CHAPTER 5

5.0   <u>REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS</u>

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Amendment No. 11

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1   SUMMARY DESCRIPTION

The Reactor Coolant System (RCS) consists of three similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a steam generator, a pump, loop piping, and instrumentation. The pressurizer surge line is connected to one of the loops. Auxiliary system piping connections into the reactor coolant piping are provided as necessary.

The principal heat removal systems which are interconnected with the RCS are the Steam and Power Conversion, Safety Injection, and Residual Heat Removal Systems. The RCS is dependent upon the steam generators, and the steam, feedwater, and condensate systems for stored and residual heat removal from normal operating conditions to a reactor coolant temperature of approximately 350ºF.

The RCS transfers the heat generated in the core to the steam generators where steam is generated to drive the turbine generator. Borated demineralized light water is circulated at the flow rate and temperature consistent with achieving the reactor core thermal hydraulic performance presented in Section 4. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber used in chemical shim control.

The RCS provides a boundary which contains the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits to acceptable values its uncontrolled release to the secondary system and to other parts of the plant under conditions of either normal or abnormal reactor operation. During transient operation the systems heat capacity attenuates thermal transients generated by the core or extracted by the steam generators. The RCS accommodates coolant volume changes within the protection system criteria.

The RCS design and operating pressure are listed in Table 5.1.0-1 together with the safety, power relief and pressurizer spray valves set points, and the protection system set point pressures. 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.

Coolant enters the reactor vessel through inlet nozzles in a plane just below the vessel flange and above the core. The coolant flows downward through the annular space between the vessel wall and the core barrel into a plenum at the bottom of the vessel. There it reverses direction to flow upward through the core. Approximately ninety-five percent of the total coolant flow is effective for heat removal from the core. The remainder of the flow includes the flow through the rod cluster control guide thimbles, the leakage across the reactor pressure vessel nozzles, and the flow deflected into the head of the vessel for cooling the upper flange. All the coolant is united and mixed in the upper plenum, and the mixed coolant stream then flows out of the vessel through exit nozzles located on the same plane as the inlet nozzles.

A one piece thermal shield, concentric with the reactor core, is located between the core barrel and the reactor vessel. The shield is bolted and welded to the top of the core barrel. The shield, which is cooled by the coolant on its downward pass, protects the vessel by attentuating much of the gamma radiation and some of the fast neutrons which escape from the core. This shield minimizes thermal stresses in the vessel which result from heat generated by the absorption of gamma energy.

Fifty core instrumentation nozzles are located on the lower head.

Elbow taps are used in the primary coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. The correlation between flow reduction and elbow tap read out has been well established by the following equation:

$$\frac{\Delta P}{\Delta P_o} = \frac{\omega}{\omega_o} 2$$

where: ΔP_o =   the referenced pressure differential at ω_o
        ω_o =   reference flow rate
        ΔP =   the pressure differential for ω at extrapolation point
        ω =   flow rate at extrapolation point

The full flow reference point is established during initial plant startup. The low flow trip point is then established by extrapolating along the correlation curve. The technique has been well established in providing core protection against low coolant flow in Westinghouse pressurized water reactor plants. The expected absolute accuracy of the channel is within 10 percent and field results have shown the repeatability of the trip point to be within 1 percent.

Pressure in the system is controlled by the pressurizer, where water and steam pressure is maintained through the use of electrical heaters and sprays. To minimize pressure variations due to contraction and expansion of the coolant, steam can either be formed by the heaters or condensed by a pressurizer spray. Instrumentation used in the pressure control system is described in Chapter 7. Spring loaded steam safety valves and power operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank, where the discharged steam is condensed and cooled by mixing with water.

By appropriate selection of the inertia of the reactor coolant pump (which affects pump coastdown), the thermal hydraulic effects which result from a loss of flow situation are reduced to a safe level. The layout of the system ensures natural circulation capability following a loss of flow to permit plant cooldown without overheating the core. Part of the system's piping is used by the Emergency Core Cooling System to deliver cooling water to the core during a loss-of-coolant accident.

In the event the condensers are not available to receive the steam generated by residual heat, the water stored in the feedwater system may be pumped into the steam generators and the resultant steam vented to the atmosphere. The auxiliary feedwater system will supply water to the steam generators in the event that the main feedwater pumps are inoperative. The system is described in Section 10.4.8.

During Online Cycle 28, a zinc injection program was implemented at RNP. The zinc injection tie-in is located upstream of the Volume Control Tank in the chemical volume control system.

Although the primary purpose of the RNP zinc addition program is source term reduction, an additional expected benefit is PWSCC mitigation.

TABLE 5.1.0-1

<u>REACTOR COOLANT SYSTEM DESIGN PARAMETERS AND PRESSURE SETTINGS</u>

Total Primary Heat Output, Mwt	2348*
Total Primary Heat Output, Btu/hr	8014×10^6
Number of Loops	3
Coolant Volume (liquid), Including Pressurizer	
Volume, ft^3	9343
Total Reactor Coolant Flow, gpm	265,500

	Pressure, psig
Design Pressure	2485
Operating Pressure (at pressurizer)	2235
Safety Valves	2485
Power Relief Valves	2335/2340
Pressurizer Spray Valves (open)	2260
High Pressure Trip	2376
High Pressure Alarm	2310
Low Pressure Trip	1844
Low Pressure Alarm	2185
Hydrostatic Test Pressure	3110

* Includes 2339 MWt from the reactor core and 9 MWt from non-nuclear heat sources.

5.1.1 SCHEMATIC FLOW DIAGRAM

The Schematic Flow Diagram for the RCS is shown in Figure 5.1.1-1.

5.1.2 PIPING AND INSTRUMENTATION DIAGRAM

A simplified Piping and Instrumentation Diagram for the RCS is shown in Figures 5.1.2-1 and 5.1.2-2.

5.1.3 ELEVATION DRAWING

The Elevation Drawing for the RCS is shown in Figure 5.1.3-1.
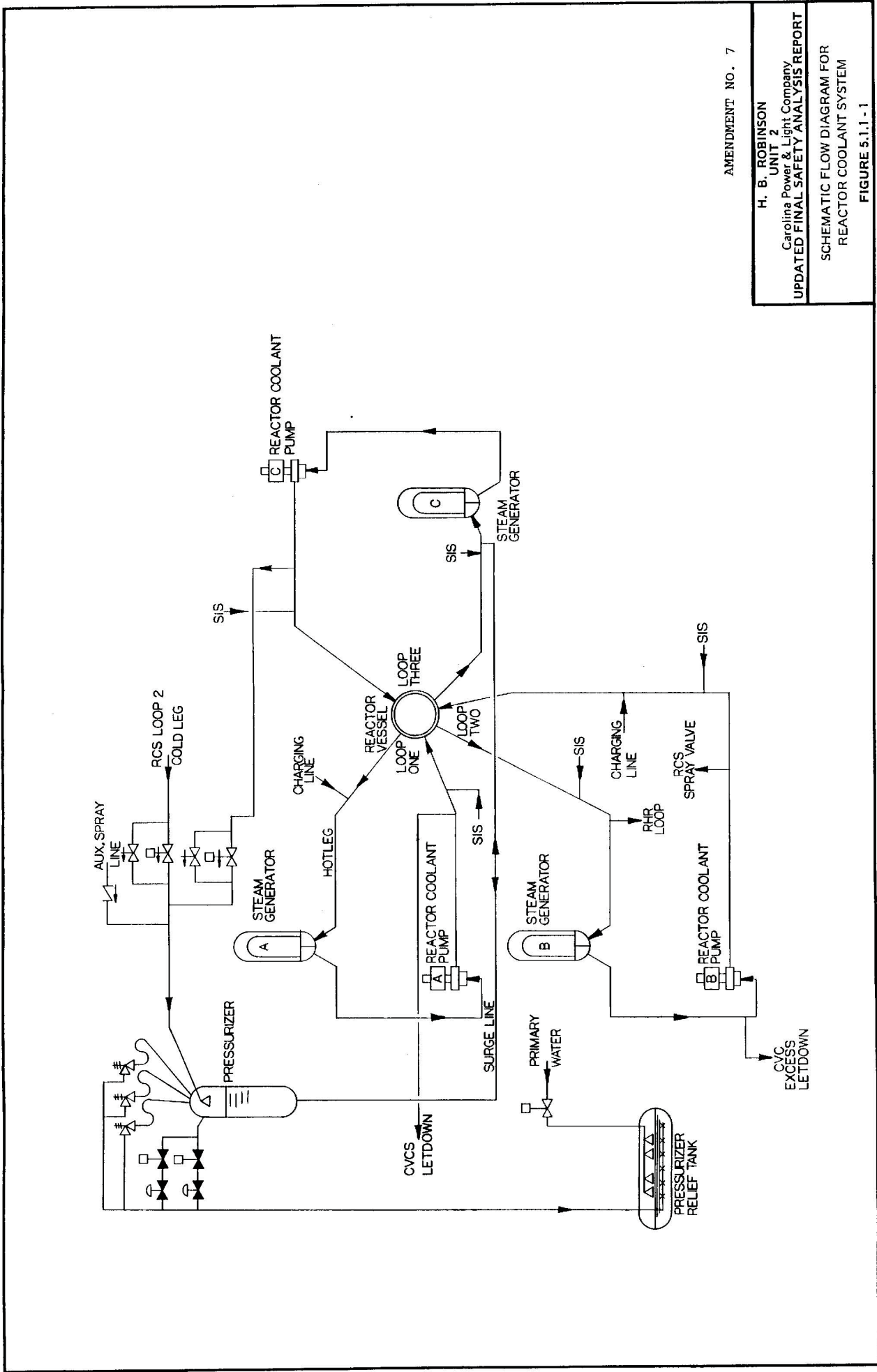
REACTOR COOLANT PUMP

RCS LOOP 2 COLD LEG

AUX. SPRAY LINE

PRESSURIZER

CVCS LETDOWN

PRIMARY WATER

PRESSURIZER RELIEF TANK

STEAM GENERATOR

CHARGING LINE

HOTLEG

REACTOR VESSEL

LOOP ONE

LOOP TWO

LOOP THREE

SiS

SiS

C

STEAM GENERATOR

A

REACTOR COOLANT PUMP

SURGE LINE

SIS

STEAM GENERATOR

B

REACTOR COOLANT PUMP

CVC EXCESS LETDOWN

SIS

CHARGING LINE

RHR LOOP

RCS SPRAY VALVE

SIS
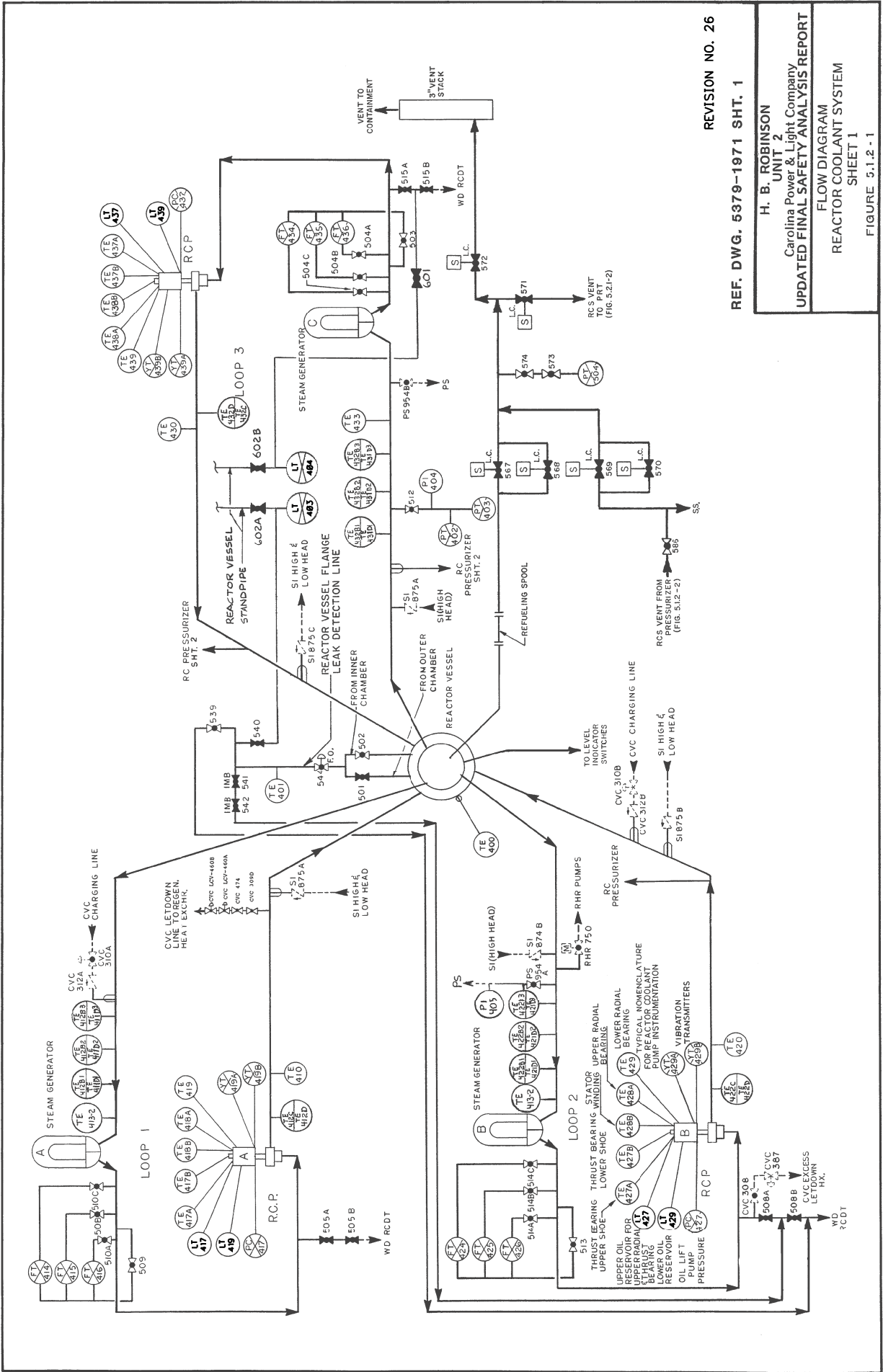
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FLOW DIAGRAM
REACTOR COOLANT SYSTEM
SHEET 1

FIGURE 5.1.2-1
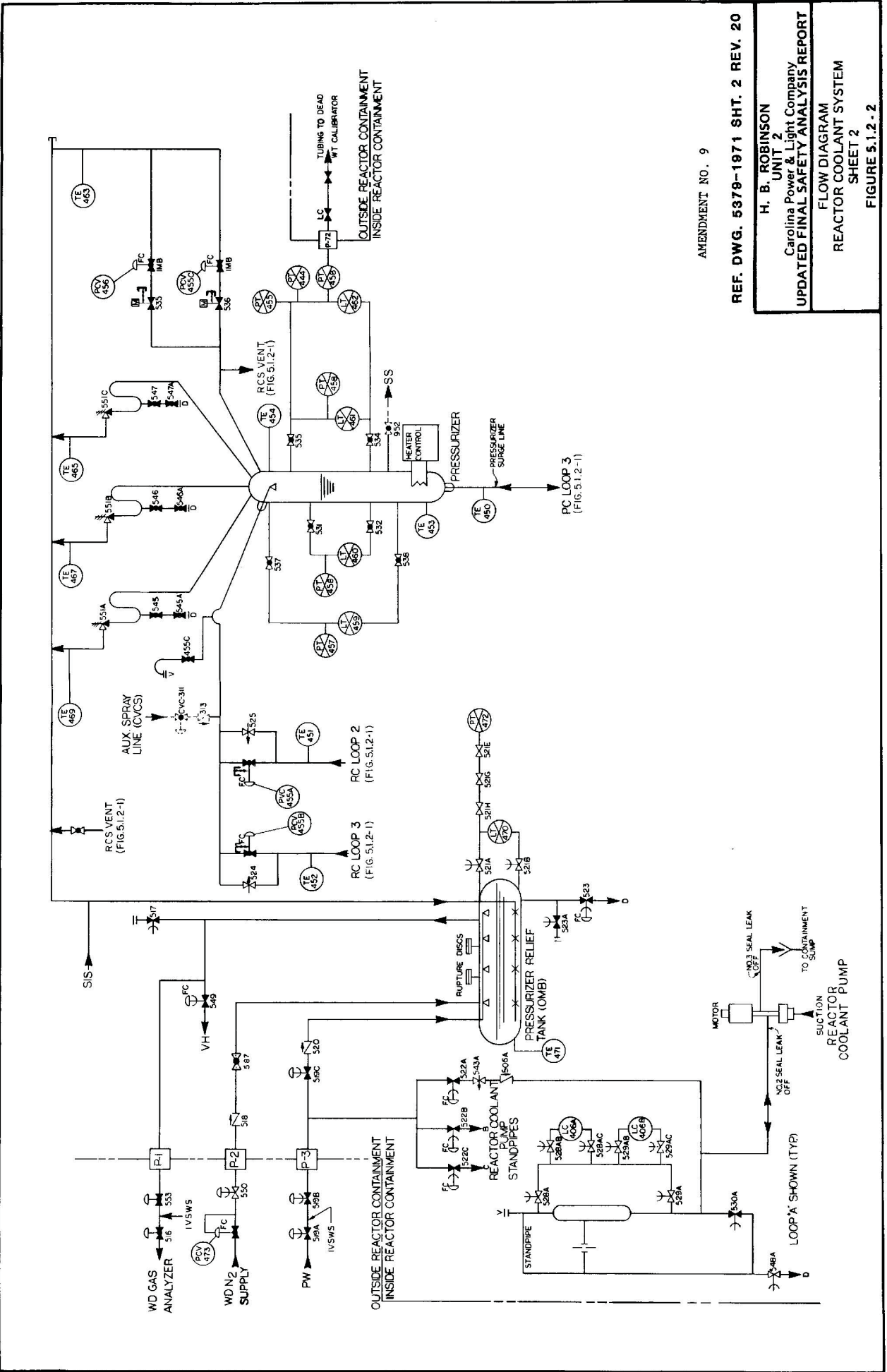
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FLOW DIAGRAM
REACTOR COOLANT SYSTEM
SHEET 2

FIGURE 5.1.2-2

AF 20448A-B

STEAM
GENERATOR
(LOOP 1)

FEED INLET
EL.285'-C

STEAM
GENERATOR
(LOOP 3)

EL.248-10 7/8

EL.245-2 7/8

EL.240-1 1/4

REACTOR
COOLANT
PUMP
(LOOP 1)

EL.248-107/8

EL.245-2 7/8

℄ EL. 242-2 1/16

7'-8 15/32

EL.233'-8 3/16
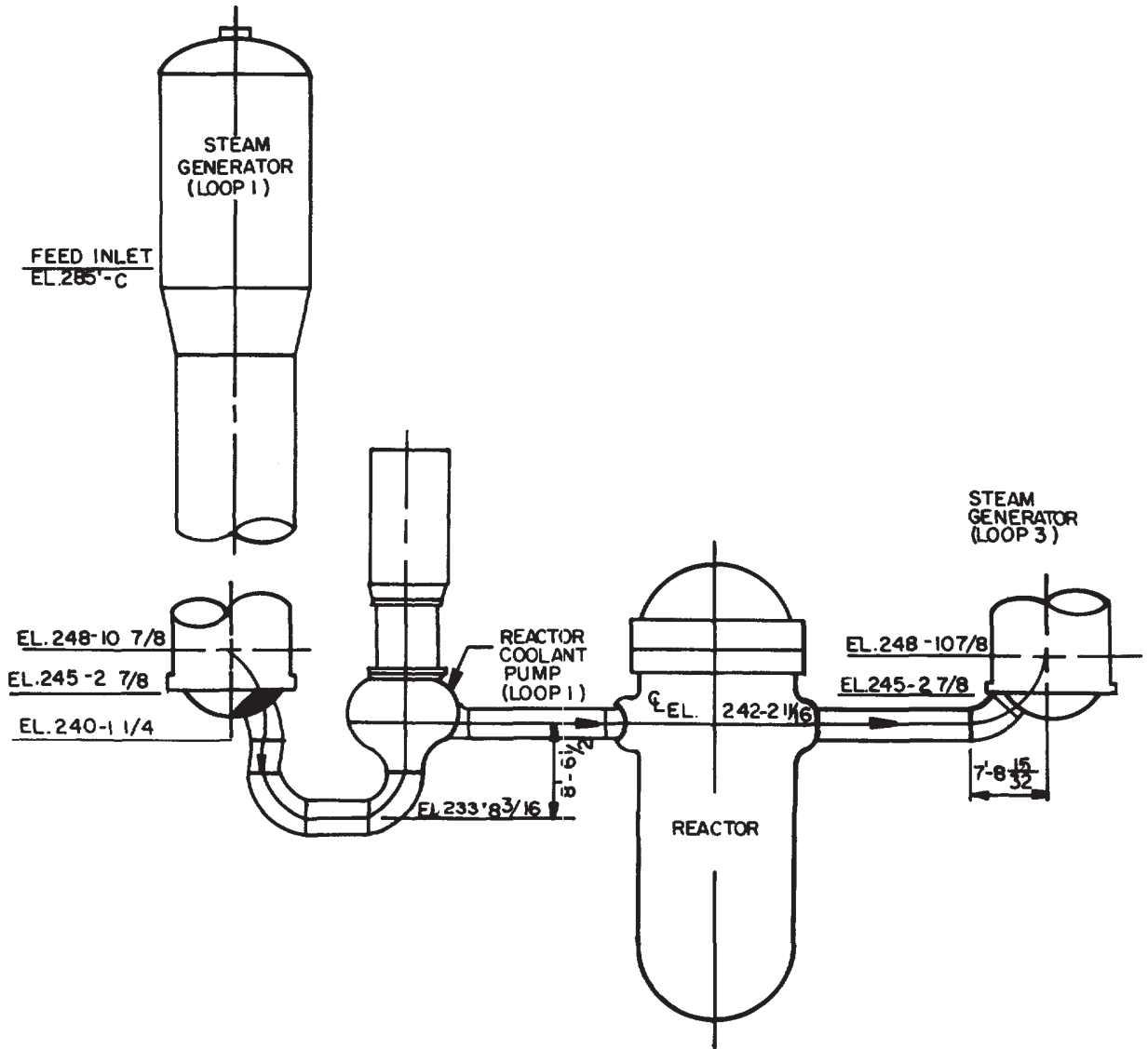
REACTOR

AMENDMENT NO.5

| H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT | REACTOR COOLANT SYSTEM ELEVATION | FIGURE 5.1.3 - 1 |

5.2   INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

5.2.1   COMPLIANCE WITH CODE

Section 3.2 shows the original construction code requirements for the reactor coolant system.

5.2.2   OVERPRESSURIZATION PROTECTION

The Reactor Coolant System (RCS) is protected against overpressure by control and protective circuits such as the high pressure trip and by code relief valves connected to the top head of the pressurizer. These power-operated relief valves and code safety valves are provided to protect against pressure surges which are beyond the pressure limiting capacity of the pressurizer spray.

5.2.2.1   Design Basis

The safety valves on the pressurizer are sized to prevent system pressure from exceeding the design pressure by more than 10 percent, in accordance with Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code. The capacity of the pressurizer safety valves is determined from considerations of the reactor protective system and accident or transient conditions which may potentially cause overpressure.

Details of the analysis are reported in Chapter 15.0. Experience has shown that the safety valve capacity so determined is adequate for all the other transients, as the results of Chapter 15.0 show.

5.2.2.2   Design Evaluation

The combined capacity of the safety valves is equal to or greater than the maximum surge rate resulting from complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve setting. The pressurizer safety and relief valves discharge into the pressurizer relief tank which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in Figure 5.1.2-2, and the valve design parameters are given in Table 5.4.10-1. Valve sizes are determined as indicated above. Reference 5.2.2-2 provides an evaluation of the safety and relief valve piping.

The normal source of motive power for PORVs is the plant nitrogen system. An accumulator in series with each PORV provides multiple cycles of operation if the nitrogen supply is lost. The alternate source of motive power is the instrument air system. If the nitrogen pressure to the valve operator falls below a preset value, instrument air is automatically aligned to the valve. An alarm signal alerts the operator whenever the PORVs start to open. This is in addition to the lights on the RTGB. Instrumentation with a display in the Control Room shows when the Safety Relief Valve opens.

The pressurizer relief tank is protected against a steam discharge exceeding the design pressure value by rupture discs which discharge into the reactor containment.

5.2.2.3   Operation Below 350ºF

The pressurizer power operated relief valves (PORV) are utilized to protect against exceeding safe pressure limits under low temperature conditions (Reference 5.2.2-1). A manual permissive switch is utilized to arm overpressure protection channel (one for each PORV) before the plant is below 350ºF and capable of a solid water condition (no steam bubble in the pressurizer). Operating solid can produce extreme pressure spikes that are

not encountered when there is the cushioning effect present from the steam bubble during normal plant operation. A nonredundant temperature comparator provides an annunciator signal when the loop temperature drops below 365ºF.

During normal plant operation, the permissive switch is not armed and the PORV are not operable in the low temperature/overpressure mode. With the system not armed, a reduction in temperature below 365ºF (normal cooldown procedures or abnormal conditions) causes the annunciator to energize signaling the operator to arm the overpressure protection system via the permissive switch. Redundancy for the arming function is provided by the plant operating procedures which require arming the system prior to decreasing RCS temperature to 350°F. When armed, the PORV become operable. Exceeding the setpoint with the system armed causes the PORV to open. The annunciator does not light under normal conditions if the system was armed prior to the temperature dropping below 365ºF. With the system armed below 365ºF, the next energizing of the annunciator indicates the setpoint has been exceeded and the PORV will open.

5.2.3   REACTOR COOLANT BOUNDARY MATERIALS

5.2.3.1   <u>aterial Specifications</u>

Each of the materials used in the RCS is selected for the expected environment and service conditions. The major component materials are listed in Table 5.2.3-1.

5.2.3.2   <u>Compatibility with Reactor Coolant</u>

All reactor coolant system materials which are exposed to the coolant are corrosion resistant. They consist of stainless steels and Inconel, and they are chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant. The chemical composition of the reactor coolant is maintained within the specification given in Table 5.2.3-2. The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of reactor coolant system surfaces.

All materials exposed to reactor coolant are corrosion resistant. Periodic analyses of the coolant chemical composition are performed to monitor the adherence of the system to the reactor coolant water quality listed in Table 5.2.3-2. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System (CVCS) and Sampling System which are described in Section 9.3.4.

5.2.3.3   <u>Fabrication and Processing of Ferritic and Austenitic Stainless Steel Materials</u>

Table 5.2.4-1 summarizes the quality assurance program with regard to inspections performed on primary system components. In addition to the inspections shown in Table 5.2.4-1, there are those which the equipment supplier performs to confirm the adequacy of material he receives, and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator were governed by ASME code requirements. The inspection procedures and acceptance standards required on pipe materials and piping fabrication were governed by United States American Standards (USAS) B31.1 and Westinghouse requirements and are equivalent to those performed on ASME coded vessels.

Procedures for performing the examinations were consistent with those established in the ASME Code Section III and were reviewed by qualified Westinghouse engineers. These procedures were developed to provide the highest assurance of quality material and fabrication. They considered not only the size of flaws, but equally as important, how the material was fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the surfaces most subject to damage as a result of the heat treating, rolling, forging, forming and fabricating processes, received a 100 percent surface inspection by magnetic particle or liquid penetrant testing after all these operations were completed. All reactor coolant plate material was subjected to shear as well as longitudinal ultrasonic testing to give maximum assurance of quality. (All forgings received the same inspection.) In addition, 100 percent of the material volume was covered in these tests as an added assurance over the grid basis required in the code.

Field erection and field welding of the reactor coolant system were performed such as to permit exact fit up of the 31 in. inside diameter (ID) closure pipe subassemblies between the steam generator and the reactor coolant pump (RCP). After installation of the pump casing and the steam generator, measurements were taken of the pipe length required to close the loop. Based on these measurements, the 31 in. ID closure pipe subassembly was properly machined and then erected and field welded to the pump suction nozzle and to the steam generator exit nozzle.

The phenomena of stress corrosion cracking and corrosion fatigue are not encountered unless a specific combination of conditions is present. The necessary conditions are a susceptible alloy, an aggressive environment, a stress, and time. It is characteristic of stress corrosion that combinations of alloy and environment which result in cracking are usually quite specific. Environments which have been shown to cause stress corrosion cracking of stainless steels are free alkalinity in the presence of a concentrating mechanism and the presence of chlorides and free oxygen. Another environment was identified by IE Bulletin 79-17, July 26, 1979, as stagnant borated water. The program at HBR 2 developed in response to this bulletin is discussed in Section 5.2.4 for Class 1 and Section 6.6 for Class 2 and 3.

Experience has shown that deposition of chemicals on the surface of tubes can occur in a steam blanketed area within a steam generator. In the presence of this environment, stress corrosion cracking can occur in stainless steels having the nominal residual stresses resulting from normal manufacturing procedures. However, the steam generator contains Inconel tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel Alloy has excellent resistance to general and pitting type corrosion in severe operating water conditions.

The selection of Inconel as the tube material for the HBR 2 steam generator was based on considerable experience with Inconel in steam generator and heat exchanger applications. Since 1962, widespread adoption of Inconel for steam generator tubes in nuclear stations was evident: as for example, Connecticut-Yankee; San Onofre; PM-1, Sundance; PM-3A, McMurdo Sound; Carolina-Virginia Tube Reactor; NPD; and Hanford N-Reactor. Materials with lead traces in the overall composition were present in the secondary side of the referenced plants. The use of lead in the materials of the secondary side of this plant has been minimized to the practical limit of that occurring as trace elements in metallurgical alloys, and is insignificant.

All external insulation of RCS components is compatible with the component materials. The cylindrical shell exterior and closure head, including flanges, of the reactor vessel, as well as the pressurizer upper head, upper shell barrel, and pressurizer relief discharge piping between the pressurizer and safety valves, are insulated with metallic reflective insulation. The channel head region of the steam generators was insulated with metallic reflective insulation in 1984 as part of the steam generator replacement. All other external corrosion resistant surfaces in the RCS are insulated with low or halide-free insulating material as required.

Cleaning of RCS piping and equipment was accomplished before and/or during erection of various equipment. Stainless steel piping was cleaned in sections as specific portions of the systems were erected. Pipe and units large enough

to permit entry by personnel were cleaned by locally applying approved solvents (Stoddard solvent, acetone and alcohol), and demineralized water, and by using a rotary disc sander or 18-8 wire brush to remove all trapped foreign particles.

Equipment specifications for fabrication required that suppliers submit the manufacturing procedures (welding, heat treating, etc.) to Westinghouse where they were reviewed by qualified Westinghouse engineers. This was also done on the field fabrication procedures to assure that installation welds were of equal quality.

Section III of the ASME B&PV Code requires that nozzles carrying significant external loads shall be attached to the shell by full penetration welds. This requirement has been carried out in the reactor coolant piping, where all auxiliary pipe connections to the reactor coolant loop were made using full penetration welds.

The RCS components were welded under procedures which required the use of both preheat and post heat. Preheat requirements, non-mandatory under Code rules, were performed on all weldments, including P1 and P3 materials which are the materials of construction in the reactor vessel, pressurizer and steam generators. Preheat and post heat of weldments both serve a common purpose; the production of tough, ductile metallurgical structures in the completed weldment. Preheating produces tough ductile welds by minimizing the formation of hard zones; whereas, post heating achieves this by tempering any hard zones which may have formed due to rapid cooling.

TABLE 5.2.3-1

MATERIALS OF CONSTRUCTION OF THE
<u>REACTOR COOLANT SYSTEM COMPONENTS</u>

<u>COMPONENT</u>	<u>SECTION</u>	<u>MATERIALS</u>
Reactor Vessel	Pressure Plate	SA-302, Gr. A SA-302, Gr. B
	Flange and Nozzle Forgings	A-508 Class II
	Cladding, Stainless Weld Rod	Type 304 equivalent
	Thermal Shield and Internals	A-240, Type 304
	Insulation	SS-SS Foil - SS
	Head Closure Bolting	A-540*
	Closure Head Forging	A-508 Gr. 3 Class 1
Steam Generator	Shell, Original	SA-302, Gr. B
	Channel Head Castings, Original	SA-216 WCC
	Heat Transfer U-Tubes	SB-163 thermally treated Code Case N-20
	Tube Plate	SA-508 Class 2A
	Lower Shell, Stub Barrel, and Shell Transition Class 2	SA-533 Grade A
	Cladding, Tube Sheet	Inconel
	Cladding, Channel Head	Type 304 or equivalent
Pressurizer	Shell	SA-302, Gr. B
	Heads	SA-216 WCC

_____

*Code case 1335.2 invoked.

TABLE 5.2.3-1 (Cont'd)

COMPONENT	SECTION	MATERIALS (or Equivalent)
	External Plate	SA-302, Gr. B
	Cladding, Stainless	Type 304 equivalent
	Internal Plate	SA-240 Type 304
	Internal Piping	SA-376 Type 316
Pressurizer Relief Tank	Shell	A-285 Gr. C
	Heads	A-285 Gr. C
Piping	Pipes	A-376 Type 316
	Fittings	A-351, CF8M or
		A-403, WP316
		A-403, WP316 (>2")
		A-182, F316 (1/2" - 2")
	Nozzles	A-182 F316
Pump	Shaft	Type 304
	Impeller	A-351, CF8
	Casing	A-351, CF8M
Valves	Pressure Containing Parts*	A-351, CF8M
		SA-182 F304 (RC 535 and 536 only)
		A-182 F316

_____

*Some bolting not in contact with reactor coolant is made of carbon steel,
 e.g. A-194, Gr. 7.

TABLE 5.2.3-2

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Electrical Conductivity	Determined by the concentration of boric acid and alkali present. Expected range is< 50 µmho/cm at 25ºC
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 9.5 (low boric acid concentration) at 25ºC
Oxygen, ppm, max (Mode 1, 2, 3 and Mode 4 with either the RCS or Pressurizer temperature greater than 250ºF)	0.1
Chloride, ppm, max (Mode 1, 2, 3, 4)	0.15
Fluoride, ppm, max (Mode 1, 2, 3, 4)	0.15
Hydrogen, cc (STP)/kg H_2O (Mode 1, 2)	25-50. Hydrogen levels may be reduced to 15 cc/kg 24 hours before shutdown without instituting Action Level 1. Hydrogen levels shall be ≥15 cc/kg but may be < 25 cc/kg for up to 24 hours after reaching reactor critical without instituting Action Level 1.
Total Suspended Solids, ppm, max (Prior to Power Operation)	0.2
pH Control Agent (^{7}LiOH) (Prior to to Power Operation)	Strong base alkali, 0.22 to 4.5 ppm as Li-7
Boric Acid as ppm B	Variable from 0 to 3000

5.2.4   IN-SERVICE INSPECTION AND TESTING OF REACTOR COOLANT PRESSURE
BOUNDARY

The Inservice Inspection (ISI) and Inservice Testing (IST) Programs for HBR 2 are in accordance
with the applicable rules and requirements of the ASME Section XI Code, Rules for Inservice
Inspection of Nuclear Power Plant Components. These programs are required by 10 CFR 50,
Sections 55a(f) for IST and 55a(g) for ISI. The ISI and IST programs comply with the Edition and/or
Addenda of the ASME Section XI Code specified in 10 CFR 50.55a(b)(2) or as approved by the
Nuclear Regulatory Commission (NRC) and are identified in site administrative procedures.

The Edition(s) and/or Addenda(s) of the ASME Section XI Code applicable to the ISI and IST
programs are defined in site administrative procedures.

The boundaries of Class 1 systems or portions thereof subject to examination and/or testing are
illustrated on the applicable system's Piping and Instrument Diagram (P&ID). These inspection
boundaries are illustrated by the use of flags which graphically define the system boundaries that are
subject to the ASME Section XI Code rules and requirements.

In response to IE Bulletin 79-17, the nondestructive examination (NDE) program to be implemented
for Class 1 portions of systems containing stagnant borated water is presented in References 3.9.6-
1 and 3.9.6-2. No evidence of intergranular stress corrosion cracking was found during any of the
inspections (Reference 3.9.6-2).

Written relief requests are granted for deviations to the ASME Section XI Inservice Inspection
requirements by the Commission pursuant to
10 CFR 50.55a(f)(6)(i) for IST and 10 CFR 50.55a(g)(6)(i) for ISI.

The requirements for Inservice Inspection of Class 2 and 3 components are described in Section
6.6.

The requirements for Inservice Testing of Class 1, 2, and 3 components are described in Section
3.9.6.

The Quality Assurance Program for the RCS is shown in Table 5.2.4-1.

During the design phase of the RCS, careful consideration was given to provide access for both
visual and nondestructive in-service inspection of primary loop components. The following
components and areas are available for visual and/or nondestructive inspection.

a)   Reactor Vessel - The entire inside surface

b)   Reactor Vessel Nozzles - The entire inside surface

c)   Closure Head - The entire inside and outside surface

d)   Reactor Vessel Studs, Nuts and Washers

e)   Field Welds between the Reactor Vessel, Steam Generators, and RCP and the Main Coolant
Piping

f)   Reactor Internals

g)   Reactor Vessel Flange Seal Surface

h)   Fuel Assemblies

i)   Rod Cluster Control Assemblies

j)   Control Rod Drive Shafts

k)   Control Rod Drive Mechanism Assemblies

l)   Main Coolant Pipe External Surfaces (except for the five ft penetration of the primary shield)

m)   Steam Generator - The external surface, the internal surfaces of the steam drum, and channel head

n)   Pressurizer - The internal and external surfaces

o)   RCP - The internal and external surfaces, motor and impeller

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

a)   All reactor internals are completely removable. The storage space required to permit these inspections is provided.

b)   The closure head is stored dry in the Reactor Building at elevation 226 ft during refueling to facilitate direct visual inspection.

c)   All reactor vessel studs, nuts and washers are removed to dry storage during refueling.

d)   Removable plugs are provided in the primary shield just above the coolant nozzles, and the insulation covering the nozzle welds is removable.

e)   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.

f)   A removable plug is provided in the lower core support plate to allow access for inspection of the bottom head without removal of the lower internals.

g)   The storage stands provided for storage of the internals allow for inspection access to both the inside and outside of the structures. The stand for the upper internals was provided as original equipment. The stand for the lower internals was purchased in 1981.

h)   The station provided for changeout of control rod clusters from one fuel assembly to another is specially designed to allow inspection of both fuel assemblies and control rod clusters.

i)   The control rod mechanism is specially designed to allow removal of the mechanism assembly from the reactor vessel head.

j)   Manways are provided in the steam generator, primary and secondary sides to allow access for internal inspection.

k)   A manway is provided in the pressurizer top head to allow access for internal inspection.

l)   All insulation on primary system components (except the reactor vessel) and piping (except for the penetration in the primary shield) is removable.

The use of conventional nondestructive, direct visual and remote visual 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. Because of these limitations on access to the reactor vessel, several steps have been incorporated into the design and manufacturing procedures in preparation for nondestructive testing. These are:

a)   Shop ultrasonic examinations were performed on all internally clad surfaces to an acceptance and repair standard to assure an adequate cladding bond to allow later ultrasonic testing of the base metal. Size of cladding bonding defect allowed was 3/4 in.

b)   The design of the reactor vessel shell in the core area is a clean, uncluttered cylindrical surface to permit positioning of test equipment without obstruction.

c)   During the manufacturing stage, selected areas of the reactor vessel were ultrasonic tested and mapped to facilitate possible future in-service inspection. The areas which were ultrasonically mapped include:

1)   Vessel flange radius, including the vessel flange to upper shell weld

2)   Middle shell course

3)   Lower shell course above the radial core supports

4)   Exterior surface of the closure head from the flange knuckle to the cooling shroud

5)   Nozzle to upper shell weld

6)   Middle shell to lower shell weld

7)   Upper shell to middle shell weld

The preoperational ultrasonic testing of these areas was performed after final stress relief.

Plans for in-service inspection of the reactor coolant system pressure envelope are discussed in Section 5.5 of the Technical Specifications and Section 3.9 of the Updated FSAR.

TABLE 5.2.4-1

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

COMPONENT	RT*	UT*	PT*	MT*	ET*
I. Steam Generator					
1.1 Tube Sheet					
1.1.1 Forging		yes		yes	
1.1.2 Cladding		yes(+)	yes(++)		
1.2 Channel Head					
1.2.1 Casting	yes			yes	
1.2.2 Cladding			yes		
1.3 Secondary Shell and Head					
1.3.1 Plates		yes			
1.4 Tubes		yes			yes
1.5 Nozzles (forgings)		yes		yes	
1.6 Weldments					
1.6.1 Shell, longitudinal	yes			yes	
1.6.2 Shell, circumferential	yes			yes	
1.6.3 Cladding (channel head- tube sheet joint cladding restoration)			yes		
1.6.4 Steam and feedwater nozzle to shell	yes			yes	
1.6.5 Support brackets				yes	

(+)   Flat Surfaces Only

(++)   Weld Deposit Areas Only

    *See last page of this table for explanation

TABLE 5.2.4-1 (Cont'd)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

COMPONENT	RT*	UT*	PT*	MT*	ET*
1.6.6 Tube to tube sheet			yes		
1.6.7 Instrument connections (primary and secondary)				yes	
1.6.8 Temporary attachments after removal				yes	
1.6.9 After hydrostatic test (all welds and complete channel head - where accessible)				yes	
1.6.10 Nozzle safe ends (forgings)	yes		yes		
1.6.11 Nozzle safe ends (weld deposit)			yes		
2. Pressurizer					
2.1 Heads					
2.1.1 Casting	yes			yes	
2.1.2 Cladding			yes		
2.2 Shell					
2.2.1 Plates		yes		yes	
2.2.2 Cladding			yes		
2.3 Heaters					
2.3.1 Tubing		yes	yes		
2.3.2 Centering of element	yes				
2.4 Nozzle	yes	yes			
2.5 Weldments					
2.5.1 Shell, longitudinal	yes			yes	
2.5.2 Shell, circumferential	yes			yes	

TABLE 5.2.4-1 (Cont'd)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

COMPONENT	RT*	UT*	PT*	MT*	ET*
2.5.3 Cladding			yes		
2.5.4 Nozzle safe end (forgings)	yes		yes		
2.5.5 Nozzle safe end (weld deposit)			yes		
2.5.6 Instrument connections			yes		
2.5.7 Support Skirt				yes	
2.5.8 Temporary attachments after removal				yes	
2.5.9 All welds after hydrostatic test				yes	
2.6 Final Assembly					
2.6.1 All accessible surfaces after hydrostatic test				yes	
3. Piping					
3.1 Fittings (castings)	yes		yes		
3.2 Fittings (forgings)		yes	yes		
3.3 Pipe		yes	yes		
3.4 Weldments					
3.4.1 Circumferential	yes		yes		
3.4.2 Nozzle to run pipe (No RT for nozzles less than 3 inches)	yes		yes		
3.4.3 Instrument connections			yes		
4. Pumps					
4.1 Castings	yes		yes		

TABLE 5.2.4-1 (Cont'd)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

COMPONENT	RT*	UT*	PT*	MT*	ET*
4.2 Forgings		yes			
4.2.1 Main shaft		yes	yes		
4.2.2 Main studs		yes	yes		
4.2.3 Flywheel (rolled plate)		yes			
4.3 Weldments					
4.3.1 Circumferential	yes		yes		
4.3.2 Instrument connections			yes		
5. Reactor Vessel					
5.1 Forgings					
5.1.1 Flanges		yes		yes	
5.1.2 Studs		yes		yes	
5.1.3 Head adapters		yes	yes		
5.1.4 Head adapter tube		yes	yes		
5.1.5 Instrumentation tube		yes	yes		
5.1.6 Main nozzles		yes		yes	
5.1.7 Nozzle safe ends		yes	yes		
5.2 Plates	yes		yes		
5.3 Weldments					
5.3.1 Main seam	yes			yes	
5.3.2 CRD head adapter connection			yes		
5.3.3 Instrumentation tube connection			yes		
5.3.4 Main nozzles	yes			yes	

TABLE 5.2.4-1 (Cont'd)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

COMPONENT	RT*	UT*	PT*	MT*	ET*
5.3.5 Cladding			yes[+++]	yes	
5.3.6 Nozzle safe ends (forging)		yes		yes	
5.3.7 Nozzle safe ends (weld deposit)				yes	
5.3.8 Head adaptor forging to head adaptor tube		yes		yes	
5.3.9 All welds after hydrotest					yes
6. Valves					
6.1 Castings	yes		yes		
6.2 Forgings (no UT for valves two inch and smaller)		yes	yes		

(+++) UT of Clad Bond-to-Base Metal

_____

*   RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle
ET - Eddy Current

5.2.5   DETECTION OF LEAKAGE THROUGH REACTOR COOLANT PRESSURE
        BOUNDARY

The existence of leakage from the RCS to the containment, regardless of the source of leakage, is detected by one or more of the following conditions:

a)   Two radiation sensitive instruments provide the capability for detection of leakage from the RCS. The containment air particulate monitor is quite sensitive to low leak rates. The containment radiogas monitor is much less sensitive but can be used as a backup to the air particulate monitor.

b)   The humidity detector is a third instrument used in leak detection. This provides a means of measuring overall leakage from all water and steam systems within the containment but is less sensitive than the radiation monitors. The humidity monitoring method is therefore used as a backup to the radiation monitoring methods.

c)   A leakage detection system is included which determines leakage losses from all water and steam systems within the containment, including that from the RCS. This system collects and measures moisture condensed from the containment atmosphere by the cooling coils of the containment air recirculation cooling units. It relies on the principle that all leakages up to sizes permissible with continued plant operation will be evaporated into the containment atmosphere. This system provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary.

d)   An increase in the amount of coolant makeup water which is required to maintain normal level in the pressurizer, or an increase in containment sump level are also used as leakage detection systems. However, these are less sensitive means of detection leakage.

To support the application of Leak Before Break methodology, at least one leakage detection system must be operable with a sensitivity capable of detecting a 1 gallon per minute leak within 4 hours in accordance with Reference 5.2.2.4.

5.2.5.1   System Description

5.2.5.1.1   Radiation Monitors

The sensitivity of the air particulate monitor to an increase in reactor coolant leak rate is dependent upon the magnitude of the normal base-line leakage into the containment. The sensitivity is greatest where base-line leakage is low as was demonstrated by the experience of Indian Point Unit 1, Yankee Rowe and Dresden Unit 1. Where containment air particulate activity is below the threshold of detectability, operation of the monitor with stationary filter paper would increase leak sensitivity to a few cubic centimeters per minute. Assuming a low background of containment air particulate radioactivity, a reactor coolant corrosion product radioactivity (Fe, Mn, Co, Cr) of 0.2 µCi/cc (a value consistent with little or no fuel cladding leakage), and complete dispersion of the leaking radioactive solids into the containment air, the air particulate monitor is capable of detecting leaks as small as approximately 0.013 gpm (50 cc/min) within twenty minutes after they occur. If only ten percent of the particulate activity is actually dispersed in the air, leakage rates of the order of 0.13 gpm (500 cc/min) are well within the detectable range.

For cases where base-line reactor coolant falls within the detectable limits of the air particulate monitor, the instrument can be adjusted to alarm on leakage increases from two to five times the base-line value.

The containment air particulate monitor together with the other radiation monitors mentioned in this Section are further described in Section 11.

The containment radioactive gas monitor is inherently less sensitive (threshold at 10^{-6} μCi/cc) than the containment air particulate monitor, and would function in the event that significant reactor coolant gaseous activity exists from fuel cladding defects. The measuring range is 10^{-6} to 10^{-3} μCi/cc.

Assuming a reactor coolant activity of 0.3 μCi/cc, the occurrence of a leak of 2 to 4 gpm would double the zero leakage background in less than 1 hour. In these circumstances this instrument is a useful backup to the air particulate monitor.

5.2.5.1.2   Humidity Detector

The humidity detection instrumentation offers another means of detection of leakage into the containment. Although this instrumentation has not nearly the sensitivity of the air particulate monitor, it has the characteristics of being sensitive to vapor originating from all sources within the containment including the reactor coolant and steam and feedwater systems. Increasing trends on the containment and dew point recorder should indicate incremental leakage.

The sensitivity of this method depends on cooling water temperature, containment air temperature variation and containment air recirculation rate.

5.2.5.1.3   Condensate Measuring System

This leak detection method is based on the principle that the condensate collected by the cooling coils matches, under equilibrium conditions, the leakage of water and steam from systems within the containment. This principle applies because conditions within the containment promote complete evaporation of leaking water from hot systems. The air and internal structure temperatures are normally held near 120ºF, the relative humidity of the air is well below the saturation point, and the cooling coils provide the only significant surfaces at or below the dew point temperature.

The containment cooling coils are designed to remove the sensible heat generated within the containment. The resulting large coil surface area has the effect that the exit air from the coils has a dew point temperature which is very nearly equal to the cooling water temperature.

Measurement of the condensate drained from each of the fan cooler units is made to determine condensation rate and thus leak rate.

Should a leak occur, the condensation rate will increase above the previous steady state due to the increased vapor content of the fan cooler air intake. A new equilibrium rate will be approached within approximately 30 minutes

after the start of the leak. Detection of the increasing condensation rate is possible, however, within 5 to 10 minutes for condensation rates in the order of 1/2 gpm and larger. Readout of the condensate flow measuring device and high flow alarms is provided in the control room.

5.2.5.2   Design and Operation

When significant leakage from the RCS is detected, action is taken to prevent the release of radioactivity to the atmosphere outside the plant.

If either the containment air particulated gamma activity or the radioactive gas activity exceeds pre-set levels on the containment air particulate and radioactive gas monitors, respectively, the containment purge supply and exhaust duct valves and pressure relief line valves are closed.

If a leak from the RCS to the component cooling loop was a gross leak, or if the leak could not be isolated from the component cooling loop before the inflow completely filled the surge tank, the relief valve on the surge tank would rise. The discharge from this valve is routed to the waste holdup tank in the Auxiliary Building.

A large leak in the RCS pressure boundary which does not flow into another closed loop would result in reactor coolant flowing into the containment sump.

5.2.5.2.1   Leakage Prevention

RCS components were manufactured to exacting specification which exceeded normal code requirements. In addition, because of the welded construction of the RCS and the extensive nondestructive testing to which it was subjected (as outlined in Section 5.2.4), leakage through metal surfaces or welded joints is very unlikely.

However, some leakage from the RCS is permitted by the RCP seals. Also, all sealed joints are potential sources of leakage even though the most appropriate sealing device is selected in each case. Thus, because of the large number of joints and the difficulty of assuring complete freedom from leakage in each case, a small integrated leakage is considered acceptable. Leakage from the reactor through its head flange will leak off between the double O-ring seal and actuate an alarm in the Control Room.

5.2.5.2.2   Locating Leaks

Experience has shown that hydrostatic testing is successful in locating leaks in a pressure containing system.

Methods of leak location which can be used during plant shutdown include visual observation for escaping steam or water or for the presence of boric acid crystals near the leak. The boric acid crystals are transported outside the RCS in the leaking fluid and deposited by the evaporation process. Portable sonic detectors sensitive to ultrasonic frequencies provide another means for locating small leaks.

5.2.5.3   Normal Leakage Paths

In considering possible leakage from the RCS containing primary coolant at high pressure, four categories were considered:

a)   Leakage to the reactor coolant drain tank
b)   Leakage to the pressurizer relief tank
c)   Leakage to the containment environment
d)   Leakage to the interconnecting systems

For clarity, each of these paths is discussed in turn.

5.2.5.3.1   Paths Directed To The Reactor Coolant Drain Tank (RCDT)

The paths directed to the RCDT are as follows:

a)   RCS Loop Drains
b)   Accumulator Drains
c)   Auxiliary System Equipment Drains
d)   Excess Let-Down
e)   RCP Seal Leakage
f)   Reactor Flange Leakoff

Of these paths, a) through d) do not present a leakage load on the RCDT during normal operation; leakage from the high pressure systems is not expected due to the use of double isolation valves. Leakage from paths e) through f) are discussed in detail below:

5.2.5.3.1.1   Deleted by Amendment No. 9.

5.2.5.3.1.2 Reactor Coolant Pump Seals

Charging flow is directed to the RCP via a seal water injection filter. It enters the pumps at a point between the labyrinth seals and the No. 1 face seal. Here the flow splits and a portion (normally about 5 gpm) enters the reactor coolant system via the labyrinth seals and thermal barrier cooler cavity. The remainder of the flow (normally about 3 gpm) flows up the pump shaft cooling the lower bearing, and leaves the pump via the No. 1 seal where its pressure is reduced to about 25-30 psig and its temperature is increased from 130ºF to about 136ºF. The labyrinth flows (15 gpm total for three RCP) flow to a common manifold and then via a filter (seal water filter) through the seal water heat exchanger (where the temperature is reduced to about 130ºF) to the volume control tank.

In the event of a loss of all seal cooling, reactor coolant begins to travel along the RCP shaft and displace the cooler seal injection water. The shutdown seal (SDS) actuates once the No. 1 seal package temperature reaches the SDS actuation temperature. SDS actuation limits the loss of reactor coolant through the RCP seal package.

The leakoff system between No. 2 and No. 3 seals is considered to be part of the RCS. The leakoff system collects leakage passed by the No. 2 seal, provides a constant backpressure on the No. 2 seal and constant pressure on the No. 3 seal. A standpipe is provided to give a constant backpressure during normal operation. The first outlet from the standpipe is orificed to permit normal No. 2 seal leakage to flow to the reactor coolant drain tank; excessive No. 2 leakage will result in a rise in the standpipe level and eventual overflow to the reactor coolant drain tank via a second overflow connection.

The normal No. 2 seal flow is 3 gph/pump. This is the value specified in the RCP Equipment Specification.

Level instrumentation on the standpipes is provided to alert the operator to abnormal conditions. The standpipe consists of a pipe with an orificed overflow at the mid-point, a normally closed drain (for service) at the bottom, and a free flowing overflow at the top. Normal No. 2 seal leakage will flow freely out the mid-point overflow. Excessive leakage will "back-up" in the standpipe until it overflows out the top. A level switch in the upper

standpipe actuates an annunciator indicating excessive flow. A level switch in the lower standpipe causes annunciation of the opposite condition which could result in undesirable dry operation of the No. 3 seal.

5.2.5.3.1.3   Reactor vessel flange leakoff

The reactor vessel flange and head is sealed by two metallic O-rings. To facilitate leakage detection, a leakoff connection is placed beyond the outer O-ring. Piping and associated valving is provided to direct any leakage to the reactor coolant drain tank.

During normal operation, it is anticipated that the leakage will be negligible since it is specified in the Reactor Vessel Equipment Specification that there is to be zero leakage past the outer O-ring under normal operating and transient conditions.

A temperature detector will indicate leakage by a high temperature alarm. The operator is further alerted by the associated increase in drain tank water temperature and eventually the change in tank level.

5.2.5.3.2   Paths directed to Pressurizer Relief Tank (PRT)

The PRT condenses and cools the discharge from the pressurizer safety and relief valves. Discharges from smaller relief valves located inside the containment are also piped to the PRT. During normal operation, leakage from the pressurizer safety valves, the pressurizer relief valves, SI Cold Leg Injection Header Relief Valve, or the CVCS letdown station relief valve is piped to the PRT.

During normal operation, the leakage to the PRT is expected to be negligible since the valves are designed for essentially zero leakage at the normal system operating pressure, as specified in the respective valve equipment specifications.

For the pressurizer safety and relief valves, temperature detectors are provided in the discharge piping to alert the operator to possible leakage.

The rate of increase of the water temperature in the PRT and the level change will indicate to the operator the magnitude of the leakage. In the event of excessive leakage into an interconnecting system causing lifting of the local relief valves, the operator would again be alerted to the situation by a rising tank water temperature.

5.2.5.3.3   Releases to the containment environment

The main contributors of leakage to the containment environment may be listed as follows:

1.   Valve stem leakage

2.   RCP seal No. 3 leakage

3.   Flange leakages

5.2.5.3.3.1   Valve Stem Leakage

The modulating valves within the containment have been live loaded and had leakoff lines capped. Of the remaining valves which serve lines and components containing reactor coolant, only two are not normally fully open or fully closed; i.e., the continuous spray bypass needle valves around the main spray valves. The remaining valves are of the back-seated type which prevent the valve stem packings from being subjected to high pressures when in the open position.

Normally open valves have backseats which can limit leakage to less than one cubic centimeter per hr per in. of stem diameter assuming no credit for packing in the valve except LCV-460A&B which limit leakage to less than one cubic centimeter per hour per inch of backseat diameter. Normally closed globe valves are installed with recirculation flow under the seat to prevent stem leakage from the more radioactive fluid side of the seat.

On the basis of these pessimistic assumptions, the leakage from valves is estimated to be approximately 52.125 cc/hr.

5.2.5.3.3.2   Reactor Coolant Pump Seal No. 3 Leakage

During normal operation, a small continuous leakage is anticipated past the No. 3 seal to the containment environment; this fluid will be charging water. The No. 3 seal leakoff is diverted to the local open drains and is thus released to the containment environment.

The fluid will be charging water and is anticipated to be of the order 100 cc/hr per pump. This is the value specified in the RCP Equipment Specification.

5.2.5.3.3.3   Flange Joints

There are a number of flanged joints in the system all of which will be subjected to leak testing before power operation. Experience has shown that hydrostatic testing is successful in locating leaks in a pressure containing system.

Methods of leak location which can be used during plant shutdown include visual observation for escaping steam or water or for the presence of boric acid crystals near the leak. The boric acid crystals are transported outside the RCS in the leaking fluid and deposited by the evaporation process.

5.2.5.3.3.4   Conclusion

On the basis of the above, the analysis of the situation indicated a total leak rate to the containment environment on the order 352.125 cc/hr. For design purposes, 40 lb/day (i.e. 800 cc/hr) is assumed.

5.2.5.3.4   Leakage to Interconnecting Systems

Each of the interconnecting systems are dealt with in turn in Table 5.2.5-1.

Although leakage of primary fluid to the secondary system via the steam generator primary/secondary boundary is not expected during normal operation because of the conservative design of the U-tubes in the steam generator, any such leakage would result in an increase of activity level in the secondary system and would be detected by the condenser air ejector gas monitor, by the main steam line N-16 monitor, or by the steam generator liquid blowdown monitor. (Reference is made to Section 11.5 of the FSAR).

TABLE 5.2.5-1

<u>LEAKAGE TO INTERCONNECTING SYSTEMS</u>

<u>SYSTEM</u>	<u>DISCUSSION</u>
CVCS	This is a normally operating interconnecting system; with redundancy for isolating purposes, if required.
Sampling System	In the event of sample valves failing to close or seat, adequate redundancy is provided by containment isolation valves; the piping between the sets of valves is designed for RCS pressure.
RHR Hot Leg Connection	Two isolation valves are provided. In the unlikely event of leakage past the two valves, interconnecting piping is provided to enable pressure relief via the RHR loop relief valve to the pressurizer relief tank.
RHR Cold Leg Connection	In the unlikely event of leakage past the accumulator check valves, RHR loop check valves and the motorized isolation valves, pressure relief will take place via the RHR loop relief valve to the pressurizer relief tank.
RHR Cold Leg Connection	In the unlikely event of leakage past the accumulator check valves, RHR loop check valves and the motorized isolation valves, pressure relief will take place via the RHR loop relief valve to the pressurizer relief tank.

TABLE 5.2.5-1 (Continued)

SYSTEM	DISCUSSION
SIS High Head Pump Injection Lines	In the event of leakage past two check valves in the cold leg lines, and one check valve and a normally closed globe valve in the hot leg lines, pressure relief will take place to the PRT via relief valves off of a SI cold leg line and a SIS Test Line, respectively.
SIS Accumulator Connections	Provisions have been made to check the leak tightness of the accumulator check valves. The implications of leakage past these valves are discussed in Section 6.2 of the FSAR.

REFERENCES: SECTION 5.2

5.2.2.1   Letter No. NG-77-1215, "Reactor Vessel Overpressurization Protection," CP&L to US NRC, dated October, 31, 1977.

5.2.2.2   Letter LAP-83-161, dated August 30, 1983, CP&L to NRC, "Pressurizer Safety and Relief Valve Piping System Evaluation"

5.2.2.3   WCAP-9558, "Mechanistic Fracture Evaluation of Reactor Coolant Piping Containing a Postulated Circumferential Through-Wall Crack," dated May 1981.

5.2.2.4   WCAP-9787, "Tensile and Toughness Properties of Primary Piping Weld Metal For Use In Mechanistic Fracture Evalaution," dated May 1981.

5.2.2.5   Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Report Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," dated February 1, 1984.

5.3   REACTOR VESSEL

The reactor vessel is cylindrical in shape with a hemispherical bottom and a flanged and gasketed removable upper head. 5.3.0-1 is a schematic of the reactor vessel. The reactor vessel design data are listed in Table 5.3.0-1.

TABLE 5.3.0-1

REACTOR VESSEL DESIGN DATA (2339 MW)

Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure, psig	3110
Design Temperature, °F	650
Overall Height of Vessel and Closure Head, ft-in.	
(Bottom Head OD to top of Control Rod Mechanism	
Housing)	41-6
Water Volume, (with core and internals in place),	
ft^3	3668
Thickness of Insulation, min, in.	3
Number of Reactor Closure Head Studs	50
Diameter of Reactor Closure Head Studs, in.	7
Flange, ID, in.	149.6
Flange, OD, in.	189
ID at Shell, in.	155.5
Inlet Nozzle ID, in.	Tapered 27-7/16 to 33-3/4
Outlet Nozzle ID, in.	28.97
Clad Thicknes, min, in.	0.156
Lower Head Thickness, min, in.	5.187
Vessel Belt-Line Thickness, min, in.	9.312
Closure Head Thickness, min, in.	7.75
Reactor Coolant Inlet Temperature, °F	546.6
Reactor Coolant Outlet Temperature, °F	604.2
Reactor Coolant Flow, lb/hr	1.01×10^8

5.3.1   REACTOR VESSEL MATERIALS

5.3.1.1   Material Specifications

The materials of construction of the reactor pressure vessel (RPV) are given in Table 5.2.3-1 and 5.3.1-1.

5.3.1.2   Special Processes Used for Manufacturing and Fabrication

The RPV material was heat treated specifically to obtain good notch ductility which ensured a low nil-ductility transition temperature (RT_{NDT}), and thereby gave assurance that the finished vessel could be initially hydrostatically tested and operated as near to room temperature as possible without restrictions. The stress limits established for the RPV are dependent upon the temperatures at which the stresses are applied. As a result of fast neutron irradiation in the region of the core, the material properties will change, including an increase in the RT_{NDT}. An initial maximum value of RT_{NDT} in this region had been established during fabrication and is listed in Reference 5.3.1-13.

5.3.1.3   Special Methods for Nondestructive Examination

Westinghouse required, as part of its RPV specification, that certain special tests which were not specified by the applicable codes be performed. These tests are listed below:

a)   Ultrasonic Testing - Westinghouse required that a 100 percent volumetric ultrasonic test of RPV plate for both shear wave and longitudinal wave be performed. Section III Class A vessel plates were required by code to receive only a longitudinal wave ultrasonic test on a 9 in. x 9 in. grid. The 100 percent volumetric ultrasonic test was a severe requirement, but it assured that the plate was of the highest quality.

b)   Radiation Surveillance Program (discussed in Section 5.3.1.6)

5.3.1.4   Special Controls for Ferritic and Austenitic Stainless Steels

The HBR 2 RPV was manufactured by Combustion Engineering and shipped to the site on July 12, 1968. It was fabricated from A302B and A302A plate material, with three vertical weld seams in each shell course, and three shell courses, as shown in Figure 5.3.1-1. The plate in the core region (beltline) is A302A. The numerical designations of the plates and welds used in the vessel are also shown in Figure 5.3.1-1, along with the location of the reactor core, relative to these materials.

The chemistry of the three plates used on the intermediate shell course was reported in Reference 5.3.1-2. The copper and nickel content of all beltline plates are presented in Table 5.3.1-3.

Identification of the beltline region welds are presented in Table 5.3.1-1. This information was obtained from weld inspection records from Combustion Engineering. As noted in this table, the circumferential seams were welded using the practice of adding nickel to the weld by a separate cold wire nickel feed process. The nickel addition welds were commonly referred to as Ni-200 welds. The longitudinal weld seams were made in 1964 prior to the practice of adding nickel to welds so that nickel content of the welds is low, < 20 percent.

There are no records of "as deposited" weld chemistry available for the longitudinal welds. All the longitudinal welds were made at Combustion Engineering during a time period when low nickel welds were the standard practice. Welds during this time period were made with RACO 3 weld wire and ARCOS B-5 flux. The basis for the chemistries established as representative of the longitudinal weld is defined in Reference 15.3.1-12 and the representative chemistry is shown in Table 5.3.1-1.

The circumferential welds in the core beltline region were both fabricated using the "Ni-200" cold wire nickel feed process. However, as discussed in detail in Reference 5.3.1-15, these nickel addition welds can be subdivided into two subsets which are characterized by a major difference in the welding process. Certain welds were made using a single arc primary electrode feed in combination with a single cold wire feed into the molten weld pool; however, most Ni-200 welds were fabricated using two primary electrodes (tandem arc) with the addition of the single cold wire nickel feed. The effects of this difference on the relative nickel content of the resulting welds are discussed in Reference 5.3.1-15. Table 5.3.1-1 identifies which processes are applicable to the HBR2 welds of concern and what resulting chemistries are considered representative for each weld.

The weld seams in the Robinson vessel were constructed with backchipping at the inner surface of the vessel which removed material from the inner surface to a stated depth of 3/8 in. for the circumferential weld. The backchipped regions were subsequently rewelded so that the welds in this region contain approximately 0.05 percent copper. The justification of this weld chemistry is given in Reference 5.3.1-4. This effect is used as a secondary argument for circumferential weld integrity. Backchipping is not considered for the longitudinal welds since the inner surface was machined, thereby removing the backchipped region.

5.3.1.5   Vessel Integrity

The ability of the steel pressure vessel that contains the reactor core and its primary coolant to resist fracture constitutes an important safety issue. The beltline region of the RPV is the most critical region of the vessel because it is subjected to neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel steels such as ASTM-A302 Grade A and B parent material of the H. B. Robinson Unit 2 (HBR2) RPV are reported in the literature. Generally, low alloy ferritic materials show an increase in hardness and other strength properties and a decrease in ductility. In pressure vessel material, the most important mechanical property changes are the reduction in the upper shelf impact strength and an increase in the temperature for the transition from brittle to ductile fracture.

The method for guarding against fast fracture in reactor pressure vessels presented in Appendix G, "Protection Against Non-Ductile Failure," to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code utilizes fracture mechanics concepts and is based on the reference nil-ductility temperature, RT_{NDT}.

RT_{NDT} is defined as the greater of:

a)   The drop weight nil-ductility transition temperature (NDTT per ASTM E-208), or

b)   The temperature 60ºF less than the 50 ft-lb (or 35 mils lateral expansion) temperature as determined from Charpy specimens oriented in a direction normal to the major working direction of the material.

In addition to the Appendix G requirements, the NRC staff developed in 1983, proposed generic screening criteria for resolution of the Pressurized Thermal Shock (PTS) issue (Reference 5.3.1-1). Methods for determining the original reference temperature for welds were included. A proposed rule published on February 7, 1984 and revised on December 19, 1995 (Reference 5.3.1-10) was refined from the 1984 version and issued as 10 CFR 50.61.

CP&L has documented compliance with 10 CFR 50.61 in the submittal in response to Generic Letter 92-01, Revision 1, and Supplement 1 and related correspondence.

5.3.1.5.1   Reactor Vessel Materials Program

The Reactor Vessel Materials Program described in Reference 5.3.1-12 concluded that the reactor vessel beltline welds are significantly less sensitive to embrittlement from neutron radiation than previously assumed. This program included research in examining weld records associated with the HBR2 beltline welds, review of available documentation for welds similar to the HBR2 beltline welds, and sampling and analyzing welds similar to the HBR2 beltline welds. Subsequent records research detailed in Reference 5.3.1-15 has further refined the assigned weld chemistry of the HBR2 vessel beltline circumferential welds and demonstrated that the welds were even less susceptible to radiation embrittlement than previously predicted by the Materials Program. Based on the results of this program, and the test results from surveillance Capsule X, CP&L has recalculated the end of current license RT_{NDT} values for the HBR2 beltline welds. The actual calculations are contained in References 5.3.1-19 and 5.3.1-20 and utilize the methodology outlined in the PTS rule published in 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." The fluence values utilized are outlined in Reference 5.3.1-20.

The results show that the upper to intermediate shell girth weld is the limiting weld for PTS concerns and the RT_{PTS} at end of license will not exceed the PTS screening criteria.

Projected 60-year Upper Shelf Energy (USE) values were calculated using 60-year fluence values. The projected USE values for plate materials exceed the 42 ft-lb minimum acceptance criteria provided in WCAP-15828 (Reference 5.3.1-20). The projected USE values for welds and for forged nozzles exceed the 50 ft-lb screening criteria established by 10 CFR 50, Appendix G.

Also, Reference 5.3.1-20 contains calculations made in accordance with the rule on PTS to show that the reactor vessel plate material will not approach the proposed PTS screening criteria prior to the expiration of the HBR2 operating license.

The NRC issued a Safety Evaluation Report (Reference 5.3.1-21) containing the details of the materials program through the period of license extension and including the results of Capsule X for pressurized thermal shock.

5.3.1.6   Material Surveillance

In the reactor vessel surveillance program, the evaluation of radiation damage is based on pre-irradiation and post-irradiation testing of Charpy V-notch, tensile and wedge opening loading (WOL) test specimens. A description of the program basis including the material to be tested, specimen, and capsule design, and pre-irradiation test results is presented in Reference 5.3.1-6. This program is 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, and will be conducted to be in accordance with ASTM E185, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors."

The original reactor vessel surveillance program used eight specimen capsules which are located about 3 in. from the vessel wall directly opposite the center portion of the core (see Figure 5.3.1-4). The capsules contain reactor vessel steel specimens of shell plate as located in the core region of the reactor and associated surveillance weld made with the same filler metal and flux as used in the upper circumferential weld. In addition, correlation monitors made from fully documented specimens of SA302 Grade B material obtained through ASTM Subcommittee E10 on Radioisotopes and Radiation Effects are inserted in the capsules. The capsules contained at least 27 tensile specimens, 200 Charpy V-notch specimens and 36 WOL specimens. Dosimeters including pure Ni, Al-Co (0.15 percent), Cd shielded Al-Co, Cd shielded Np-237 and Cd shielded U-238 are used in some of the capsules (see Reference 5.3.1-2 for details). The dosimeters permit evaluation of the flux seen by the specimens and vessel wall. In addition, thermal monitors made of low melting alloys can provide proof of high temperatures experienced by specimens. All of the materials are enclosed in a tight-fitting stainless steel sheath to prevent corrosion.

The analytical methodology and the design basis currently used to predict time averaged fast neutron flux and fluence levels within the pressure vessel/surveillance capsule geometry are discussed in detail in Reference 5.3.1-3. Additional HBR2 plant specific analyses have been performed to include geometric, material, and power distribution information fully consistent with the above methodology and to provide a sound basis for the prediction of the long term fast neutron environment to which the pressure vessel will be exposed (Reference 5.3.1-19).

Geometric information for use in neutron transport calculations is provided in Figures 5.3.1-2 through 5.3.1-4. In Figure 5.3.1-2, a plan view of the reactor at the core midplane is depicted. This figure shows the reactor core, lower internals, pressure vessel, and the inner diameter of the primary biological shield. Pertinent dimensional information is also included in Figure 5.3.1-2. In Figure 5.3.1-3, a detailed description of the surveillance capsule geometry and associated structure is provided. This information is sufficient to allow accurate determinations of capsule lead factors as well as spectrum averaged reaction cross-sections for dosimetry applications. In Figure 5.3.1-4, the azimuthal location of each of the capsules included in the reactor vessel surveillance program is illustrated.

Since initial startup when the surveillance program was established based on Reference 5.3.1-6, five surveillance capsules have been withdrawn from the HBR2

reactor. In 1973, Capsules S and Z were removed from the 280º and 230º azimuthal position while in the 1975-1976 outage, Capsule V was withdrawn from the 290º location. Neutron dosimetry from both Capsules S and V were evaluated by Southwest Research Institute (SWRI), and the results were documented in SWRI 02-3574 and SWRI 02-4397, respectively. Early in 1977, the results for Capsule V (Reference 5.3.1-2) were revised in a letter from E. B. Norris to T. Clements (Reference 5.3.1-7). Capsule Z was not analyzed. During the 1982 refueling outage (Cycle 9), Capsule T was removed. Also, two capsules were moved from lag positions and placed in lead positions, and an additional capsule was installed on the thermal shield, which has been used to confirm the calculated reduction in flux from the low leakage core. In 2001, Capsule X was removed and evaluated by Westinghouse. The results were documented in WCAP-15805 (Reference 5.3.1-19).

Capsule X was shipped to the George Westinghouse Science and Technology Center for testing. The results of these tests are documented in Reference 5.3.1-19. Charpy V notch test points were grouped in the transition regions as opposed to the shelf regions to better define the 30 foot pound shift. The RT_{NDT} shift for weld metal irradiated to 4.49×10^{19} n/cm^2 was 266ºF at 30 foot pounds and 252ºF at 50 foot pounds. The comparison of actual RT_{NDT} versus the predicted using the methods of Regulatory Guide 1.99 Revision 2 for weld metal, plate material, and monitor correlation material is given by Table 5.3.1-5.

The following tabulates the remaining schedule for removal of the Reactor Vessel Materials Surveillance Capsules U, Y, and W:

Capsule	Effective Full Power Years
U (See Note 1)	38.0
Y (See Note 2)	Reserve
W (See Note 2)	Reserve

Note 1:   Capsule to location 280º at Cycle 8. Capsule will reach a fluence of approximately 7.84×10^{19} n/cm^2 (66 EFPY peak fluence) at approximately 38.0 EFPY. Thus, Capsule U should be withdrawn at 38.0 EFPY, or the next scheduled outage after the 80 year peak vessel fluence is reached.

Note 2:   Capsule Y was relocated to the 290º holder position and Capsule W was relocated to the 270º holder position during RO-27.

For any surveillance capsules removed and tested after January 2004, the test specimens are to be stored in lieu of disposal after testing (NTM 84955, License Renewal Commitment 22 and Operating License Condition 3.L).

A comparison of calculated and measured results based on the Fe^{54} (n, p) Mn^{54} reaction is provided in Table 5.3.1-4.

The temperature monitors in Capsule T did not melt showing the temperature remained below 579ºF for the first eight fuel cycles.

5.3.1.7   Reactor Vessel Fasteners

The reactor closure head and the reactor vessel flange are joined by fifty (50) 7-inch diameter studs. Two metallic O-rings seal the reactor vessel when the reactor closure head is bolted in place. A leakoff connection is provided between the two O-rings to monitor leakage across the inner O-ring. In addition, a leakoff connection is also provided beyond the outer O-ring seal.

The stud material is ASTM A-540, which at design temperature has a minimum yield strength of 104,400 psi in accordance with Code Case 1335.2. The membrane stress in the studs when they are at the steady state operational condition is approximately 37,500 psi. This means that as few as nineteen of the fifty studs are required in order to withstand the hydrostatic end load on vessel head without the membrane stress exceeding yield strength of the stud material at design temperature.

TABLE 5.3.1-1

IDENTIFICATION OF H. B. ROBINSON UNIT 2 VESSEL WELD MATERIALS

Weld Location[a]	Seam No. [a]	Weld Process	Weld Wire Type/Heat No.	Fkux Type/Lot	CU Content %	Ni Content %
Upper Shell Longitudinal Seams	1-273A 1-273B 1-273C	Note B	RACO 3/86054B	ARC0SB5/ 4D5F	0.22	0.054
Intermediate Shell Longitudinal Seams	2-0273A 2-0273B 2-0273B	Note B	RACO 3/86054B	ARC0SB5 4E5F	0.22	0.054
Lower Shell Longitudinal Seams	3-0273A 3-0273A 3-0273A	Note b	RACO 3/86054B	ARC0SB5	0.22	0.054
Upper Circumferential Weld	10-0273A	Note C	RACO 3/W5214 Ni-200/N7753A	Linde 1092/ 3617	0.208	1.01
Lower Circumferential Weld	11-0273A	Note D	RACO 3/34B009 Ni-200/N9879A	Linde 1092/ 3724	0.19	0.98
Surveillance Weld		Note C	RACO 3/W5214 Ni-200/N7753A	Linde 1092/ 3617	0.34	0.66

Note A – See Figure 5.3.1.1 for seam locations.
Note B - Submerged arc; no Ni wire added.
Note C - Submerged arc; tandem arc process; Ni wire added.
Note D - Submerged arc; single arc process; Ni wire added.
Note E - See Reference 5.3.1-15.

TABLE 5.3.1-2

CHEMISTRY FOR REPRESENTATIVE LOW NICKEL WELDS
(RACO3 WELD WIRE, ARCOS B-5 FLUX)

	C	Mn	Si	P	S	Mo	Cu	V	Ni
San Onofre 1	.11	1.50	.35	.017	.013	.47	.19	.03	-
Connecticut Yankee	.17	1.25	.32	.015	.011	.53	.22	-	.11
Big Rock Point	.12	1.25	.28	.014	.012	.53	.27	-	.10
Jose Cabrera	.047	1.36	.35	.020	.019	.48	.22	-	.046

COPPER AND NICKEL ANALYSES OF H. B. ROBINSON UNIT 2 VESSEL PLATE MATERIALS

Plate ID[a]	Type	Cu Content %	Ni Content %
W10201-1	A302A	.13	.11
W10201-2	A302A	.15	.25
W10201-3	A302A	.11	.08
W10201-4	A302A	.12	.09
W10201-5	A302A	.10	.12
W10201-6	A302A	.09	.09
W9807-3	A302A	.12	.10
W9807-5	A302A	.15	.10
W9807-9	A302A	.14	.15

Note (a) - See Figure 5.3.1-1 for plate locations.

TABLE 5.3.1-4

RESULTS OF FAST NEUTRON DOSIMETRY FOR SURVEILLANCE CAPSULES

Capsule	Fast Neutron Flux $(n/cm^2 \ sec)$	
	Calculated	Best Estimate
V	5.29×10^{10}	6.56×10^{10}
S	1.19×10^{11}	1.25×10^{11}
T	1.69×10^{11}	1.98×10^{11}
X[a]	9.31×10^{10}	9.88×10^{10}

(a) Exposure rates for Capsule X are determined for the 0•• location average over fuel Cycles 9 through 20.

TABLE 5.3.1-5

Comparison of the H.B. Robinson Unit 2 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions						
Material	Capsule	Fluence (x 10^{19} n/cm^2)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F)[a]	Measured (°F)[b]	Predicted (%)[a]	Measured (%)[c]
Inter. Shell Plate WI020I-4 (Longitudinal)	S	0.479	45.39	32.51	18	10
	X	4.49	78.86	104.73	30	1
Surveillance Program Weld Metal	V	0.530	179.17	209.32	39	38
	T	3.87	293.68	288.15	52	46
	X	4.49	300.64	265.93	54	29
Heat Affected Zone Material	V	0.530	---	59.21	---	26
	T	3.87	---	(d)	---	24
	X	4.49	---	210.13	---	19
Correlation Monitor Material	S	0.479	---	72.79	---	3
	V	0.530	---	69.39	---	5
	T	3.87	---	156.83	---	5
	X	4.49	---	125.21	---	0

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.
(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1.
(c) Values are based on the definition of Upper Shelf Energy given in ASTM E185-82.
(d) Only 2 specimens were tested from Capsule T to confirm the Upper Shelf Energy. Thus, there is insufficient data to determine the measured 30 ft-lb shift.

5.3.2 PRESSURE - TEMPERATURE LIMITS

5.3.2.1 Limit Curves

The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IC} curve) which appears in Appendix G of the ASME Code. The K_{IC} curve is a lower bound static fracture toughness result obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IC} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

The value of RT_{NDT}, and in turn the operating limits of nuclear power plants, can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of a given reactor pressure vessel are monitored by a surveillance program (see Section 5.3.1.6). The increase in the Charpy V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} ($RT_{NDT\ initial} + \Delta RT_{NDT}$) is utilized to index the material to the K_{IC} curve and in turn to set operating limits for the nuclear power plant which take into account the effects of irradiation on the reactor vessel materials. Allowable pressure temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G to Section XI of the ASME B&PV Code and ASME Code Case N-641, which allows the use of K_{IC} and alternative methodology when a circumferential weld has the highest adjusted RT_{NDT}. The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IC} curve in Appendix G for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients.

Following the generation of pressure temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point by point comparison of the steady state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the OD to the ID location; and the pressure limit must, at all times, be based on the most conservative case. The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at the ID position. The thermal gradients induced during cooldown tend to produce tensile stresses at the ID location and compressive stresses at the OD position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The reactor system operating cycles used for design purposes are given in Table 3.9.1-1. The normal system heating and cooling rate is 50ºF/hr. Sufficient electrical heaters are installed in the pressurizer to permit a heatup rate, starting with a minimum water level of 55ºF/hr. This rate takes into account the small continuous spray flow provided to maintain the pressurizer liquid homogeneous with the coolant.

The fastest cooldown rates result from the hypothetical case of a break of a main steam line which is discussed in Chapter 15.

The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) are in accordance with the Technical Specifications (Appendix A to the Facility Operating License).

These limiting rates (Technical Specifications Heatup and Cooldown Pressure/Temperature Limit figures) are updated periodically in accordance with the requirements with the following criteria and procedures before the calculated exposure of the vessel exceeds the exposures for which they apply.

1.   At least 60 days before the end of the integrated power period for which Technical Specifications Heatup and Cooldown Pressure/Temperature Limit figures apply, the limit lines on the figures shall be updated for a new integrated power period utilizing methods derived from the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, and in accordance with applicable appendices of 10 CFR 50. These limit lines shall reflect any changes in predicted vessel neutron fluence over the integrated power period or changes resulting from the irradiation specimen measurement program.

REFERENCES: SECTION 5.3

5.3.1-1   NRC Policy Issue, SECY-82-465, Pressurized Thermal Shock (PTS) November 23, 1982.

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5.3.1-3   Meyer, T. A., "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants," (WCAP - 10019), December, 1981.

5.3.1-4   C. E. Power Systems (L. C. Miller) letter to Carolina Power & Light Company (B. Furr), "Backwelding of Longitudinal and Circumferential Weld Seams and Transmittal of Typical Weld Metal Chemical Analysis," December, 1981.

5.3.1-5   Carolina Power & Light Company letter, "Thermal Shock to Reactor Pressure Vessels," E. E. Utley to D. G. Eisenhut (NRC), January 25, 1982, with attachments.

5.3.1-6   Yanichko, S. E., "Carolina Power & Light Company, H. B. Robinson Unit 2 Reactor Vessel Radiation Surveillance Program," Westinghouse Nuclear Energy System (WCAP-7373), (January, 1970).

5.3.1-7   SRI letter to Carolina Power & Light Co. (Revision to Reference 5.3.1-1), E. B. Norris to T. Clements, April 4, 1977.

5.3.1-8   Deleted.

5.3.1-9   Deleted.

5.3.1-10   "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Volume 60, No. 243, dated December 19, 1995.

5.3.1-11   Yanichko, S.E., "Analysis of Capsule T from the H. B. Robinson Unit 2 Reactor Vessel Radiation Surveillance Program (WCAP-10304)," Westinghouse Electric Corporation, March 1983.

5.3.1-12   Carolina Power & Light letter "Pressurized Thermal Shock - Materials Properties," A. B. Cutter to H. R. Denton (NRC), June 29, 1984.

5.3.1-13   Carolina Power & Light letter "Pressurized Thermal Shock - Reactor Vessel Materials," M. A. McDuffie to S. Varga (NRC), October 30, 1984.

5.3.1-14   Deleted.

5.3.1-15   Carolina Power & Light letter "Revised Pressurized Thermal Shock Response,"
            S. R. Zimmerman to NRC, January 16, 1987.

5.3.1-16   Deleted.

5.3.1-17   Letter LAP-83-79, dated April 1, 1983, CP&L to NRC, "Pressurized
            Thermal Shock."

5.3.1-18   Letter LAP-83-345, dated July 25, 1983, CP&L to NRC, "Flux Reduction
            Program."

5.3.1-19   T.J Laubham, et. al., "Analysis of Capsule X from the Carolina Power and Light
            Company, H. B. Robinson, Unit 2 Reactor Vessel Radiation Surveillance
            Program (WCAP-15805)," Westinghouse Electric Company LLC, March 2002.

5.3.1-20   T.J Laubham, "Evaluation of Pressurized Thermal Shock for H. B. Robinson, Unit
            2 (WCAP-15828)," Westinghouse Electric Company LLC, May 2002.

5.3.1-21   "NRC Safety Evaluation Report Related to the License Renewal of the H. B.
            Robinson Steam Electric Plant, Unit 2," U.S. Nuclear Regulatory Commission,
            January, 2004.
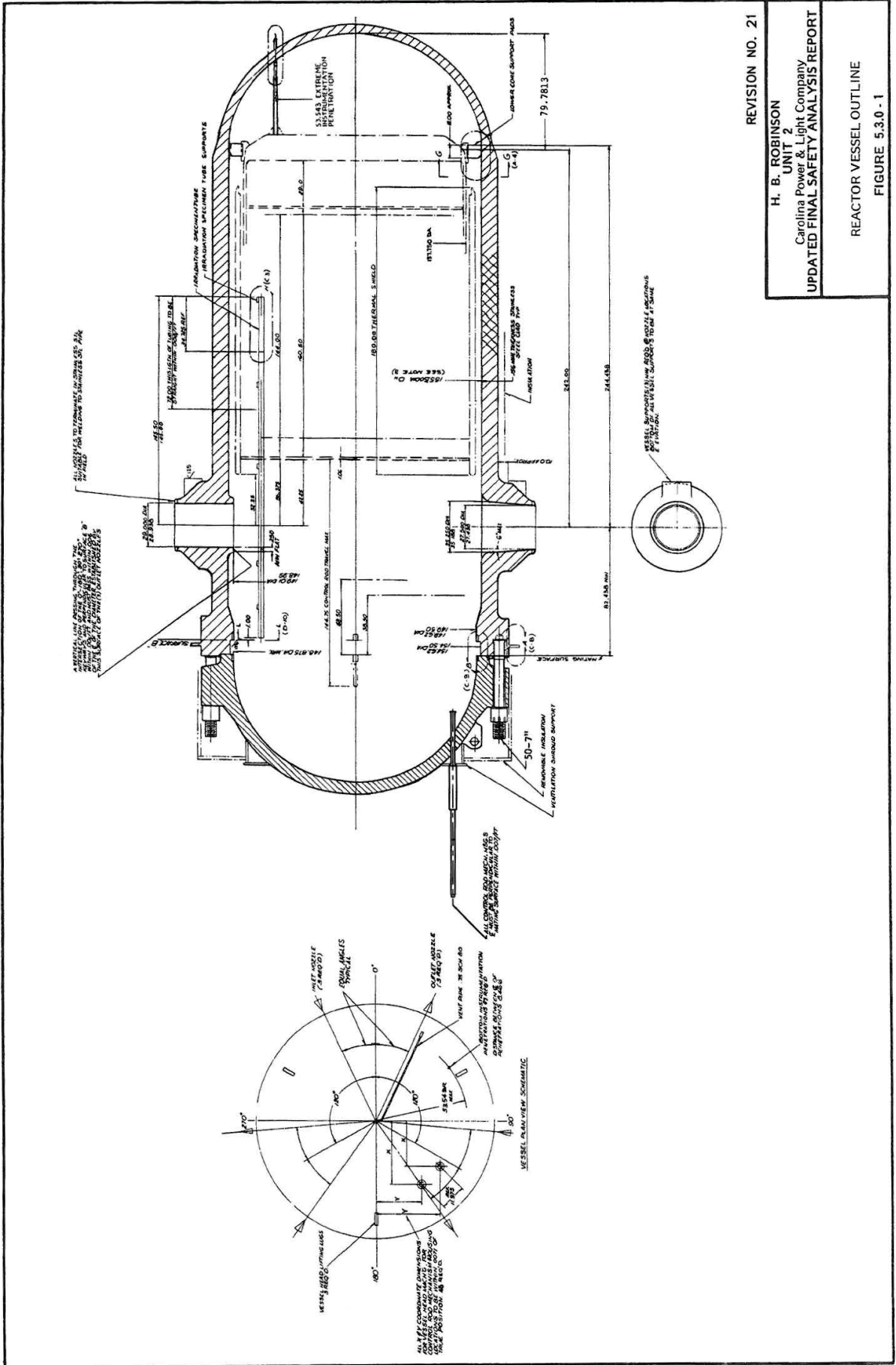
H. B. ROBINSON
UNIT 2
Carolina Power & Light Company
UPDATED FINAL SAFETY ANALYSIS REPORT

REACTOR VESSEL OUTLINE

FIGURE 5.3.0 - 1

CROSS SECTION

CORE

UPPER SHELL

5 3/8

W10201-2

180°

I-273B

EOL FLUENCE=
2.50 X10¹⁹ MAX.

EOL FLUENCE=
2.50 X10¹⁹ MAX.

10°

W10201-1

90°

270°

20°

40°

I-273A

I-273C

10-273*
CIRCUMFERENTIAL WELD
EOL FLUENCE=2.50X10¹⁹ MAX.

W10201-3

EOL FLUENCE=
2.50 X10¹⁹ MAX.

0°

INTERMEDIATE SHELL

180°

EOL FLUENCE=
6.01X10¹⁹ MAX.

*2-273B

W10201-4

50°

2-273C

90°

270°

EOL FLUENCE=
6.01 X10¹⁹ MAX.

20°

EOL FLUENCE=
6.01 X10¹⁹ MAX.

W10201-6

10°

W10201-5

2-273A
WELD EOL FLUENCE=
4.46 X10¹⁹ MAX.

0°

11-273*
CIRCUMFERENTIAL WELD
EOL FLUENCE=2.05X10¹⁹ MAX.

LOWER SHELL

21 3/4

180°

EOL FLUENCE=
2.05X10¹⁹ MAX.

*3-273B

W9807-5

3-273C*

90°

30°

30°

270°

W9807-3

W9807-9

EOL FLUENCE=
2.05X10¹⁹ MAX.

EOL FLUENCE=
2.05X10¹⁹ MAX.

0°

3-273A*
LONG. WELD
EOL FLUENCE=
2.05X10¹⁹ MAX.

*SEE TABLE 5.3.1-1
EOL = 50 EFPY

Revision No. 20

| H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT | IDENTIFICATION, LOCATION, AND END OF LICENSE FLUENCE OF BELTLINE REGION MATERIAL FOR THE REACTOR VESSEL | FIGURE 5.3.1-1 |

$r_1 = 170.021 \text{ cm}$

$r_2 = 175.182 \text{ cm}$

$r_3 = 181.134 \text{ cm}$

$r_4 = 187.959 \text{ cm}$

$r_5 = 197.485 \text{ cm}$

$r_6 = 217.488 \text{ cm}$

$r_7 = 237.808 \text{ cm}$

1 Primary Shield
2 Air Gap
3 Pressure Vessel
4 Water
5 Thermal Shield
6 Core barrel
7 Core baffle
8 Core
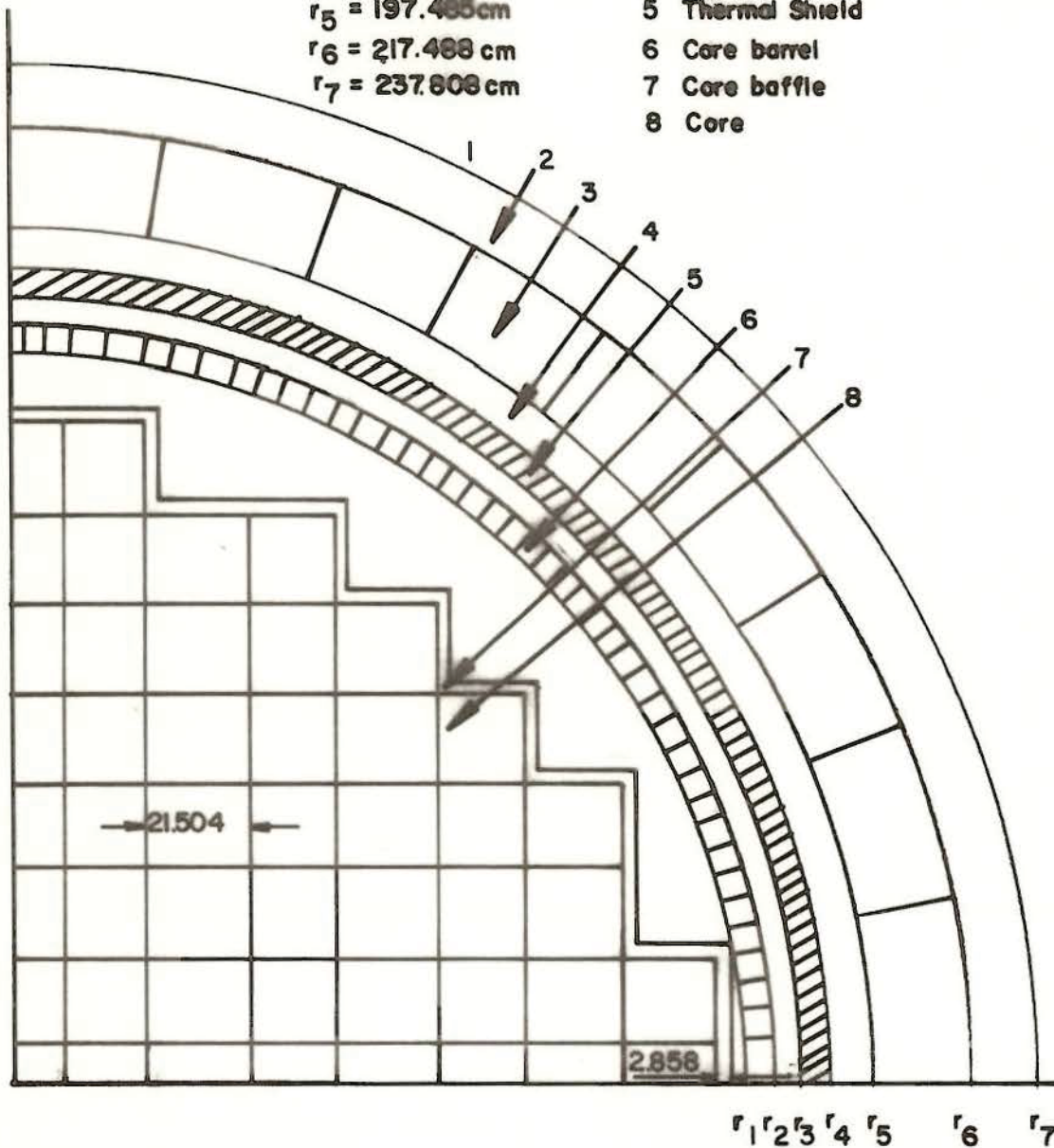
21.504

2.858

r_1 r_2 r_3 r_4 r_5 r_6 r_7

(r, θ) Reactor Geometry

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	PLAN VIEW OF REACTOR AT CORE MIDPLANE	FIGURE 5.3.1 - 2
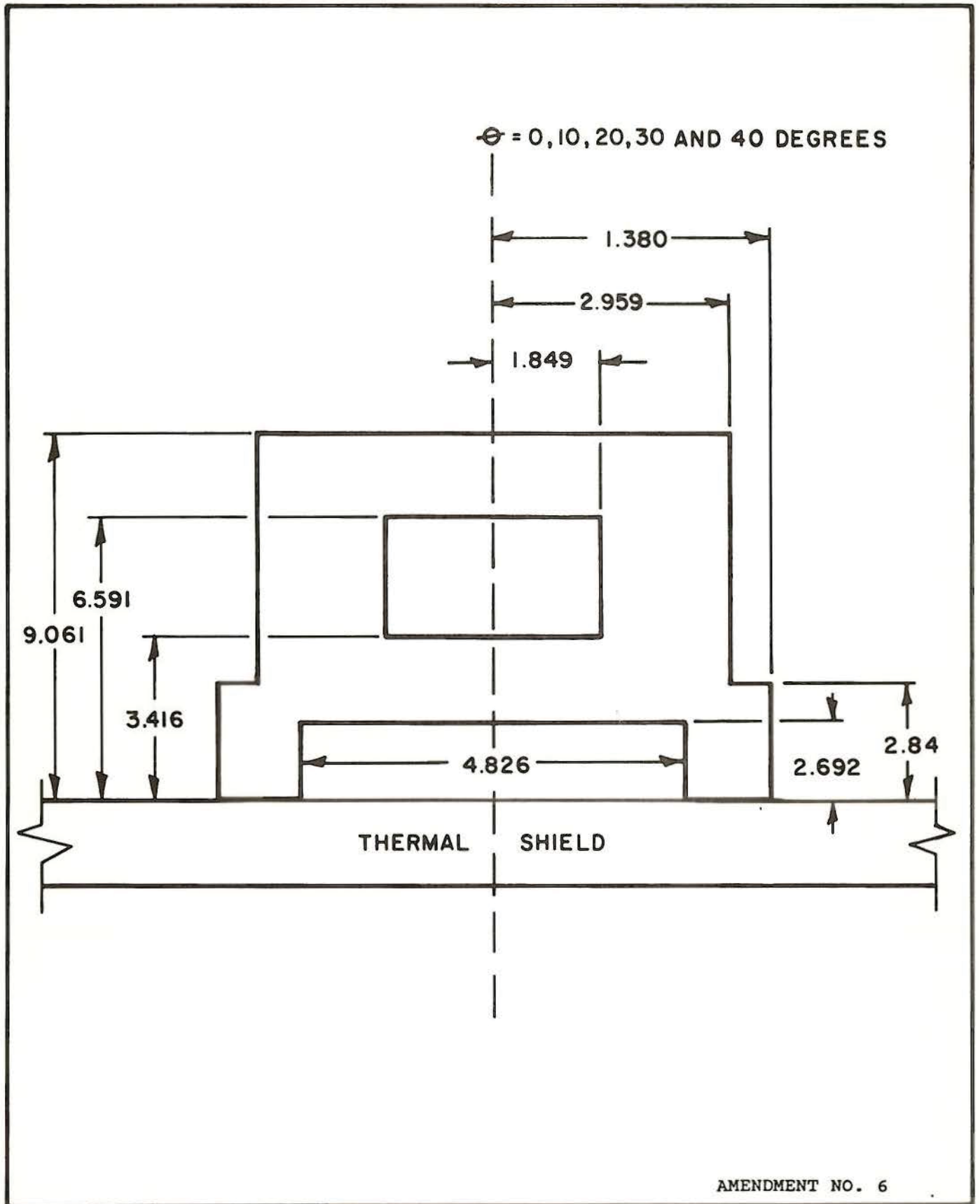
Θ = 0, 10, 20, 30 AND 40 DEGREES

1.380

2.959

1.849

6.591

9.061

3.416

4.826

THERMAL SHIELD

2.692

2.84

H. B. ROBINSON UNIT 2		FIGURE
Carolina Power & Light Company	SURVEILLANCE CAPSULE HOLDER DIMENSIONS (cm)	5.3.1 - 3
UPDATED FINAL SAFETY ANALYSIS REPORT		
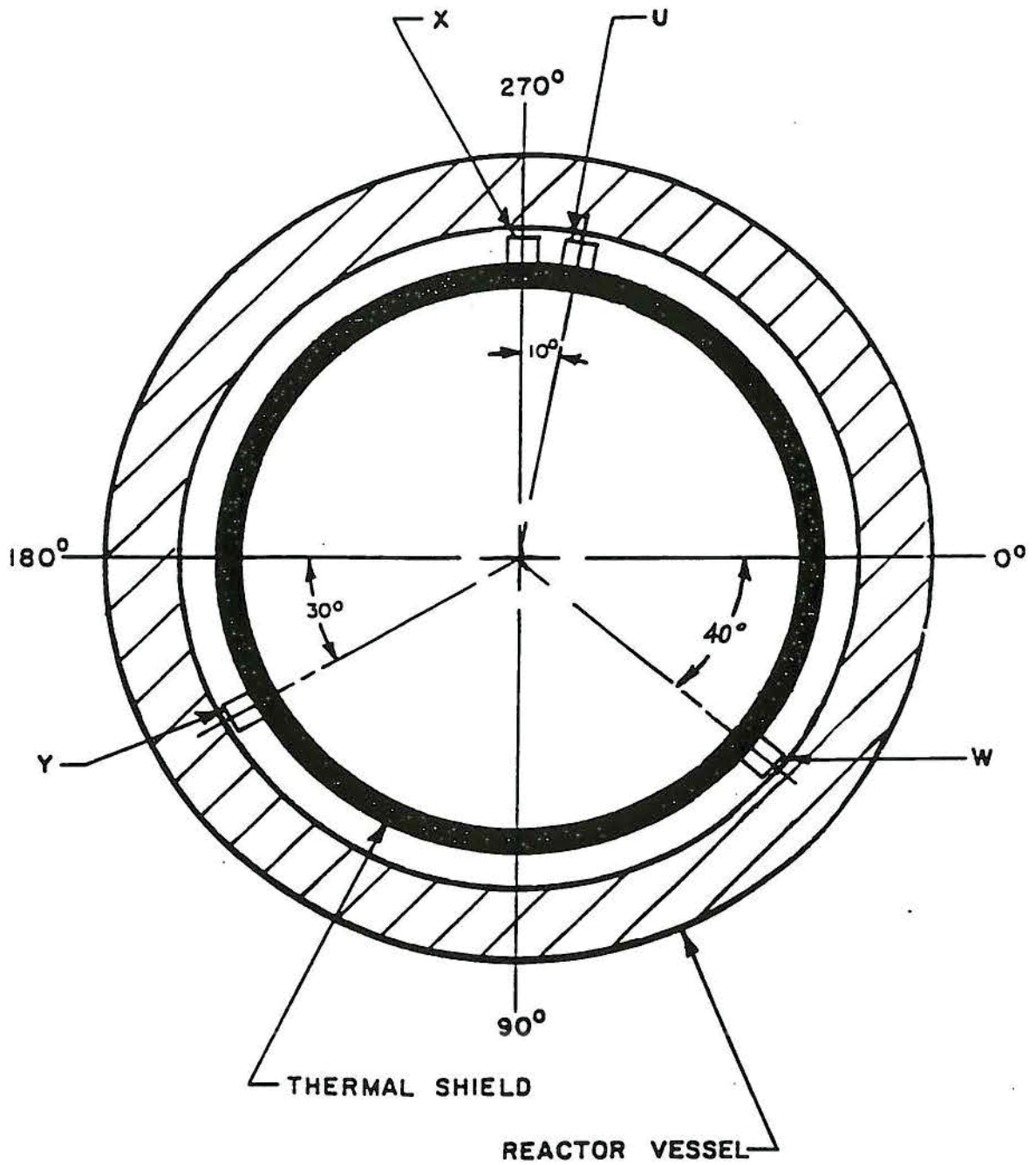
X
U
270°
10°
180°
30°
40°
0°
Y
W
90°
THERMAL SHIELD
REACTOR VESSEL

| H.B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT | ARRANGEMENT OF SURVEILLANCE CAPSULES | FIGURE 5.3.1-4 |

FIGURE 5.3.1-5

Deleted in Revision No. 20

DELETED REVISION NO. 15
FIGURE 5.3.1-6

DELETED REVISION NO. 15
FIGURE 5.3.1-7

DELETED REVISION NO. 15
FIGURE 5.3.1-8

DELETED REVISION NO. 15
FIGURE 5.3.1-9

DELETED REVISION NO. 15
FIGURE 5.3.1-10

DELETED REVISION NO. 15
FIGURE 5.3.1-11

5.4   COMPONENT AND SUBSYSTEM DESIGN

5.4.1   REACTOR COOLANT PUMPS

5.4.1.1   Design Description

Each reactor coolant loop contains a vertical single stage centrifugal pump which employs a controlled leakage seal assembly. A view of a controlled leakage pump is shown in Figure 5.4.1-1 and the principal design parameters for the pumps are listed in Table 5.4.1-1. The reactor coolant pump (RCP) estimated performance and net positive suction head characteristics are shown in Figure 5.4.1-2. The performance characteristic is common to all of the higher specific speed centrifugal pumps and the 'knee' at about 45 percent design flow introduces no operational restriction, since the pumps operate at full speed.

Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the impeller, discharged through passages in the diffuser and out through a discharge nozzle in the side of the casing. The motor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the pumps in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

The pump employs a controlled leakage film-riding face seal (i.e., number 1 seal) and two rubbing-face seals (i.e., numbers 2 and 3). The number 1 seal restricts leakage along the pump shaft. The number 2 seal helps direct controlled leakage out of the pump, and the number 3 seal minimizes the leakage of water and vapor from the pump into the containment atmosphere. A fourth sealing device called a shutdown seal is housed within the No. 1 seal area and is passively actuated by high temperature if seal cooling is lost.

A portion of the high pressure water flow from the charging pumps is injected into the RCP between the impeller and the controlled leakage seal. Part of the flow enters the Reactor Coolant System through a labyrinth seal in the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump. A very small amount which leaks through the secondary seal is also collected and removed from the pump.

Component cooling water is supplied to the motor bearing cooler and the thermal barrier cooling coil.

The squirrel cage induction motor driving the pump is air cooled and has oil lubricated thrust and radial bearings. A water lubricated bearing provides radial support for the pump shaft.

A flywheel and an anti-reverse rotation device are located at the top of the RCP motor. The flywheel, whose design and testing is discussed in Section 3.5.1, provides additional inertia to increase the RCP coastdown time, thereby reducing the consequences of loss-of-coolant accident. The anti-reverse rotation device prevents backflow, which may occur during LOCA, from turning the RCP in the reverse direction.

5.4.1.2 Pump Support Structure

Each RCP is supported on a three-legged structural system consisting of three connected columns fabricated of carbon steel members, structural sections and pipe. Provisions for limited movement of the structure in any horizontal direction to accommodate piping expansion are accomplished with a sliding "Lubrite" base plate arrangement and a system of tie rods and anchor bolts which restrain the structure from movement beyond the calculated limits. Sliding slot at the top of the support structures permits radial thermal growth of the pumps during heatup.

TABLE 5.4.1-1

REACTOR COOLANT PUMPS DESIGN DATA

Number of Pumps	3
Design Pressure/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3110
Design Temperature (casing), °F	650
RPM at Nameplate Rating	1180
Suction Temperature, °F	546.1
Net Positive Suction Head, ft	170
Developed Head, ft	261
Capacity, gpm	88,500
Seal Water Injection, gpm	8
Seal Water Return, gpm	3
Pump Discharge Nozzle, ID, in.	27-1/2
Pump Suction Nozzle, ID, in.	31
Overall Unit Height, ft	28.242
Water Volume, ft^3	192
Pump-Motor Moment of Inertia, lb-ft^2	70,000
Motor Data:	
Type	AC Induction Single Speed, Air Cooled
Voltage	4000
Insulation Class	Class B or Better
Phase	3
Frequency, cps	60

TABLE 5.4.1-1 (Cont'd)

Starting

Current, Maximum, amp	4900
Input (hot reactor coolant), kW	4000
Input (cold reactor coolant), kW	5300
Power, HP (nameplate)	6000

5.4.2   STEAM GENERATORS

5.4.2.1   Design Description

Each loop contains a vertical shell and U-tube steam generator. A steam generator of this type is shown in Figure 5.4.2-1. Principal design parameters are listed in Table 5.4.2-1.

Reactor coolant enters the inlet side of the channel head at the bottom of the steam generator through the inlet nozzle, flows through the U-tubes to an outlet channel and leaves the generator through another bottom nozzle. The inlet and outlet channels are separated by a partition. Manways are provided to permit access to the U-tubes and moisture separating equipment.

Feedwater to the steam generator enters just above the top of the U-tubes through a feedwater ring. The water flows downward through an annulus between the tube wrapper and the shell and then upward through the tube bundle where part of it is converted to steam.

The steam-water mixture from the tube bundle passes through a steam swirl vane assembly which imparts a centrifugal motion to the mixture and separates the water particles from the steam. The water spills over the edge of the swirl vane housing and combines with the feedwater for another pass through the tube bundle.

The steam rises through additional separators which limit the moisture content of the steam to one fourth of one percent or less under all design load conditions.

The steam generator is constructed primarily of carbon steel. The heat transfer tubes are Inconel. The interior surfaces of the channel heads and nozzles are clad with austenitic stainless steel, and the side of the tube sheet in contact with the reactor coolant is clad with Inconel.

5.4.2.2   Support Structure

The steam generators are supported on a structural system consisting of four connected columns all welded together, fabricated of carbon steel members, with provisions for limited movement of the structure in a horizontal direction to accommodate piping expansion with a system of "Lubrite" plates, hydraulic snubbers, guides, and stops. The "Lubrite" plates, hydraulic snubbers, guides, and stops are designed as damped support to resist the action of seismic and pipe break loads. Sliding shoes at the top of the support structure permit radial thermal growth of the steam generators during heatup.

5.4.2.3   Inservice Inspection

The requirements for Inservice Testing of Class 1, 2, and 3 components are described in Section 3.9.6. The requirements for Inservice Inspection of Class 1 components are described in Section 5.2.4. The requirements for Inservice Inspection of Class 2 and 3 components are described in Section 6.6. The inspection requirements for steam generator tubing are addressed in the Plant Technical Specifications.

TABLE 5.4.2-1

<u>STEAM GENERATOR DESIGN DATA (2339 MWt)</u>[*]

Number of Steam Generators	3
Design Pressure, Reactor Coolant/Steam, psig	2485/1085
Reactor Coolant Hydrostatic Test pressure (tube side-cold), psig (original)	3106
Design Temperature, Reactor Coolant/Steam, ºF	650/556
Reactor Coolant Flow, lb/hr	33.8×10^6
Total Heat Transfer Surface Area per S/G, ft^2	43,467
Steam Conditions at Full Load, Outlet Nozzle:	
Steam Flow, lb/hr	3.43×10^6
Steam Temperature, ºF	521.4
Steam Pressure, psia	821
Feedwater Temperature, ºF	440
Overall Height, ft-in.	63-1.6
Shell OD, upper/lower, in.	166/127
Shell Thickness, upper/lower, in.	3.5/2.62
Number of U-tubes	3214
U-tube Diameter, in. (OD)	0.875
Tube Wall Thickness, (average), in.	0.050
Number of Manways/ID, in.	3/16
Number of handholes/ID, in.	6/6

[*] Steam Generator Lower Assembly Replacement 1984
  All parameters are for nominal operation conditions at 100% load.

TABLE 5.4.2-1 (Cont'd)

Primary Side Pressure, psia	2,250
Primary Side Pressure Drop (Nominal), psi	28.7
Secondary Side Pressure Drop (Nominal), psi psi (Feedwater Inlet to Steam Outlet)	18.9
Heat Transferred, BTU/hr	2.660×10^9
Primary Volume, ft^3	925
Secondary Volume, ft^3	4,755

5.4.3   REACTOR COOLANT PIPING

The general arrangement of the Reactor Coolant System piping is shown on the plant layout drawings in Section 5.1. Piping design data are presented in Table 5.4.3-1.

The reactor coolant piping layout is designed on the basis of providing "Floating" supports for the steam generator and RCP in order to absorb the thermal expansion from the fixed or anchored reactor vessel.

The austenitic stainless steel reactor coolant piping and fittings which make up the loops are 29 in. ID in the hot legs, 27-1/2 in. ID in the cold legs and 31 in. ID between each loop's steam generator outlet and its RCP suction. The pressurizer relief line, which connects the pressurizer safety and relief valves' outlets to the inlet nozzle flange on the pressurizer relief tank, is constructed of carbon steel.

Smaller piping, including the pressurizer surge and spray lines, drains, and connections to other systems are austenitic stainless steel. All joints and connections are welded except for stainless steel flange connections to the carbon steel pressurizer relief tank and the connections at the relief and safety valves.

Thermal sleeves are installed at the following locations where high thermal stresses could otherwise develop due to rapid changes in fluid temperature during normal operational transients:

a)   Return line from the residual heat removal loop

b)   Both ends of the pressurizer surge line

c)   Pressurizer spray line connection to the pressurizer

d)   Charging lines and auxiliary charging line connections

TABLE 5.4.3-1

REACTOR COOLANT PIPING DESIGN DATA

Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure, (cold) psig	3110
Design Temperature, ºF	650
Design Temperature, (pressurizer surge line), ºF	680
Reactor Inlet Piping, ID, in.	27-1/2
Reactor Inlet Piping, nominal thickness, in.	2.375
Reactor Outlet Piping, ID, in.	29
Reactor Outlet Piping, nomimal thickness, in.	2.50
Coolant Pump Suction Piping, ID, in.	31
Coolant Pump Suction Piping, nominal thickness, in.	2.625
Pressurizer Surge Line Piping, ID, in./Pipe Schedule	12/Sch 140
Pressurizer Surge Line Piping, nominal thickness, in.	1.125
Water Volume, (all 3 loops) ft^3	783

* Surge line fitted with a 14 in./12 in. adapter at the pressurizer

5.4.4   RESIDUAL HEAT REMOVAL SYSTEM

The Residual Heat Removal (RHR) system is one of three loops of the Auxiliary Coolant System. The other two loops are the Component Coolant system (Section 9.2.2) and the Spent Fuel Pool Coolant System (Section 9.1.3).

The residual heat removal loop removes residual and sensible heat from the core and reduces the temperature of the Reactor Coolant System during the second phase of plant cooldown.

5.4.4.1   Design Basis

The residual heat removal loop is designed to remove residual and sensible heat from the core and reduce the temperature of the Reactor Coolant System during the second phase of plant cooldown. During the first phase of cooldown, the temperature of the Reactor Coolant System is reduced by transferring heat from the Reactor Coolant System to the Steam and Power Conversion System.

All piping and components of the RHR are designed to the applicable codes and standards listed in Section 3.2. Austenitic stainless steel piping is used in the residual heat removal loop, which contains reactor coolant.

All active loop components which are relied upon to perform their function are redundant.

The loop design precludes any significant reduction in the overall design reactor shutdown margin when the loop is brought into operation for residual heat removal or for emergency core cooling by recirculation.

The loop design includes provisions to enable hydrostatic testing to applicable code test pressures during shutdown.

Loop components, whose design pressure and temperature are less than the Reactor Coolant System design limits, are provided with overpressure protective devices and redundant isolation means.

5.4.4.2   System Design

Figure 5.4.4-1 shows the residual heat removal system.

5.4.4.2.1   Residual Heat Exchangers

The two residual heat exchangers, located within the auxiliary building, are of the shell and U-tube type with the tubes welded to the tube sheet. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell side. The tubes and other surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

5.4.4.2.2   Residual Heat Removal Pumps

The two residual heat removal pumps are in-line, centrifugal units with special seals to minimize reactor coolant leakage to the atmosphere. All pump parts in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

5.4.4.2.3   Residual Heat Removal Valves

The valves used in the residual heat removal loop are constructed of austenitic stainless steel or equivalent corrosion resistant material.

Manual motor operated stop valves are provided to isolate equipment for maintenance. Throttle valves are provided for remote and manual control of the residual heat exchanger tube side flow. Check valves prevent reverse flow through the residual heat removal pumps.

Double, remotely operated series stop valves are provided at the inlet to isolate the residual heat removal loop from the Reactor Coolant System.

When Reactor Coolant System pressure exceeds the design pressure of the residual heat removal loop, an interlock between the Reactor Coolant System narrow range pressure channel and the inlet isolation valves prevents the valves from opening. A remotely operated stop valve and two series check valves isolate each line to the Reactor Coolant System cold legs from the residual heat removal loop. Overpressure in the residual heat removal loop is prevented by a relief valve which discharges to the pressurizer relief tank.

Valves that perform a modulating function are equipped with two sets of packing and standard bolting, or are configured with a live load packing arrangement. The valves include an intermediate leak off connection or have had their leak off line removed and capped. The leak off pipes which are not capped discharge to the Auxiliary Building Atmosphere and enter the waste disposal system via the floor drain.

Manually operated valves have backseats to facilitate repacking and can be used to limit the stem leakage when the valves are open.

5.4.4.2.4   Residual Heat Removal Piping

All residual heat removal loop piping is austenitic stainless steel. The piping is welded except for flanged connections at the control valves.

5.4.4.3   System Performance

Two pumps and two residual heat exchangers perform the decay heat cooling functions for the reactor unit. After the Reactor Coolant System temperature and pressure have been reduced to less than 350ºF and 375 psig respectively, decay heat cooling is initiated by aligning the pumps to take suction from the reactor outlet line and discharge through the heat exchangers into the reactor inlet line. If only one pump and one heat exchanger are available reduction of reactor coolant temperature is accomplished at a lower rate.

An additional function of the residual heat removal pumps is to assist in the mitigation of the LOCA in the refill of the vessel and to ultimately return the core to a subcooled state. A complete discussion of this function is provided in Section 6.3.

5.4.5   MAIN STEAM LINE AND FEEDWATER PIPING

Refer to Chapter 10.0.

5.4.6   PRESSURIZER

The general arrangement of the pressurizer is shown in Figure 5.4.6-1 and the design data are listed in Table 5.4.6-1.

The pressurizer maintains the required reactor coolant pressure during steady-state operation, limits the pressure changes caused by coolant thermal expansion and contraction during normal load transients, and prevents the pressure in the Reactor Coolant System from exceeding the design pressure.

The pressurizer contains replaceable direct immersion heaters, multiple safety and relief valves, a spray nozzle and interconnecting piping, valves and instrumentation. The electric heaters located in the lower section of the vessel maintain the pressure of the Reactor Coolant System by keeping the water and steam in the pressurizer at saturation temperature corresponding to the system pressure. Three pressurizer heater banks (1 control and 2 backup) with a total design capacity of 1300 kW are installed. A minimum total capacity of 800 kW is required for normal operating conditions. A minimum of 125 kW of heater capacity is capable of being powered from emergency power supplies. This capacity is sufficient to maintain the Reactor Coolant System near normal operating pressure and to aid natural circulation. This is automatically tripped off from the emergency bus in the event of a safety injection signal to prevent overloading of the diesel generators.

The pressurizer is designed to accommodate positive and negative surges caused by load transients. The surge line which is attached to the bottom of the pressurizer connects it to the hot leg of a reactor coolant loop. During a positive surge, caused by a decrease in plant load, the spray system, which is fed from the cold leg of a coolant loop, condenses steam in the pressurizer to prevent the pressurizer pressure from reaching the set point of the power operated relief valves. Power operated spray valves on the pressurizer limit the pressure during load transients. In addition, the spray valves can be operated manually by a switch in the control room. A small continuous spray flow is provided to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excess cooling of the spray and surge line piping.

During a negative pressure surge, caused by an increase in plant load, flashing of water to steam and generation of steam by automatic actuation of the heaters keeps the pressure above the minimum allowable limit. Heaters are also energized on high water level during positive surges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop.

The pressurizer is constructed of carbon steel with internal surfaces clad with austenitic stainless steel. The heaters are sheathed in austenitic stainless steel.

The pressurizer vessel surge nozzle is protected from thermal shock by a thermal sleeve. A thermal sleeve also protects the pressurizer spray nozzle connection.

TABLE 5.4.6-1

PRESSURIZER AND PRESSURIZER RELIEF TANK DESIGN DATA

Pressurizer

Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3110
Design/Operating Temperature, ºF	680/653
Water Volume, Full Power, ft^3*	780
Steam Volume, Full Power, ft^3	520
Surge Line Nozzle Diameter, in./Pipe Schedule	14/Sch 140
Shell ID, in./Minimum Shell Thickness, in.	84/4.1
Minimum Clad Thickness, in.	0.188
Electric Heaters Capacity, kW (design)	1300
kW (minimum required operation)	800
Heatup Rate of Pressurizer Using Heaters Only, ºF/hr	55 (approximately)

Power Relief Valves

Number	2
Set Pressure, psig	2335/2340
Capacity, lb/hr Saturated Steam/Valve	210,000

Safety Valves

Number	3
Set Pressure, psig	2485
Capacity, lb/hr Saturated Steam Valve	293,330

* 60 percent of net internal volume (maximum calculated power)

TABLE 5.4.6-1 (Cont'd)

<u>Pressurizer Relief Tank</u>

Design Pressure, psig	100
Rupture Disc Release Pressure, psig	100
Design Temperature, ºF	340
Normal Water Temperature, ºF	120
Total Volume, ft^3	1300
Rupture Discs Relief Capacity,	
Saturated Steam (combined) lb/hr	900,000

5.4.7 PRESSURIZER RELIEF DISCHARGE SYSTEM

The power operated relief and code safety valves associated with this system are discussed in Section 5.2.2.   This section discusses the pressurizer relief tank (PRT) which constitutes the remainder of the system.

5.4.7.1 Design Basis

The principal design parameters of the PRT are given in Table 5.4.6-1.

5.4.7.2 System Description

Steam discharged from the power relief and safety valves passes to the PRT which is partially filled with water at or near ambient containment conditions. The PRT normally contains water in a predominantly nitrogen atmosphere. Steam is discharged under the water level to condense and cool by mixing with the water. To reduce the likelihood of PRT overpressurization following a discharge, the PRT is equipped with a spray to add cooling water and a drain to the Waste Disposal System to remove excess heated water.

The PRT size is 1300 ft^3 with a design temperature and pressure of 340ºF and 100 psig respectively.

The PRT is piped to the pressurizer safety and power operated relief valves by a 12 in. line. It is protected from overpressurization by 2 rupture discs that relieve pressure to the containment vessel at approximately 100 psig.

The rupture discs are designed to pass 900,000 lb/hr saturated steam.

The PRT also collects leakage and liquid from various system relief valves located inside the containment.

The PRT is normally filled to about 70 percent with primary water and also has approximately 3 psig nitrogen atmosphere in it. A nitrogen regulator outside containment maintains this pressure in the tank along with the ability to vent the PRT to the vent header. Primary water may be added to the tank by use of the primary water pumps and valves. Water may be pumped from the tank by utilizing the "B" reactor coolant drain tank pump and valves or gravity drained to the containment sump via WD-1708.

5.4.7.3 Instrumentation Requirements

The PRT is equipped with level, pressure and temperature sensing devices which provide indication and alarms.

5.4.8   VALVES

All valve surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials. Connections to stainless steel piping are welded. The leak off lines for RHR-759A, 759B, 757A, 757B, 757C, & 757D are capped via a threaded joint.

Valves that perform a modulating function are equipped with two sets of packing and standard bolting, or are configured with a live load packing arrangement. The valves include an intermediate leak off connection or have had their leak off lines removed and capped

5.4.9   SAFETY AND RELIEF VALVES

Three spring loaded safety valves and two power operated relief valves are provided for overpressure protection.

5.4.9.1   Safety Valves

The safety valves, set for the system design pressure of 2485 psig are spring loaded and enclosed pop type with back pressure compensation. The combined capacity of the valves is equal to, or greater than, the maximum surge rate resulting from complete loss of load without reactor trip or any other control except that the secondary plant safety valves are assumed to operate when steam pressure reaches their setpoint. A water seal is maintained below each valve seat to inhibit leakage. A resistance temperature detector is installed in the discharge piping. It will provide indication and a high temperature alarm in the Control Room to warn the operator of an actuated or leaky safety valve.

Acoustic monitors are also installed on each of the three safety valves. These monitors provide valve position indication in the Control Room. Again, an alarm is provided to warn the operator of an actuated or leaky safety valve.

5.4.9.2   Power Operated Relief Valves

The two power operated relief valves (PORV) are set to open at 2335/2340 psig. Their opening is actuated on signals generated by the pressurizer pressure transmitters. A resistance temperature detector installed in the discharge line common to both valves, serves the same purpose as those for the safety valves. A separate alarm, in parallel with the position indication light circuit, is provided to annunciate whenever either PORV opens (valve limit switch closes). The PORV are provided to limit any pressure excursion and thus limit the operation of the spring loaded pressurizer safety valves. Motor operated stop valves, located upstream of the PORV, are provided in order to remove the PORV from service should they leak excessively. This is allowable since the safety valves are sized to protect the Reactor Coolant System without the aid of the PORV.

Based upon the Westinghouse Owners' Group report submitted in response to NUREG-0737, Item II.K.3.2 (Reference 5.4.9-1) Carolina Power & Light Company has determined that the automatic PORV block valve isolation system discussed in Item II.K.3.1 is not required for H. B. Robinson, Unit 2.

5.4.10 REACTOR COOLANT GAS VENT SYSTEM

5.4.10.1 Design Basis

All piping and components are designed to meet or exceed the criteria of   USAS B31.1.

All pipe support components were designed to ASME Section III, Subsection NF.

All piping and fittings are constructed of Type 304 austenitic stainless steel or equivalent.

All solenoid valves are constructed of Type 316L austenitic stainless steel.

All pipe support components are constructed of Type 304 austenitic stainless steel.

The system is designed for 2500 psia and 700ºF.

5.4.10.2 System Description

The Reactor Coolant Gas Vent System (RCGVS) is designed to be used to remotely vent non-condensable gases from the reactor vessel head and pressurizer steam space during post-accident situations when large quantities of non-condensable gases may collect in these high points. The purpose of venting is to prevent possible interference with core cooling. Small amounts of gas can be vented to the pressurizer relief tank (PRT) and thus not enter the containment atmosphere. Larger volumes will require venting directly to the containment. Pressure instrumentation is included in the design to monitor system leakage during normal plant operation.

The system also aids in RCS venting procedures following a maintenance outage. Although designed for accident conditions, the system may be used to aid in the pre- and post-refueling venting of the reactor coolant system. Both the pressurizer and reactor vessel may be vented by using this system.

The system is designed to permit the operator to vent the reactor vessel head or pressurizer steam space from the control room under post-accident conditions, and is operable following all design basis events except those requiring evacuation of the control room or a complete loss of all AC power. The vent path from either the pressurizer or reactor vessel head is single active failure proof with active components powered from emergency power sources. Parallel valves powered off alternate power sources are provided at both vent sources to assure a vent path exists in the event of a single failure of either a valve or the power source. The system provides a redundant vent path either to the containment directly or to the PRT.

5.4.10.3 Operation

The RCGVS is not used during normal power operations. All remotely operated solenoid valves are administratively controlled in the control room to assure the system is used only when necessary under accident conditions.
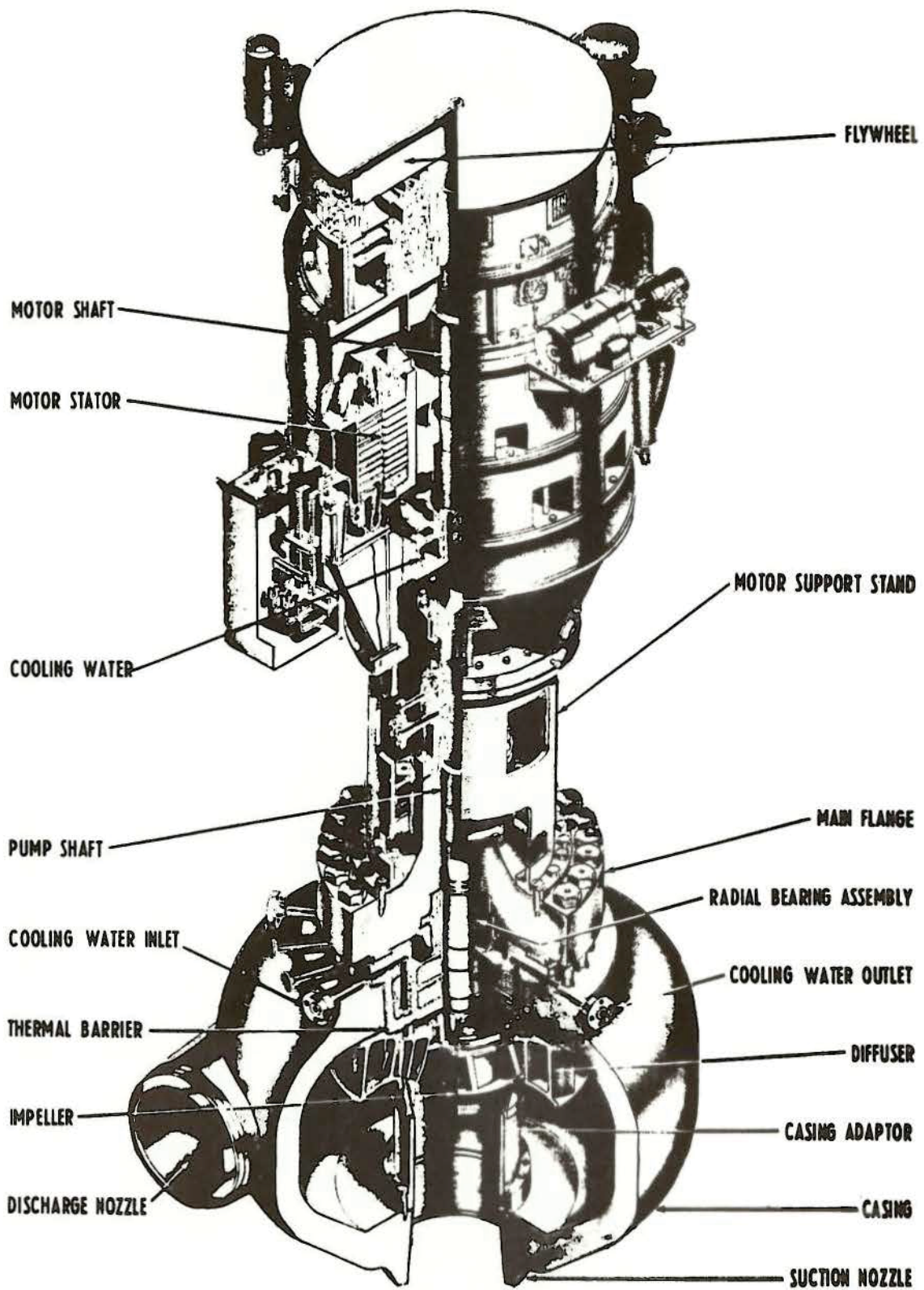
In the unlikely event that an accident results in the generation of significant quantities of non-condensable gases within the RCS, the RCGVS will be used to remove the gases, thereby mitigating the effect of the accident.

REFERENCES: SECTION 5.4

5.4.9-1   "Probabilistic Analysis and Operational Data in Response to NUREG-0737, Item II.K.3.2 for Westinghouse NSSS Plants," Westinghouse Electric Corporation (WCAP-9804), (February 1981).

FLYWHEEL

MOTOR SHAFT

MOTOR STATOR

COOLING WATER

MOTOR SUPPORT STAND

PUMP SHAFT

MAIN FLANGE

RADIAL BEARING ASSEMBLY

COOLING WATER INLET

COOLING WATER OUTLET

THERMAL BARRIER

DIFFUSER

IMPELLER

CASING ADAPTOR

DISCHARGE NOZZLE

CASING

SUCTION NOZZLE

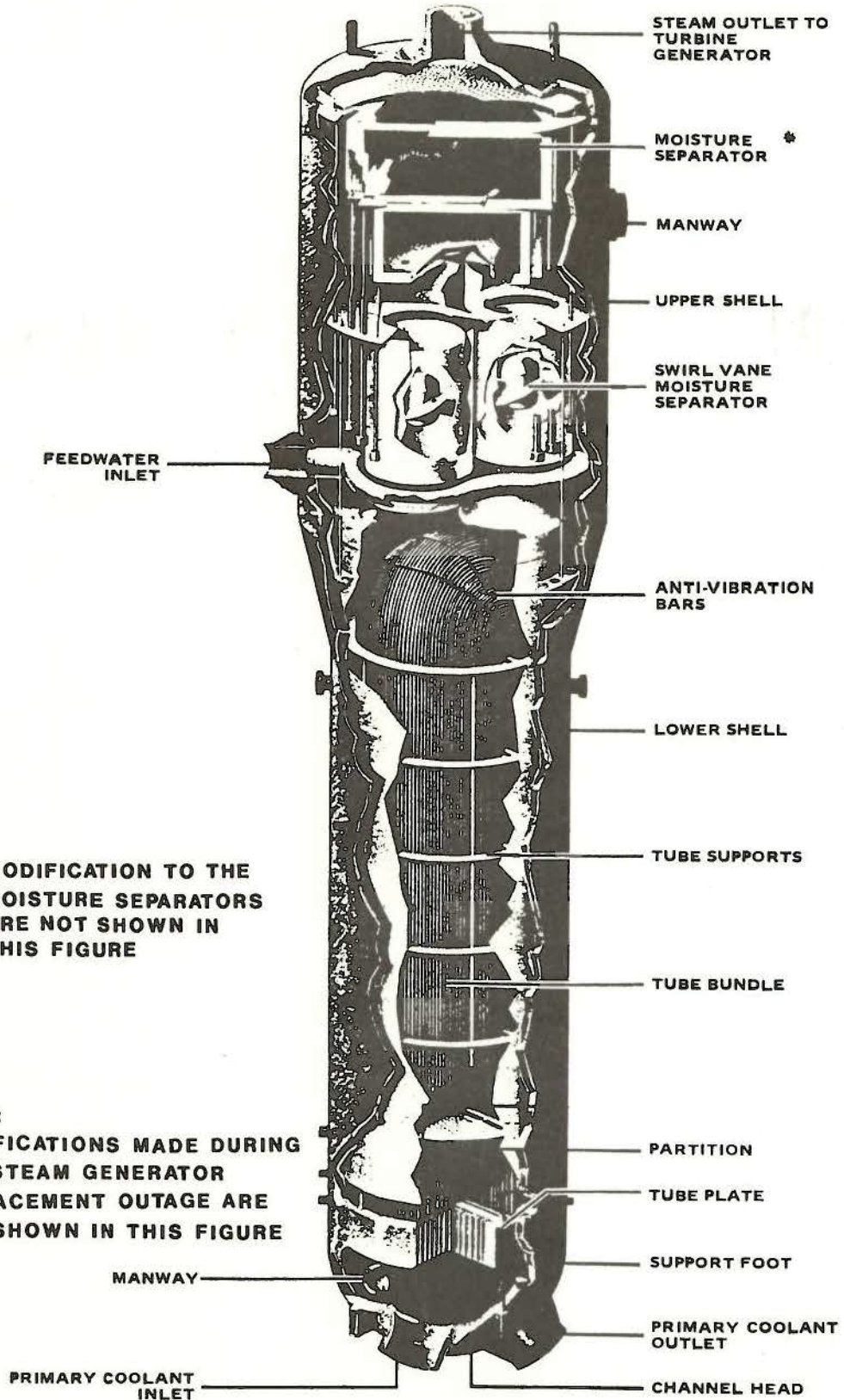| H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT | REACTOR COOLANT CONTROLLED LEAKAGE PUMP | FIGURE 5.4.1 - 1 |
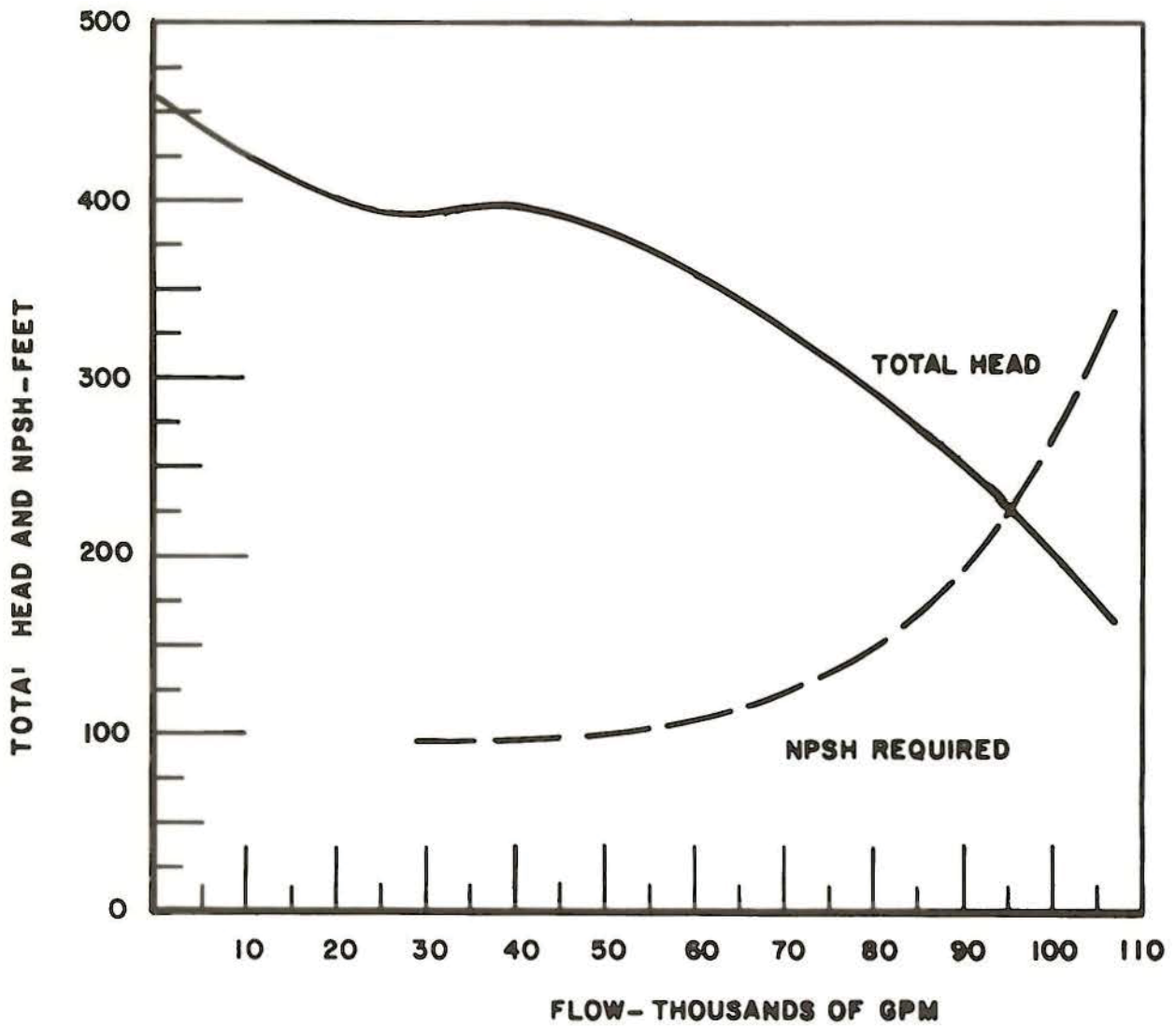
REACTOR COOLANT PUMP
ESTIMATED PERFORMANCE
CHARACTERISTICS

H. B. ROBINSON
UNIT 2
Carolina Power & Light Company
UPDATED FINAL
SAFETY ANALYSIS REPORT

FIGURE

5.4.1 - 2

STEAM OUTLET TO TURBINE GENERATOR

MOISTURE SEPARATOR ✾

MANWAY

UPPER SHELL

SWIRL VANE MOISTURE SEPARATOR

FEEDWATER INLET

ANTI-VIBRATION BARS

LOWER SHELL

✿ MODIFICATION TO THE MOISTURE SEPARATORS ARE NOT SHOWN IN THIS FIGURE

TUBE SUPPORTS

TUBE BUNDLE

NOTE:
MODIFICATIONS MADE DURING THE STEAM GENERATOR REPLACEMENT OUTAGE ARE NOT SHOWN IN THIS FIGURE

PARTITION

TUBE PLATE

SUPPORT FOOT

MANWAY

PRIMARY COOLANT OUTLET

PRIMARY COOLANT INLET

CHANNEL HEAD

AMENDMENT 3

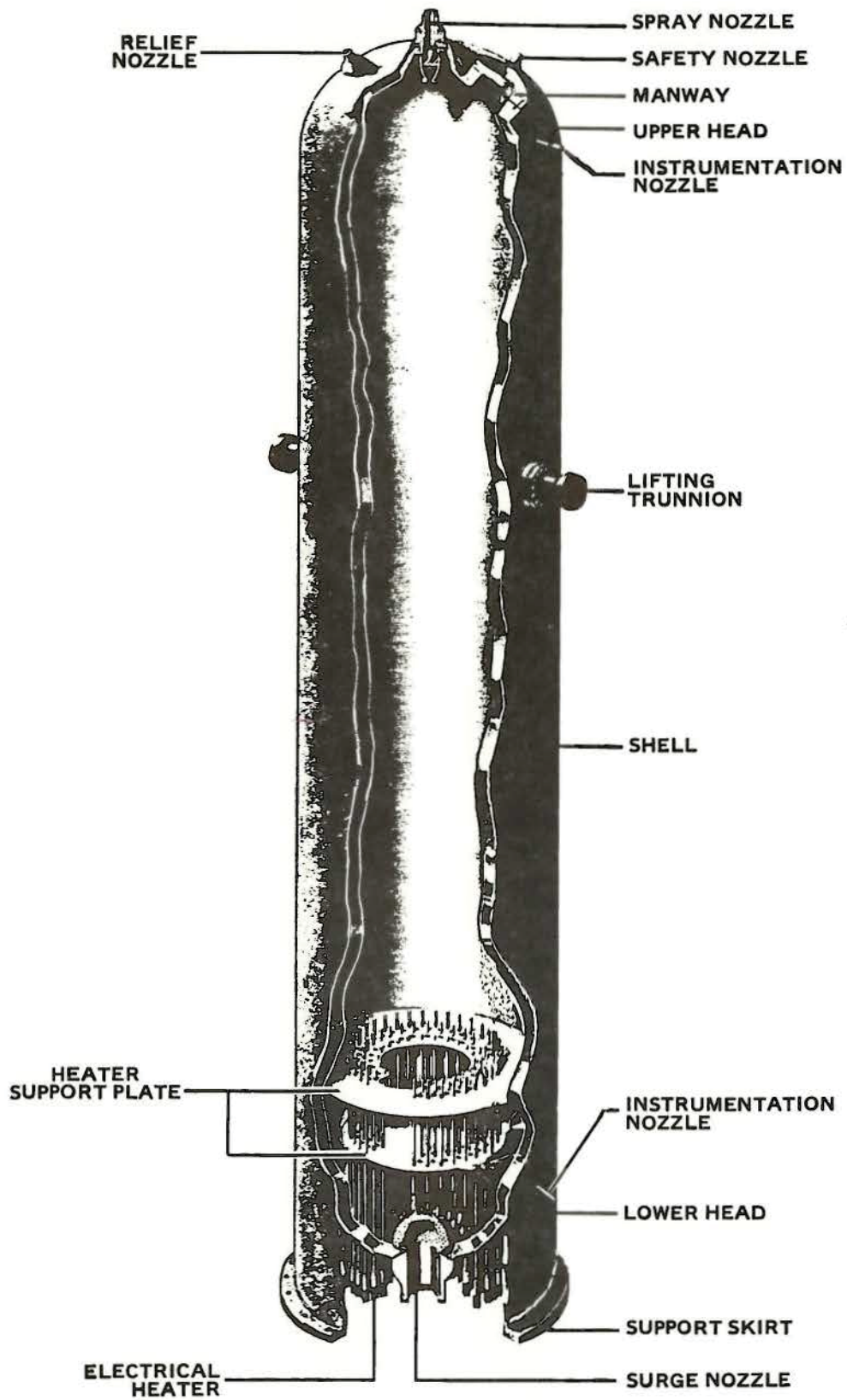| H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT | STEAM GENERATOR | FIGURE 5.4.2 - 1 |
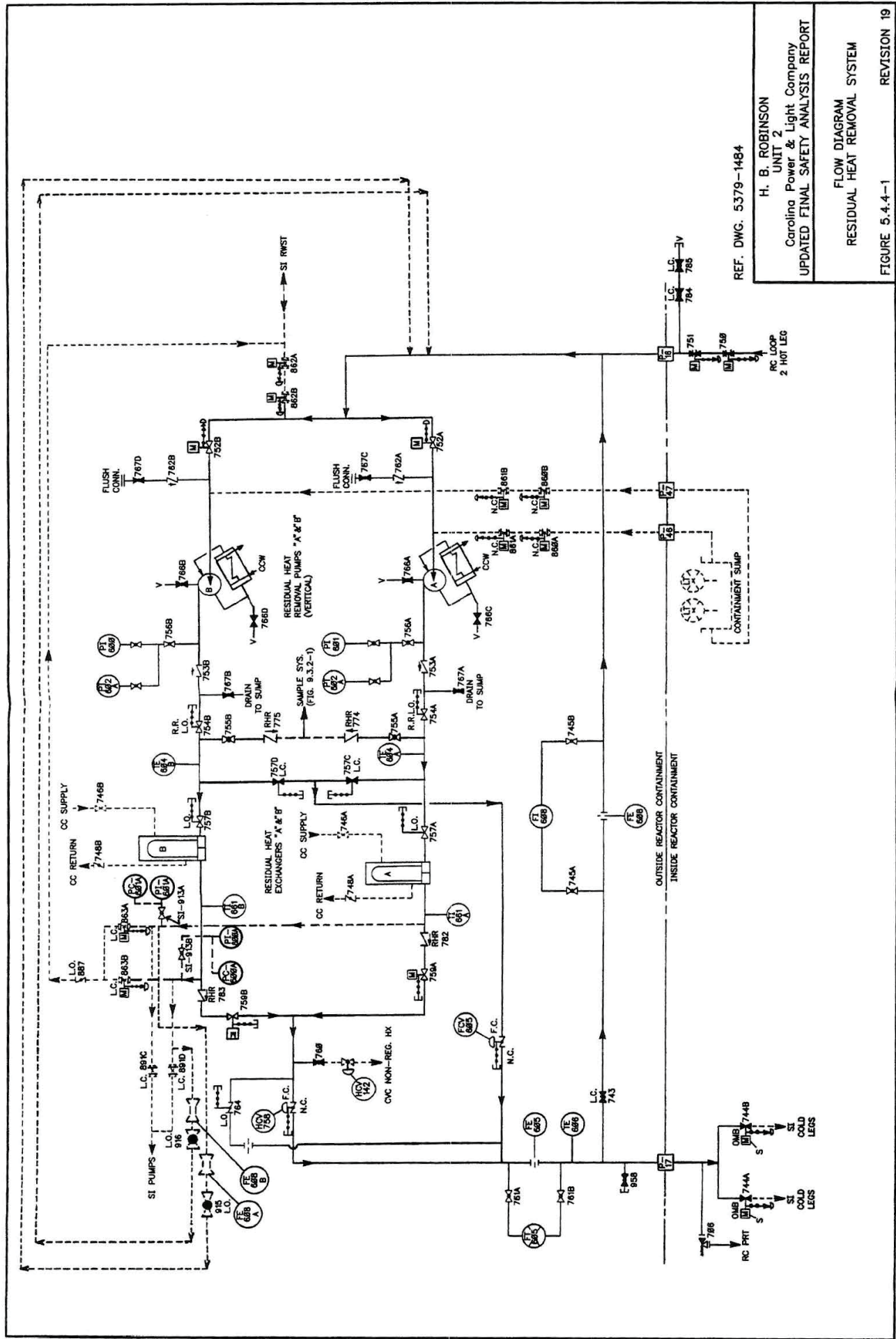
SI RWST

RC LOOP
2 HOT LEG

RESIDUAL HEAT
REMOVAL PUMPS "A" & "B"
(VERTICAL)

FLUSH CONN.

CONTAINMENT SUMP

DRAIN TO SUMP

SAMPLE SYS.
(FIG. 9.3.2-1)

DRAIN TO SUMP

R.R.L.O.

RESIDUAL HEAT
EXCHANGERS "A" & "B"

CC SUPPLY

CC RETURN

CC SUPPLY

CC RETURN

OUTSIDE REACTOR CONTAINMENT
INSIDE REACTOR CONTAINMENT

RHR

CVC NON-REG. HX

ST PUMPS

SI COLD LEGS

SI COLD LEGS

RC PRT

RELIEF NOZZLE

SPRAY NOZZLE
SAFETY NOZZLE
MANWAY
UPPER HEAD
INSTRUMENTATION NOZZLE

LIFTING TRUNNION

SHELL

HEATER SUPPORT PLATE

INSTRUMENTATION NOZZLE

LOWER HEAD

SUPPORT SKIRT

ELECTRICAL HEATER

SURGE NOZZLE

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SAFETY ANALYSIS REPORT | PRESSURIZER | FIGURE

5.4.6 - 1 |