or smaller valves in liquid filled lines connected to the RCS are Class 2 since the interface with the Class 1 piping is provided with suitable orificing for such valves. Pressurizer instrument root valves connected to the steam space above the normal water level have been upgraded to Safety Class 1, since failure of these valves could cause automatic operation of the ECCS systems. Design data for the RCS valves are given in Table 5.5-12. RN 07-012

To ensure that the valves meet the design objectives, the materials specified for construction must minimize corrosion/erosion and ensure compatibility with the environment. Leakage is minimized to the extent practicable by design and Class 1 stresses are maintained within the limits of the ASME Code, Section III and the limits specified in Section 5.5.1.

5.5.12.2 Design Description

Valves in the RCS which are in contact with the coolant are constructed primarily of stainless steel. Other materials in contact with the coolant, such as for hard surfacing and packing, are special materials.

Manual and motor operated valves of the RCS which are 3 inches and larger are provided with double-packed stuffing boxes and intermediate lantern ring leakoff connections. Throttling control valves, regardless of size, are provided with double-packed stuffing boxes and with stem leakoff connections. Leakoff connections on valves, normally operated in a radioactive fluid, are piped to a closed collection system. Leakage to the atmosphere is essentially 0 for these valves.

Gate valves at the engineered safety features interface are wedge design and are essentially straight through. The wedges are flex-wedge or solid. All gate valves have backseats. Globe valves are "T" and "Y" style. Check valves are swing type for sizes 2-1/2 inches and larger. All check valves which contain radioactive fluid are stainless steel and do not have body penetrations other than the inlet, outlet, and bonnet. The check hinge is serviced through the bonnet.

The accumulator check valve is designed such that at the required flow the resulting pressure drop is within the specified limits. All operating parts are contained within the body. The disc has limited rotation to provide a change of seating surface and alignment after each valve opening.

5.5.12.3 Design Evaluation

The design/analysis requirements for Class 1 valves, as discussed in Section 5.2, limit stresses to levels which ensure the structural integrity of the valves. In addition, the testing programs described in Section 3.9.2.4 demonstrate the ability of the valves to operate as required during anticipated and postulated plant conditions.

Reactor coolant chemistry parameters are specified in the design specifications to assure the compatibility of valve construction materials with the reactor coolant.

5.5.12.4 Tests and Inspections

RCS valves are tested in accordance with the requirements of the ASME Code, Section III. The tests and inspections discussed in Section 3.9.2.4 are performed to ensure the operability of active valves. In-place operational testing is performed on valves as required by the Technical Specifications. RN 04-012

There are no full penetration welds within valve body walls. Valves are accessible for disassembly and internal visual inspection to the extent practical. Plant layout configurations determine the degree of inspectability. The valve quality assurance program is given in Table 5.5-13.

Inservice inspection is discussed in Section 5.2.8.

5.5.13 SAFETY AND RELIEF VALVES

5.5.13.1 Design Bases

The combined capacity of the pressurizer safety valves is designed to accommodate the maximum surge resulting from complete loss of load. This objective is met without reactor trip or any operator action by the opening of the steam safety valves when steam pressure reaches the steam side safety setting.

The pressurizer power operated relief valves are designed to limit pressurizer pressure to a value below the fixed high pressure reactor trip setpoint for most transients. Conservative analyses indicate that, for the 95% load rejection transient, the high pressure reactor trip setpoint would be reached. No credit is taken for automatic operation of these valves to mitigate the consequences of an accident. They are designed to fail to the closed position on loss of air supply.

5.5.13.2 Design Description

The pressurizer safety valves are totally enclosed pop type. The valves are spring loaded, self-actuated, and with backpressure compensation features.

The pipe connecting the pressurizer nozzles to their respective code safety valves are shaped in the form of a loop. The low point of the loops is drained continuously to the

pressurizer liquid space to remove any condensation formed in the safety valve inlet pipes.

The pressurizer power operated relief valves are pneumatic actuated valves which respond to a signal from a pressure sensing system or to manual control. Remotely operated stop valves are provided to isolate the power operated relief valves if excessive leakage develops.

Pressurizer safety and relief valve leakage monitoring is discussed in Section 5.5.10.2.2.4. Refer to Section 5.2 for a discussion of discharge line supports.

Design parameters for the pressurizer spray control, safety, and power relief valves are given in Table 5.5-14.

5.5.13.3 Design Evaluation

The pressurizer safety valves prevent RCS pressure from exceeding 110% of system design pressure, in compliance with the ASME Nuclear Power Plant Components Code.

The pressurizer power relief valves prevent actuation of the fixed reactor high pressure trip for all design transients except the design step load decreases with steam dump. For this event, peak pressure will exceed the trip setpoint but remains below the safety valve setpoint. The relief valves limit undesirable opening of the spring-loaded safety valves. Note that setpoint studies to date indicate that the pressure rise in a 3-loop plant for the design step load decrease of 10% from full power is limited to 60 psi. These studies also indicate that the design step load decrease of 10% under N-1 loop operation is limited to approximately 50 psi. In both cases, the pressure rise is not sufficient to actuate the power operated relief valves, and thus, this design is conservative.

5.5.13.4 Tests and Inspections

All safety and relief valves are subjected to hydrostatic tests, seat leakage tests, operational tests, and inspections as required. For safety valves that are required to function during a faulted condition, additional tests are performed. These tests are described in Section 3.9.2.4.

There are no full penetration welds within the valve body walls. Valves are accessible for disassembly and internal visual inspection.

5.5.14 COMPONENT SUPPORTS

Component supports allow virtually unrestrained lateral thermal movement of the loops during plant operation and provide restraint to the loops and components during accident conditions. The loading combinations and design stress limits are discussed in Section 5.2.1.10. The design maintains the integrity of the Reactor Coolant System boundary for normal and accident conditions and satisfies the requirements of the piping code. Results of piping and supports stress evaluation will be presented in Section 5.2.1.10. The normal operating temperatures for the Reactor Coolant System supports are 50°F to 400°F.

5.5.14.1 Description

The support structures are of welded steel construction and are either linear type or plate and shell type. Vessel skirts and saddles are fabricated from plate and shell elements to accommodate a biaxial stress field. Linear supports are tension and compression struts, beams and columns. Attachments are of integral and nonintegral types. Integral attachments are welded, cast, or forged to the pressure boundary component by lugs, shoes, rings, and skirts. Nonintegral attachments are bolted, pinned, or bear on the pressure boundary component.

The supports permit unrestrained thermal growth of the supported systems, but restrain vertical, lateral, and rotational movement resulting from seismic and pipe break loadings. This is accomplished using pin-ended columns for vertical support and girders, bumper pedestals, and tie rods for lateral support. 02-01

Shimming and grouting enable adjustment of all support elements during erection to achieve correct fit-up and alignment.

1. Vessel

   Supports for the reactor vessel (see Figure 5.5-7) are individual air cooled rectangular box structures beneath the vessel nozzles bolted to the primary shield wall concrete. Each box structure consists of a horizontal top plate that receives loads from the reactor vessel shoe, a horizontal bottom plate supported by and transferring loads to the primary shield wall concrete, and connecting vertical plates. The supports are air cooled to maintain the supporting concrete temperature within acceptable levels.

2. Steam Generator

The lower supports for the steam generator (see Figure 5.5-8) consist of 4 vertical pin-ended columns bolted to the bottom of the steam generator support pads; and lateral support girders and pedestals that bear against horizontal bumper blocks bolted to the side of the generator support pads. The upper lateral steam generator support consists of a ring girder around the generator shell supported by struts. Loads are transferred from the equipment to the ring girder by means of a number of bumper blocks between the girder and generator shell. 02-01

3. Pump

The reactor coolant pump supports (see Figure 5.5-9) consist of 3 pin-ended structural steel columns and 3 lateral tie bars. A large diameter bolt connects each column and tie rod to a pump support pad. The outer ends of all 3 tie rods have slotted pin holes to permit unrestrained lateral movement of the pump during plant heatup and cooldown, but provide lateral restraint for accident loadings.

4. Pressurizer

The pressurizer (see Figure 5.5-10) is supported at its base by bolting the flangering to the supporting concrete slab. In addition, upper lateral support is provided near the vessel center of gravity by 4 "V frames" or struts extending horizontally from the compartment walls and bearing against the vessel lugs.

5.5.14.2 Evaluation

Detailed evaluation ensures the design adequacy and structural integrity of the reactor coolant loop and the primary equipment supports system. This detailed evaluation is made by comparing the analytical results with established criteria for acceptability. Structural analyses are performed to demonstrate design adequacy for safety and reliability of the plant in case of a large or small seismic disturbance and/or loss of coolant accident conditions. Loads that the system is designed to encounter, of 10 during its lifetime (thermal, weight, pressure, and operating basis earthquake), are applied and stresses are compared to allowable values, as described in Section 5.2.1.10.

All Reactor Coolant System (RCS) supports are designed for concurrent loadings from deadweight, pressure, postulated pipe ruptures in the RCS as identified in WCAP-8082 and WCAP-13206, and the safe shutdown earthquake. As indicated in Section 3.6, postulated breaks in the reactor coolant loop piping, except for branch line connections, have been eliminated. Reactor coolant loop piping branch nozzle (i.e., accumulator connection, pressurizer surge line, residual heat removal, etc.) breaks are postulated. Additionally, steam line breaks, and safety and relief line breaks are considered in the design of steam generator and pressurizer supports respectively. Included under the considerations for pipe rupture loads are the effects of RCS hydraulic loads, jet impingement, thrust, and asymmetric pressures about the supported components.

The pipe rupture analysis which is performed for the reactor vessel support loads includes nonaxisymmetric pressure distributions on the internals.

A detailed dynamic model of the reactor vessel and internals includes the stiffnesses of the reactor vessel support and the attached piping. Hydraulic forces are developed in the internals for the postulated break; these forces are characterized by time dependent forcing functions on the vessel and core barrel. In the derivation of these forcing functions, the fluid-structure (or hydroelastic) interaction in the downcomer region between the barrel and the vessel is taken into account. As a result of the loop branch pipe breaks, loop mechanical loads are also applied to the vessel. The loads from these 3 sources, the internals reactions, reactor cavity pressure loads, and the loop mechanical forces are applied simultaneously in a nonlinear elastic dynamic time history analysis on the model of the vessel, supports, and internals. The results of this analysis are the dynamic loads on the reactor vessel supports and vessel time history displacements. The maximum loads are combined with other applicable loads, such as seismic and deadweight and applied statically to the vessel support structure.

Although considered in the design, it is expected on the basis of past experience that asymmetric pressure loadings on the major components will be negligible.

For public health and safety, the safe shutdown earthquake and design basis loss of coolant accident, resulting in a rapid depressurization of the system, are required design condition. For these loadings, the basic criteria ensure that the severity of the accident will not be increased, thus maintaining the system in a safe condition. The rupture of a reactor coolant loop pipe will not violate the integrity of the unbroken leg of the loop. To ensure the integrity and stability of the reactor coolant loop support system and a safe shutdown of the system under loss of coolant accident and the worst combined (Normal + SSE + LOCA) loadings, the stresses in the unbroken piping of a broken loop, the unbroken loop piping, and the supports system are analyzed. The results of design analysis are provided in Section 5.2.1.10.

5.5.14.3 Tests and Inspections

The design and fabrication is specified in accordance with the AISC Specifications for the "Design, Fabrication, and Erection of Structural Steel for Buildings," 1969 Edition and applicable portions of the ASME Boiler and Pressure Vessel Code. Welder qualifications, welding procedures, and inspection of welded joints is specified to be in accordance with Section IX of the ASME Code.

5.5.15 REACTOR VESSEL HEAD VENT SYSTEM

5.5.15.1 Design Basis

The basic function of the Reactor Vessel Head Vent System (RVHVS) is to remove non-condensable gases or steam from the reactor vessel head. This system is designed to mitigate a possible condition of inadequate core cooling or impaired natural circulation resulting from the accumulation of noncondensable gases in the RCS. The design of the RVHVS is in accordance with the requirements of NUREG-0578 and subsequent definitions and clarifications (References [2] and [3]).

5.5.15.2 Design Description and Evaluation

5.5.15.2.1 General Description

The RVHVS is designed to remove non-condensable gases or steam from the Reactor Coolant System via remote manual operations from the control room. The system discharges to the pressurizer relief tank. The RVHVS is designed to vent a volume of hydrogen, noncondensable gases, etc., at system design pressure and temperature approximately equivalent to 1/2 of the Reactor Coolant System volume in 1 hour.

An H_2 burn from a 100% zircaloy/H_2O reaction has been addressed and it has been determined that containment integrity would not be breached. Therefore, contained venting outside containment is not considered necessary.

The flow diagram of the RVHVS is shown in Figure 5.5-13. The RVHVS consists of 2 parallel flow paths with redundant isolation valves in each flow path. The venting operation uses only 1 of the flow paths at any time.

The physical layout of the RCS hot leg piping is such that its entire volume can be vented via the RVHV system.

The equipment design parameters are listed in Table 5.5-16.

As indicated above, normally the venting from the RVHVS is contained by the pressurizer relief tank. However, venting to containment could occur if the rupture disc ruptures. In that case, the location of the PRT is such that excellent gas communication exists within the secondary shield area that any gas that escapes from the PRT will be readily mixed with the containment atmosphere with additional mixing being promoted by the Reactor Building Ventilation System.

The active portion of the system consists of 4 two-inch motor operated isolation valves connected to the reactor vessel head vent pipe. The isolation valves in series in each flow path are powered by opposite vital power supplies. The isolation valves are fail as is, active valves. One (1) normally closed isolation valve and 1 normally open valve are located in each flow path. Leakage past the vent valves during normal plant operation is detected by the pressurizer relief tank instrumentation. All of the isolation valves are qualified to IEEE - 323-1974, 344-1975, and 382-1972 and to the requirements of Regulatory Guide 1.48 as described in Appendix 3A. RN 14-037

If one single active failure prevents a venting operation through 1 flow path, the redundant path is available for venting. Similarly, the 2 isolation valves in each flow path provide a single failure method of isolating the venting system. With 2 valves in series, the failure of any 1 valve or power supply will not inadvertently open a vent path. Thus, the combination of safety grade train assignments and valve failure modes will not prevent vessel head venting nor venting isolation with any single active failure.

The RVHVS has 2 normally de-energized valves in series in each flow path. Power lockout capability to all 4 isolation valves is provided by administrative control at the motor control load center. This arrangement eliminates the possibility of a spuriously opened flow path.

The system is operated from the control room. The position indication from each valve is monitored in the control room by status lights. Position indication is unavailable in the control room when the motor control breaker is open.

The RVHVS is connected to the head vent pipe as shown on Figure 5.5-13. The system is orificed to limit the blowdown from a break downstream of either of the orifices to within the capacity of 1 of the centrifugal charging pumps.

A break of the RVHVS line upstream of the orifices would result in a small LOCA of not greater than 1 inch diameter. Such a break is similar to those analyzed in WCAP-9600 (Reference [4]). Since a break in the head vent line would behave similarly to the hot leg break case presented in WCAP-9600, the results presented therein are applicable to a RVHVS line break. This postulated vent line break, therefore, results in no calculated core uncovery.

All piping and equipment from the vessel head vent up to and including the second isolation valve in each flowpath are designed and fabricated in accordance with ASME, Section III, Class 1 requirements. The remainder of the piping and equipment is non-nuclear safety, but is seismically supported up to the 12" pressurizer relief line.

The system provides for venting the reactor vessel head by using only safety grade equipment. The RVHVS satisfies applicable requirements and industry standards including ASME Code classification, safety classification, single-failure criteria, and environmental qualification.

5.5.15.2.2 Supports

The vent system piping is supported to ensure that the resulting loads and stresses on the piping and on the vent connection to the vessel head are acceptable.

The support design for attaching the head vent system piping to the reactor vessel head lifting leg is shown in Figure 5.5-14. The support is a 2-part clamp configuration, called a double bolt riser clamp. The clamp and associated bolts, nuts, spacers, and washers are made of stainless steel. A gap exists between the 1 inch head vent pipe and the support clamp to allow for thermal expansion in the vertical direction.

The support design for attaching the head vent system piping to the CRDM seismic support platform is shown in Figure 5.5-15. This support is a 2-part clamp configuration, called a double bolt clamp bracket. This clamp support is used to rigidly support the piping in the radial direction. The clamp and associated bolts, nuts, spacers, and washers are made of stainless steel, with high strength hold down bolts threaded into the deck of the CRDM Seismic support platform. A gap exists between the one inch head vent pipe and the support clamp to allow for thermal expansion in the axial direction.

All supports and support structures comply with the requirements of the AISC Code, Part II.

5.5.15.3 Analytical Considerations

The analysis of the reactor vessel head vent piping is based on the following plant operation conditions defined in the ASME Code Section III:

1. Normal Condition:

   Pressure, deadweight, and thermal expansion analysis of the vent pipe during

   a) normal reactor operation with the 2 inboard vent isolation valves closed and

   b) post-refueling venting.

2. Upset Condition:

   Loads generated by the operating basis earthquake (OBE) response spectra.

3. Faulted Condition:

   Loads generated by the safe shutdown earthquake (SSE) and by valve thrust during venting. In accordance with ASME Code Section III, faulted conditions are not included in fatigue evaluations.

The Class I piping used for the reactor vessel head vent is one inch schedule 160 and, therefore, in accordance with ASME Code Section III, is analyzed following the procedures of NC-3600 for Class II piping.

For plant operating conditions listed above, the piping stresses are shown to meet the requirements of equations (8), (9), (10), or (11) of ASME III, Section NC-3600 with a design temperature of 650°F and a design pressure of 2485 psig.

5.5.16 REFERENCES

1. "Reactor Coolant Pump Integrity in LOCA," WCAP-8163, September, 1973.

2. Letter from D. B. Vassallo (NRC) to all Applicants for an Operating License, "Followup Actions Resulting From the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident," and Enclosure 4: Installation of Remotely Operated High Point Vents in the Reactor Coolant System, September 27, 1979.

3. Letter from D. B. Vassallo (NRC) to all Applicants for an Operating License, "Discussion of Lessons Learned Short Term Requirements," Enclosure 1, pp. 44-49, Reactor Coolant System Venting, November 9, 1979.

4. "Report on Small Break Accidents for Westinghouse NSSS System," WCAP-9600, June 1979, (specifically Case F, Section 3.2).

5. "Basis for Heatup and Cooldown Limit Curves," WCAP-7924-A, April 1975.

6. WCAP-13480, Revision 1, "Westinghouse Delta-75 Steam Generator Design and Fabrication Information for the V. C. Summer Nuclear Station," October 1993.

TABLE 5.5-1

REACTOR COOLANT PUMP DESIGN PARAMETERS

Unit Design Pressure, psig	2485
Unit Design Temperature, °F	650[(1)]
Unit Overall Height, ft	26
Seal Water Injection, gpm	8
Seal Water Return, gpm	3
Cooling Water Flow, gpm	195
Maximum Continuous Cooling Water Inlet Temperature, °F	105

Pump

Capacity, gpm	100,700
Developed Head, ft	279
NPSH Required, ft	Figure 5.5-2
Suction Temperature, °F	554.6
Pump Discharge Nozzle, Inside Diameter, in	27-1/2
Pump Suction Nozzle, Inside Diameter, in	31
Speed, rpm	1180
Water Volume, ft^3	81
Weight (dry), lbs.	199,000

Motor

Type	Drip proof, squirrel cage induction air-cooled
Power, Hp	7000
Voltage, volts	6900
Phase	3
Frequency, Hz	60
Insulation Class	Class F, thermoplastic epoxy insulation
Starting	
Current	3000 amp @ 6900 volts
Input, hot reactor coolant	501 amp
Input, cold reactor coolant	635 amp

TABLE 5.5-1 (Continued)

REACTOR COOLANT PUMP DESIGN PARAMETERS

Motor (Continued)

Pump Moment of Inertia, lb-ft^2 maximum	
Flywheel	70,000
Motor	22,500
Shaft	520
Impeller	1,980

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(1) Design temperature of pressure retaining parts of the pump assembly exposed to the reactor coolant and injection water on the high pressure side of the controlled leakage seal shall be that temperature determined for the parts for a primary loop temperature of 650°F.

TABLE 5.5-2

REACTOR COOLANT PUMP QUALITY ASSURANCE PROGRAM

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

(1) RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle

TABLE 5.5-3

STEAM GENERATOR DESIGN DATA

Design Pressure, reactor coolant side, psig	2485	
Design Pressure, steam side, psig	1185	
Design Temperature, reactor coolant side, °F	650	
Design Temperature, steam side, °F	600	
Total Heat Transfer Surface Area, ft^2	75,185	99-01
Maximum Moisture Carryover, wt percent	0.10	
Overall Height, ft-in	67-8	
Number of U-Tubes	6307	
U-Tube Outer Diameter, in	0.688	
Tube Wall Thickness, nominal, in	0.040	
Number of Manways	4	
Inside Diameter of Manways, in	16	
Number of Handholes	8	
Inside Diameter of Handholes, in	4" and 6"	
Design Fouling Factor hr-ft^2-°F/BTU	0.00011	99-01

TABLE 5.5-4

STEAM GENERATOR QUALITY ASSURANCE PROGRAM

	RT(1)	UT(1)	PT(1)	MT(1)	ET(1)
Tubesheet					
1. Forging		Yes		Yes	
2. Cladding		Yes(2)	Yes(3)		
Channel Head					
1. Casting	Yes			Yes	
2. Cladding			Yes		
Secondary Shell & Head					
1. Plates		Yes			
Tubes		Yes			Yes
Nozzles (Forgings)		Yes		Yes	
Weldments					
1. Shell, longitudinal	Yes			Yes	
2. Shell, circumferential	Yes			Yes	
3. Cladding (channel head-tubesheet joint cladding restoration)			Yes		
4. Steam and feedwater nozzle to shell	Yes			Yes	
5. Support brackets				Yes	
6. Tube to tubesheet			Yes		
7. Instrument connections (primary and secondary)				Yes	
8. Temporary attachments after removal				Yes	
9. After hydrostatic test (all welds and complete channel head - where accessible)				Yes	

TABLE 5.5-4 (Continued)

STEAM GENERATOR QUALITY ASSURANCE PROGRAM

	RT(1)	UT(1)	PT(1)	MT(1)	ET(1)
Weldments (Continued)					
10. Nozzle safe ends (if forgings)	Yes		Yes		
11. Nozzle safe ends (if weld deposit)			Yes		

(1) RT- Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle
ET - Eddy Current

(2) Flat Surfaces Only

(3) Weld Deposit Areas Only

TABLE 5.5-5
REACTOR COOLANT PIPING DESIGN PARAMETERS

Reactor Inlet Piping, inside diameter, in	27-1/2
Reactor Inlet Piping, nominal wall thickness, in	2.32
Reactor Outlet Piping, inside diameter, in	29
Reactor Outlet Piping, nominal wall thickness, in	2.45
Coolant Pump Suction Piping, inside diameter, in	31
Coolant Pump Suction Piping, nominal wall thickness, in	2.60
Pressurizer Surge Line Piping, nominal pipe size, in	14
Pressurizer Surge Line Piping, nominal wall thickness, in	1.406
Reactor Coolant Loop Piping	
Design/Operating Pressure, psig	2485/2235
Design Temperature, °F	650
Pressurizer Surge Line	
Design Pressure, psig	2485
Design Temperature, °F	680
Pressurizer Safety Valve Inlet Line	
Design Pressure, psig	2485
Design Temperature, °F	680
Pressurizer (Power-Operated) Relief Valve Inlet Line	
Design Pressure, psig	2485
Design Temperature, °F	680
Pressurizer Relief Tank Inlet Line	
Design Pressure, psig	600
Design Temperature, °F	600

TABLE 5.5-6

REACTOR COOLANT PIPING QUALITY ASSURANCE PROGRAM

	RT(1)	UT(1)	PT(1)
	Yes		Yes
Fittings and Pipe (Forgings)		Yes	Yes
Weldments			
1. Circumferential	Yes		Yes
2. Nozzle to runpipe (Except no RT for nozzles less than 6 inches)	Yes		Yes
3. Instrument Connections			Yes
Castings	Yes		Yes (after finishing)
Forgings		Yes	Yes (after finishing)

(1) RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant

TABLE 5.5-7

DESIGN BASES FOR RESIDUAL HEAT REMOVAL SYSTEM OPERATION

Residual Heat Removal System Startup	~ 4 hours after Reactor Shutdown.
Reactor Coolant System Initial Pressure, psig	~ 425
Reactor Coolant System Initial Temperature, °F	~ 350
Component Cooling Water Design Temperature, °F	105
Cooldown Time, Hours After Initiation of Residual Heat	~ 16 [(1)]
Removal System Operation (CCWS Supply Temperature not exceeding 120 °F)	~ 20 [(2)]
Reactor Coolant System Temperature At End of Cooldown, °F	140
Decay Heat Generation at 20 Hours After Reactor Shutdown, BTU/hr	60.6×10^6 [(1)]
Decay Heat Generation at 24 Hours After Reactor Shutdown, BTU/hr	60.6×10^6 [(2)]

(1) Core Power = 2775 MWt
(2) Core Power = 2900 MWt

TABLE 5.5-8

RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DATA

Residual Heat Removal Pumps

Number	2
Design Pressure, psig	600
Design Temperature, °F	400
Design Flow, gpm	3750
Design Head, ft	240
NPSH at 3750 gpm, ft	14
Motor Power, HP	300

Residual Heat Exchangers

Number	2	
Design Heat Removal Capacity	30.3×10^6 BTU/hr	
	Tube Side	Shell Side
Design Pressure, psig	600	150
Design Temperature, °F	400	200
Design Flow, lb/hr	1.86×10^6	2.8×10^6
Inlet Temperature, °F	139	105
Outlet Temperature, °F	123	116
Material	Austenitic Stainless Steel	Carbon Steel
Fluid	Reactor Coolant	Component Cooling Water

TABLE 5.5-9

PRESSURIZER DESIGN DATA

Design Pressure, psig	2485
Design Temperature, °F	680
Surge Line Nozzle Diameter, in	14
Heatup Rate of Pressurizer Using Heaters Only, °F/hr	55
Internal Volume, ft^3	1400

TABLE 5.5-10

PRESSURIZER QUALITY ASSURANCE PROGRAM

	RT(1)	UT(1)	PT(1)	MT(1)	
Heads					
1. Plates		Yes			
2. Cladding			Yes		
Shell					
1. Plates		Yes			02-01
2. Cladding			Yes		
Heaters					
1. Tubing (2)		Yes	Yes		
2. Centering of element	Yes				
Nozzle (Forgings)		Yes	Yes (3)	Yes (3)	
Weldments					
1. Shell, longitudinal	Yes			Yes	
2. Shell, circumferential	Yes			Yes	
3. Cladding			Yes		
4. Nozzle Safe End (if forging)	Yes		Yes		02-01
5. Instrument Connection			Yes		
6. Support Skirt		Yes		Yes	
7. Temporary Attachments (after removal)				Yes	
8. All external pressure boundary welds after shop hydrostatic test				Yes	

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(1) RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle 02-01

(2) Or a UT and ET

(3) MT or PT

TABLE 5.5-11

PRESSURIZER RELIEF TANK DESIGN DATA

Design Pressure, psig		100
Rupture Disc Release Pressure, psig	Nominal:	91
	Range:	86-100
Design Temperature, °F		340
Total Rupture Disc Relief Capacity, lb/hr at 100 psig		1.6×10^6

TABLE 5.5-12

REACTOR COOLANT SYSTEM VALVE DESIGN PARAMETERS

Design/Normal Operating Pressure, psig	2485/2235
Preoperational Plant Hydrotest, psig	3107
Design Temperature, °F	650

TABLE 5.5-13

REACTOR COOLANT SYSTEM VALVES QUALITY ASSURANCE PROGRAM

	RT[1]	UT[1]	PT[1]
Boundary Valves, Pressurizer Relief and Safety Valves			
Castings - (larger than 4 inches)	Yes		Yes
(2 inches to 4 inches)	Yes[2]		Yes
Forgings - (larger than 4 inches)	[3]	[3]	Yes
(2 inches to 4 inches)			Yes

(1) RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
(2) Weld Ends Only
(3) Either RT or UT

TABLE 5.5-14

PRESSURIZER VALVES DESIGN PARAMETERS

Pressurizer Spray Valves

Number	2
Design Pressure, psig	2485
Design Temperature, °F	650
Design Flow, for valves full open, each, gpm	350

Pressurizer Safety Valves

Number	3
Maximum Relieving Capacity, ASME rated flow, lb/hr	420,000
Set Pressure, psig	2485
Design Temperature, °F	650
Fluid	Saturated steam
Transient Condition, °F	(Superheated steam) 680
Backpressure:	
Normal, psig	3 to 5
Expected during discharge, psig	350

Pressurizer Power Relief Valves

Number	3
Design Pressure, psig	2485
Design Temperature, °F	650
Relieving Capacity, at 2350 psig, lb/hr (per valve)	210,000
Fluid	Saturated steam
Transient Condition, °F	(Superheated steam) 680

TABLE 5.5-15

SINGLE ACTIVE FAILURE ANALYSES OF THE RESIDUAL HEAT REMOVAL SYSTEM

	Component	Malfunction	Comments
1.	RHR suction isolation valve (8701 A or B or 8702 A or B)	Fails to open	RHR cooling restricted to one train. Cooldown time extended.
2.	RWST/RHR suction isolation valve (8809 A or B)	Fails to close	RHR cooling restricted to one train. Cooldown time extended.
3.	RHR pump (no. 1 or 2)	Fails to operate	RHR cooling restricted to one train. Cooldown time extended.
4.	Flow instrument (602A or 602B)	Fails low	Miniflow valve (FCV 602A or FCV 602B) opens. Slight increase in RHR pump flow in one train. Slight extension in cooldown time.
5.	Miniflow valve (FCV 602A or FCV 602B)	Fails to close	Slight increase in RHR pump flow in one train. Slight extension in cooldown time.
6.	Flow instrument (605A or 605B)	Fails low	Bypass valve opens. Cooldown in one train reduced. Cooldown time extended.
7.	Flow instrument (605A or 605B)	Fails high	Bypass valve closes. Constant total return flow in one train to the RCS cannot be maintained. No adverse effects.
8.	Bypass valve (FCV 605A or FCV 605B)	Fails to close	Cooling in one train reduced. Cooldown time extended.

TABLE 5.5-15 (Continued)

SINGLE ACTIVE FAILURE ANALYSES OF THE RESIDUAL HEAT REMOVAL SYSTEM

	Component	Malfunction	Comments
9.	Bypass valve (FCV 605A or FCV 605B)	Fails to open	Constant total return flow in one train to the RCS cannot be maintained. No adverse effects.
10.	RHR heat exchanger (no. 1 or 2)	Fails to function	RHR cooling restricted to one train. Cooldown time extended.
11.	Flow control valve (HCV 603A or HCV 603B)	Fails to open	RHR cooling restricted to one train. Cooldown time extended.
12.	Flow control valve (HCV 603A or HCV 603B)	Fails open	Cannot control cooldown rate in one train, thus, requiring that cooling be restricted to the other train. Cooldown time extended.

See Sections 5.5.7.3.3 and 5.5.7.3.4, respectively, for a discussion of Overpressurization Protection and Prevention of Exposure of the RHR System to Normal RCS Pressures.

TABLE 5.5-16

REACTOR VESSEL HEAD VENT SYSTEM EQUIPMENT DESIGN PARAMETER

Valves

Number (include two manual valves)	6
Design pressure, psig	2485
Design temperature,°F	650

Piping

Normal vent line nominal diameter, in.	1 - 3/4
RVHVS flow path line, nominal diameter, in	1
Design pressure, psig	2485
Design temperature, °F	650

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Amendment 02-01

TABLE 5.5-17

FRACTURE TOUGHNESS DATA FOR SA533 GRADE A CLASS 2 MATERIAL

Component	Component Part	Test Number	Charpy V-Notch (ft-lb)	Lateral Expansion (in)	Orientation Longitudinal (L) Transverse (T)	Test Temperature (°F)	T_{NDT} (°F)	RT_{NDT} (°F)
Pressurizer (1581)	Lower Head	T03639	82, 85, 80 90, 90, 78	.074, .071, .068 .063, .072, .072	T L	70 10	10	10
Pressurizer (1581)	Upper Head	T03625	53, 54, 53 75, 74, 91	.052, .051, .048 .060, .070, .071	T L	70 10	10	10
Pressurizer (1581)	Shell Barrel	T03328	78, 82, 81	.056, .057, .049	L	10	-*	-
Pressurizer (1581)	Shell Barrel	T03704	65, 59, 56	.051, .056, .059	T	70	0	10
Pressurizer (1581)	Shell Barrel	T03605	78, 78, 85	.065, .071, .066	T	70	10	10

02-01

* Drop weight test results not available.

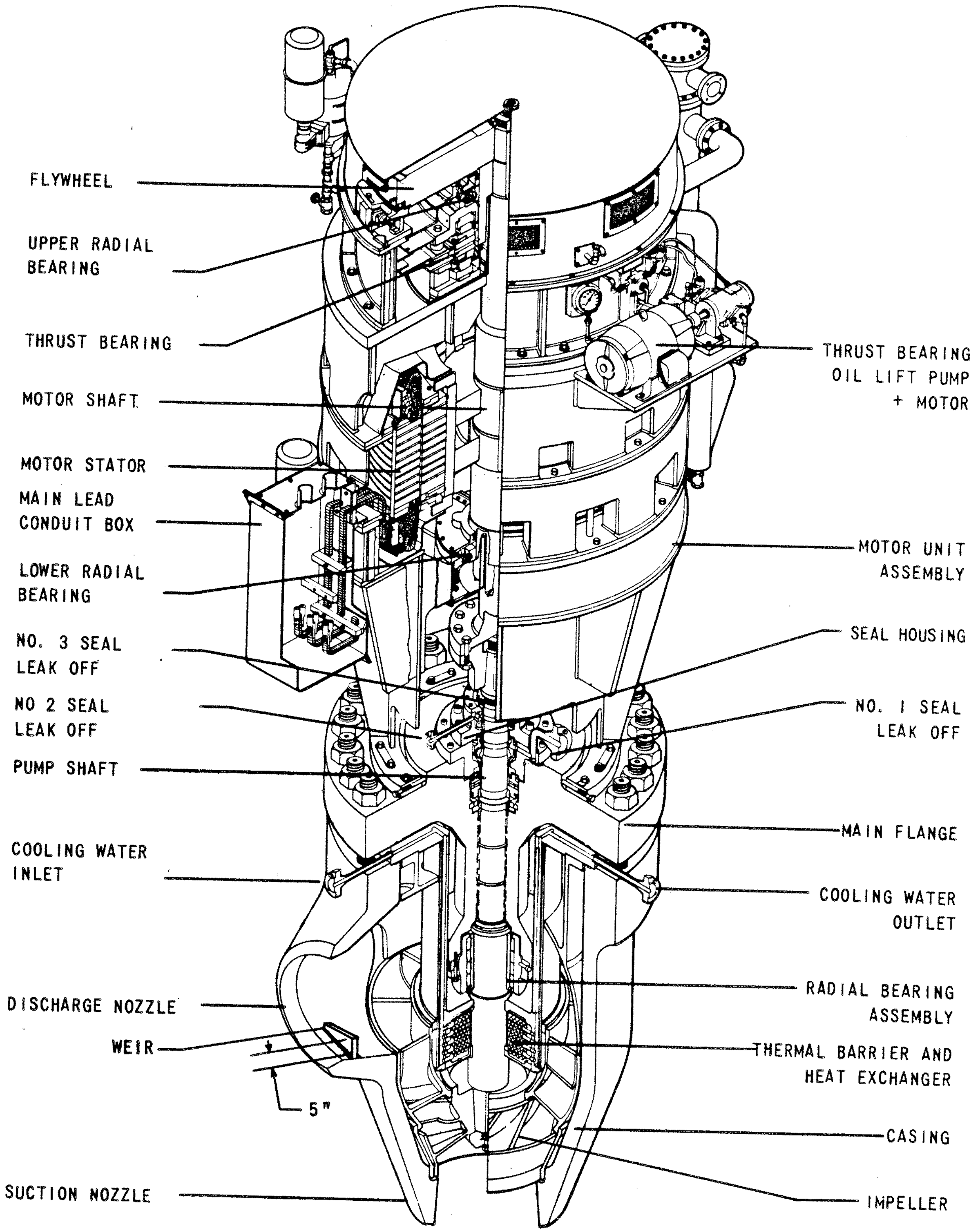
TABLE 5.5-18

PRESSURIZER WELD DATA

Material	Weld Process	Test Number Electrode	Test Number Flux	Drop WT @ 20 °F[1]	Charpy V-Notch FT-LB[1]	TestTemp. (°F)	Lat. Expansion (Mils)	RT_{NDT} °F
SFA 5.18	SAW	2755	3970	2 NB	130-127-134	70	59-57-61	10
SFA 5.18	SAW	3864	3970	2 NB	121-111-130	70	80-71-73	10
SFA 5.18	SAW	4113	4553	2 NB	111-114-112	70	84-78-79	10
SFA 5.18	SAW	3882	4080	2 NB	126-126-124	70	61-61-52	10
SFA 5.18	SAW	4098	4553	2 NB	108-106-106	70	80-78-78	10
SFA 5.18	SAW	2881	4080	2 NB	120-138-144	70	93-92-64	10
SFA 5.18	SAW	2755	3561	-	94-94-87	10	66-66-61	10
SFA 5.5	SMAW	3418	-	2 NB	210-257-219	70	91-71-80	10
SFA 5.5	SMAW	3991	-	2 NB	73-78-57	70	57-62-41	10
SFA 5.5	SMAW	3415	-	2 NB	82-84-84	70	61-67-61	10
SFA 5.5	SMAW	3419	-	2 NB	91-92-101	70	70-71-76	10
SFA 5.5	SMAW	3401	-	-	Average 65	10	-	30
SFA 5.5	SMAW	4507	-	2 NB	80-79-74	70	67-76-64	10
SFA 5.5	SMAW	3993	-	2 NB	102-104-100	70	81-77-80	10
SFA 5.5	SMAW	4503	-	2 NB	52-75-76	70	43-61-51	10
SFA 5.5	SMAW	2165	-	-	Average 52	10	-	30

NOTE:

1. Where complete charpy or drop weight data was not obtained, RT_{NDT} was estimated in accordance with the guidelines of MTEB 5-2 and Westinghouse WCAP-7924A, “Basis for Heat-up and Cooldown Curves”.

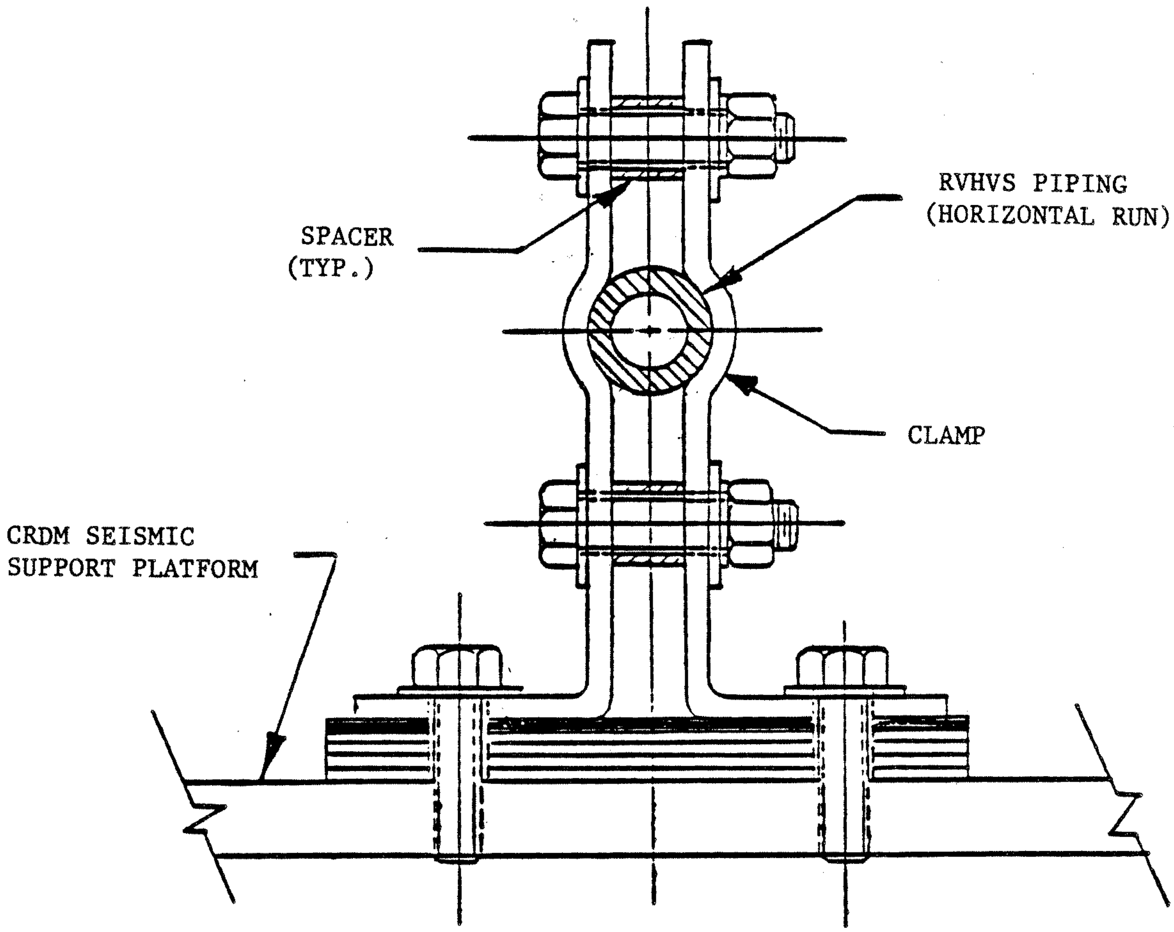


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Coolant Controlled Leakage Pump

Figure 5.5-1

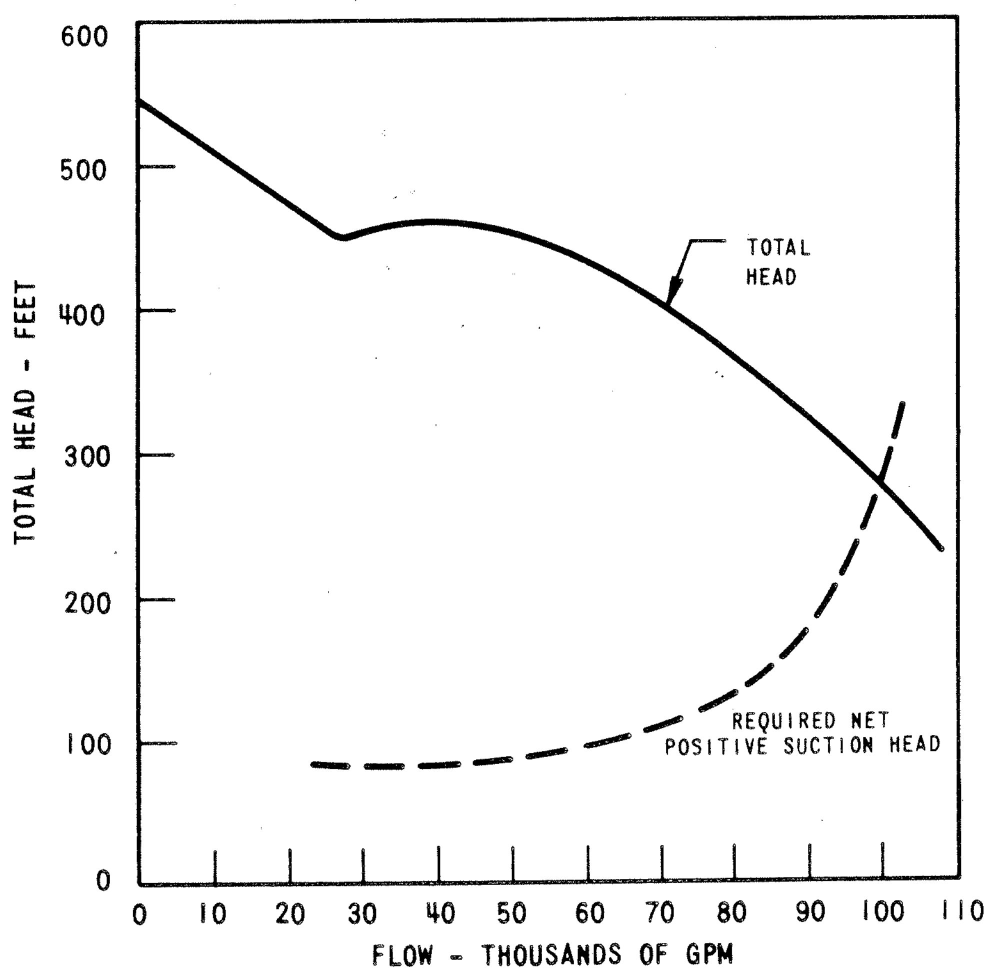
Amendment 0
August 1984



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Vessel Head Vent System
Pipe Support Clamp to CRDM
Seismic Support Platform

Figure 5.5-15



Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Coolant Pump Estimated Performance Characteristic

Figure 5.5-2

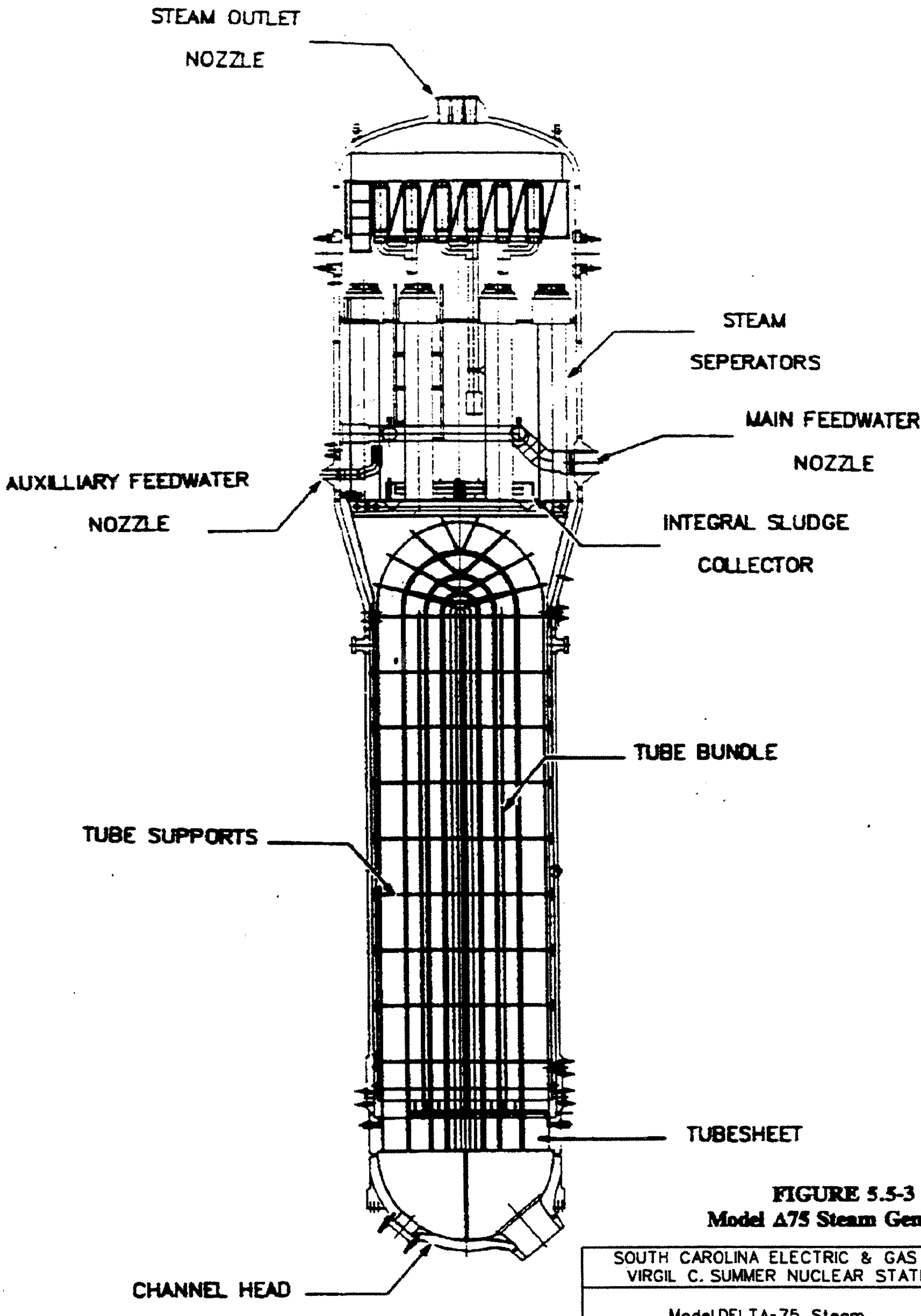
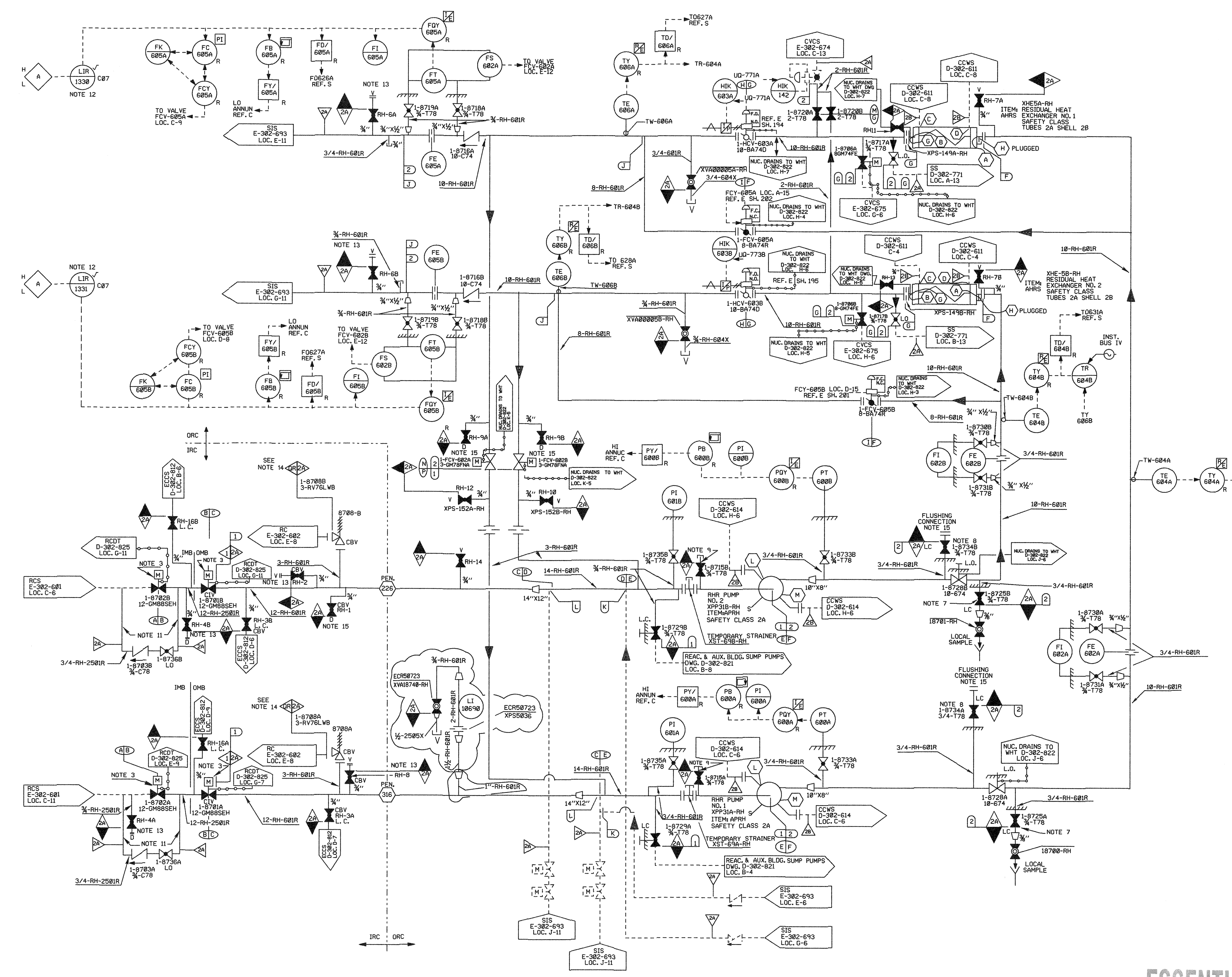


FIGURE 5.5-3
Model Δ75 Steam Generator

SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Model DELTA-75 Steam Generator Figure 5.5-3

AMENDMENT 96-02
JULY 1996



NOTES:

1. THIS DRAWING IS BASED UP DWG. 114E074, REVISION 12 ("BASE DRAWING") OF WESTINGHOUSE ELECTRIC CORPORATION, NUCLEAR ENERGY SYSTEMS PITTSBURGH, PA. WHO IS SOLELY RESPONSIBLE FOR THE ACCURACY OR THE RELIABILITY OF THE DESIGN INFORMATION SET FORTH IN THE BASE DRAWING.
2. FOR ALPHA REFERENCES, SEE DWG. E-302-002, FLOW DIAGRAM LEGEND.
3. VALVE INTERLOCKED WITH REACTOR COOLANT SYSTEM PRESSURE SIGNAL.
4. FOR CONVENTIONAL PIPING SPECIFICATIONS, SEE GAI SPECIFICATION SP-329-4461-00, PAGE 29, (WESTINGHOUSE PIPE CLASS CONVERSION TO ENGINEER'S PIPE LINE SPECIFICATOIN).
5.
6. P (WITH LETTER DESIGNATION) INDICATES THERMAL TRANSIENT DESIGN FLAG.
7. LOCATE VALVE OUTSIDE OF SHIELD WALL. SAMPLE LINE MUST BE AT A LOWER ELEVATION THAN RHR PIPE TO ALLOW GRAVITY FLOW.
8. LOCATE VALVE ABOVE RESIDUAL HEAT REMOVAL PUMP SHIELDING.
9. TEMPORARY STRAINER IS PLACED IN SPOOL PIECE DURING INITIAL FLUSHING OPERATIONS. STRAINER MUST BE REMOVED BEFORE PLANT START UP. STRAINER AND SPOOL PIECE SUPPLIED BY OTHERS. ¾" CAPPED LINE IS CONNECTED TO PRESSURE GAUGE DURING INITIAL FLUSHING.
10.
11. PROVIDE ⅜" ID FLOW RESTRICTOR PER NOTE 4, SEE DWG. E-302-002, FLOW DIAGRAM LEGEND.
12. LIR-1330 & LIR-1331 ARE ONLY USED DURING MID-LOOP (NON POWER OPERATION).
13. THIS FLANGE HAS BEEN DRILLED AND TAPPED TO ACCEPT A ⅜" COMPRESSION FITTING TO FACILITATE LLRT WORK. THE FITTING SHALL BE CAPPED AT OTHER TIMES.
14. PIPE IS QUALITY RELATED "RELIEF AND SAFETY VALVE DISCHARGE" PIPING PER TRP-13.
15. THIS FLANGE HAS BEEN DRILLED AND TAPPED TO ACCEPT A ¾" COMPRESSION FITTING TO FACILITATE DRAIN. THE FITTING SHALL BE CAPPED AT OTHER TIMES.

THIS IS A
NUCLEAR SAFETY RELATED
DOCUMENT. NO DEVIATION SHALL BE INITIATED OR PERFORMED WITHOUT PRIOR DOCUMENTATION AND WRITTEN APPROVAL

ESSENTIAL

RN 09-003

2	535*	350	600	350	750	35
1	400	350	450	350	563	35
#	PSIG	°F	PSIG	°F	PSIG	°F
	NORMAL		UPSET		HYDROTEST	
DESIGN DATA						

* SYSTEM MAY BE OPERATED AT PRESSURE UP TO 600 PSIG DURING NORMAL PLANT OPERATIONS. REFER TO DESIGN SPECIFICAION DSP-544DA, TABLE 3.

1/4"=1'-0" 0' 5' 10' 15' 20'

DRAWING LEGIBILITY
CLASS 1
SCE&G CAD ENHANCED

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
18	5/4/05	TGB	REVISED PER ECR-50099	TGB	MWD
17	11/3/03	RHM	REVISED PER ECR-50260	RHM	LEK
16	10/23/03	RHM	REVISED PER ECR-50099	RHM	WK
			AND ECR-50260		LEK
15	5/28/03	JMR	CADD ENHANCED PER ECR-50239	MGR	DDJ
20	11/06/09	JTS	REVISED PER ECR-50723	DDJ	WTW
19	5/13/05	DDJ	REVISED PER ECR-50099	JCL	MWD

FSAR Figure 5.5-4

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

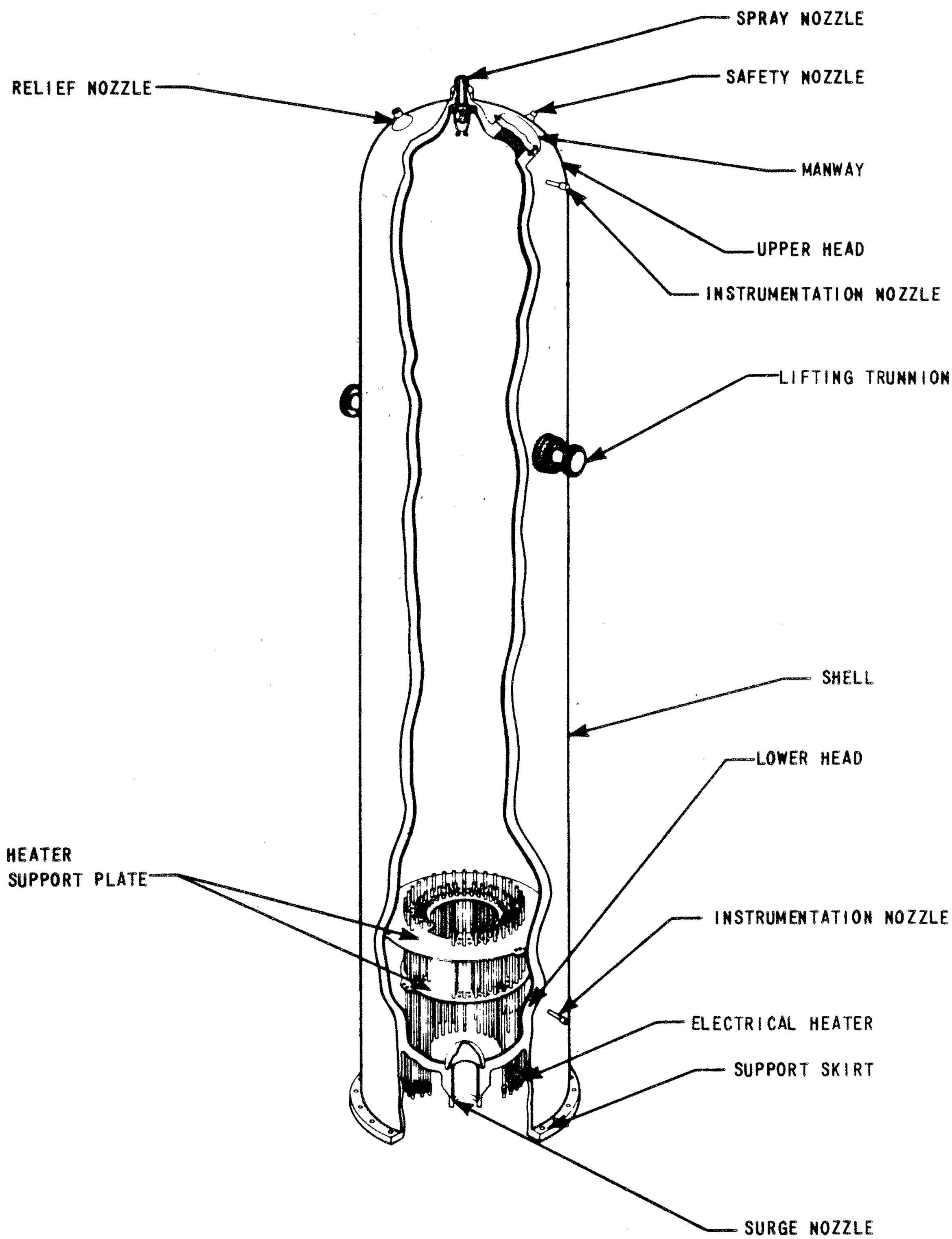
PIPING SYSTEM FLOW DIAGRAM

RESIDUAL HEAT REMOVAL

SCE&G A SCANA COMPANY — DESIGN ENGINEERING — V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. ACI	2. RHM	3. ALB

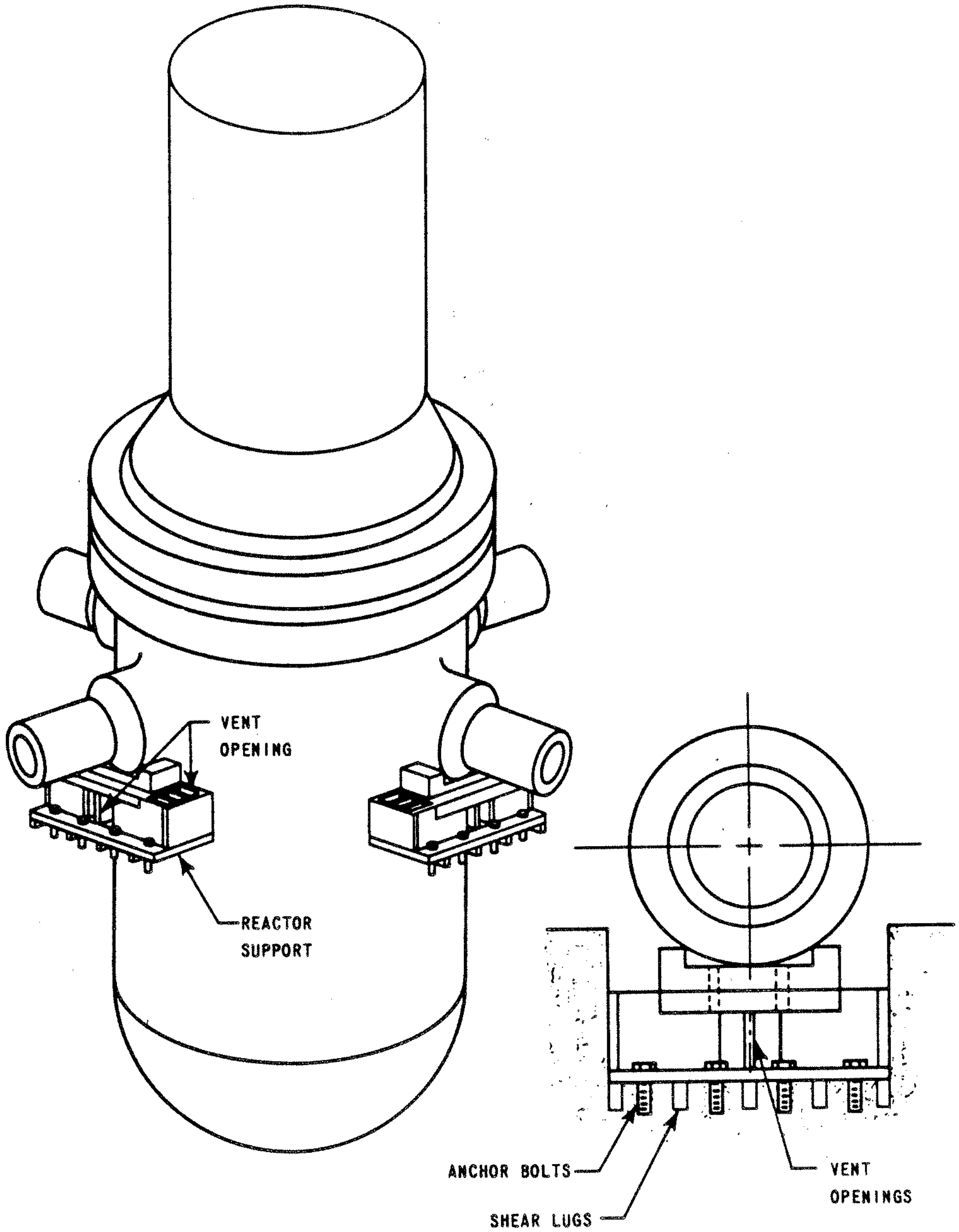
SCALE	DRAWING NUMBER	SHT. NUMBER	REV
NO SCALE	E-302-641		20



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Pressurizer

Figure 5.5-6

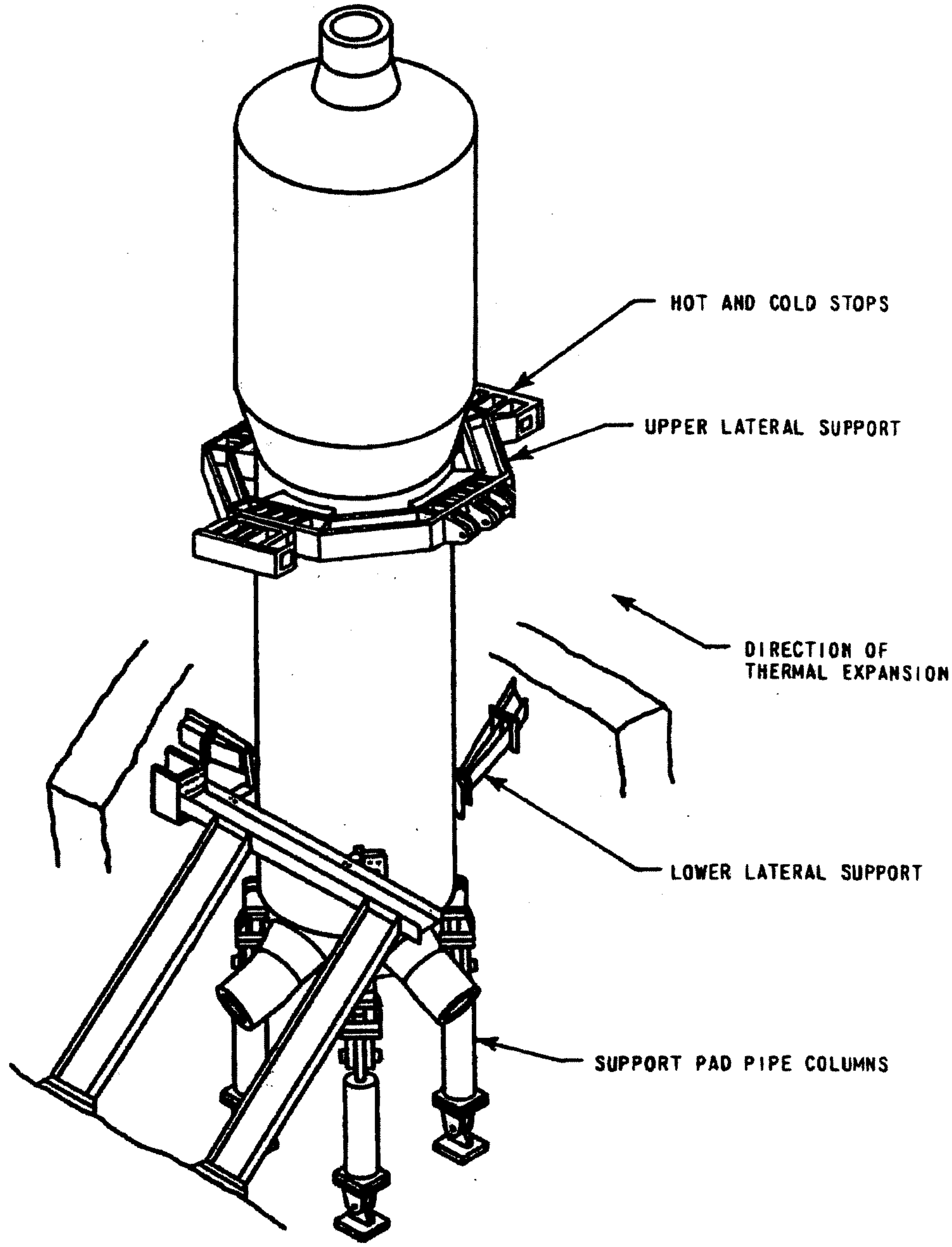


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Vessel Supports

Figure 5.5-7

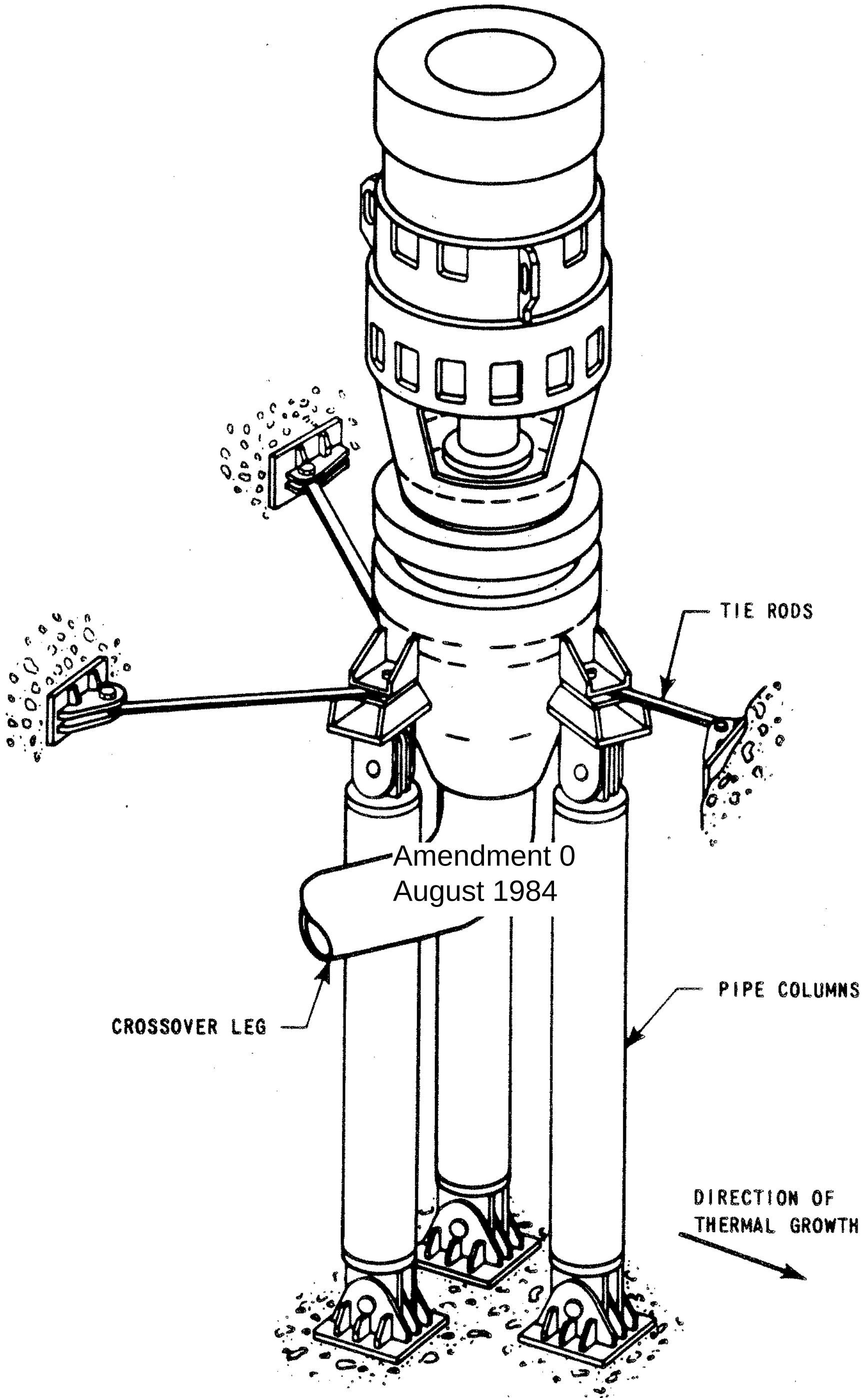


SOUTH CAROLINA ELECTRIC AND GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Steam Generator Supports

Figure 5.5-8

Amendment 0
August 1984

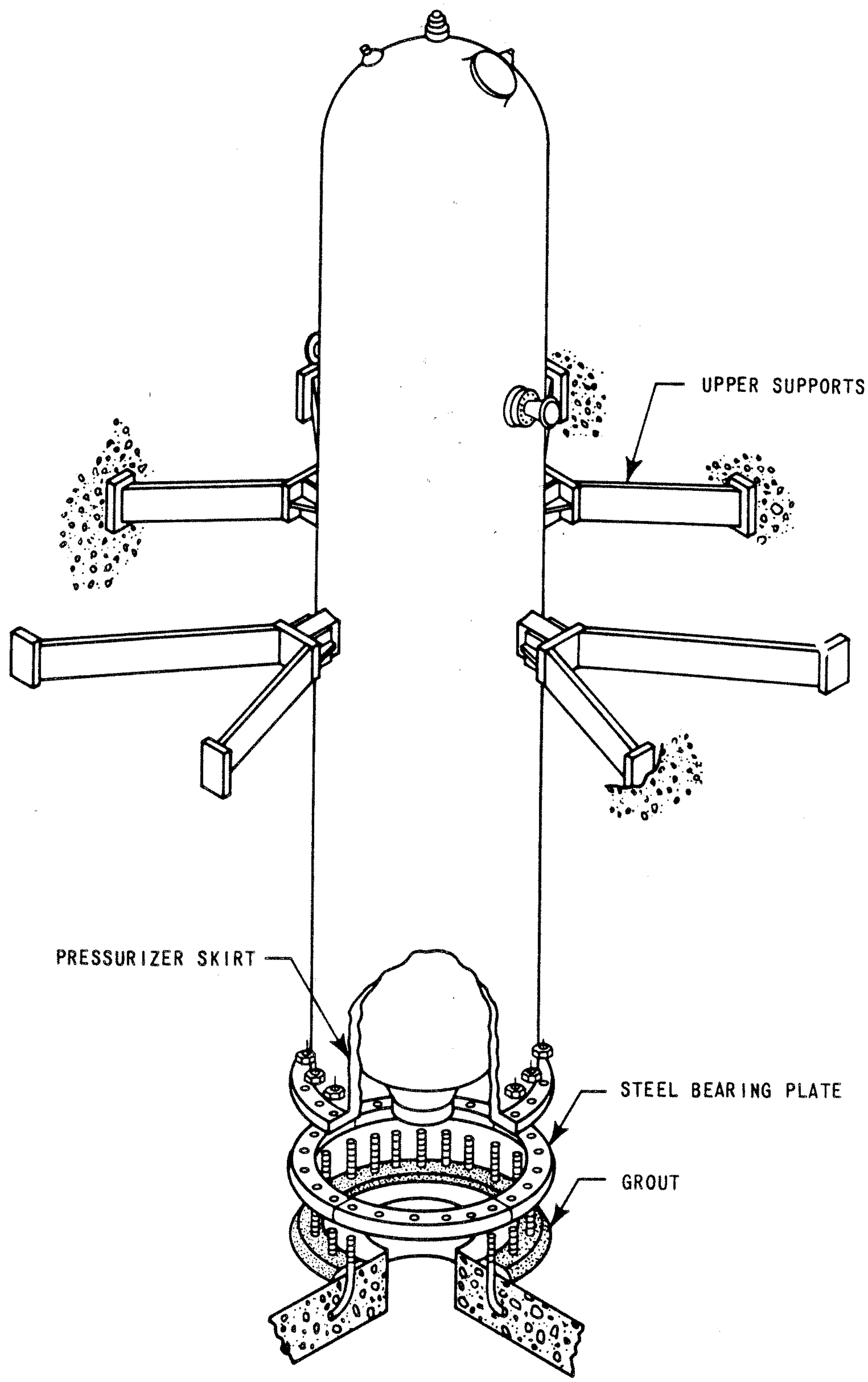


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Coolant Pump Supports

Figure 5.5-9



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Pressurizer Supports

Figure 5.5-10

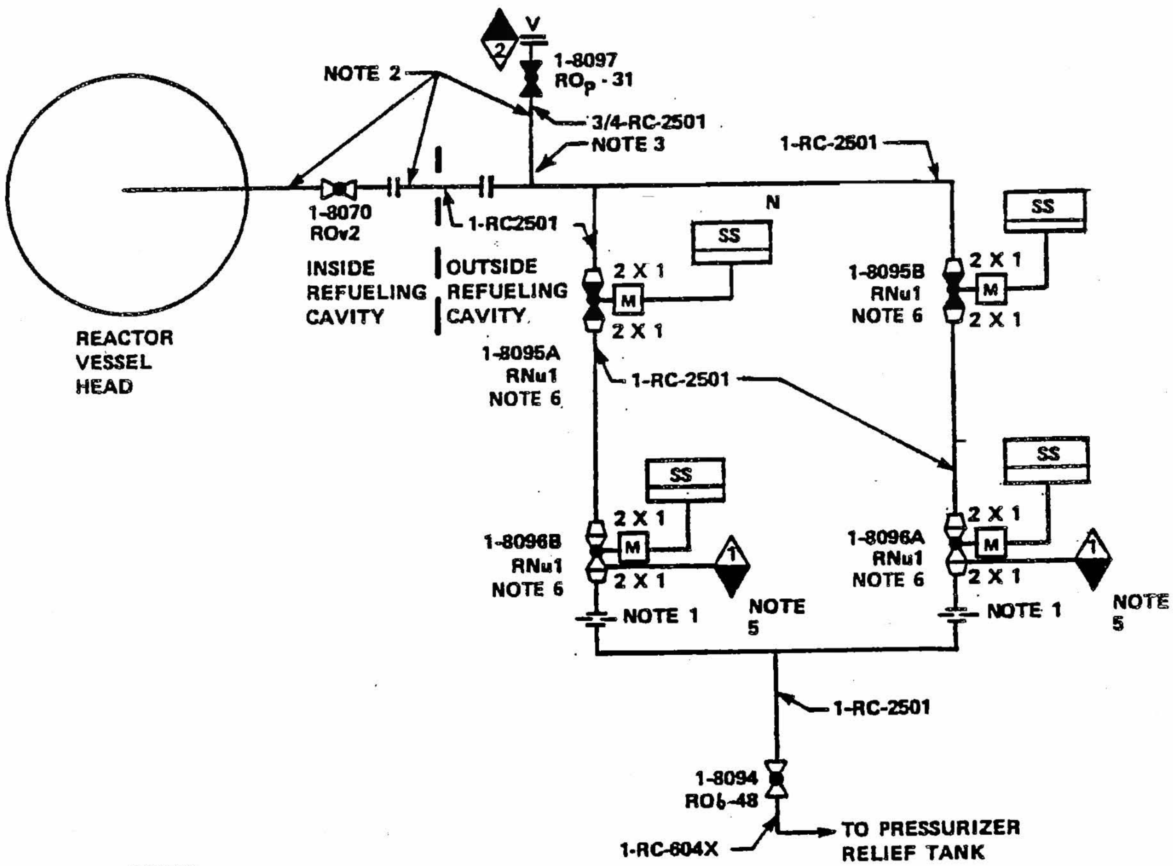
Amendment 0
August 1984

Figure 5.5-11

Deleted By Amendment 95-04

Figure 5.5-12

Deleted By Amendment 95-04



NOTES:

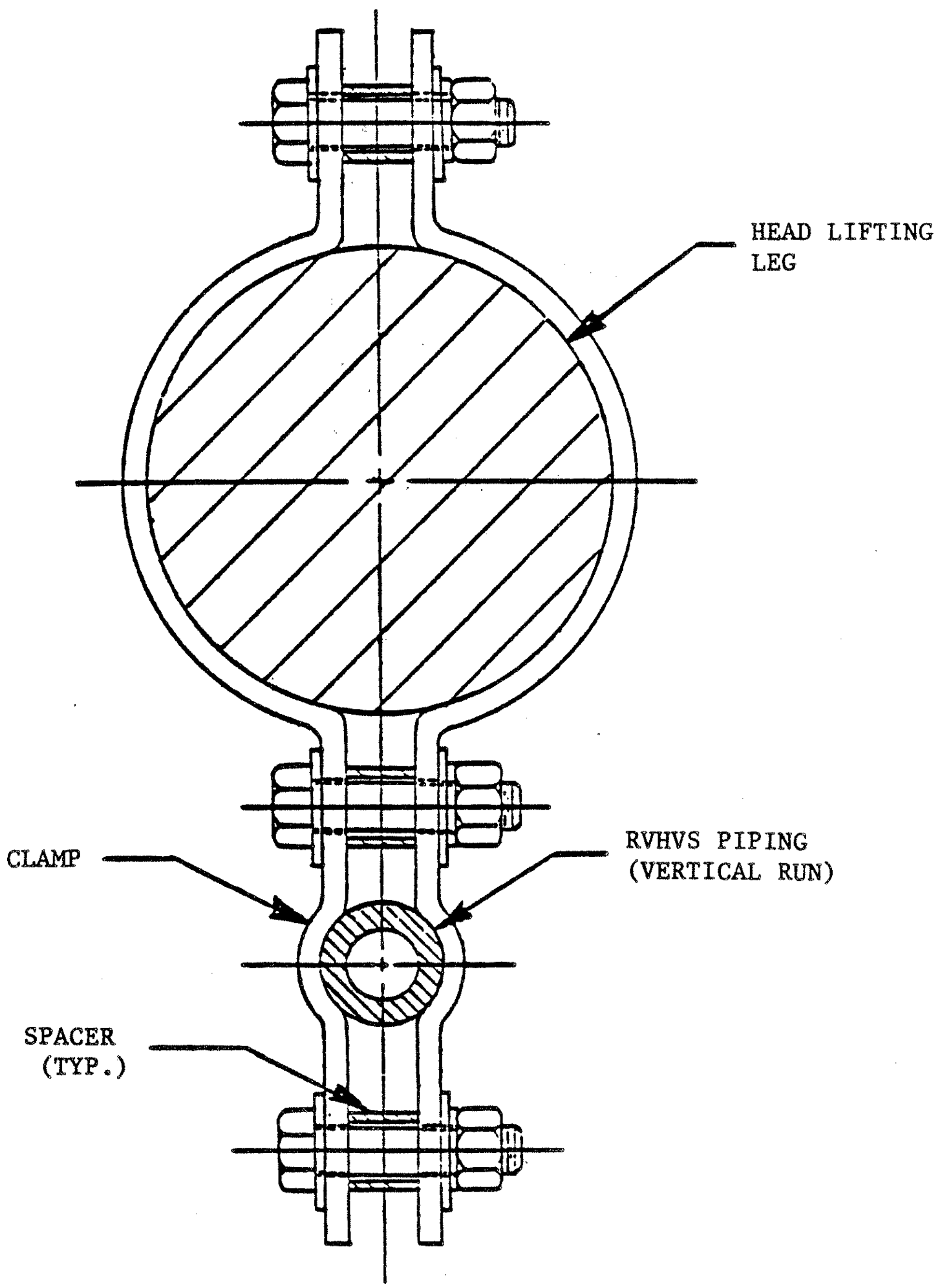
1. 3/8 INCH FLOW RESTRICTING ORIFICE.
2. NORMAL VENT LINE.
3. 3/8 INCH FLOW RESTRICTION.
4. DELETED
5. PIPE FABRICATED TO CODE CLASS 2 BUT NOT STAMPED.
6. VALVES 8095A, B AND 8096A, B MAY BE IN EITHER OF THE FOLLOWING LINE-UPS:
   A. XVT-8095A & B CLOSED; XVT-8096A & B OPEN.
   B. XVT-8095A & B OPEN; XVT-8096A & B CLOSED.

RN 14-037

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Schematic Flow Diagram of the Reactor Vessel Head Vent System

Figure 5.5-13

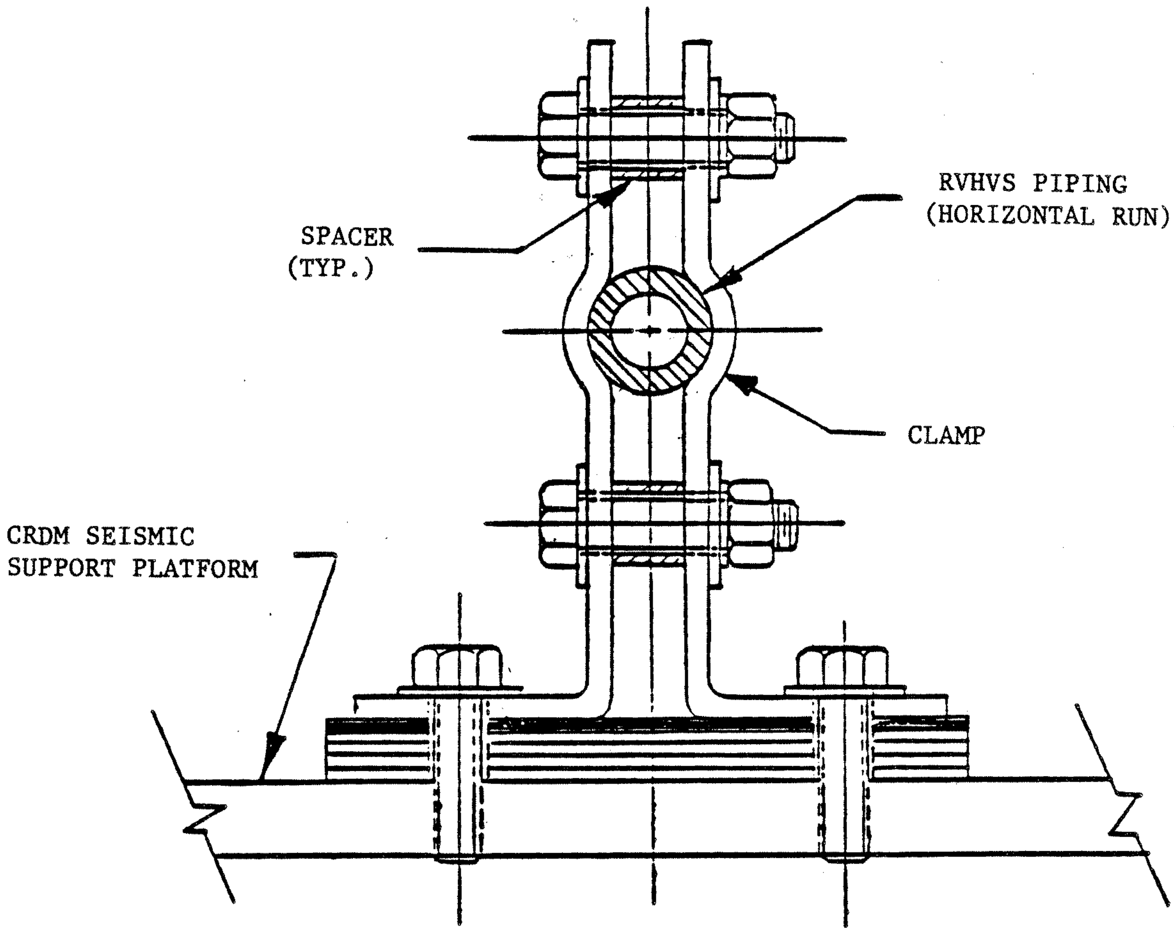


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Vessel Head Vent System
Pipe Support Clamp to Head
Lifting Leg

Figure 5.5-14

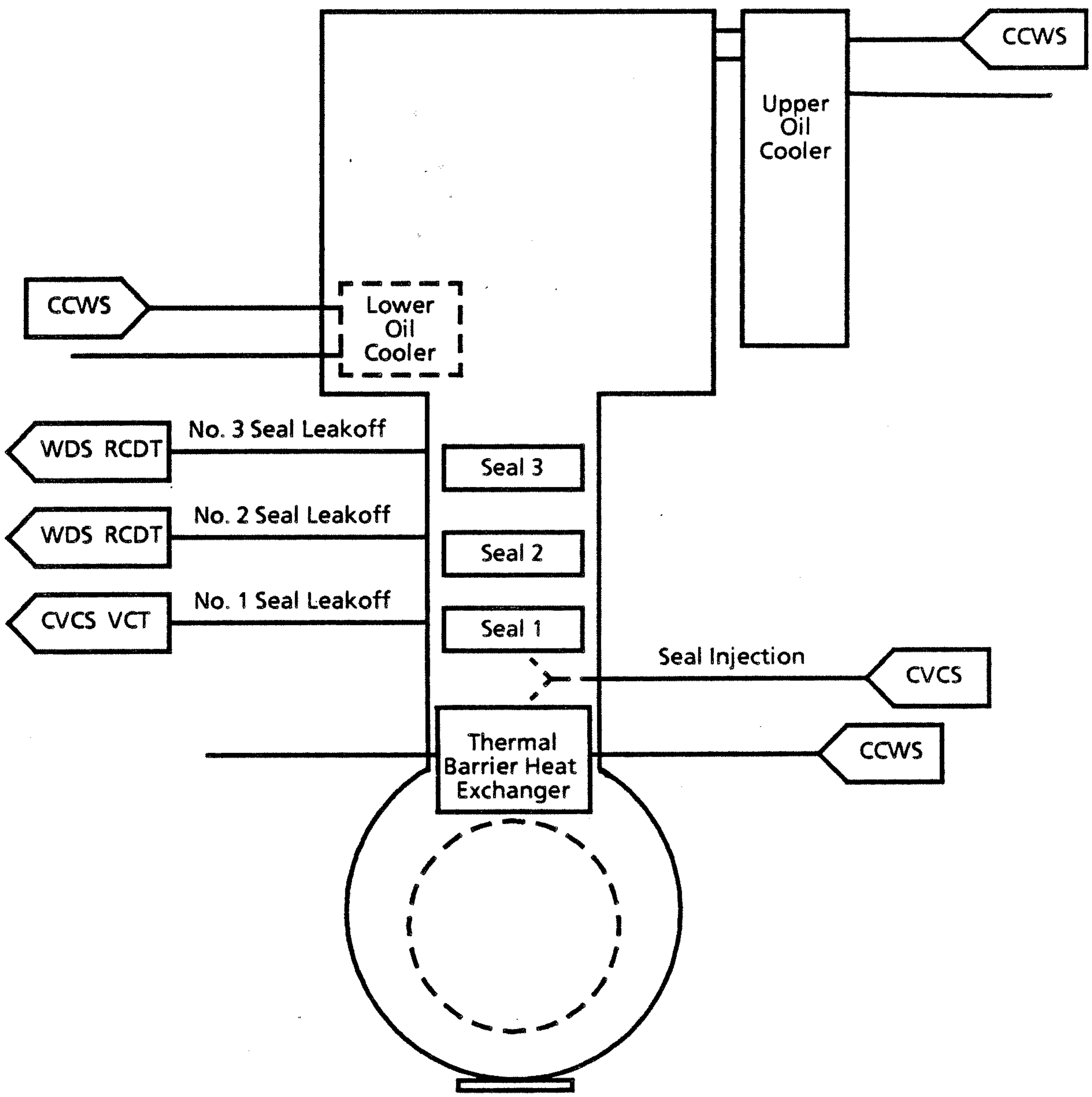


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Vessel Head Vent System
Pipe Support Clamp to CRDM
Seismic Support Platform

Figure 5.5-15

Amendment 0
August 1984



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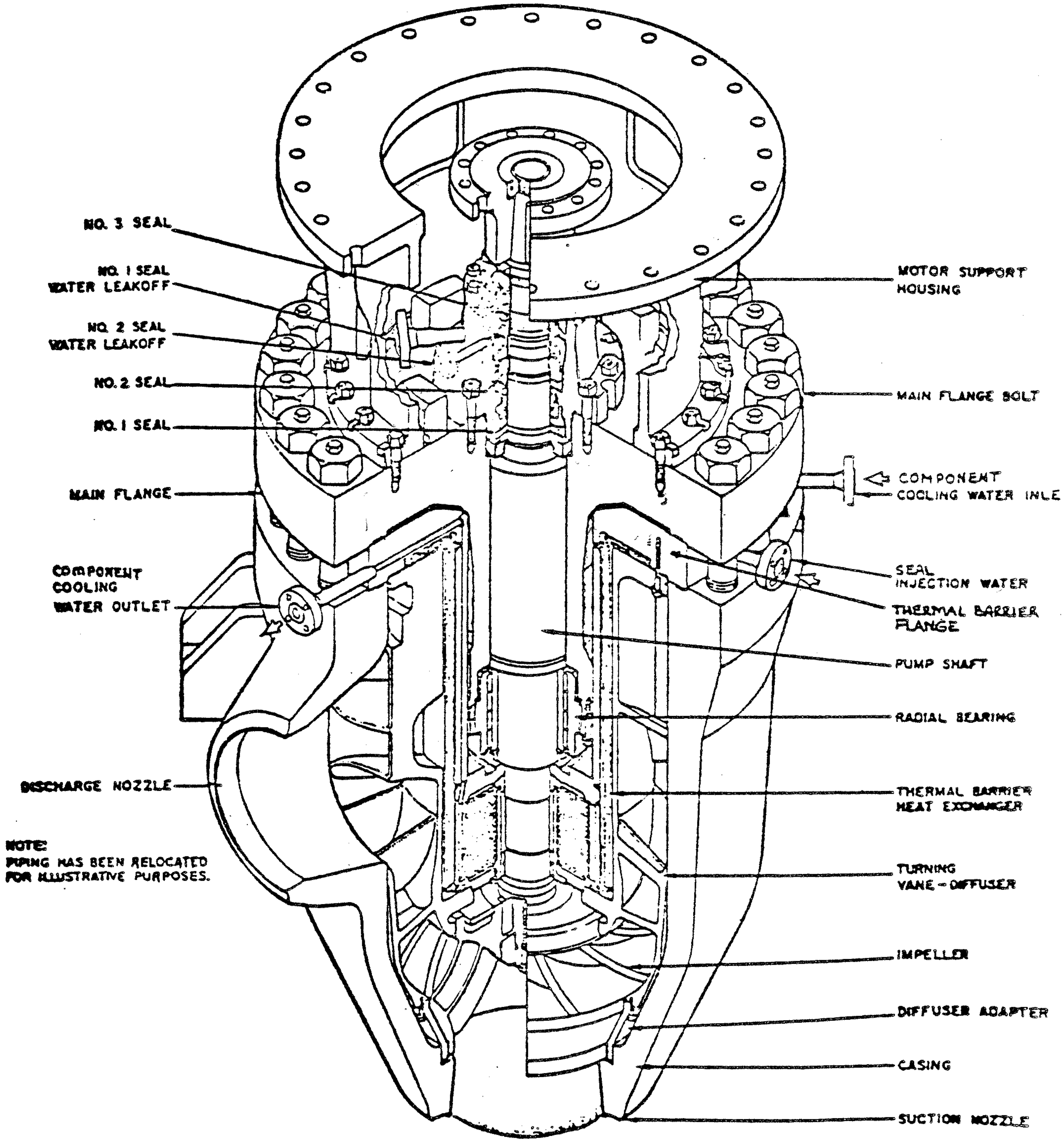
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

RCP Cooling Supplies

Figure 5.5-16

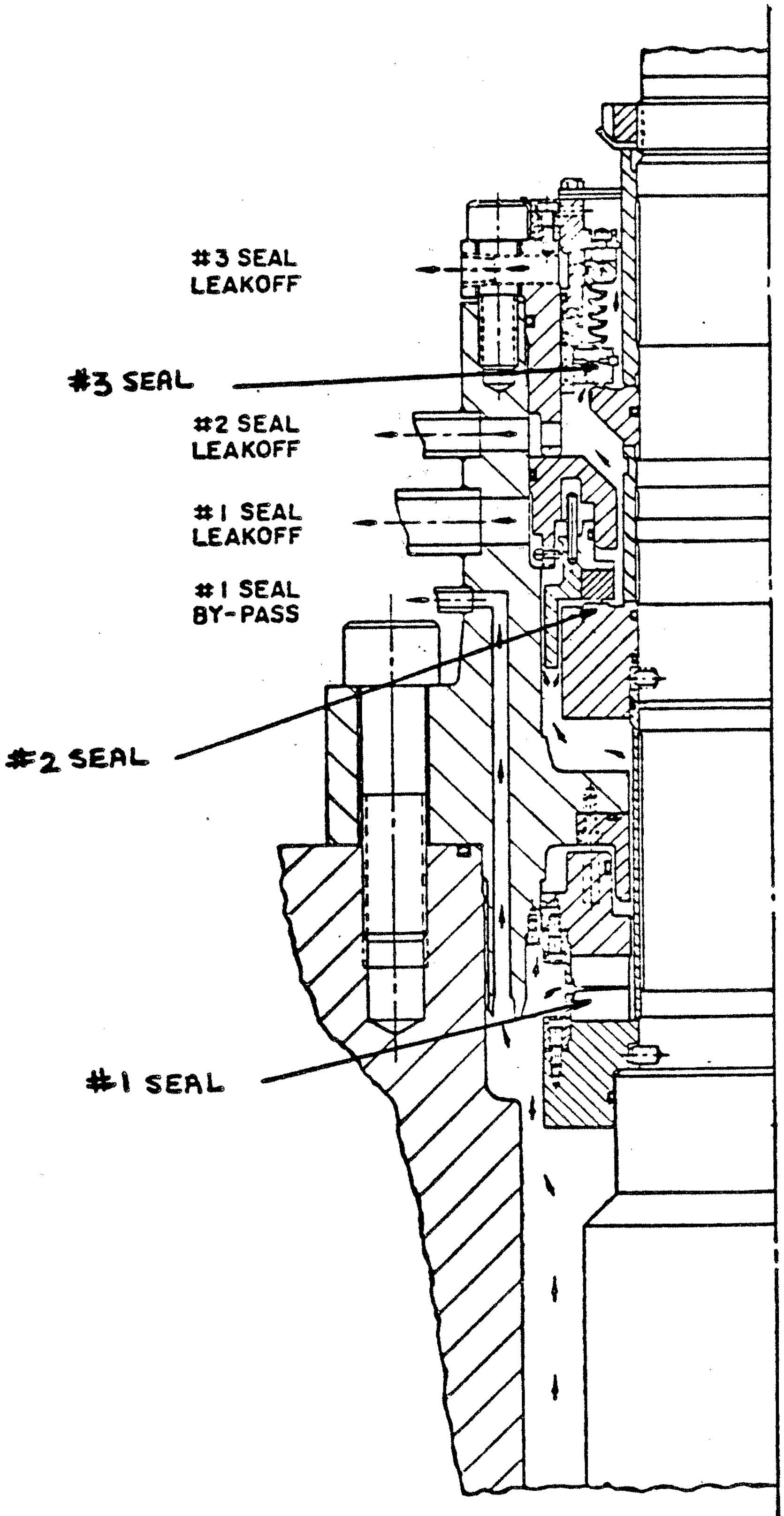


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

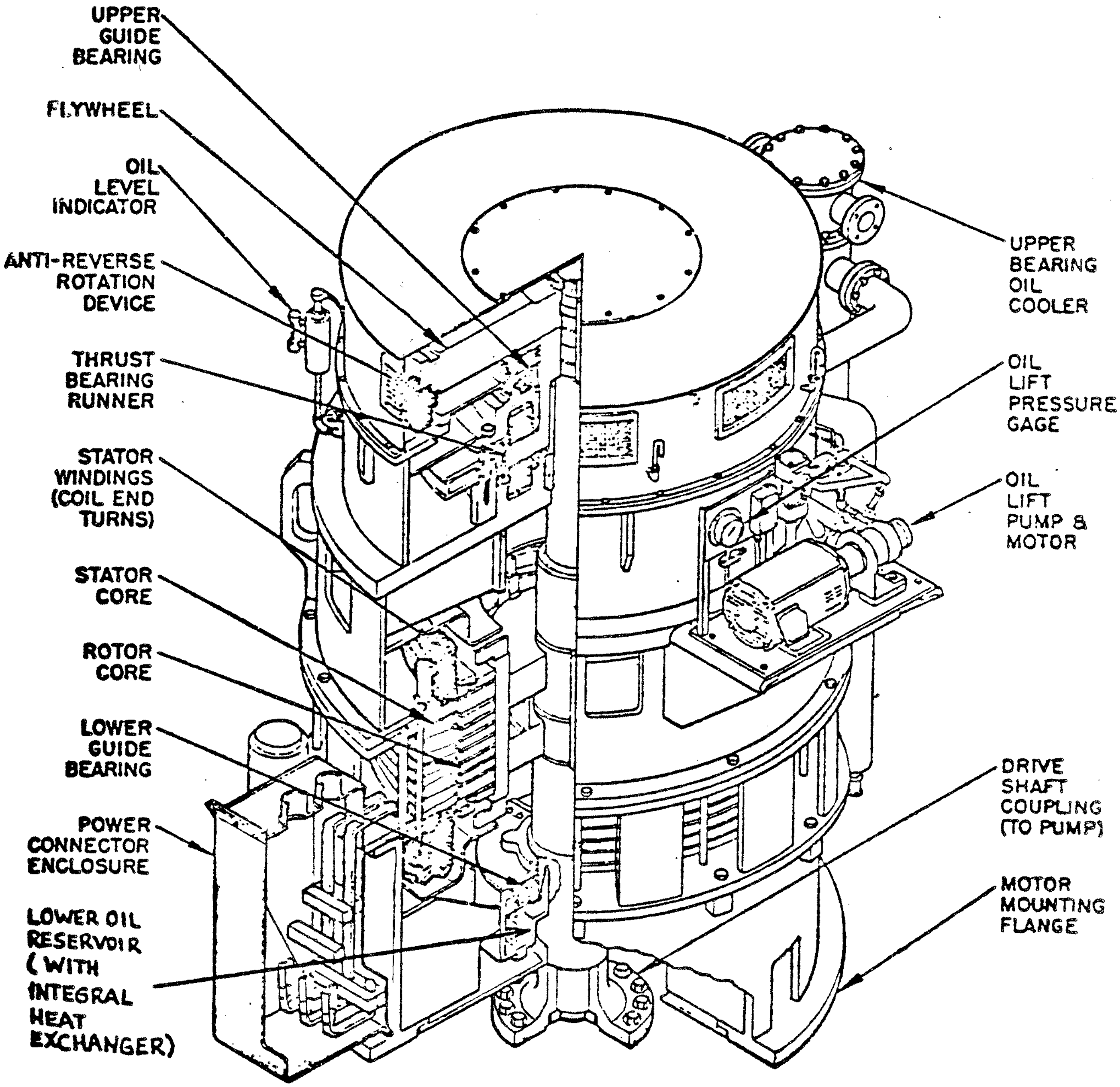
Reactor Coolant Pump

Figure 5.5-17



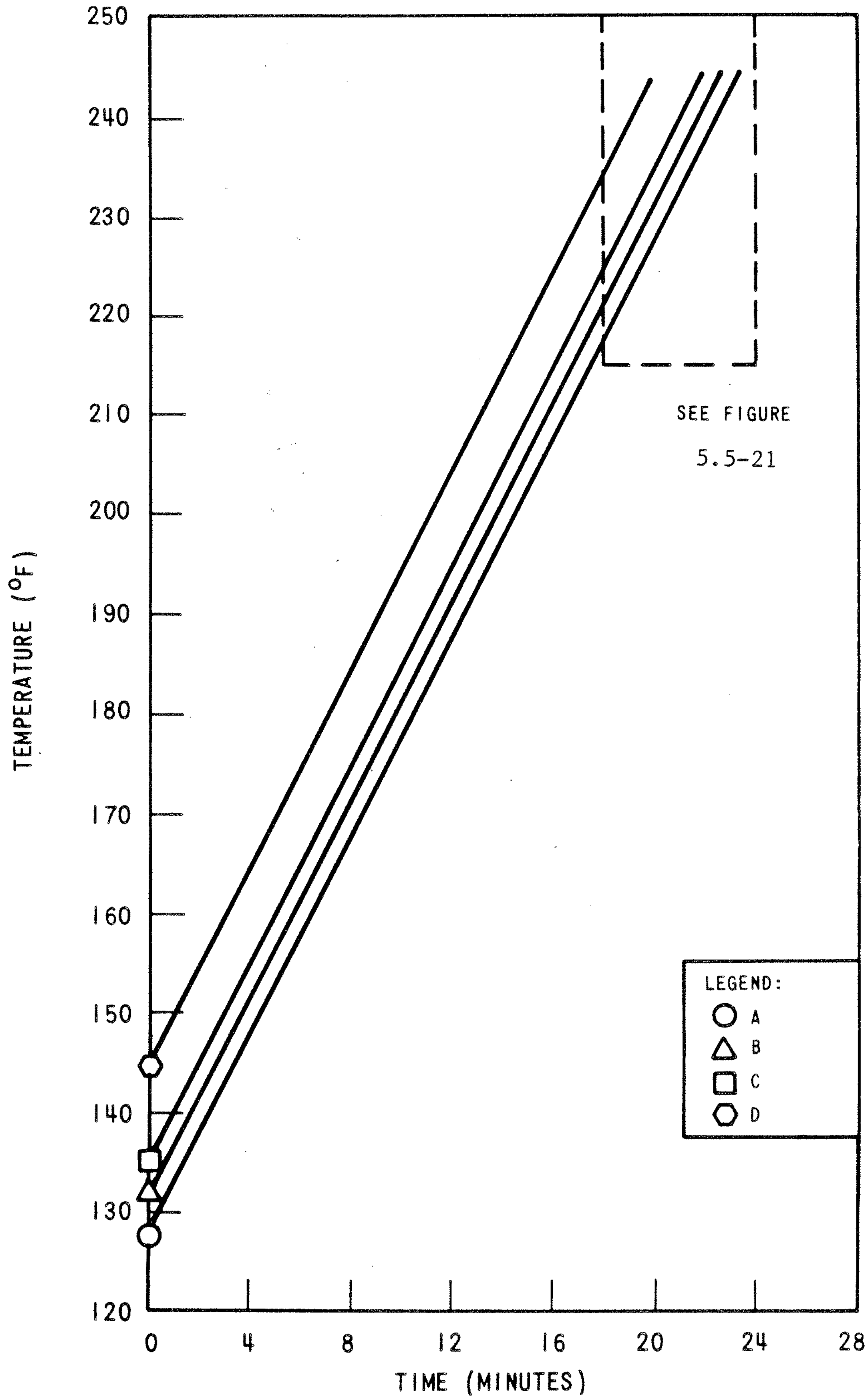
Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Seal Flow Diagram Figure 5.5-18



Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Reactor Coolant Pump Motor Figure 5.5-19

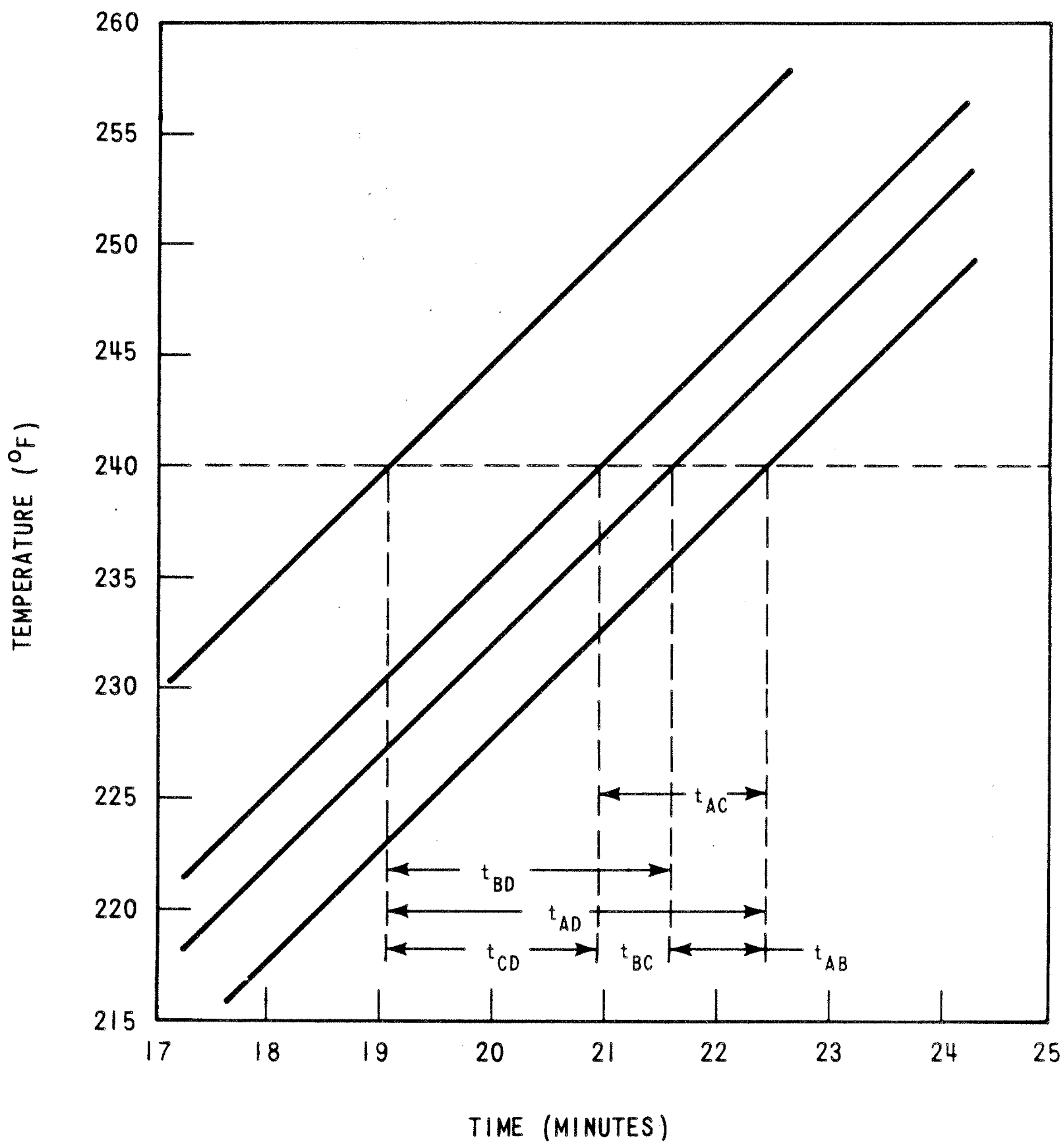


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

RCP Motor Bearing Heatup Data for Loss of CCW

Figure 5.5-20

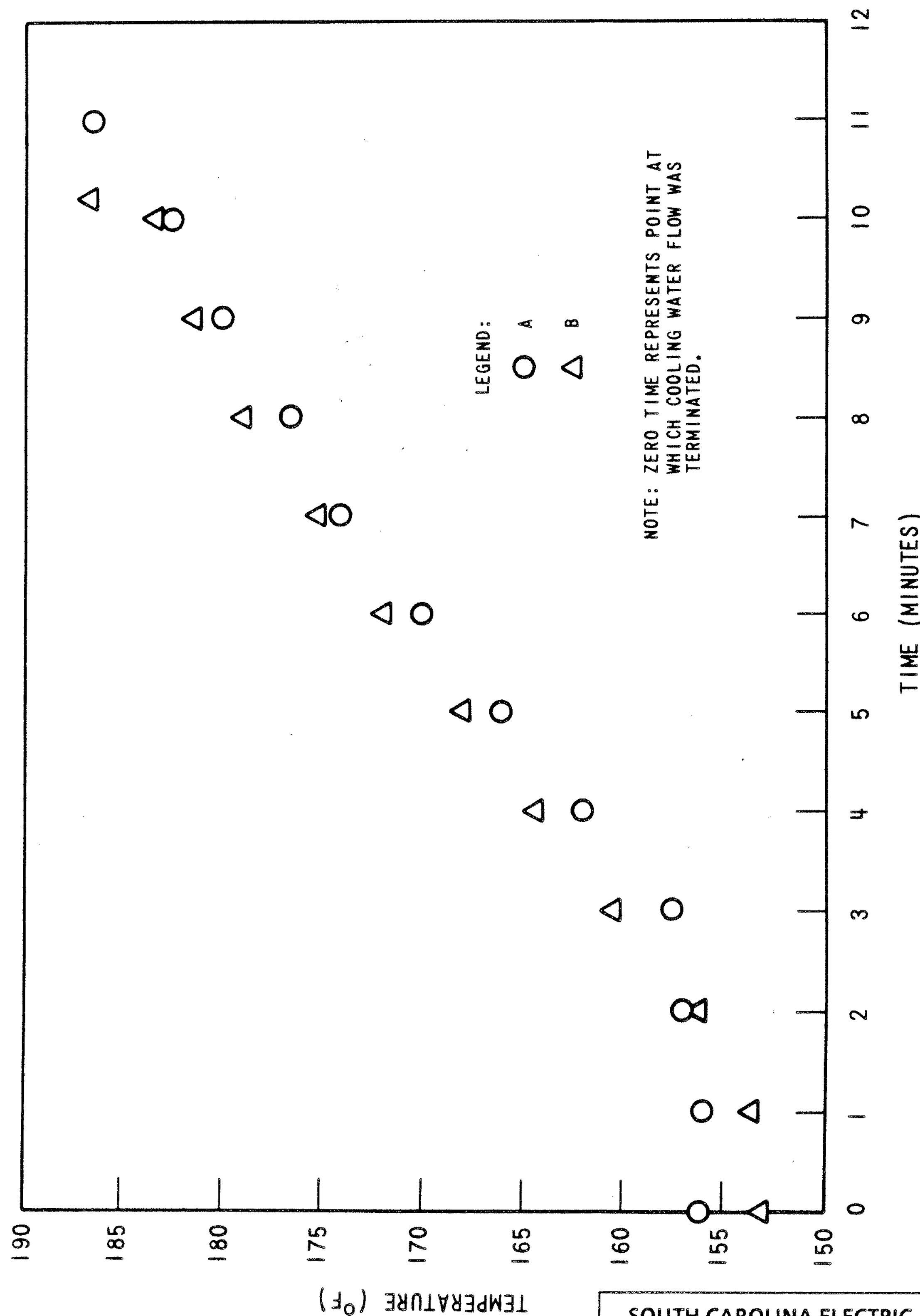


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

RCP Motor Bearing Heatup Data
for Loss of CCW - Details

Figure 5.5-21

Amendment 0
August 1984



Motor Upper Thrust Bearing Temperatures after Termination of Cooling Water Flow

Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Motor Upper Thrust Bearing Temperatures After Termination of Cooling Water Flow

Figure 5.5-22

5.6 INSTRUMENTATION REQUIREMENTS

Process control instrumentation is provided for the purpose of acquiring data on the pressurizer and on a per loop basis for the key process parameters of the Reactor Coolant System (including the reactor coolant pump motors), as well as for the Residual Heat Removal System. The pick-off points for the Reactor Coolant System are shown in Figure 5.1-1 (Sheets 1 through 3); and for the Residual Heat Removal System in Figures 5.5-4 and 6.3-1, Sheet. 3. In addition to providing input signals for the protection system and the plant control systems, the instrumentation sensors furnish input signals for monitoring and/or alarming purposes for the following parameters: 99-01

1. Temperatures
2. Flows
3. Pressures
4. Water levels
5. Valve position

In general these input signals are used for the following purposes:

1. Provide input to the reactor trip system for reactor trips as follows:
   a. Overtemperature ΔT
   b. Overpower ΔT
   c. Low pressurizer pressure
   d. High pressurizer pressure
   e. High pressurizer water level
   f. Low primary coolant flow

It is noted that the following parameters, which are also sensed so as to generate an input to the reactor trip system, while not part of the Reactor Coolant System, are included here for purposes of completeness:

   g. Feedwater flow
   h. Multiple steam generator water level
   i. Turbine first stage pressure

2. Provide input to the Engineered Safety Features Actuation System as follows:

   a. Pressurizer low pressure

It is noted that the following parameters, which are also sensed so as to generate an input to the Engineered Safety Features Actuation System, while not part of the Reactor Coolant System, are included here for purposes of completeness:

   b. High steamline differential pressure

   c. High steamline flow coincident with lo-lo reactor coolant average temperature (T_{avg})

   d. High containment pressure

   e. Low steamline pressure

3. Furnish input signals to the non-safety-related systems, such as the plant control systems and surveillance circuits so that:

   a. T_{avg} is maintained within prescribed limits. The resistance temperature detector instrumentation is identified on Figure 5.1-1, Sheet 3.

   b. Pressurizer level control, using T_{avg} to program the setpoint, maintains the coolant level within prescribed limits.

   c. Pressurizer pressure is controlled within specified limits.

   d. Steam dump control, using T_{avg} control, accommodates sudden loss of generator load.

   e. Information is furnished to the control room operator and to local stations for monitoring.

The following is a functional description of the system instrumentation. Unless otherwise stated, all indicators, recorders, and alarm annunciators are located in the control room.

1. Temperature Monitoring Instrumentation

   a. Narrow Range Hot and Cold Leg Temperature

      The hot and cold loop temperature signals required for input to the protection and control functions are obtained using thermowell mounted RTDs installed in each reactor coolant loop.

      The hot leg temperature measurement in each loop is accomplished using 3 fast response, narrow range, dual element RTDs mounted in thermowells. The hot leg thermowells are located within the 3 scoops previously used for the RTD bypass manifold at locations 120° apart in the cross sectional plane. The scoops were modified by drilling a flow hole in the top of the scoops so that water flows in through the existing holes in the leading edge of the scoop, past the RTD, and out through the new drilled hole.

      Due to temperature streaming, the 3 fast response hot leg RTDs are electronically averaged to generate the hot leg temperature.

      The cold leg temperature measurements in each loop are accomplished by 1 fast response, narrow range, dual element RTD. The existing cold leg RTD bypass penetration nozzle was modified to accept the thermowell and RTD. Temperature streaming in the cold leg is not a concern due to the mixing action of the reactor coolant pump

      Signals from these instruments are used to compute the reactor coolant ΔT (temperature of the hot leg, T_{hot}, minus the temperature of the cold leg, T_{cold}) and the T_{avg}. These temperatures are indicated on the main control board for each loop. In addition, high T_{avg}, T_{ref} deviation, overtemperature setpoint, overpower setpoint, and ΔT setpoint, are recorded on the main control board.

   b. Cold Leg and Hot Leg Temperatures

      Temperature detectors, located in the thermometer wells in the cold and hot leg piping of each loop, supply signals to wide range temperature recorders and indicators for loops A and B on the main control board. This information is used by the operator to control coolant temperature during startup and shutdown.

c. Pressurizer Temperature

There are 2 temperature detectors in the pressurizer, 1 in the steam phase and 1 in the water phase. Both detectors supply signals to temperature indicators and high temperature alarms in the control room. The steam phase detector, located near the top of the vessel, is used during startup to determine water temperature when the pressurizer is completely filled with water. The water phase detector, located at an elevation near the center of the heaters, is used during cooldown when the steam phase detector response is slow due to poor heat transfer.

d. Surge Line Temperature

This detector supplies a signal for a temperature indicator and a low temperature alarm in the control room. Low temperature is an indication that the continuous spray rate is too small.

e. Safety and Relief Valve Discharge Temperatures

Temperatures in the pressurizer safety and relief valve discharge lines are measured and supply signals for a high temperature alarm and indicator in the control room. An increase in a discharge line temperature is an indication of leakage through the associated valve.

f. Spray Line Temperatures

Temperatures in the spray lines from 2 loops are measured and indicated on the main control board. Alarms from these signals are actuated by low spray water temperature. Alarm conditions indicate insufficient flow in the spray lines.

g. Pressurizer Relief Tank Water Temperature

The temperature of the water in the pressurizer relief tank is indicated on the main control board. An alarm actuated by high temperature informs the operator that cooling of the tank contents is required.

h. Reactor Vessel Flange Leakoff Temperature

The temperature in the leakoff line from the reactor vessel flange o-ring seal leakage monitor connections is indicated on the main control board. An increase in temperature above ambient is an indication of o-ring seal leakage. High temperature actuates an alarm.

i. Reactor Coolant Pump Motor Temperature Instrumentation

(1) Thrust Bearing Upper and Lower Shoes Temperature

Resistance temperature detectors are provided with 1 located in the shoe of the upper and 1 in the shoe of the lower thrust bearing. Monitoring of these detectors is provided by the plant computer which actuates a high temperature alarm at the computer.

(2) Stator Winding Temperature

The stator windings contain 6 resistance type detectors, 2 per phase, imbedded in the windings. A signal from 1 of these detectors is monitored by the plant computer which actuates a high temperature alarm at the computer.

(3) Upper and Lower Bearing Temperature

Resistance temperature detectors are located 1 in the upper and 1 in the lower radial bearings. Monitoring of these detectors is provided by the plant computer which actuates a high temperature alarm at the computer.

2. Flow Indication

a. Reactor Coolant Loop Flow

Flow in each reactor coolant loop is monitored by 3 differential pressure measuring detectors at a piping elbow tap in each reactor coolant loop and indicated on the main control board. These measurements on a 2 out of 3 coincidence circuit provide a low flow signal to actuate a reactor trip.

b. Safety Valve Flow

An acoustical type sensor is provided downstream of each safety valve to detect flow through the safety valves. The sensor activates an alarm in the control room when flow is detected. Control room indication is provided to enable the plant operator to determine which valve is open.

3. Pressure Indication

a. Pressurizer Pressure

Three (3) pressurizer pressure transmitters provide signals for individual indicators in the control room for actuation of both a low pressure trip and a high pressure trip. One (1) of the signals may be selected by the operator for indication on a pressure recorder. Three (3) transmitters provide low pressure signals for safety injection initiation. Three (3) transmitters provide signals for safety injection signal unblock during plant startup. An additional transmitter is used, along with a reference pressure signal, to develop a demand signal for a 3-mode controller. The lower portion of the controller's output range operates the pressurizer heaters. For normal operation, a small group of heaters is controlled by variable power to maintain the pressurizer operating pressure. If the pressure error signal falls towards the bottom of the variable heater control range all pressurizer heaters are turned on.

The upper portion of the controller's output range operates the pressurizer spray valves and 1 power relief valve. The spray valves are proportionally controlled in a range above normal operating pressure with spray flow increasing as pressure rises. If the pressure rises significantly above the proportional range of the spray valves, a power relief valve (interlocked with a separate transmitter so as to prevent spurious operation) is opened. A further increase in pressure will actuate a high pressure reactor trip. A separate transmitter (interlocked also with another transmitter to prevent spurious operation) provides power relief valve operation for 2 additional valves upon high pressurizer pressure. A signal interlock between transmitter is required to open (or keep open) the power relief valve.

b. This item deleted.

c. Reactor Coolant Loop Pressures

Two (2) transmitting channels are provided. Both transmitters provide a recorder signal and an indication of reactor coolant pressure on 2 of the hot legs. These are wide range transmitters which provide pressure indication over the full operating range. The recorder, together with the indicators, serve as a guide to the operator for manual pressurizer heater and spray control and letdown to the Chemical and Volume Control System during plant startup and shutdown. An amplifier signal from the lower portion of the range of one of the channels is indicated on the main control board to provide improved legibility at the lower pressures.

The 2 wide range channels provide the permissive signals for the residual heat removal loop suction line isolation valve interlock circuit.

d. Pressurizer Relief Tank Pressure

The pressurizer relief tank pressure transmitter provides a signal to an indicator and a high pressure alarm in the control room.

e. Reactor Coolant Pump Motor Oil Pressure and Level

(1) Oil Pressure

Pressure switches are provided on the high pressure oil lift system. Low oil pressure actuates an alarm on the main control board. In addition, an interlock system prevents starting of the pump until the oil lift pump is started manually prior to starting the reactor coolant pump motor and actuates a status light to indicate adequate oil pressure. A local pressure gauge is also provided.

(2) Lower Oil Reservoir Liquid Level

A level switch is provided in the motor lower radial bearing oil reservoir. The switch actuates a high or low level alarm in the control room

(3) Upper Oil Reservoir Liquid Level

A level switch is provided in the motor upper radial bearing and thrust bearing oil reservoir. The switch actuates a high or low level alarm in the control room.

4. Liquid Level Indication

a. Pressurizer Water Level

Three (3) pressurizer liquid level transmitters provide signals for use in the Reactor Control and Protection System, the Emergency Core Cooling System, and the Chemical and Volume Control System. Each transmitter provides an independent high water level signal that is used to actuate an alarm and a reactor trip. The transmitters also provide independent low water level signals that activate an alarm. Each transmitter also provides a signal for a level indicator that is located on the main control board.

In addition to the above, signals may be selected for specific functions as follows:

(1) Any 1 of the 3 level transmitter signals may be selected by the operator for display on a level recorder located on the main control board. This same recorder is used to display a pressurizer reference liquid level signal.

(2) Two (2) of the 3 transmitters perform the following functions: (A selector switch allows the third transmitter to replace either of these 2.)

(a) One (1) transmitter provides a signal which actuates an alarm when the liquid level falls to a fixed level setpoint. The same signal trips the pressurizer heaters "off" and close the letdown line isolation valves.

(b) One (1) transmitter supplies a signal to the liquid level controller for charging flow control and initiation of a low flow (hi demand) alarm. This signal is compared to the reference level. If the actual level is lower than the reference level, a low alarm is actuated. If the actual level exceeds the reference level, a high level alarm and all pressurizer backup heaters are energized.

A fourth independent pressurizer level transmitter that is calibrated for low temperature conditions, provides water level indication on the main control board during startup, shutdown and refueling operations.

b. Pressurizer Relief Tank Level

The pressurizer relief tank level transmitter supplies a signal for an indicator on the main control board for high and low level alarms.

c. Reactor Vessel Level

The reactor vessel level system provides a direct reading of reactor vessel level on the main control board which can be used by the operator in conjunction with the core subcooling monitor to identify the possibility of inadequate core cooling conditions. Details of the reactor vessel level system are provided in Section 7.6.9.

5. Valve Position

The pressurizer power operated relief valves are provided with limit switches for safety grade indication of valve open/closed position in the control room.

6. Core Subcooling Indication

The core subcooling monitor provides continuous monitoring of the margin to saturation in the reactor core (i.e., the amount of subcooling) on the main control board. The core subcooling monitor utilizes inputs from the hot leg RTDs, Reactor Coolant System pressure sensors, and selected incore thermocouples. Details of the core subcooling monitor system are provided in Section 7.5.5.

The Reactor Coolant System design and operating pressure together with the safety, power relief and pressurizer spray valve setpoints, as well as the protection system setpoint pressures are listed in Table 5.2-7.

The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops instrumentation and control accuracy and response characteristics, and system relief valve characteristics.

Process control instrumentation for the Residual Heat Removal System is provided for the following purposes:

a. Furnish input signals for monitoring and/or alarming purposes for:

   (1) Temperature indications

   (2) Pressure indications

   (3) Flow indications

b. Furnish input signals for control purposes of such processes as follows:

   (1) Control valve in the residual heat removal pump bypass line so that it opens at flows below a preset limit and closes at flows above a preset limit.

   (2) Residual heat removal inlet valves control circuitry. See Section 7.6 for the description of the interlocks and requirements for automatic closure.

   (3) Control valve in the residual heat removal heat exchanger bypass line to control temperature of reactor coolant returning to reactor coolant loops during plant cooldown.

   (4) Residual heat removal pump circuitry for starting residual heat removal pumps on "S" signal.

5.7 INSERVICE INSPECTION PROGRAM (INCLUDING PRESERVICE INSPECTION)

During the design phase of the Virgil C. Summer Nuclear Station, careful consideration has been given to access, where practical, for the performance of preservice and inservice examinations.

5.7.1 APPLICABLE CODES AND STANDARDS

The inservice inspection program (including preservice inspection) is administered in accordance with the applicable codes and standards as addressed in 10CFR50.55a.

Deviations from Code requirements are allowed and/or prescribed by various means. These deviations may be permitted via NRC approved Relief Requests, Code Cases which are adopted by the NRC in various Regulatory Guides, Generic Letters, and exceptions described in 10CFR50.55a. Any deviations from the Code requirements will be documented in the inservice inspection program (i.e., ISI, IST, and Repair and Replacement). RN 04-012

5.7.1.1 Preservice Inspection

The initial preservice inspection for Class 1, 2, and 3 components was performed to the extent practical in accordance with IWB-2100, IWC-2100, and IWD-2100 of the ASME B&PV Code Section XI (1974 edition through Summer 1975 Addenda).

5.7.1.2 First Inspection Interval - January 1, 1984, Through December 31, 1993

During the first 10 year inspection interval, for Class 1, 2, and 3 components (except as noted below), the preservice inspections for repairs, replacements, and inservice inspections were performed to the extent practical to the ASME B&PV Code Section XI (1977 edition through Summer 1978). Per 10CFR50.55a(b)(2)(iv), examination of Code Class 2 pipe welds in the Residual Heat Removal System, Emergency Core Cooling System, and Containment Heat Removal System shall be determined by the requirements of paragraph IWC-1220, Table IWC-2520 Category C-F and C-G, and paragraph IWC-2411 of the ASME B&PV Code Section XI (1974 edition through Summer 1975 addenda). For V. C. Summer, these systems are defined as Residual Heat Removal (RH), Safety Injection (SI), Service Water (SW), Reactor Building Spray (SP), and Component Cooling Water (CC). For ASME Class 1, 2, and 3 component supports, preservice inspections for repairs and replacements, and inservice inspection are performed to the extent practical in accordance with the ASME B&PV Code Section XI (1977 edition through Summer 1978) and subsection IWF of the ASME B&PV Code Section XI (1980 edition through Winter 1981). Inservice testing of Class 1, 2, and 3 valves is performed to the extent practical in accordance with subsection IWV of the ASME B&PV Code Section XI (1977 edition through Summer 1978). Inservice testing of Class 1, 2, and 3 pumps is performed to the extent practical in accordance with the ASME B&PV Code Section XI (1980 edition).

5.7.1.3 Second Inspection Interval - January 1, 1994, through December 31, 2003

The second 10 year interval was performed to the requirements of Section XI of the ASME B&PV Code as prescribed by 10CFR50.55a. The latest edition of Section XI approved for use on January 1, 1994, was the 1989 edition.

5.7.1.4 Third Inspection Interval - January 1, 2004, through December 31, 2013

The third 10 year inservice inspection interval will meet the requirements of Section XI of the ASME B&PV Code as prescribed by 10CFR50.55a. The latest edition of Section XI approved for use on January 1, 2004 was the 1998 Edition, with Addenda through 2000. However, as later editions are incorporated into 10CFR50.55a, the provisions of the later Code may be incorporated, in part or in whole, as allowed by 10CFR50.55a during the interval. The inservice inspection plans will designate the applicable Edition and Addenda to the corresponding inspection items. The inservice testing program for ASME Class 1, 2, and 3 pumps and valves will be implemented in accordance with the ASME OM Code, 1998 Edition, with Addenda through 2000 as prescribed by 10CFR50.55a. RN 04-012

Periodic inspection of the Reactor Building (Containment) Structure at the Virgil C. Summer Nuclear Station will be conducted in accordance with the requirements of the 2001 Edition with the 2003 Addenda of the ASME B&PV Code, Section XI, Division 1, Subsections IWE and IWL, as modified by NRC Final Rulemaking to 10CFR50.55a published in the Federal Register on August 8, 1996. The current Containment Inspection interval is effective from January 1, 2007 to December 31, 2016. RN 11-020

Note to 5.7.1.4:

During the third and subsequent 10 year inspection intervals, preservice inspections for repairs and replacements and inservice inspection will be performed to the extent practical to the ASME B&PV Code Section XI [edition and addenda to be determined by the latest edition and addenda incorporated by reference in 10CFR50.55a(a), 12 months prior to the start of the inspection interval as per 10CFR50.55a(g)(4)(ii)]. Per IWA-2430(b), "The inspection interval shall be determined by calendar years following placement of the power unit into commercial service" as determined by the regulation of the Federal Power Commission, Chapter I - Title 18, Code of Federal Regulations Part 191, paragraph 9.D. Sub paragraph IWA-2430(c) further states that, "Each inspection interval may be decreased or extended (but not cumulatively) by as much as 1 year." For the purpose of determining the beginning and end of future ten year inspection intervals, the plant began commercial operation on January 1, 1984. RN 04-012 14-002

5.7.1.5 Fourth Inspection Interval - January 1, 2014, through December 31, 2023

The fourth 10 year inservice inspection interval will meet the requirements of Section XI of the ASME B&PV Code as prescribed by 10CFR50.55a. The latest edition of Section XI approved for use on January 1, 2014 was the 2007 Edition, with Addenda through 2008. However, as later editions are incorporated into 10CFR50.55a, the provisions of the later Code may be incorporated, in part or in whole, as allowed by 10CFR50.55a during the interval. The inservice inspection plans will designate the applicable Edition and Addenda to the corresponding inspection items. The inservice testing program for ASME Class 1, 2, and 3 pumps, valves and snubbers will be implemented in accordance with the ASME OM Code, 2004 Edition, with Addenda through 2006 as prescribed by 10CFR50.55a. RN 14-002

5.7.2 PROVISIONS FOR ACCESS TO REACTOR COOLANT PRESSURE BOUNDARY

Provisions for adequate access have been verified by a review of all the drawings applicable to the layout and arrangement of the reactor coolant and associated auxiliary systems within the boundaries established in accordance with the requirements of IWB-1200 and IWC-1200.

Specific provisions made for inspection access in the design of the reactor vessel, system layout, and other major primary coolant components to ensure compliance with the requirements of IWA-1500 and IWB-2600 are as follows:

1. Reactor internals are completely removable. The tools and storage space required to permit reactor internals removal for these inspections are provided.

2. The reactor vessel shell in the core area is designed with a clean, uncluttered cylindrical inside surface to permit future positioning of test equipment without obstruction.

3. The reactor vessel cladding is improved in finish by the vessel manufacturer by grinding to the extent necessary to permit meaningful examination of the vessel welds and adjacent base metal in accordance with the Code.

4. The cladding to base metal interface is ultrasonically examined by the vessel manufacturer to assure satisfactory bonding to allow the volumetric inspection of the vessel welds and base metal from the vessel inside surface.

5. The reactor closure head is stored dry on the operating floor during refueling, allowing direct access for inspection.

6. The insulation on the vessel closure and lower heads is removable allowing access for the visual and nondestructive examination of head welds.

7. Reactor vessel studs, nuts, and washers are removed for dry storage during refueling, allowing inspection in parallel with refueling operations.

8. Access holes in the core barrel flange allow access for remote visual examination of the clad surface of the vessel without removal of the lower internals assembly.

9. Access holes are provided in the primary shield providing access for the surface and visual examination of the primary nozzle safe end welds.

10. Manways are provided in the steam generators to provide access for internal inspection.

11. A manway is provided in the pressurizer top head to allow access for internal inspection.

12. Where practical, the insulation covering component and piping welds and adjacent base metal is designed for ease of removal and replacement in areas where external inspection is planned.

13. Openings are provided in the operating floor concrete above the reactor coolant pumps to permit removal of the pump motor to provide internal inspection access to the pumps.

14. The primary loop compartments are designed to allow personnel entry during refueling operations to permit direct inspection access to the external portion of piping and components.

Provisions are made for inspection access in the design of Class 2 components to ensure compliance with the requirements of IWC-2600.

The use of conventional nondestructive volumetric test techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component.

As indicated above, the only sophisticated remote inspection equipment currently required is for inspection of the reactor vessel. The baseline preservice inspection is performed onsite by Westinghouse utilizing a remote reactor vessel ultrasonic inspection tool which will perform the Code required inspection of the circumferential and longitudinal shell welds, the flange to vessel weld, the ligaments between the flange holes, the nozzle to vessel welds, and the nozzle to safe end to pipe welds.

5.7.3 EQUIPMENT FOR INSERVICE INSPECTIONS

The vessel inspection tool has 2 major structural components, the center column assembly which is supported and aligned from the vessel internals support flange by the head assembly. These 2 structures form the rigid base assembly from which the scanning arm and drive assemblies are mounted and operated. Two (2) configurations of the tool can be arranged depending on the areas of the vessel to be examined. In Figure 5.7-1, the scanning arm and transducer array, necessary to examine the vessel shell welds, are mounted on the main carriage assembly. The main carriage assembly can traverse the full length of the vessel up and down the center column and also provide 360 degrees circumferential rotation around the inside of the vessel. In Figure 5.7-2, the nozzle scanning assembly is mounted beneath the main carriage and can be rotated for 360 degrees to provide alignment with any of the vessel nozzles. The tool, in this configuration, can be installed in the vessel without removal of the lower internals package. The scanning attachment for the examination of the vessel flange weld and ligaments can be installed on the tool in either of the 2 configurations described above. Figures 5.7-1 and 5.7-2 show the inspection tool in both these configurations.

5.7.4 RECORDING AND COMPARING DATA

For reactor vessel scanning, a data acquisition system is used which combines simplicity with operational reliability and adaptability. It employs an electronic system with a receiver or data channel for each transducer unit. The signal received from the transducer is transmitted through an electronic distance amplitude correction device and, depending on the amplitude, gated through individual transigates to a printer. Simultaneously with the gated signal, information is received from the position indication encoder system which provides position coordinates. The resultant indication is printed on a recording device together with its position reference.

Following the completion of the scan of the areas scheduled for inspection, the tool operator(s) will, operating on manual control, return the transducer array to each of the indication position coordinates and evaluate the nature of the indication to sufficient extent to be able to determine the size, shape, and location. Special transducer arrays with variable angle search units are provided to assist in this activity.

For manual ultrasonic scanning, such as examination of a circumferential pipe weld, Westinghouse Nuclear Energy Systems has prepared a set of standard operator procedures to ensure that inspection results are recorded in a manner which will avoid any ambiguity in interpretation. The procedure specifies how any indication must be recorded with respect to scan direction and weld orientation, and how datum points, from which the location of indications are measured, are to be selected. Figure 5.7-3 shows a typical data sheet for recording instrument calibration settings for the volumetric examination of a circumferential pipe weld.

The data from the various baseline examinations, collected in accordance with the above procedures, were assembled into a comprehensive report tabulating the results.

The report describes the scope of the inspection, the procedures utilized, the equipment utilized, names and qualifications of the operators, and the examination results including instrument calibration criteria in sufficient detail to ensure repeatability for each examination.

5.7.5 REACTOR VESSEL ACCEPTANCE STANDARDS

The reactor vessel acceptance standards to be used during preoperational mapping of the vessel by ultrasonic examination met the requirements of IWB-3000.

5.7.6 COORDINATION OF INSPECTION EQUIPMENT WITH ACCESS PROVISIONS

The only areas where it is expected that high radiation levels will prohibit the access of personnel for direct examination of component areas or systems is the reactor vessel. The special design provisions and tooling required to perform the Code required examinations in these areas have been discussed above. Westinghouse is carrying out a continuing program of radiation surveys during refueling programs in operating plants to ensure that possible future problem areas are detected at an early stage. Should additional experience in the maintenance and inspection of operating plants indicate that other areas exist where access will be either limited or impossible, steps will be taken to develop any remotely operated inspection equipment considered necessary to meet the commitments of examinations required by Section XI of the Code.

HISTORICAL DATA ONLY

The detailed inservice inspection program is presented in the Technical Specifications, Section 3/4.4.10. This program is based on Section XI of the ASME Code. It is anticipated that the results obtained from compliance with the Technical Specifications, and the state of the art will be evaluated before establishing a long term inspection program. The data collected during the first 5 years of operation, along with the overall operating experiences, will be reviewed to determine the overall inspection program to be implemented for the lifetime of the facility. 00-01

5.7.7 PRESERVICE AND INSERVICE INSPECTION OF CLASS 1, 2, AND 3 COMPONENTS

5.7.7.1 Component Subject to Examination or Inspection

All ASME Code, Section III, Class 1 components subject to examination or inspection will be examined or tested, as applicable, in accordance with the requirements of Article IWB-2000 ASME Code Section XI.

All ASME Code, Section III, Class 2 components subject to examination or inspection will be examined or tested, as applicable, in accordance with the requirements of Article IWC-2000 ASME Code Section XI. The Technical Specifications specify Code Class 2

components requiring weld examinations, including the type(s) of examination as specified by Article IWC-2000 ASME Code Section XI.
ASME Code, Section III, Class 3 components, if any, subject to examination or inspection will be examined or tested in accordance with Article IWD-2000 ASME Code Section XI and any additional requirements of the Technical Specifications.

Code Class 2 and 3 pumps will be inservice tested in accordance with the ASME Code of Record, prescribed under 10CFR50.55a. RN 04-012

Code Class 1, 2, and 3 valves will be inservice tested in accordance with the ASME Code of Record, prescribed under 10CFR50.55a. RN 04-012

5.7.7.2 Accessibility

The design and arrangement of Code Class components, where possible, provides suitable access to welds for nondestructive testing and visual inspections. Special platforms are provided at the circumferential welds on the steam generator shells. Weld preparation, spacing, and clearance for welds on Code Class 2 components that are to be nondestructively examined are in accordance with ASME Section XI.

In general, the plant piping and equipment arrangement is designed with consideration for maintenance and inspection. Should accessibility prevent a particular inspection or examination from being performed adequately, this condition will be resolved in accordance with ASME Section XI and/or 10CFR50.55a. RN 99-110

5.7.7.3 Examination Techniques and Procedures

The examination techniques and procedures utilized are in accordance with the applicable requirements of ASME Section XI. Nondestructive examinations and visual procedures are in accordance with the requirements of ASME Section XI Code and its applicable references.

5.7.7.4 Inspection Frequency

Class 1 components are inspected in accordance with a schedule which satisfies the requirements of Article IWB-2000 ASME Code Section XI.

Class 2 components are inspected in accordance with a schedule which satisfies the requirements of Article IWC-2000 ASME Code Section XI.

Class 3 components, if any, are inspected in accordance with a schedule which satisfies the requirements of Article IWD-2000 ASME Code Section XI.

5.7.7.5 Examination Categories and Requirements

The inspection categories and requirements for Class 1 components are in compliance with Article IWB-2000 ASME Code Section XI.

The inspection categories and requirements for Class 2 components are in compliance with Article IWC-2000 ASME Code Section XI.

The inspection categories and requirements for Class 3 components, if any, are in compliance with Article IWD-2000 ASME Code Section XI.

5.7.7.6 Evaluation of Examination Results

The evaluation of Class 1, 2, and 3 component examination results complies with the requirements of Articles IWA-3000 and IWB-3000, IWC-3000, or IWD-3000 of the ASME Code Section XI, as appropriate. The acceptance standard(s) for indications detected during the examination are those of ASME Code Section XI. When acceptance standards are not addressed in the ASME Code Section XI, those of the construction code, ASME Section III, are used. When specified, Section III Edition and addenda are applied for the acceptance standard. 00-01

Repair procedures for flaws discovered during examination are developed in accordance with the requirements of Article IWA-4000 ASME Code Section XI and its applicable references for Class 1, 2, and 3 components.

5.7.7.7 System Pressure Tests

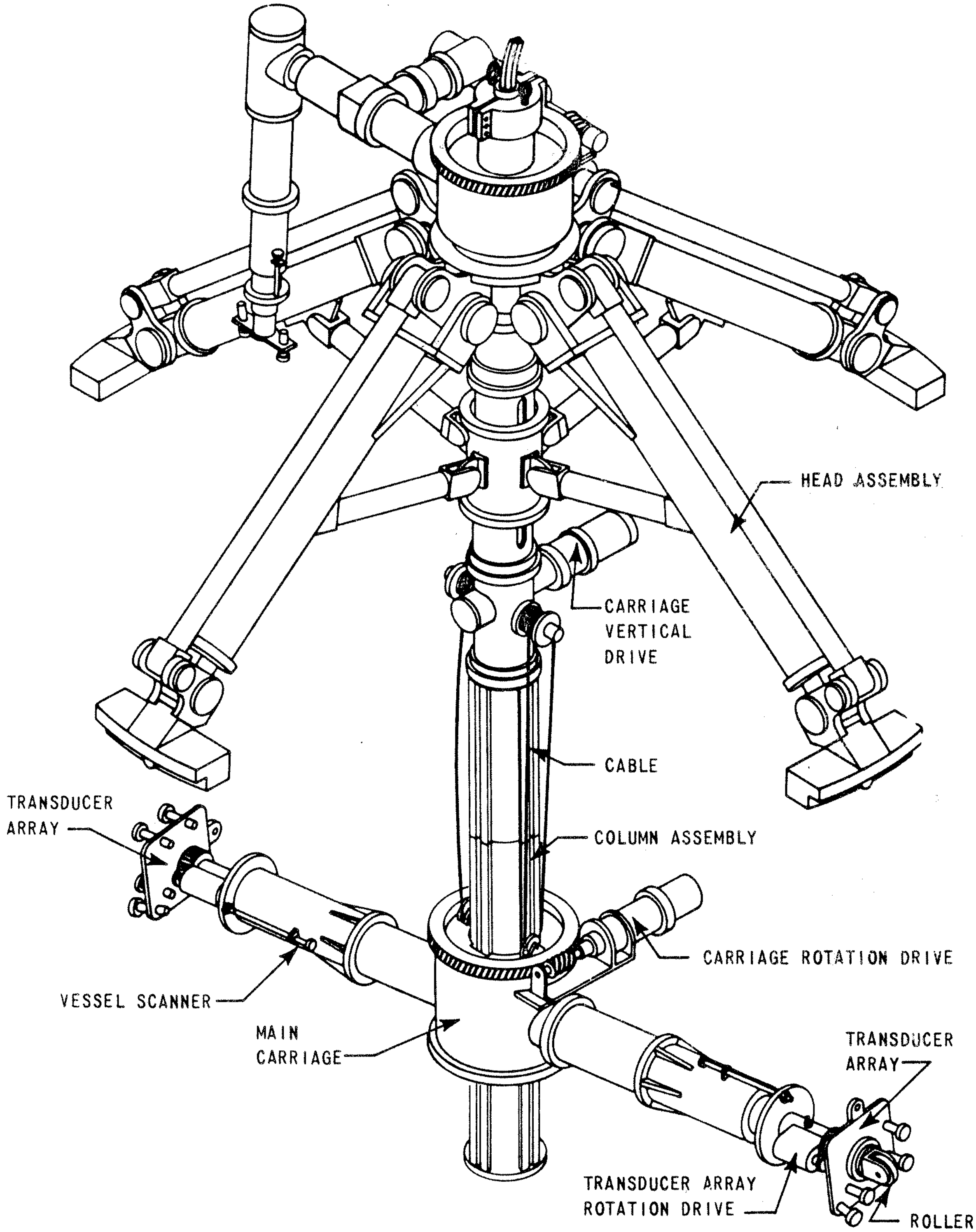
System pressure tests for Class 1 items comply with the testing and examination criteria of Articles IWA-5000 and IWB-5000, ASME Code Section XI.

System pressure tests for Class 2 items comply with the testing and examination criteria of Articles IWA-5000 and IWC-5000, ASME Code Section XI.

System pressure tests for Class 3 items, if any, comply with the testing and examination criteria of Articles IWA-5000 and IWD-5000, ASME Code Section XI.

5.7.8 AUGMENTED INSERVICE INSPECTION TO PROTECT AGAINST POSTULATED PIPING FAILURES

Design measures have been taken to ensure that the reactor vessel and essential equipment within or outside of the containment, including components of the reactor coolant pressure boundary, and other safety-related components have been adequately protected against the effects of blowdown jet, reactive forces, and pipe whip resulting from postulated rupture of piping located inside or outside of containment. These measures are discussed in Section 3.6. As explained earlier in this section, high energy fluid system piping is inspected in accordance with ASME XI. Augmented inspection of high energy fluid system piping between containment isolation valves is not deemed necessary.

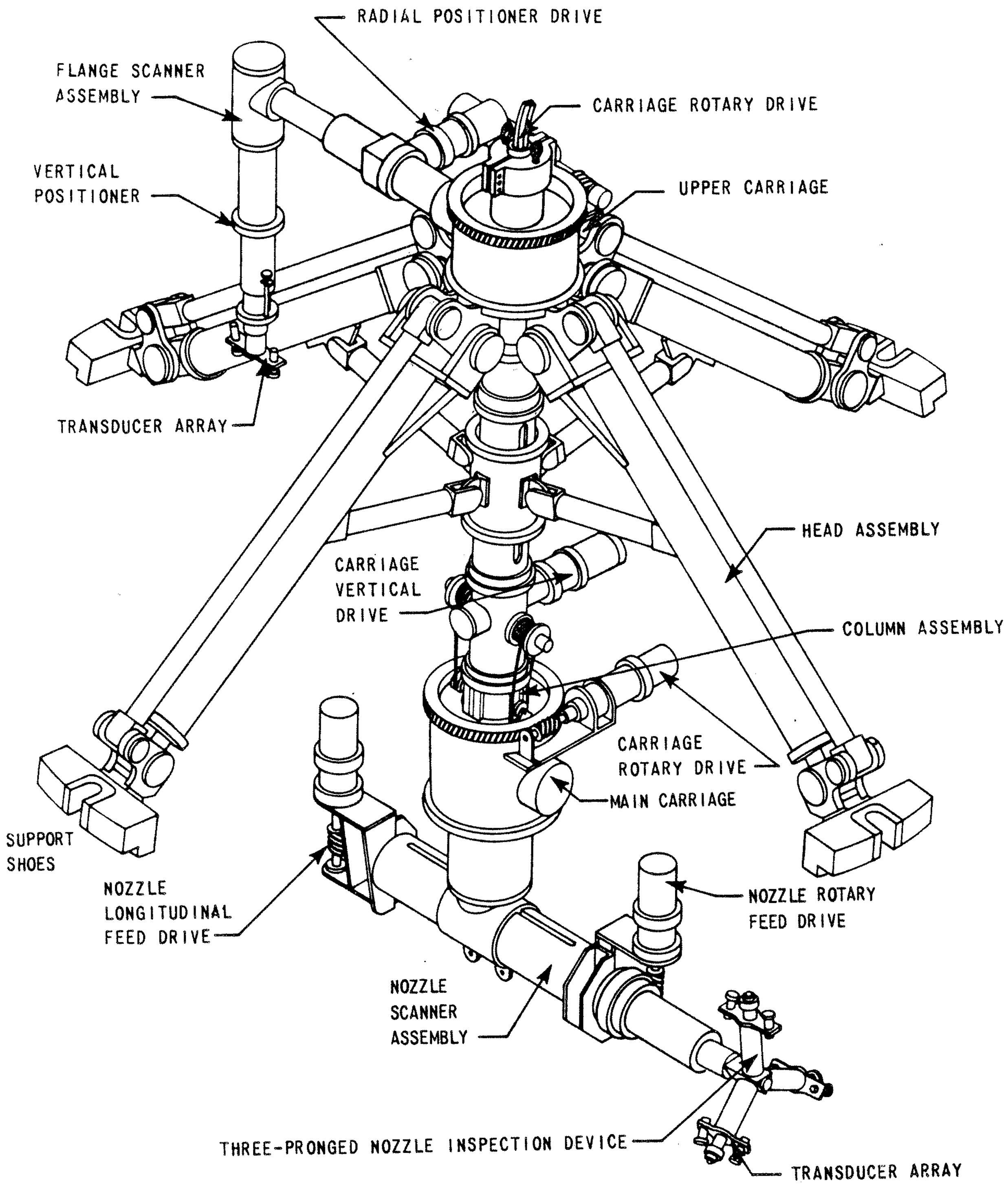


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Tool Details
(Vessel Scanner)

Figure 5.7-1



Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Tool Details
(Nozzle and Flange Scanner)

Figure 5.7-2

WELD DATA SHEET

Date:

Operators:

Weld No.________________

Instrument Serial No. ________________

Check List: Reject____ Damping____ Filter Out Rep. Rate 1000 T-Jack

SU Cable Length____ Reference Block____ Serial No.____

Search Unit ____ ____ ____ ____ ____ ____

Type MHz Size Beam Style Serial No.

Primary Reference Response ________________dB__________Gain

Calibration

Set S/RT hole to 7.5 div. amplitude, mark distance amplitude curve from other holes.

Transfer

1. On Ref. Block but off holes. Perform transfer method using straight beam search unit to give correction or a second angle beam search unit.

Search Unit ____ ____ ____ ____ ____ ____

Type 2X above MHz Size Beam Style Serial No.

2. Attenuate (dB) response for good signal. Note Signal amplitude__________
3. On vessel wall adjust gain to set signal amplitude as in 2 above.

   Record gain: __________

   gain

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VIRGIL C. SUMMER NUCLEAR STATION

Sample Weld Data Sheet
(Sheet 1 of 2)

Figure 5.7-3

WELD DATA SHEET (Continued)

4. Record transfer corrected gain________________________
____

5. Scanning at 4. -14 dB______________dB________________Gain
Indications exceeding DAC line to be investigated and recorded showing their position and dimensions in the weldment.

6. Evaluate Indication at 5. +14 dB_______________dB__________Gain
couplant: