TABLE 5.2-1

APPLICABLE CODE ADDENDA FOR REACTOR COOLANT SYSTEM COMPONENTS

Reactor Vessel	ASME III, 1971 Edition, through Winter 72
Steam Generator	ASME III, 1971 Edition, through Summer 73
Pressurizer	ASME III, 1971 Edition, through Summer 73
CRDM Housing Full Length	ASME III, 1974 Edition, through Summer 74
CRDM Head Adapter	ASME III, 1971 Edition, through Winter 72
Reactor Coolant Pump	ASME III, 1971 Edition, through Summer 73
Reactor Coolant Pipe	ASME III, 1974 Edition, through Summer 75
Class 1 Interconnecting Piping to the RCS*	ASME III, 1971 Edition, through Winter 72
Surge Line	ASME III, 1974 Edition, through Summer 75
Valves	
Pressurizer Safety	ASME III, 1971 Edition, through Winter 72
Motor Operated	ASME III, 1974 Edition, through Summer 74
Manual (3" and larger)	ASME III, 1974 Edition, through Summer 74
(2" and smaller)	ASME III, 1974 Edition, through Summer 75
Control	ASME III, 1974 Edition, through Summer 75
Pressurizer Spray	ASME III, 1971 Edition, through Summer 72

* Piping provided by A/E.

TABLE 5.2-2
(Sheet 1 of 3)

CLASS 1 PRIMARY COMPONENTS MATERIAL SPECIFICATIONS

Reactor Vessel Components	
Shell and head plates (other than core region)	SA-533, Gr. B, Class 1 (vacuum treated)
Shell plates (core region)	SA-533, Gr. B, Class 1 (vacuum treated)
Shell, flange and nozzle forgings nozzle safe ends	SA-508, Class 2; SA-182, Type F316
CRDM and/or ECCS appurtenances, upper head	SB-167; SA-182, Type F304
Instrumentation tube appurtenances, lower head	SB-166
Closure studs, nuts, washers	SA-540, Class 3, Gr. B24 threaded portions of the closure studs have been treated with an anti galling coating
Core support pads	SB-166 with carbon less than 0.10%
Monitor tubes	SA-312, Type 316; SB-166
Vent pipe	SB-166; SB-167
Vessel supports, seal ledge and head lifting lugs	SA-516, Gr. 70 quenched and tempered; SA-533, Gr. B, Class 1 (vessel supports may be of weld metal build up of strength equivalent to nozzle material)
Cladding and buttering	Stainless steel weld metal analysis A-7* and Ni-Cr-Fe weld metal F-Number 43
Steam Generator Components	
Pressure plates	SA-533, Gr. A, B or C, Class 1 or 2
Pressure forgings (including nozzles and tubesheet)	SA-508, Class 2, 2a or 3

* Designated A-8 in the 1974 edition of the ASME Code

TABLE 5.2-2
(Sheet 2 of 3)

CLASS 1 PRIMARY COMPONENTS MATERIAL SPECIFICATIONS

Nozzle safe ends	Stainless steel weld metal analysis A-7*
Channel heads	SA-533, Gr. A, B or C, Class 1 or 2 or SA-216, Gr. WCC
Tubes	SB-163, Ni-Cr-Fe annealed
Cladding and buttering	Stainless steel weld metal analysis A-7* and Ni-Cr-Fe weld metal F-Number 43
Closure bolting	SA-193, Gr. B7 threaded portions may be treated with an anti galling coating.
Pressurizer Components	
Pressure plates	SA-533, Gr. A, Class 2
Pressure forgings	SA-508, Class 2a
Nozzle safe ends	SA-182, Type 316L and Ni-Cr-Fe weld metal F-Number 43
Cladding and buttering	Stainless steel weld metal analysis A-8 and Ni-Cr-Fe weld metal F-Number 43
Closure bolting	SA-193, Gr. B7
Reactor Coolant Pump	
Pressure forgings F348	SA-182, Type F304, F316, F347 or
Pressure casting	SA-351, Gr. CF8, CF8A or CF8M
Tube and pipe	SA-213, 376 or 312, seamless Type 304 or 316
Pressure plates	SA-240, Type 304 or 316
Bar material	SA-479, Type 304 or 316

* . Designated A-8 in the 1974 edition of the ASME Code

TABLE 5.2-2
(Sheet 3 of 3)

CLASS 1 PRIMARY COMPONENTS MATERIAL SPECIFICATIONS

Closure bolting	SA-193, 320, 540 or 453, Gr. 660
Flywheel	SA-533, Gr. B, Class 1
Reactor Coolant Piping	
Reactor coolant pipe	SA-376, Gr. 304N or SA-351, Gr. CF8A centrifugal casting
Reactor coolant fittings	SA-351, Gr. CF8A
Branch nozzles	SA-182, Gr. F316N
Surge line	SA-376, Gr. 304, 316 or F304N
Auxiliary piping ½ through 12 inch and wall schedules 40S through 80S (ahead of second isolation valve)	ANSI B36.19
All other auxiliary piping (ahead of second isolation valve)	ANSI B36.10
Socket weld fittings	ANSI B16.11
Piping flanges	ANSI B16.5
Control Rod Drive Mechanism	
Latch housing	SA-182, Gr. F304; SA-351, Gr. CF8
Rod travel housing	SA-182, Gr. F304; SA-336, Gr. F8; SA-312, Gr. Type 304
Cap	SA-479, Type 304
Welding materials	Stainless steel weld metal analysis A-8

TABLE 5.2-7

REACTOR COOLANT PRESSURE BOUNDARY VALVE NUMBERS

CSV-0002	RCV-0080	RHV-0063	SIV-0130*
CSV-0018	RCV-0081	RHV-0065	SIV-0140*
CSV-0034	RCV-0087*	SIV-0003	SIV-0143
CSV-0050	RCV-0088*	SIV-0005*	SIV-0144*
CSV-0175	RCV-0109	SIV-0006*	SIV-0147
CSV-0176	RCV-0110	SIV-0017	SIV-0148*
CSV-0178	RCV-0115	SIV-0020*	SIV-0151
CSV-0179	RCV-0116	SIV-0021*	SIV-0152*
CSV-0181	RCV-0117	SIV-0032	SIV-0155
CSV-0182	RCV-0122	SIV-0035*	SIV-0156*
CSV-0185	RCV-0124	SIV-0036*	RC-PCV-456A
CSV-0186	RCV-0475*	SIV-0047	RC-PCV-456B
CSV-0471	RCV-0479*	SIV-0050*	RC-LCV-459
CSV-0472	RHV-0015*	SIV-0051*	RC-LCV-460
CSV-0473	RHV-0029*	SIV-0081*	RC-PCV-455A
CSV-0474	RHV-0030*	SIV-0082*	RC-PCV-455B
CSV-0752	RHV-0031*	SIV-0086*	
RCV-0001	RHV-0050*	SIV-0087*	
RCV-0017	RHV-0051*	SIV-0106*	
RCV-0022*	RHV-0052*	SIV-0110*	
RCV-0023*	RHV-0053*	SIV-0118*	
RCV-0050	RHV-0059	SIV-0122*	
RCV-0079	RHV-0061	SIV-0126*	

* Reactor coolant pressure isolation valves which require leakage testing in accordance with the Technical Specifications.

TABLE 5.2-8

TYPICAL PLANT THERMAL-HYDRAULIC PARAMETERS

	Units	2-Loop	3-Loop	4-Loop	Seabrook
Heat Output, Core	MWt	1,780	2,652	3,411	3,411
System Pressure	psia	2,250	2,250	2,250	2,250
Coolant Flow	gpm	178,000	265,500	354,000	382,800
Average Core Mass Velocity	10^6lb./hr-ft^2	2.42	2.33	2.50	2.62
Inlet Temperature	°F	545	544	552.5	558.8
Core Average T_{mod}	°F	581	580	588	591.8
Core Length	Ft	12	12	12	12
Average Power Density	kW/1	102	100	104	104
Maximum Fuel Temperature	°F	4,100	4,200	4,200	4,200
Pressurizer Volume	Ft^3	1,000	1,400	1,800	1,800
Pressurizer Volume Ratioed to Primary System Volume		0.157	0.148	0.148	0.146
Peak Surge Rate for Pressurizer Safety Valve Sizing Transient	Ft^3/sec	21.8	33.2	41.0	36.733
Pressurizer Safety Valve Flow at 2500 psia - +3% Accumulation	Ft^3/sec	26.1	36.1	43.3	43.2
Ratio of Safety Valve Flow to Peak Surge Rate		1.197	1.087	1.056	1.179
Full Power Steam Flow per Loop	lb./sec	1,078	1,076	1,038	1,051
Nominal Shell-Side Steam Generator Water Mass per Loop	lb.	100,300	106,000	106,000	107,000

Section III and 10 CFR 50, Appendix G. Compliance with Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," is discussed in Section 1.8. Nondestructive examinations are performed in accordance with the ASME Code, Section III. Fracture toughness data for the Seabrook Unit 1 reactor vessel bolting materials is presented in Table 5.3-4.

Westinghouse refueling procedures require the studs, nuts and washers to be removed from the reactor closure and be placed in storage racks during preparation for refueling. The storage racks are then removed from the refueling cavity and stored at convenient locations on the containment operating deck prior to removal of the reactor closure head and refueling cavity flooding. Therefore the reactor closure studs are never exposed to the borated refueling cavity water. Threaded portions of the studs have been treated with an anti galling coating.

The stud holes in the reactor flange are sealed with special plugs before removing the reactor closure thus preventing leakage of the borated refueling water into the stud holes.

5.3.2 Pressure-Temperature Limits

5.3.2.1 Limit Curves

Startup and shutdown operating limitations will be based on the properties of the core region materials of the reactor pressure vessel. Actual material property test data will be used. The methods outlined in Appendix G to Section III of the ASME Code will be employed for the shell regions in the analysis of protection against nonductile failure. The initial operating curves are calculated assuming a period of reactor operation such that the beltline material will be limiting. The heatup and cooldown curves are given in the Technical Specifications. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the adjusted reference nil-ductility temperature shift (ΔRT_{NDT}).

Predicted RT_{NDT} values are derived using the maximum fluence at ¼T (thickness) and ¾T location (tips of the code reference flaw when flaw is assumed at inside diameter and outside diameter locations, respectively) curve. This curve is presented in the Technical Specifications. For a selected time of operation, this shift is assigned a sufficient magnitude so that no unirradiated ferritic materials in other components of the Reactor Coolant System (RCS) will be limiting in the analysis.

The operating curves including pressure-temperature limitations are calculated in accordance with 10 CFR 50, Appendix G and ASME Code, Section III, Appendix G, requirements. Changes in fracture toughness of the core region plates or forgings, weldments and associated heat-affected zones due to radiation damage will be monitored by a surveillance program which conforms with ASTM E185-79, 10 CFR 50, Appendix H. The evaluation of the radiation damage in this surveillance program is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and ½T compact tension specimens. The post-irradiation testing will be carried out during the lifetime of the reactor vessel. Specimens are irradiated in capsules located near the core midheight and removable from the vessel at specified intervals.

The results of the radiation surveillance program will be used to verify that the RT_{NDT} predicted from the effects of the fluence and copper content curve is appropriate and to make any changes necessary to correct the fluence and copper curves if RT_{NDT} determined from the surveillance program is greater than the predicted RT_{NDT}. Temperature limits for preservice hydrotests and in-service leak and hydrotests will be calculated in accordance with 10 CFR 50, Appendix G.

Compliance with Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," is discussed in Section 1.8.

5.3.2.2 Operating Procedures

The transient conditions that are considered in the design of the reactor vessel are presented in Subsection 3.9(N).1.1. These transients are representative of the operating conditions that should prudently be considered to occur during plant operation. The transients selected form a conservative basis for evaluation of the RCS to insure the integrity of the RCS equipment.

Those transients listed as upset condition transients are given in Table 3.9(N)-1. None of these transients will result in pressure-temperature changes which exceed the heatup and cooldown limitations as described in Subsection 5.3.3.1 and in the Technical Specifications.

5.3.3 Reactor Vessel Integrity

5.3.3.1 Design

The reactor vessel is cylindrical with a welded hemispherical bottom head and a removable, bolted, flanged and gasketed, hemispherical upper head. The reactor vessel flange and head are sealed by two hollow metallic O-rings. Seal leakage is detected by means of two leakoff connections: one between the inner and outer ring and one outside the outer O-ring. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. The reactor vessel closure head contains head adaptors. These head adaptors are tubular members, attached by partial penetration welds to the underside of the closure head. The upper end of these adaptors contain acme threads for the assembly of control rod drive mechanisms or instrumentation adaptors. The seal arrangement at the upper end of these adaptors consists of a welded flexible canopy seal. Inlet and outlet nozzles are located symmetrically around the vessel. Outlet nozzles are

Periodic analysis of the coolant chemical composition is performed to monitor the adherence of the system to desired reactor coolant water quality. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System and sampling system which are described in Chapter 9.

b. Sensitized Stainless Steel

Sensitized stainless steel is discussed in Subsection 5.2.3.

c. Contaminant Control

Contamination of stainless steel and Inconel by copper, low melting temperature alloys, mercury and lead is prohibited. Colloidal graphite is the only permissible thread lubricant.

Except for those times when thermal insulation is removed for the installation of new thermal insulation (due to mechanical damage or age), the austenitic stainless steel surfaces are cleaned and analyzed to a halogen limit specified by the NSSS vendor.

5.4.3.4 Tests and Inspections

The RCS piping quality assurance program is given in Table 5.4-6.

Volumetric examination is performed throughout 100 percent of the wall volume of each pipe and fitting, in accordance with the applicable requirements of Section III of the ASME Code for all pipe 27½ inches and larger. All unacceptable defects are eliminated in accordance with the requirements of the same section of the code.

A liquid penetrant examination is performed on both the entire outside and inside surfaces of each finished fitting in accordance with the criteria of the ASME Code, Section III. Acceptance standards are in accordance with the applicable requirements of the ASME Code, Section III.

The pressurizer surge line conforms to SA-376, Grade 304, 304N, or 316 with supplementary requirements S2 (transverse tension tests), and S6 (ultrasonic test). The S2 requirement applies to each length of pipe. The S6 requirement applies to 100 percent of the piping wall volume.

The end of pipe sections, branch ends and fittings are machined back to provide a smooth weld transition adjacent to the weld path.

5.4.4 Main Steam Line Flow Restrictor

5.4.4.1 Design Basis

The outlet nozzle of the steam generator is provided with a flow restrictor designed to limit steam flow in the unlikely event of a break in the main steam line. A large increase in steam flow will create a backpressure which limits further increase in flow. Several protective advantages are thereby provided: rapid rise in containment pressure is prevented, the rate of heat removal from the reactor coolant is kept within acceptable limits, thrust forces on the main steam line piping are reduced, and stresses on internal steam generator components, particularly the tube sheet and tubes, are limited. The resistor is also designed to minimize the unrecovered pressure loss across the restrictor during normal operation.

5.4.4.2 Design Description

The flow restrictor consists of seven Inconel (ASME SB-163) venturi inserts which are inserted into the holes in an integral steam outlet nozzle forging. The inserts are arranged with one venturi at the centerline of the outlet nozzle and the other six equally spaced around it. After insertion into the nozzle forging holes, the Inconel venturi inserts are welded to the Inconel cladding on the inner surface of the forgings.

5.4.4.3 Design Evaluation

The flow restrictor design has been sufficiently analyzed to assure its structural adequacy. The equivalent throat diameter of the steam generator outlet is 16 inches, and the resultant pressure drop through the restrictor at 100 percent steam flow is approximately 2.75 psi. This is based on a design flow rate of 3.785×10^6 lb/hr. Materials of construction and manufacturing of the flow restrictor are in accordance with Section III of the ASME Code.

5.4.4.4 Tests and Inspections

Since the restrictor is not a part of the steam system boundary, no tests and inspection beyond those during fabrication are anticipated.

5.4.5 Main Steam Isolation System

The Main Steam Isolation System is composed of the main steam isolation valves, actuators, logic cabinets and portions of the Engineered Safety Features Actuation System. The MSIVs are located in the mainsteam and feedwater pipe chases adjacent to the Containment.

5.4.5.1 Design Bases

a. Isolate the steam generators and main steam lines in the event of a main steam or feedwater line rupture to prevent the uncontrolled blowdown of more than one steam generator.

b. Isolate the Containment from the outside environment in the event of a design basis accident, upon receipt of containment isolation signals as described in Subsection 6.2.4.

5.4.5.2 Description

The main steam isolation system components are described in Section 10.3 and Subsection 6.2.4.

5.4.5.3 System Operation

The isolation system operation is described in Subsection 6.2.4. The isolation valve operation is described in Subsection 10.3.2.

5.4.5.4 Design Evaluation

The main steam isolation valves will close in 3 to 5 seconds after receipt of a signal from the Engineered Safety Features Actuation System, and are capable of stopping flow from either the forward or reverse directions.

Immediately after a pipe break and until isolation occurs, all steam generators will be partially blown down. After isolation occurs, only the inventory portions of the main steam piping will be blown down, if the break is downstream of an isolation valve. If the break occurs upstream of an isolation valve, either inside or outside Containment, one steam generator will be blown down.

5.4.5.5 Tests and Inspections

Each MSIV is given an operability test every three months. This involves partially stroking the valve. See Subsection 10.3.2 for additional discussion of MSIV testing.

5.4.6 Reactor Core Isolation Cooling System

Not applicable to Seabrook.

5.4.7 Residual Heat Removal System

The Residual Heat Removal System (RHRS) transfers heat from the Reactor Coolant System (RCS) to the Component Cooling System (CCS) to reduce the temperature of the reactor coolant to the cold shutdown temperature at a controlled rate during the second part of normal plant cooldown, and maintains this temperature until the plant is started up again.

Parts of the RHRS also serve as parts of the ECCS during the injection and recirculation phases of a LOCA (see Section 6.3).

The RHRS also is used to transfer refueling water between the refueling cavity and the refueling water storage tank at the beginning and end of the refueling operations.

Nuclear plants employing the same RHRS design as the Seabrook Station are given in Section 1.3.

5.4.7.1 Design Bases

RHRS design parameters are listed in Table 5.4-7.

The RHRS is placed in operation approximately four hours after reactor shutdown, when the temperature and pressure of the RCS are approximately 350°F and 365 psig, respectively. Assuming that two heat exchangers and two pumps are in service and that each heat exchanger is supplied with component cooling water at 3000 gpm, the RHRS is designed to reduce the temperature of the reactor coolant from 350°F to 125°F within 20 hours. The time required under these conditions to reduce the reactor coolant temperature from 350°F to 212°F is less than 3 hours. Maximum anticipated component cooling water temperature is 102°F during initial RHRS operation; however, it will approach 85°F at 24 hours after reactor shutdown. The heat load handled by the RHRS during the cooldown transient includes residual and decay heat from the core, coolant sensible heat, and reactor coolant pump heat. The design heat load is based on the decay heat fraction that exists at 24 hours following reactor shutdown from an extended run at full power. The normal cooldown curve is shown in Figure 5.4-5.

Assuming that only one heat exchanger and pump are in service and that the heat exchanger is supplied with component cooling water at 3000 gpm, the RHRS is capable of reducing the temperature of the reactor coolant from 350°F to 200°F within 33.5 hours. The time required, under these conditions, to reduce reactor coolant temperature from 350°F to 212°F is less than 23 hours. Maximum anticipated component cooling water temperature is 102°F during initial RHRS operation; however, it will approach 85°F at 120 hours after reactor shutdown. The single train RHR cooldown curve is shown on Figure 5.4-6.

Assuming 3000 gpm water at 85°F with 2 pumps and 2 heat exchangers in operation, the system is cooled from 350°F to 125°F so that 125°F is achieved within 24 hours after shutdown. As the cooldown proceeds, the CCW temperature may rise as high as 102°F but will drop off again to 85°F.

The RHRS is designed to be isolated from the RCS whenever the RCS pressure exceeds the RHRS design pressure. The RHRS is isolated from the RCS on the suction side by two motor-operated valves in series on each suction line. Each motor-operated valve is interlocked to prevent its opening if RCS pressure is greater than that which could result in the RHR system design pressure being exceeded. (This includes the effects of instrument uncertainty and bistable deadband.) Refer to the Station Technical Specifications. (See Figure 5.4-7.) The RHRS is isolated from the RCS on the discharge side by two check valves in each return line. Also provided on the discharge side is a normally open motor-operated valve downstream of each RHRS heat exchanger. (These check valves and motor-operated valves are not considered part of the RHRS; they are shown as part of the ECCS, see Figures 6.3-1 through 6.3-5.)

Those valves used for hot leg injection are in the residual heat removal pump and safety injection pump discharge and do not qualify as high-to-low pressure isolation barriers since they are backed up by normally closed motor-operated gate valves which form an additional high-to-low pressure barrier. Those valves used for charging/safety injection are in the high pressure discharge piping of the charging pumps and do not quality as high-to-low pressure isolation barriers. Additionally, these check valves are backed up by normally closed motor-operated gate valves.

Although they are backed up by normally closed motor-operated gate valves, periodic leakage testing is currently performed for the following hot leg and hi-head check valves as stated in Table 5.2-7 and Section 6.3.4.2:

RHV-0050, RHV-0051, RHV-0052, RHV-0053,
SIV-0081, SIV-0082, SIV-0086, SIV-0087,
SIV-0106, SIV-0110,
SIV-0140, SIV-0144, SIV-0148, SIV-0152 and SIV-0156

The check valves in the cold leg injection paths do form high-to-low pressure isolation barriers.

Appropriate connections for leak testing these valves are shown in Figures 5.4-9 through 5.4-11, and 6.3-1 through 6.3-5.

Each inlet line to the RHRS is equipped with a pressure relief valve designed to relieve the combined flow of all the charging pumps at the relief valve set pressure. These relief valves also protect the system from inadvertent overpressurization during plant cooldown or startup.

Relief valves have been installed to provide thermal relief protection for water trapped between the redundant RHR suction isolation valves (RC-V22, -V23 in line 13-1-2501-12, and RC-V87, -V88 in line 58-1-2501-12). These lines and valving will relieve excessive pressure to the pressurizer relief tank.

Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve designed to relieve the maximum possible back leakage through the valves isolating the RHRS from the RCS.

The RHRS is designed for a single nuclear power unit, and is not shared between the two units.

The RHRS is designed to be operable from the control room with limited action outside the control room for normal operation. Manual operations required of the operator are: unlocking and closing the circuit breakers for the RHR suction isolation valves, opening the suction isolation valves, positioning the flow control valves downstream of the RHRS heat exchangers, and starting the RHRS pumps. Because of its redundant two train design, the RHRS is designed to accept all major component single failures, with the only effect being an extension in the required cooldown time. For two low probability electrical system single failures i.e., failure in the suction isolation valve interlock circuitry, or diesel generator failure in conjunction with loss of offsite power; limited operator action outside the control room is required to open the suction isolation valves. Manual actions are discussed in further detail in Subsections 5.4.7.2f and 5.4.7.2g. The RHRS motor-operated isolation valves located inside Containment are not susceptible to flooding following a steam line break or a loss-of-coolant accident. Although Westinghouse considers it to be of low probability, spurious operation of a single motor-operated valve can be accepted without loss of function, as a result of the redundant two train design.

In its normal function, the RHRS transfers heat from the Reactor Coolant System to the Component Cooling Water System during normal plant cooldown (two trains operating) and during plant maintenance/refueling (one train operating). For this function, connections between the two trains of residual heat removal are isolated to prevent a passive failure from affecting more than one loop.

The RHRS also serves as the low pressure pump suction of the Emergency Core Cooling System (ECCS). As discussed in Subsection 6.3.2.5, the system has been designed and proven by analysis as having the ability to withstand any single credible active failure during the injection phase (post-LOCA), or any active or passive failure during the recirculation phase. This design approach is consistent with ANSI/ANS 58.9-1981 "Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems."

Additional information about ECCS design protection for a passive failure is contained in Subsection 6.3.2.5b and a single passive failure analysis is presented in Table 6.3-6.

Missile protection, protection against dynamic effects associated with the postulated rupture of piping, and seismic design are discussed in Sections 3.5, 3.6(N), respectively.

5.4.7.2 System Design

a. Schematic Piping and Instrumentation Diagrams

The RHRS, as shown in Figures 5.4-9 through 5.4-12, consists of two residual heat exchangers, two residual heat removal pumps, and the associated piping, valves, and instrumentation necessary for operational control. The inlet lines to the RHRS are connected to the hot legs of two reactor coolant loops, while the return lines are connected to the cold legs of each of the reactor coolant loops. These return lines are also the ECCS low head injection lines (see Figures 6.3-1 through 6.3-5).

The RHRS suction lines are isolated from the RCS by two motor-operated valves in series located inside the Containment. Each discharge line is isolated from the RCS by two check valves located inside the Containment and by a normally-open motor-operated valve located outside the Containment. (The check valves and the motor-operated valve on each discharge line are not part of the RHRS; these valves are shown as part of the ECCS, see Figure 6.3-1.)

During RHRS operation, reactor coolant flows from the RCS to the residual heat removal pumps, through the tube side of the residual heat exchangers, and back to the RCS. The heat is transferred to the component cooling water circulating through the shell side of the residual heat exchanger.

Coincident with operation of the RHRS, a portion of the reactor coolant flow may be diverted from downstream of the residual heat exchangers to the Chemical and Volume Control System (CVCS) low pressure letdown line for cleanup and/or pressure control. By regulating the diverted flowrate and the charging flow, the RCS pressure may be controlled. Pressure regulation is necessary to maintain the pressure range dictated by the fracture prevention criteria requirements of the reactor vessel and by the Number 1 seal differential pressure and net positive suction head requirements of the reactor coolant pumps.

The RCS cooldown rate is manually controlled by regulating the reactor coolant flow through the tube side of the residual heat exchangers. The flow control valve in the bypass line around each residual heat exchange automatically maintains a constant return flow to the RCS. Instrumentation is provided to monitor system pressure, temperature and total flow.

The RHRS is also used for filling the refueling cavity before refueling. Also, after refueling operations, the RHRS is used to pump the water back to the refueling water storage tank until the water level is brought down to the flange of the reactor vessel. The remainder of the water is removed via a drain connection at the bottom of the refueling canal.

When the RHRS is in operation, the water chemistry is the same as that of the reactor coolant. Provision is made for the Process Sampling System to extract samples from the flow of reactor coolant downstream of the residual heat exchangers. A local sampling point is also provided on each residual heat removal train between the pump and heat exchanger.

The RHRS also functions, in conjunction with the high head portion of the ECCS, to provide injection of borated water from the refueling water storage tank into the RCS cold legs during the injection phase following a loss-of-coolant accident.

In its capacity as the low head portion of the ECCS, the RHRS provides long-term recirculation capability for core cooling following the injection phase of the loss-of-coolant accident. This function is accomplished by aligning the RHRS to take fluid from the containment sump, cool it by circulation through the residual heat exchangers, and supply it to the core directly as well as via the centrifugal charging pumps and safety injection pumps.

The use of the RHRS as part of the ECCS is more completely described in Section 6.3.

The RHR pumps, in order to perform their ECCS function, are interlocked to start automatically on receipt of a safety injection signal (see Section 6.3).

Each RHR loop is isolated from the Reactor Coolant System by two redundant motor-operated valves. One valve in each loop is located inside the missile barrier and is powered from Train B. The redundant valve is located outside the missile barrier and is powered from Train A. This ensures that isolation can be maintained between the RHR and the RCS in the event of a single failure of the power supply.

The RHR suction isolation valves in each inlet line from the RCS are separately interlocked to prevent being opened when RCS pressure is greater than that which could result in the RHR system design pressure being exceeded. (This includes the effects of instrument uncertainty and bistable deadband.) Refer to the Station Technical Specifications. These interlocks are described in more detail in Subsections 5.4.7.2d and 7.6.2 (see Figure 5.4-7).

The RHR suction isolation valves are also interlocked to prevent their being opened unless the isolation valves in the recirculation lines from the residual heat exchanger outlets to the suctions of the safety injection pumps and centrifugal charging pumps are closed (see Figure 5.4-7).

Failure of an interlock will result in failure of a valve to open.

Failures of interlocks are inherently addressed in the failure modes and effects analyses by addressing the valves that are controlled by the interlocks.

The motor-operated valves in the RHR miniflow bypass lines are interlocked to open when the RHR pump discharge flow is less than 750 gpm and close when the flow exceeds approximately 1400 gpm (see Figure 5.4-7, Sheet 2).

b. Equipment and Component Descriptions

The materials used to fabricate RHRS components are in accordance with the applicable code requirements. All parts of components in contact with borated water are fabricated or clad with austenitic stainless steel or equivalent corrosion resistant material. Component parameters are given in Table 5.4-8.

1. Residual Heat Removal Pumps

Two pumps are installed in the RHRS. The pumps are sized to deliver reactor coolant flow through the residual heat removal heat exchangers to meet the plant cooldown requirements. The use of two separate residual heat removal trains assures that cooling capacity is only partially lost should one pump become inoperative.

The residual heat removal pumps are protected from overheating and loss of suction flow by miniflow bypass lines that assure flow to the pump suction. A valve located in each miniflow line is controlled by a signal from the flow switches located in each pump discharge header. The control valves open when the residual heat removal pump discharge flow is less than approximately 750 gpm and close when the flow exceeds approximately 1400 gpm.

A pressure sensor in each pump discharge header provides a signal for an indicator in the control room. A high pressure alarm is also provided in the control room.

The two pumps are vertical, centrifugal units with mechanical seals on the shafts. All pump surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

The residual heat removal pumps also function as the low head safety injection pumps in the ECCS (see Section 6.3 for further information and for the residual heat removal pump performance curves).

2. Residual Heat Exchangers

Two residual heat exchangers are installed in the system. The heat exchanger design is based on heat load and temperature differences between reactor coolant and component cooling water existing 24 hours after reactor shutdown when the temperature difference between the two systems is small.

The installation of two heat exchangers in separate and independent residual heat removal trains assures that the heat removal capacity of the system is only partially lost if one train becomes inoperative.

The residual heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell. The tubes are welded to the tube sheet to prevent leakage of reactor coolant.

The residual heat exchangers also function as part of the ECCS (see Section 6.3).

3. Residual Heat Removal System Valves

Valves that perform a modulating function are equipped with two sets of packings and an intermediate leakoff connection that discharges to the drain header.

Manual and motor-operated valves have backseats to facilitate repacking and to limit stem leakage when the valves are open.

Leakage connections are provided where required by valve size and fluid conditions.

c. System Operation

1. Plant Startup

Generally, while at cold shutdown condition, decay heat from the reactor core is being removed by the RHRS. The number of pumps and heat exchangers in service depends upon the heat load at the time.

At initiation of plant heatup, the RHRS remains aligned to the RCS in order to maintain a low pressure letdown path to the CVCS. This flowpath is used to provide RCS pressure control while the pressurizer heaters are forming the steam bubble and heating the pressurizer. As the reactor coolant pumps are started, their thermal input heats the reactor coolant inventory. Once the pressurizer steam bubble formation is complete, the RHRS is isolated from the RCS and aligned for operation as part of the ECCS.

2. Power Generation and Hot Standby Operation

During power generation and hot standby operation, the RHRS is not in service but is aligned for operation as part of the ECCS.

3. Plant Cooldown

Plant cooldown is defined as the operation which brings the reactor from no-load temperature and pressure to cold conditions.

The initial phase of plant cooldown is accomplished by transferring heat from the RCS to the steam and power conversion system through the use of the steam generators.

When the reactor coolant temperature and pressure are reduced to approximately 350°F and 362 psig, approximately four hours after reactor shutdown, the second phase of cooldown starts with the RHRS being placed in operation.

Startup of the RHRS includes a warmup period during which reactor coolant flow through the heat exchangers is limited to minimize thermal shock. The rate of heat removal from the reactor coolant is manually controlled by regulating the coolant flow through the residual heat exchangers. By adjusting these control valves downstream of the residual heat exchangers, the mixed mean temperature of the return flows is overpressurization in the RHRS. All credible events were examined for their potential to overpressurize the RHRS. These events included normal operating conditions, infrequent transients, and abnormal occurrences.

The most severe credible overpressure transient is the mass input transient resulting form one centrifugal charging pump operating in an unthrottled condition with flow to the Reactor Coolant System while letdown flow is isolated. The capacity of a single RHRS inlet relief valve is sufficient to satisfy RHRS overpressure requirements for this transient during the hot shutdown and cold shutdown operational modes. Procedures and administrative controls ensure that more severe RHRS overpressure transients do not occur during RHRS operations.

Two situations were analyzed to confirm the adequacy of the RHRS relief valve to prevent overpressurization of the RHRS. The first considers the RHRS in the initial phase of RCS cooldown. RCS temperature and pressure are 350°F and 400 psig respectively, and one centrifugal charging pump is in operation. The operator initiates RHRS operation by opening one inlet line and starting the corresponding RHR pump. At this time, a complete loss of plant air occurs, the charging line flow control valve fails open, and the low pressure letdown flow control valve fails closed. The maximum charging pump injection rate is approximately 400 gpm at the relief valve set pressure of 450 psig.

The second transient consists of the RCS in the final stages of cooldown. RCS temperature and pressure are less than 200°F and 450 psig, respectively. The additional conservatism of a second centrifugal charging pump is assumed. The combined flow from both charging pumps is less than 600 gpm.

As each RHRS inlet relief valve has a capacity of 900 gpm at a set pressure of 450 psig, sufficient margin is present to bound liquid and two-phase relief rate uncertainties.

Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve to relieve the maximum possible back-leakage through the valves separating the RHRS from the RCS. Each valve has a relief flow capacity of 20 gpm at a set pressure of 600 psig. These relief valves are located in the ECCS (see Figure 6.3-1).

The fluid discharge by the suction side relief valves is collected in the pressurizer relief tank. The fluid discharged by the discharge side relief valves is collected in the primary drain tank of the equipment and floor drain system.

The design of the RHRS includes two motor-operated gate isolation valves in series on each inlet line between the high pressure RCS and the lower pressure RHRS. They are closed, with power locked out, during normal operation and are only opened for residual heat removal during a plant cooldown after the RCS pressure is reduced to 362 psig or lower and RCS temperature is reduced to approximately 350°F. During a plant startup, the inlet isolation valves are shut after drawing a bubble in the pressurizer and prior to increasing RCS pressure above 400 psig. These isolation valves are provided with a "prevent-open" interlock that is designed to prevent possible exposure of the RHRS to normal RCS operating pressure. The two inlet isolation valves in each residual heat removal subsystem are separately and independently interlocked with pressure signals to prevent their being opened whenever the RCS pressure is greater than approximately that which could result in the RHR system design pressure being exceeded. (This includes the effects of instrument uncertainty and bistable deadband.) Refer to the Station Technical Specifications. Power to the suction line isolation valve motor operators is locked out to prevent spurious opening. An alarm is actuated if a suction isolation valve is open when RCS pressure is above 362 psig as a backup to administrative control to ensure that both valves, rather than just one, are closed at normal RCS operating pressure, thereby ensuring compliance with reactor coolant pressure boundary isolation criteria.

The use of two independently powered motor-operated valves in each of the two inlet lines, along with two independent pressure interlock signals assures a design which meets applicable single failure criteria. Not only more than one single failure but also different failure mechanisms must be postulated to defeat the function of preventing possible exposure of the RHR system to normal RCS operating pressure. These protective interlock and alarm designs, in combination with plant operating procedures, provide diverse means of accomplishing the protective function. For further information on the instrumentation and control features, see Subsection 7.6.2.

Valve position indication is provided on the main control board for both the open and closed positions for the RHR suction line isolation valves and RHR recirculation valves. On loss of suction, the RHR system would go into a recirculation mode, thus protecting the operating RHR pump. Alarms are provided that will actuate, if either suction valve for an operating RHR pump is not fully open and if the flow through the RHR pump is below the minimum expected. Indication of reduced flow and pump amperage is also provided in the control room for the operator to observe during normal system surveillance.

Isolation of the low pressure RHRS from the high pressure RCS is provided on the discharge side by two check valves in series. These check valves are located in the ECCS and their testing is described in Subsection 6.3.4.2.

e. Applicable Codes and Classifications

The entire RHRS is designed as Nuclear Safety Class 2, with the exception of the suction isolation valves which are Safety Class 1. Component codes and classifications are given in Section 3.2.

f. System Reliability Considerations

General Design Criterion 34 requires that a system to remove residual heat be provided. The safety function of this system is to transfer fission product decay heat and other residual heat from the core at a rate sufficient to prevent fuel or pressure boundary design limits from being exceeded. Safety grade systems are provided in the plant design, both NSSS scope and BOP scope, to perform this function. The safety grade systems which perform this function for all plant conditions except a LOCA are: the RCS and steam generators, which operate in conjunction with the Emergency Feedwater System; the steam generator safety and power-operated relief valves; and the RHRS which operates in conjunction with the Component Cooling Water System and the Service Water System. For LOCA conditions, the safety grade system which performs the function of removing residual heat from the reactor core is the ECCS, which operates in conjunction with the Component Cooling Water System and the Service Water System.

The Emergency Feedwater System, along with the steam generator safety and power-operated relief valves, provides a completely separate, independent, and diverse means of performing the safety function of removing residual heat, which is normally performed by the RHR system when RCS temperature is less than 350°F. The Emergency Feedwater System is capable of performing this function for an extended period of time following plant shutdown.

The RHR system is provided with two residual heat removal pumps, and two heat exchangers arranged in two separate, independent flow paths. To assure reliability, each residual heat removal pump is connected to a different vital bus. Each residual heat removal train is isolated from the RCS on the suction side by two motor-operated valves in series with each valve receiving power via a separate motor control center and from a different vital bus. Each suction isolation valve is also interlocked to prevent exposure of the RHR system to the normal operating pressure of the RCS (see Subsection 5.4.7.2d).

RHR system operation for normal conditions and for major failures is accomplished from the control room. Limited action outside the control room is required to prepare the RHRS for operation as discussed in Subsection 5.4.7.2f. The redundancy in the RHR system design provides the system with the capability to maintain its cooling function even with major single failures, such as failure of an RHR pump, valve, or heat exchanger, since the redundant train can be used for continued heat removal.

Although such major system failures are within the system design basis, there are other less significant failures which can prevent opening of the RHR suction isolation valves from the control room. Since these failures are of a minor nature, improbable to occur, and easily corrected outside the control room, with ample time to do so, they have been realistically excluded from the engineering design basis. Such failures are not likely to occur during the limited time period in which they can have any effect (i.e., when opening the suction isolation valves to initiate RHR operation); however, even if they should occur, they have no adverse safety impact and can be readily corrected. In such a situation, the Emergency Feedwater System and steam generator power-operated relief valves can be used to perform the safety function of removing residual heat. In fact, they can be used to continue the plant cooldown below 350°F, until the RHR system is made available.

One failure of this type is a failure in the interlock circuitry which is designed to prevent exposure of the RHR system to the normal operating pressure of the RCS (see Subsection 5.4.7.2d). In the event of such a failure, RHR system operation can be initiated by defeating the failed interlock through corrective action at the solid-state protection system cabinet or by taking local control at the remote safe shutdown control panels.

The other type of failure which can prevent opening the RHR suction isolation valves from the control room is a failure of an electrical power train. Such a failure is extremely unlikely to occur during the few minutes out of a year's operating time during which it can have any consequence.

To ensure operation of at least one 100 percent RHR train in the event of loss of a single power supply, a temporary power connection will be provided to the MCC compartment of the RHR valve whose power supply has failed. This temporary power connection will be a cable from a designated compartment of the unaffected redundant MCC to the compartment of the affected valve. The MCCs involved are MCC E521 and E621. See Figure 8.3-27 for location of MCCs. Terminals are provided in the MCC compartments to facilitate a temporary connection to the line side of the breaker-starter combination of valve. Control of the valve is still from the MCB. The individual MCC compartments are capable of being withdrawn from the bus bar connection and secured in the disconnected position, which precludes a faulted MCC from affecting the operation of the valve having the temporary connection.

The only impact of either of the above types of failures is some delay in initiating residual heat removal operation, while action is

TABLE 5.4-17
(Sheet 1 of 6)

FAILURE MODE AND EFFECTS ANALYSIS - RESIDUAL HEAT REMOVAL SYSTEM
ACTIVE COMPONENTS - PLANT COOLDOWN OPERATION

Component	Failure Mode	*Effect on System Operation	**Failure Detection Method	Remarks
1. Motor-operated gate valve RC-V23 (RC-V88 analogous)	a. Fails to open on demand (open manual mode CB switch selection)	a. Failure blocks reactor coolant flow from hot leg of RC loop #1 through Train A of RHRS. Fault reduces redundancy of RHR coolant trains provided. No effect on safety for system operation. Plant cooldown requirements will be met by reactor coolant flow from hot leg of RC loop #4 flowing through Train B of RHRS; however, time required to reduce RCS temperature will be extended.	a. Valve position indication (closed to open position change) at CB; RC pressure indication (PI-405) at CB; RHR Train A discharge flow indication (FI-618) at CB; and RHR pump discharge pressure indication (PI-614) at CB.	1. Valve is electrically interlocked with RHR to charging pump suction line isolation valve (RH-V35) and with a "prevent-open" RC pressure interlock (PB-405). The valve cannot be opened from the CB if the indicated isolation valve is open or if RC pressure exceeds the interlock condition.
	b. Once open, fails closed (open manual mode CB switch selection)	b. Same effect on system operation as that stated above for failure mode "Fails to open on demand."	b. Same method of detection as those stated above for failure mode "Fails to open on demand."	2. If both trains of RHRS system are unavailable for plant cooldown due to multiple component failures, the Emergency Feedwater System and S.G. power operated relief valves can be used to perform the safety function of removing residual heat.
2. Motor-operated gate valve RC-V22 (RC-V87 analogous)	a. Same failure modes as those stated for item #1.	a. Same effect on system operation as that stated for item #1.	a. Same methods of detection as those stated for item #1, except for RC pressure indication (PI-403) at CB.	1. Same remarks as those stated for item #1, except for pressure interlock (PB-403) control.

* See list at end of table for definition of acronyms and abbreviations used.

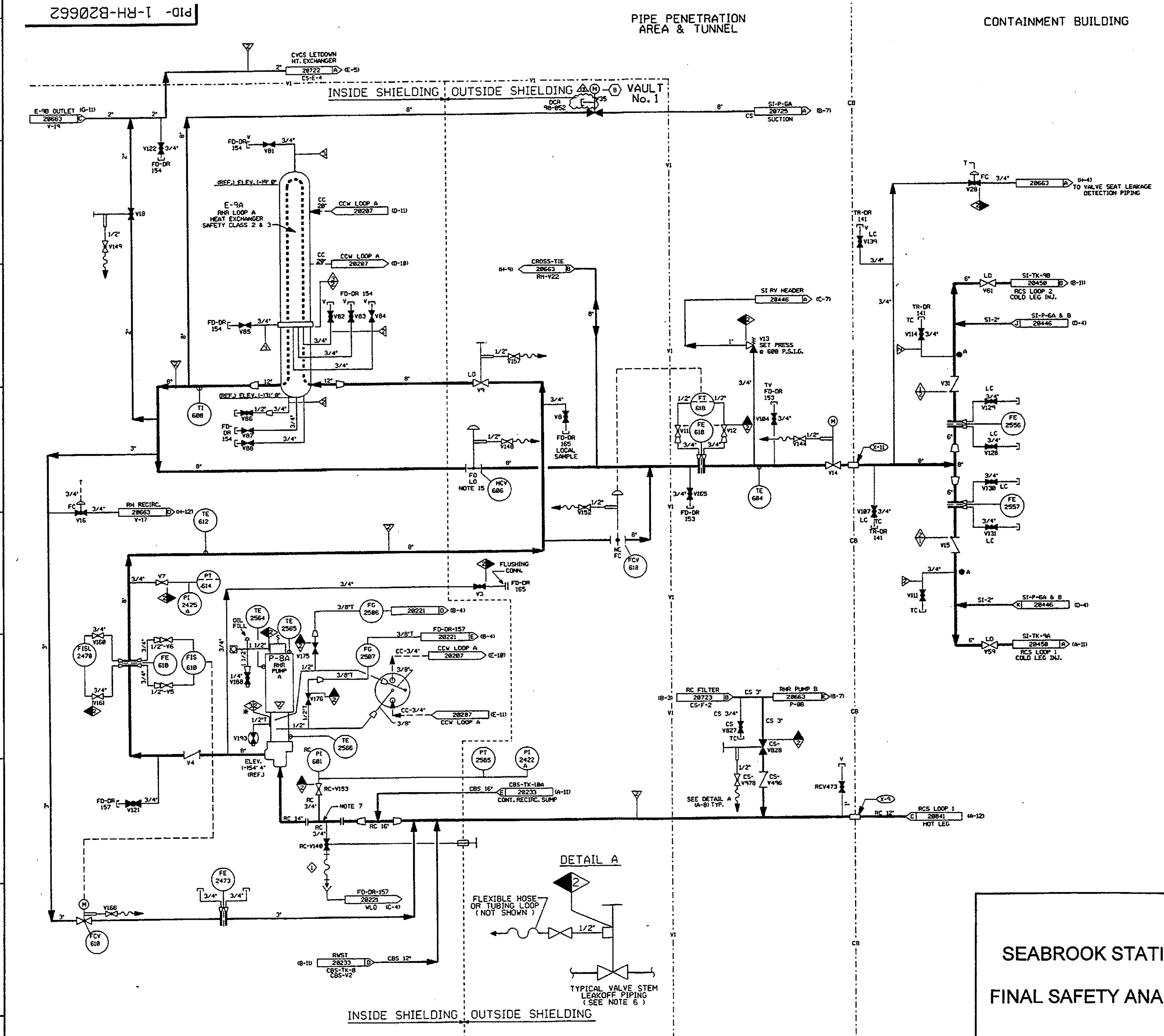
** As part of plant operation, periodic tests, surveillance inspection and instrument calibrations are made to monitor equipment and performance. Failures may be detected during such monitoring of equipment in addition to detection methods noted.

TABLE 5.4-17
(Sheet 2 of 6)

FAILURE MODE AND EFFECTS ANALYSIS - RESIDUAL HEAT REMOVAL SYSTEM
ACTIVE COMPONENTS - PLANT COOLDOWN OPERATION

Component	Failure Mode	*Effect on System Operation	**Failure Detection Method	Remarks
3. Residual heat removal pump RH-P-8A, (pump RH-P-8B analogous)	a. Fails to deliver working fluid.	a. Failure results in loss of reactor coolant flow from hot leg of RC loop #1 through Train A of RHRS. Fault reduces redundance of RHR coolant trains provided. No effect on safety for system operation. Plant cooldown requirements will be met by reactor coolant flow from hot leg of RC loop #4 flowing through Train B of RHRS; however, time required to reduce RCS temperature will be extended.	a. Open pump switchgear circuit breaker indication at CB; circuit breaker close position monitor light for group monitoring of components at CB; common breaker trip alarm at CB; RC pressure indication (PI-405) at CB; RHR Train A discharge flow indication (FI-618) (see item #11) at CB; and pump discharge pressure indication (PI-614) at CB.	1. The RHRS shares components with the ECCS. Pumps are tested as part of the ECCS testing program (see Subsection 6.3.4). Pump failure may also be detected during ECCS testing.
4. Motor-operated globe valve RH-FCV-610 (RH-FCV-611 analogous)	a. Fails to open on demand (open manual mode CB switch selection).	a. Failure blocks miniflow line to suction of RHR pump A during cooldown operation. No effect on safety for system operation. Plant cooldown requirements will be met by reactor coolant flow from hot leg of RC loop #4 flowing through Train B of RHRS; however, time required to reduce RCS temperature will be extended.	a. Valve position indication (closed to open position change) at CB.	1. Valve is automatically controlled to open when pump discharge is less than ~750 gpm and close when the discharge exceeds ~1400 gpm. The valve protects the pump from dead-heading during ECCS operation. CB switch set to "Auto" position for automatic control of valve positioning.

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

① FURTHER ENGINEERING EVALUATION TO EFFECT ECA 99/115181C.

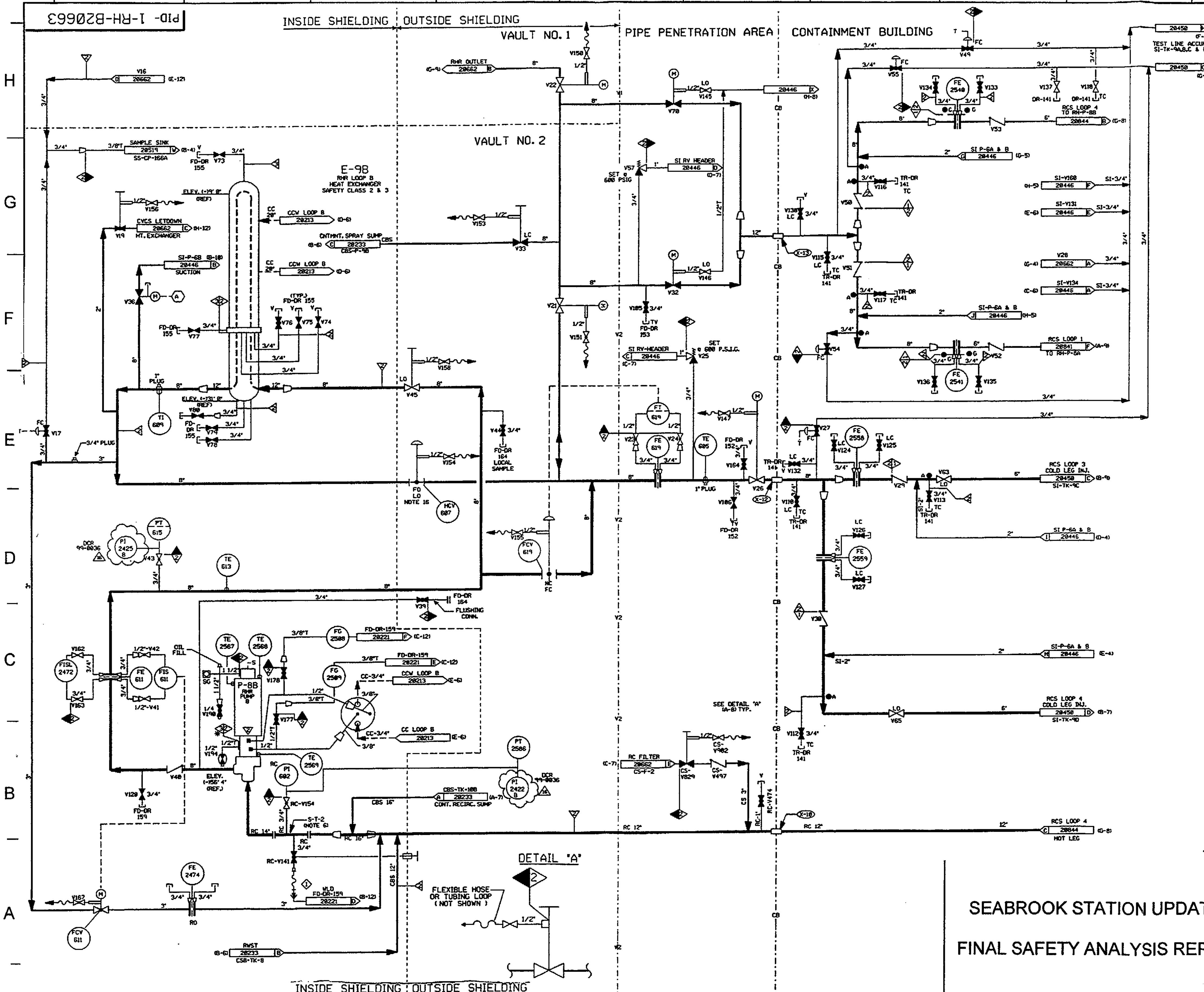
NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20660 & 20663. PORTIONS OF THIS DRAWING ARE DUPLICATED ON DRAWINGS SI-20448 & SI-20449 DUE TO DUAL SYSTEM FUNCTIONS.
2. ALL LINES, VALVES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX RH, UNLESS NOTED OTHERWISE.
3. DELETED.
4. ● INDICATES 3/8" FLOW RESTRICTOR INSTALLED. SEE APPLICABLE DETAIL ON DRAWING 1-RC-20840.
5. DELETED.
6. REFER TO DRAWING VSL-20774 FOR VALVE LEAKOFFS.
7. PERMANENT SPOOL PIECE INSTALLATION SHOWN TEMP. STRAINER IS USED IN PLACE OF SPOOL PIECE DURING INITIAL FLUSHING OPERATION. STRAINER TO BE REMOVED BEFORE START-UP.
8. INSTRUMENTATION:
   (B) INTERLOCKED WITH SI-V89, SI-V90, SI-V93,(DWG. 20446) CBS-V8,(DWG. 20233) RC-V22, & RC-V23,(DWG. 20841)
9. VENTS, DRAINS, TEST CONN. & ORIFICE TAP PIPING CODE BREAKS ARE AT THE DOWNSTREAM END OF THE OUTER ISOLATION VALVE PER 1-NHY-805111, UNLESS NOTED OTHERWISE.
10. DELETED.
11. DELETED.
12. △ - INDICATES REVISION LEVEL
13. DELETED
14. DELETED
15. HANDJACK HANDWHEEL LOCKED OPEN EXCEPT DURING MIDLOOP.

PORTIONS OF THIS DRAWING ARE
NUCLEAR SAFETY RELATED

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RESIDUAL HEAT REMOVAL SYS. TRAIN A DETAIL	
	1-RH-B20662	FIGURE 5.4-10

11/07/00 REV. 19



REV. 07

FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

◇ FURTHER ENGINEERING EVALUATION TO EFFECT ECA 99/115181C.

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20660 & 20662. PORTIONS OF THIS DRAWING ARE DUPLICATED ON DRAWING SI-20446, SI-20448 & SI-20449 DUE TO DUAL SYSTEM FUNCTIONS.
2. ALL LINES, VALVES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX RH, UNLESS NOTED OTHERWISE.
3. ● INDICATES 3/8" FLOW RESTRICTOR INSTALLED. SEE APPLICABLE DETAIL ON DRAWING RC-20848.
4. DELETED
5. REFER TO DRAWING VSL-20774 FOR VALVE LEAKOFFS.
6. PERMANENT SPOOL PIECE INSTALLATION SHOWN TEMP. STRAINER IS USED IN PLACE OF SPOOL PIECE DURING INITIAL FLUSHING OPERATION. STRAINER TO BE REMOVED BEFORE PLANT START-UP.
7. INSTRUMENTATION:
   Ⓐ INTERLOCKED WITH SI-V89, SI-V90, SI-V93 (DWG. 20446) CBS-V14 (DWG. 20233) RC-V87, & RC-V88 (DWG. 20844).
8. VENTS, DRAINS, TEST CONN. & ORIFICE TAP PIPING CODE BREAKS ARE AT THE DOWNSTREAM END OF THE OUTER ISOLATION VALVE PER 1-NHY-805111. UNLESS NOTED OTHERWISE.

9 - 11. DELETED

12. △ -INDICATES REVISION LEVEL.

13. DELETED

14. DELETED

15. ◇ INDICATES OPEN ITEM NUMBER.

16. HANDJACK HANDWHEEL LOCKED OPEN EXCEPT DURING MIDLOOP.

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RESIDUAL HEAT REMOVAL SYS. TRAIN B CROSS-TIE DETAIL
1-RH-B20663	FIGURE 5.4-11

Seabrook Station

North Atlantic

Updated Final Safety Analysis Report

Revision 7

CHAPTER 6

ENGINEERED SAFETY FEATURES

CONTENTS

CHAPTER 6

ENGINEERED SAFETY FEATURES

CONTENTS

CHAPTER 6

ENGINEERED SAFETY FEATURES

TABLES

CHAPTER 6

ENGINEERED SAFETY FEATURES

TABLES

CHAPTER 6

ENGINEERED SAFETY FEATURES

6.1(B) ENGINEERED SAFETY FEATURE MATERIALS

This section provides a discussion of the materials used in Engineered Safety Feature (ESF) components and the material interactions that potentially could impair operation of ESF.

6.1(B).1 Metallic Materials

6.1(B).1.1 Materials Selection and Fabrication

Typical material specifications applicable to components in the ESF not covered by Subsection 6.1(N).1 are listed in Table 6.1(B)-1. In some cases this list of materials may not be totally inclusive; however, the listed specifications are representative of those materials used. Materials utilized in ESF have been selected for their compatibility with core and containment spray solution, and conform with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Articles NC-2160 and NC-3120, plus all applicable addenda and code cases.

Typical specifications for principal pressure-retaining ferritic materials, austenitic stainless steels and nonferrous materials, including bolting and weld rod materials used in pressure-boundary welds in the Engineered Safety Features, are listed in Table 6.1(B)-1. These materials are qualified to the requirements of the ASME Code, Section III and Section IX, and are used in procedures which have been qualified to these same rules.

The following controls are placed on fabrication and assembly of austenitic stainless steel materials used in the ESF components to insure a high degree of quality and reliability. These controls assure compliance with Regulatory Guides 1.31, 1.37, and 1.44:

a. Significant sensitization of austenitic stainless steel is avoided by imposing the following controls:

    1. Use of low-heat input welding procedures and processes, as well as maximum interpass temperature control

    2. Use of fully annealed material heat-treated in accordance with specific parameters (e.g., water quenching)

    3. Prohibition of stress relief after welding

    4. Engineering review of welding and heat-treatment procedures.

b. Specific controls are imposed during fabrication and installation to preclude contamination of stainless steel by chlorides and low melting point constituents, particularly during welding and heat treatment. These controls are monitored by checking chemical analysis certifications of materials that contact stainless steel (such as tapes, marking crayons and cleaning solutions), and engineering review of final cleaning procedures.

c. Cold working of stainless steel is prohibited after solution annealing except in mild environments where residual stresses from bending or forming are minimal. Piping for the containment spray system spray headers undergoes moderate bending during fabrication; however, this will have no deleterious affects on system performance since internal pressure during system operation is low and the chemical environment mild. In no case is cold-worked stainless steel with a yield strength of 90,000 psi or greater used in ESF constituents.

d. Each heat or lot of filler material is required to be checked to assure the presence of 5 to 20 percent ferrite as calculated from the chemical composition and/or by a magnetic measuring check of a weld pad made with the subject filler material. Maximum interpass temperature control is also imposed during welding to minimize hot cracking.

The thermal-insulation used on ESF piping and equipment inside containment is fiberglass blanket insulation of the type commercially known as Nukon, with a stainless steel jacket over the outside surface of the insulation. Nukon is consistent with the recommendations of Regulatory Guide 1.36. Owens-Corning Fiberglass submitted Topical Reports OCF-1 on Nukon to the NRC for review in August 1977. The thermal insulation used on ESF piping and equipment outside containment shall be either fiberglass or calcium silicate molded sections with an aluminum jacket over the outside surface of the insulation.

6.1(B).1.2 Composition Compatibility and Stability of Containment Core Coolants

The pH of the coolants for the ESF during a loss of coolant is dependent upon the boron concentration of the Reactor Coolant System, the accumulators and the refueling water storage tank, and the concentration of sodium hydroxide in the spray additive tank. The concentrations of boron in the accumulators, and refueling water storage tank, and the concentration of sodium hydroxide in the spray additive tank are fixed and are verified periodically by analyzing samples of the solution in each tank and adjusting the chemical compositions if necessary. A recirculation/sample system is installed for the Spray Additive Tank to verify its concentration. The boron concentration in the Reactor Coolant System can vary from 0 to 4000 ppm depending upon the requirements for reactivity control. Depending on the various initial chemical compositions of the injected coolant and the Reactor Coolant System, the pH of the coolant can range from 8.8 to 9.5 at the beginning of the recirculation phase of ESF operation, and remains constant thereafter. The pH

(a) System Parameters and Initial Conditions

The system parameters and the initial conditions used in the pressure-temperature response analysis are presented in Tables 6.2-1 and 6.2.2. The parameters and the initial conditions are chosen to maximize the containment pressure and temperature responses unless their effects are insignificant.

(b) Actuation of Containment Sprays

The Containment Spray System is initiated by a containment spray actuation signal which is generated by the containment Hi-3 ("P" signal). The analysis limit for this setpoint is 19.8 psig. The maximum delay in signal processing and the response time of the protection system instrumentation is one second. The stroking time of the spray system valves is 20 seconds to become fully open with the exception of CBS-V38 and CBS-V43 which have a maximum stroke time of 25 seconds. The maximum delay time to bring the pumps to full speed and to fill the feed lines and headers is 38 seconds following the receipt of this signal. The valve opening and fill-up of the line takes place concurrently. Therefore, the maximum delay of 39 seconds after generation of the "P" signal consists of 1 second plus 38 seconds for fill-up. In the case of loss of offsite power concurrent with a coolant system pipe rupture, the emergency electric power from the onsite diesel generators will be available in 12 seconds. Receipt of the "P" signal by the actuation sequencer, which is discussed in detail in Subsection 8.3.1, will cause the spray valves to start opening immediately, or as soon as the emergency power is available. If the "P" signal is received within 27 seconds or between 27 and 52 seconds following an accident, the spray pumps will be started at 27 seconds or 52 seconds respectively. If the signal is received after 52 seconds, the pumps will be started immediately. Thus, for any "P" signal received before 27 seconds following an accident, which is true for all LOCA cases analyzed, the spray time is always at 65 seconds after the accident. For MSLB cases analyzed, the time to generate the "P" signal varies over a wide range. However, for all MSLB cases, a constant conservative spray delay time of 65 seconds after receipt of the signal has been assumed.

(c) Containment Passive Heat Sinks

The structure and equipment within the containment which have been modeled as passive heat sinks are listed in Table 6.2-3. Table 6.2-4 gives the thermophysical properties of the materials for the heat sinks. In Table 6.2-3, some of the heat sinks of similar structure have been lumped into various thickness groups. For internal heat sinks exposed to the containment on one side only, it is conservatively assumed that the other side is insulated. For those exposed on both sides and having a plane of symmetry, only half of the heat sink is modeled but the effective surface area is doubled. For the steel-lined structures, the steel-concrete interface resistance has been modeled as an equivalent air gap of $\frac{1}{16}$ inch. This is conservative when compared with experimental data in References 2 and 3. The interface conductance therein varies from 100,000 Btu/hr-ft^2-°F for a very good contact to 10 Btu/hr-ft^2-°F for a very poor contact. The lower limit corresponds to the equivalent conductance of an air gap approximately 20 mils thick.

The heat transfer between the containment and the passive heat sinks is calculated in the CONTRAST-S code by combining the contribution from the condensation and convection. An effective heat transfer coefficient, h_{eff}, based on the temperature difference between the containment atmosphere and the heat sink surface can be defined as follows:

$$h_{eff} = f(h_{cond}-h_{conv})(T_{sat}-T_{wall})/(T_{con}-T_{wall}) + h_{conv}$$

where,

h_{cond} = condensing heat transfer coefficient

h_{conv} = convective heat transfer coefficient

T_{sat} = containment atmosphere dew point

T_{wall} = surface temperature of heat sink

T_{con} = containment atmosphere temperature

f = 1, if $T_{sat} > T_{wall}$, 0 otherwise.

In the estimation of the condensing heat transfer coefficient, the modified Tagami correlation (Reference 4) is used for the LOCA analysis while the Uchida correlation (Reference 5) is used for the MSLB to maintain the containment pressure and temperature within the design envelope for worst case primary and secondary side ruptures.

2. Spray Additive Tank

The spray additive tank (SAT) is mounted adjacent to the RWST, and drains by gravity into the RWST mixing chamber through a six inch diameter pipe which has redundant valving. This line connects the bottoms of the SAT with the RWST mixing chamber. External heaters are provided to prevent freezing or chemical precipitation during cold weather. The mixing ratio of the spray additive tank volume to the RWST volume is such that the pH of the spray solution during the injection phase will average between 8.8 and 9.2 units. The tank is sized to provide the correct amount of sodium hydroxide solution to insure that the final containment recirculation sump pH after injection will be between 8.5 and 11.0 units for the various reactor coolant conditions. No provision is made in the design of the SAT to prevent the reaction of NaOH with atmospheric carbon dioxide during long-term storage.

Proper concentration of sodium hydroxide between 19 and 21% by weight will be verified periodically by chemical analysis.

3. Containment Spray Heat Exchangers

The containment spray heat exchangers are shell and tube-type heat exchangers with spray flow in the tube side and primary component cooling water (PCCW) on the shell side. They are sized such that one containment spray heat exchanger and one residual heat removal heat exchanger provide 100 percent of design heat removal capacity.

Heat exchanger parameters, including flow rates, were selected so that one RHR heat exchanger and one CBS heat exchanger satisfy containment cooling requirements. Table 6.2-76 contains the heat exchanger performance data used for the accident analyses.

4. Spray Headers and Nozzles

The spray headers are positioned in the containment dome to maximize coverage of the containment volume. Four separate headers are used to obtain the distribution of the flow, two for each train. Each train contains 198 nozzles with each nozzle providing a design flow of 15.2 gpm (see Figures 6.2-76 and 6.2-77).

5. Refueling Water Storage Tank

The refueling water storage tank (RWST) is designed to store 475,000 gallons of borated water. This tank is designed to supply water both for refueling operations and to the Containment Spray System and the Emergency Core Cooling System during accident operations. The RWST capacity is based on accident requirements and will supply the safety injection, the charging, residual heat removal and containment spray pumps for at least 21.9 minutes during the injection phase of a design base accident.

Margin is provided to allow time for transfer of the systems to the recirculation mode and to account for instrument errors. Analysis is based on a minimum of 350,000 gallons of water being injected. An external steam heating supply system is provided to protect against freezing. Tank temperature is indicated locally and alarmed in the main control room.

c. Material Compatibility

The components of the CBS system, including the spray nozzles, are fabricated of materials listed in Table 6.2-75.

The pH of the sump water following an accident is monitored to ensure that the pH is maintained in the correct range (discussed in part f. of this section) as the hydroxide is consumed by chemical reaction with zinc and aluminum within the containment. Two sample points exist to withdraw samples downstream of the RHR heat exchangers: the normal connection to the sample sink, and a local sample point. Sodium hydroxide can be added for pH adjustment using the chemical and volume control system tanks and pumps. The solution is prepared in the chemical mixing tanks and supplied to the suction of the charging pumps. The charging pump suction is fed from the RHR system during recirculation.

Neither the containment spray pumps or motors nor other engineered safeguard pumps or motors are exposed to the containment atmosphere; accordingly, no adverse effects are considered due to the post-accident containment environment.

j. Containment Recirculation Sump and Screen Design

The containment recirculation sump collects and screens the water available for supplying the residual heat removal, containment spray, safety injection and high head charging pumps during the recirculation mode of operation following an accident. The sump is designed to meet the intent of Regulatory Guide 1.82. Two completely independent sumps are located in the containment to maintain the "double train" concept as described in Subsection 6.2.2.2d.

One sump supplies water to Train A and the other sump supplies Train B. The arrangement of these sumps is shown in Figure 6.2-79. The minimum water level in containment during a loss-of-coolant accident is Elevation (-)23.79 ft.

Heavy particles are prevented from reaching the sumps by sloping the surrounding floor away from the sumps. This facilitates settling of debris on the floor prior to reaching the sump area. A vertical trash rack with 1"x3 11/16 inch openings is provided to protect the fine inner screen from large floating particles.

The fine (8x8 openings per inch) inner screens consist of framed panels securely attached to a structural steel frame. These prevent particles 0.097 inches and greater in diameter from passing through or bypassing the screens and entering the sumps. The screen has sufficient area to accommodate 50 percent blockage and still limit the maximum approach velocity to approximately 0.2 ft/sec. With an approach velocity of 0.2 ft/sec, all debris with a specific gravity of 1.05 or more will settle to the floor prior to reaching the sumps. The anti-vortexing criteria that were used in the sump design are discussed in References 24, 25 and 26.

The minimum physical restriction in the flow path consists of 0.075 inches, which is the effective opening of the fuel assembly debris filter bottom nozzle in combination with the P-grid. Although this opening is smaller than the sump screen mesh, the design does not represent a challenge to long-term core cooling. Debris particles only in the range of 0.075 to 0.097 inches in diameter could potentially block flow path openings into the core. There are approximately 50,000 to 60,000 flow channels in the core. The P-grid intersects each flow hole resulting in approximately 200,000 to 240,000 0.075 inch parallel openings into the core, to promote long-term cooling. Following a large break LOCA, it is only necessary to maintain the core water level above the top of the active fuel. Long-term cooling is not challenged, therefore, until most of these openings become blocked. For the small range of particles involved (0.075 to 0.097 inches), it is not likely that sufficient debris would be generated in this range to block a significant number of openings. Furthermore, due to the low flow rates under post accident conditions, debris will tend to settle out into the lower plenum of the core and not block the flow openings. The type of debris generated following a large break LOCA includes insulation, paint chips, metal and cement fragments, etc. In view of the high density of the metallic and cement debris, these particles will settle out prior to reaching the core. For the lighter material (insulation and paint chips), it is inconceivable that sufficient amounts of debris falling in the range of 0.075 to 0.097 inches could be generated in sufficient quantity to block a significant portion of the core given the number of parallel flow channels. Therefore, the debris filter bottom nozzle/P-grid design will not add significant ECCS flow path resistance and compromise long-term core cooling following a large break LOCA. Therefore, the design meets the intent of Regulatory Guide 1.82.

The potential for clogging of the sump screens by equipment and piping insulation or loose insulation in the containment is minimized by the type of insulation used.

The thermal insulation inside the containment for all the piping and equipment except the reactor pressure vessel will be fiberglass blanket insulation of the type commercially known as Nukon, with a stainless-steel jacket over the outside surface of the insulation. Nukon is consistent with the recommendations of Regulatory Guide 1.36. The reactor pressure vessel is insulated with stainless-steel reflective insulation.

d. Net Positive Suction Head Available

Adequate net positive suction head (NPSH) for the containment spray pumps is assured under all postulated operating conditions by analysis of the suction head available and vendor testing of the completed pumps.

Figure 6.2-78 shows NPSH available versus NPSH required over the range of flow. Maximum calculated flow under the most limiting NPSH conditions, i.e. during recirculation, is 3270 gpm. NPSH available at this flow is 21.04 feet versus a maximum required NPSH of 20.5 feet. The CBS pump analysis of available NPSH conservatively assumes that each residual heat removal pump (which shares a common suction on a train basis with each CBS pump) is also operating at design cold leg recirculation flow of 4700 gpm and considers the suction flow path with the highest hydraulic resistance. The formulas and flow resistance data in Reference 21 were used along with the test data for the bell-mouth sump suction piping, to compute NPSH available.

Table 6.2-78 lists the values of containment pressure head, vapor pressure head of pumped fluid, suction head, and friction head used in the analysis.

e. Integrated Energy, Content of the Containment Atmosphere and Recirculation Water

Figures 6.2-87 and 6.2-88 show the integrated energy content of the containment atmosphere and recirculation water, respectively, as functions of time following the postulated design basis loss-of-coolant accident. The integrated energy absorbed by the structural heat sinks and removed by the containment heat removal heat exchangers is shown in Figures 6.2-89 and 6.2-90, respectively.

f. Debris

The major source of debris that could be generated during a loss-of-coolant accident is insulation. The thermal insulation being used inside the containment will be both stainless steel reflective insulation and fiberglass insulation of the type commercially known as Nukon, with a stainless steel jacket over the outside surface of the insulation. Nukon is consistent with the recommendations of Regulatory Guide 1.36.

6.2.2.4 Testing and Inspection

The preoperational testing of the containment heat removal system verified the functional capability of the individual systems under operational conditions.

Testing and inspection of the ECCS systems is discussed in Section 6.3 and Chapter 14.

The preoperational testing of the Containment Spray System verified the operational parameters of the spray pumps during recirculation to the RWST. This testing included a demonstration of system response to ESF signals and the ability of the sump to supply the containment spray and residual heat removal pumps. Flow testing of the nozzles was performed by the manufacturer and was not performed in the field. An air flow test was performed to verify that no nozzles are plugged.

Operability of the gravity feed system was demonstrated during preoperational testing of the ECCS Performance Test (Table 14.2-3, Item 8). The preoperational test will demonstrate the draw-down characteristics of the RWST and SAT during the different flow conditions of the ECCS Performance Test.

The Containment Spray System will be inspected and tested periodically in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI and the requirements of the Technical Specifications.

The containment recirculation sumps will be visually inspected periodically to insure that they are free of debris and all screens are intact.

6.2.2.5 Instrumentation

The Containment Heat Removal System is provided with instruments and controls to allow the operator to monitor the status and operation of the spray system and to allow the automatic or manual initiation of the injection and recirculation modes of operation.

The manual spray actuation consists of four momentary controls (see Figure 7.2-1, Sheet 8). Actuation occurs only if two associated controls are operated simultaneously. This prevents inadvertent spray initiation as a result of operator error. The automatic initiation is by coincidence of 2 out of 4 protection set loops, monitoring the containment pressure. The spray actuation signal starts the containment spray pumps and positions all valves to their operating configuration. The design details of the Engineered Safety Features Actuation System are presented in Section 7.3.

The details of the interlocks involved in the suction valve realignment from the RWST to the containment sump during the switchover from injection to recirculation mode are presented in Table 6.3-7. RWST instrumentation is discussed in Subsection 6.3.5. Indications of pump operation are provided by pump status indication lamps and the pressure indications at the main control room. Alignment of automatic valves is indicated by the valve status indications. Additionally, a separate status monitoring indication system is provided at the control room for both modes of the spray system. This enables the operator to evaluate the extent to which the valves are open and if the system is operating effectively. Alarms are also provided to indicate that either train of the Containment Spray System is inoperative. The design vent and drain (TVD) valves associated with penetrations are not specified in Table 6.2-83, but are shown on Figure 6.2-91 (Sheets 1 through 11). All TVD valves located between the containment isolation valves are identified. For the remaining TVD valves (located outside the containment isolation valves), only the test valves are schematically shown as arrows to identify all containment isolation valves, demonstrate the ability to perform the Type C test (if required), and yet provide a clear, unobstructed schematic representation of containment penetrations. For further clarity, alphabetic suffixes were added to the individual lines of the multiple-line penetrations. These suffixes do not appear in other design documents.

c. <u>Valve Actuation Signals</u>

The design of the system providing the signals for containment isolation complies with the following general requirements:

1. The containment isolation signal overrides all signals for actuations of containment isolation valves for nonessential systems.

2. Phased isolation is used. With phased isolation, all systems except non-engineered safety features and engineered safety features-related systems are automatically isolated. Only those engineered safety feature-related systems that can be justified to remain operational shall not be automatically isolated during the initial phase.

3. Diverse parameters are used wherever possible for developing isolation signals.

4. Concurrent containment isolation occurs coincident with initiation of emergency core cooling.

5. All valves that receive a containment isolation signal cannot be reopened until the isolation signal is reset and manual action is taken to reopen the valve. The controls are separated so that only one valve, or group of valves associated with a penetration, open for each manual action.

Automatically tripped isolation valves are actuated to the closed position by one of two separate containment isolation signals. The first of these signals ("T" Signal) is derived in conjunction with automatic safety injection actuation on Hi-1 containment pressure, and trips the majority of the automatic isolation valves. The Hi-1 setpoint is the lowest practical and includes margin for containment pressurization, instrument error and operating margin. These are valves in the nonessential process lines which do not increase the potential for damage to in-containment equipment when isolated.

This is defined as "phase A" isolation, and the valves are designated by the letter "T" in the isolation diagrams of Figure 6.2-91. The second, or "phase B," containment isolation signal ("P" Signal) is derived from Hi-3 containment pressure and/or actuation of the Containment Spray System, and trips the automatic isolation valves in the other process lines (which do not include safety injection lines) penetrating the containment. These isolation valves are designated by the letter "P" in the isolation diagrams.

Containment air purge (CAP) and containment online purge (COP) system lines which provide an open path from the containment environs are equipped with radiation monitors that are capable of isolating these lines upon receipt of a high radiation signal, in addition to automatic safety injection actuation, manual containment spray actuation and manual phase "A" isolation signals. Further discussion of containment isolation signals is found in Section 7.3. (Refer to Figure 7.2-8.)

d. <u>Valve Closure Time</u>

The objective in establishing valve closure time is to limit the release of radioactivity from the containment to as-low-as-is-reasonably-achievable. Consideration is given to the fluid system requirements (e.g., water hammer) in determining the valve closure time, the effect of closure time on valve reliability, as well as the containment isolation requirements.

These considerations have been addressed in the design of the containment isolation system, within the context and requirements of the guidelines and applicable criteria presented in Subsection 6.2.4.1, Design Bases.

Isolation valve closure times for the Containment Isolation System are presented in Table 6.2-83. The valves listed there reflect the maximum time required to isolate a system so that radioactive release to the environs during a design basis accident is within limits in 10 CFR 100. Refer to Subsection 9.4.5 for discussion of containment online purge line isolation.

e. <u>Operability of Valves Inside Containment</u>

Isolation valves located inside containment are subject to the high pressure, high temperature, steam-laden atmosphere resulting from an accident. Operability of these valves in the accident environment is ensured by proper design, construction and installation, as reflected by the following considerations:

1. All components in the valve installation, including valve bodies, trim and moving parts, actuators, instrument air control and power wiring are constructed of materials

sufficiently temperature and humidity resistant to be unaffected by the accident environment. Special attention is given to electrical insulation, air operator diaphragms and stem packing material. Section 3.11 discusses the qualification of this equipment for operation in the containment atmosphere during an accident condition.

2. In addition to normal pressures, the valves are designed to withstand maximum pressure differentials in the reverse direction imposed by the accident conditions. Additionally, a review was performed to ensure that the containment isolation valves inside the containment were not overpressurized due to trapped fluid in adjacent piping in the post-LOCA environment.

3. The containment structure online purge subsystem is designed to prevent debris from entering the exhaust and supply lines to ensure the operability of the isolation valves. This is accomplished by debris screens installed in the ends of the lines. Each debris screen consists of heavy-bar stainless steel grating, banded and welded to the exhaust and supply ends of the lines. Both the exhaust and inlet piping have two 90° bends and a minimum of 14 feet. This design greatly reduces the possibility of direct impingement of debris on the valves. The pipe, screens and supports are seismic Category I. The screens will be capable of withstanding the differential pressure resulting from a LOCA up to the point of containment isolation.

Operability of valves and their operators within containment atmosphere is addressed by qualifying this equipment to IEEE Standard 382-1972, Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations, and NRC Regulatory Guide 1.73, Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants. Updated FSAR Section 3.9 provides additional information on valve operability.

Containment isolation valve operators have been provided to ensure adequate reliability for the operation of the valves. Lines penetrating the containment that serve the engineered safety features as well as their associated support systems are provided with motor operators. To verify that motor operators have sufficient torque to operate the valves, the valves are tested by opening and closing under full system pressure. Operability during and after a LOCA is thus assured.

5. Seismic activity

A detailed discussion of these tests is given in Reference 22.

During preoperational testing, the functional operability of the recombiner and its control will be demonstrated. Periodic testing of the recombiners will be performed in accordance with Technical Specification requirements.

c. Backup Purge System

The containment purge function of the CGCS will be demonstrated during preoperational testing. Initial and periodic tests of the containment enclosure emergency exhaust filters is described in Subsection 6.5.1.

d. Containment Structure Recirculation Filter System

The operability of containment atmospheric mixing fans and dampers will be verified during preoperational testing. Periodic tests of the fans and dampers will be conducted in accordance with Technical Specification requirements.

6.2.5.5 Instrumentation Requirements

With the occurrence of a LOCA, the ESF actuation signal will start the Containment Recirculation Fan System. Two independent hydrogen analyzers which monitor containment hydrogen concentrations after an accident, are located outside the containment. Off-normal conditions, such as low temperature, low sample flow and pressure, and cell failure are alarmed at the control room on a system level as a common alarm, and individually indicated at the local panel. The output of either channel of the hydrogen analyzers is available to the operator, both locally and at the MCB. When either channel indicates a hydrogen concentration at or prior to 0.5 volume percent below the limiting hydrogen concentration, this fact is alarmed at the main control board (MCB). The operator would then start the recombiners, if he had not done so already.

The recombiners are thermal electric types. Temperature sensors located at the recombiner input to the MCB temperature indication to maintain the recombiner temperature at an optimum value for efficient recombination of H_2 and O_2 into water. The power input to the recombiner is manually set from the MCB. Recombiner temperature and power input are indicated at the MCB. The temperature indication is used for equipment tests and periodic checkouts and is not required for the safety-related function of the hydrogen recombiners.

In the event that neither recombiner starts, the hydrogen concentration in the containment would continue to rise. When it has risen significantly above the first alarm point but is still below the limiting hydrogen concentration, this fact would be alarmed at the MCB as a signal to the operator to initiate the purging of the containment.

The Containment Purge System is normally closed, and is isolated from the containment by four valves: CGC-V14 and CGC-V28 inside the containment are remotely controlled, motor-operated valves that close on a "T" signal; valves CGC-V15 and CGC-V36 located outside the containment are manually operated and normally locked closed. Initiating purge flow is a manually controlled operation, with a combination of both local and remote control from the MCB. Valve status for the remotely operated valves CGC-V14 and CGC-V28 is provided near the associated control switches. Pressure and flow instruments are provided in the purge line. Purge flow indication is available at the MCB. Air for the purge system is supplied from the Compressed Air System.

6.2.6 Containment Leak Rate Testing

The reactor containment structure, the containment penetrations and the containment isolation barriers are designed to permit periodic Type A integrated leakage rate testing. The reactor containment and its leakage limiting barriers are also designed to permit periodic inspection of important areas such as penetrations. Penetrations with resilient seals or expansion bellows are designed to permit periodic leakage testing at pressures up to the containment design pressure. Piping systems penetrating the reactor containment are provided with the capability of redundant isolation, as dictated by their importance to safety functions. These systems are designed so that their isolation capabilities can be periodically tested for operability and leakage to ensure compliance with the established leakage rate limits. The foregoing are intended to be in full compliance with General Design Criteria 52, 53 and 54 of 10 CFR 50, Appendix A.

All portions of the above systems (RHR, CBS, SI and CS) are located within the containment enclosure boundary except piping associated with the injection phase of ECCS and a minor amount of charging pump piping used during the recirculation mode. Any leakage from these systems following a LOCA is therefore filtered by the containment enclosure emergency exhaust filters prior to release to the environment.

The piping that lies outside the containment enclosure boundaries includes the pump suction lines from the refueling water storage tank to the RHR, SI and CBS pumps (lines 1201-1-151-14" and 1202-1-151-14", see Figure 6.2-74) and the centrifugal charging pumps (lines 1205-1-151-8" and 1206-1-151-8", including valves LCV-112D and LCV-112E). These lines are used only during the injection phase of post-accident operation. They are isolated within the containment enclosure, will not be contaminated during recirculation, and will not present a release path. A portion of line CS-374-1-2501-4" is run outside the containment enclosure. This line is used during the recirculation phase of

TABLE 6.2-87

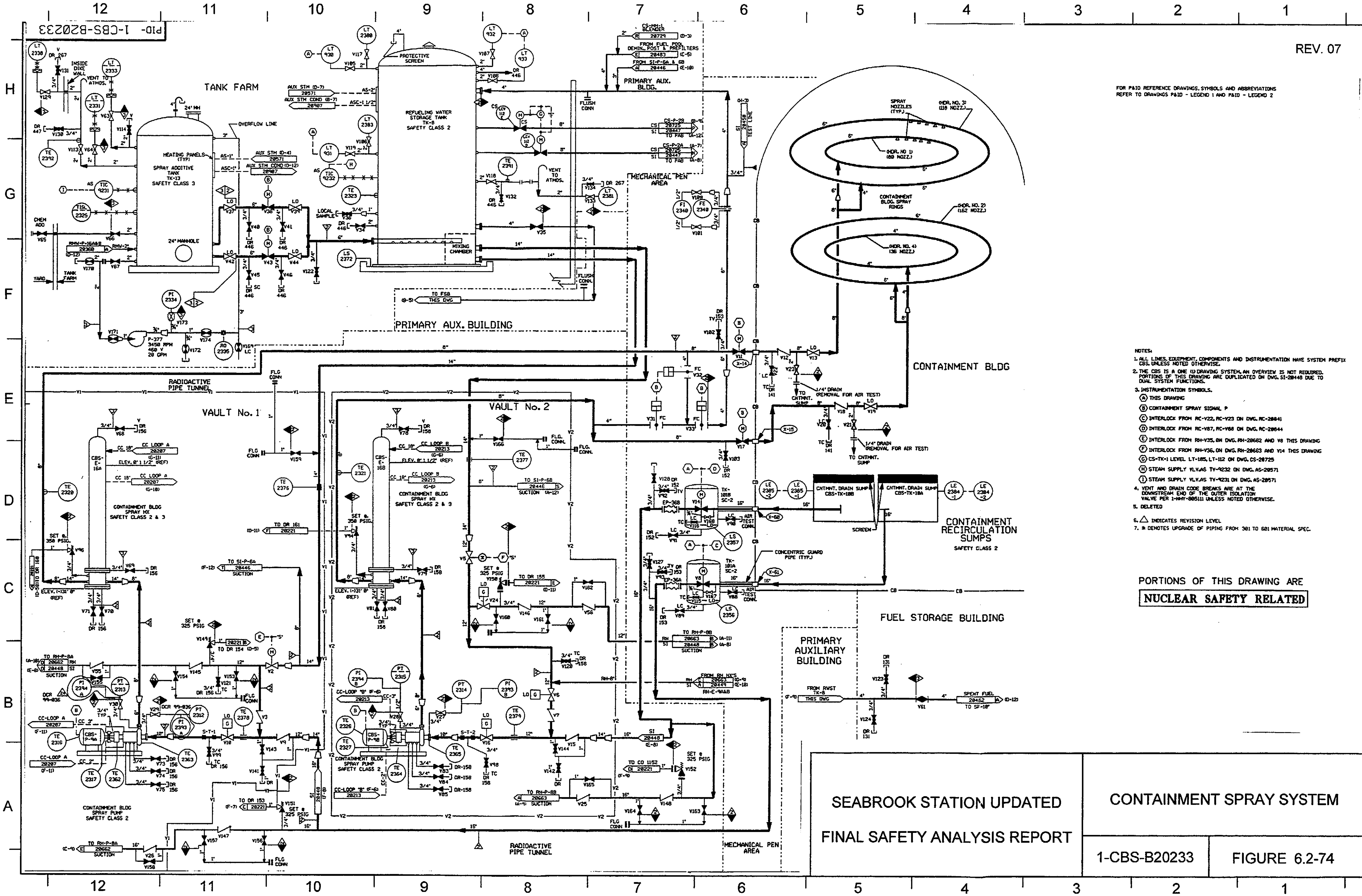
CONTAINMENT BUILDING ALUMINUM INVENTORY

Item	Exposed Surface (ft^2)
Jib Crane	3
Nuclear Instrumentation	83
Flux Map Drive System	88
Rod Position Indicator Connectors	81
CRDM Connectors	69
Miscellaneous Valves	86
Containment Elevator Fans	127.9
Excore Detector Supports	28.6
CRD Fans, Blades and Hubs	19.1
Polar Gantry Crane	28
I&C Transmitters & Misc.	29.8
Electrical Fixtures	27.6
Refueling Machine and Fuel Transfer System	26
Test Pump	3
Refueling Machine and Transfer System	26
Signs (HP, etc.)	8
Pressurizer Spray Valve Booster	3
Gaitronics System	140
Work Control Allocation	500
Contingency	449
TOTAL	1800.0

TABLE 6.2-88

CONTAINMENT BUILDING ZINC INVENTORY

Item	Exposed Surface (ft^2)
Ductwork, Angles and Supports	48,357
Decking	8,055
Grating	24,395
Cable Trays	19,295
Tray Supports	86,190
Conduit	23,387
Conduit Supports	98,256
Instrument Tube Trays and Supports	21,040
Electrical Box Supports	8,416
Wire Mesh Doors	1,212
Scaffolding Components	3,400
Wire Mesh Tool Crib	648
Misc Sheet Metal and Structural Members	700
Refueling Machine and Fuel Transfer System	56
Work Control Allocation	6,000
Contingency	5,593
TOTAL	355,000



10/04/00 REV. 24

(c) The experience derived from the check valves employed in the emergency injection systems indicates that the system is reliable and workable; check valve leakage has not been a problem. This is substantiated by the satisfactory experience obtained from operation of the Robert Emmett Ginna and subsequent plants where the usage of check valves is identical to Seabrook.

(d) The accumulators can accept some in-leakage from the RCS without affecting availability. Continuous in-leakage would require, however, that the accumulator volume be adjusted periodically to Technical Specification requirements.

4. Relief Valves

Relief valves are installed in various sections of the ECCS to protect lines which have a lower design pressure than the RCS. The valve stem and spring adjustment assembly are isolated from the system fluids by a bellows seal between the valve disc and spindle (except on the relief valves which discharge N_2 gas). The closed bonnet provides an additional barrier for enclosure of the relief valves. Table 6.3-2 lists the system's relief valves with their capacities and setpoints.!!!UFCR 99-55r0!!!

5. Butterfly Valves

Each main residual heat removal line has an air-operated butterfly valve (RH-HCV-606 and 607) which is normally open and is designed to fail in the open position. The actuator is arranged so that air pressure on the diaphragm overcomes the spring force, causing the linkage to move the butterfly to the closed position. Upon loss of air pressure, the spring returns the butterfly to the open position. These valves are left in the full-open position during normal operation to maximize flow from this system to the RCS during the injection mode of the ECCS operation. These valves are used during normal residual heat removal system (RHRS) operation to control cooldown flow rate.

Each residual heat removal heat exchanger bypass line has an air-operated butterfly valve which is normally closed and is designed to fail closed. Those valves (RH-FCV-618 and 619) are used during normal cooldown to avoid thermal shock to the residual heat exchanger.

6. Accumulator Motor-Operated Valve Controls

As part of the plant shutdown administrative procedures, the operator is required to close these valves. This prevents a loss of accumulator water inventory to the RCS, and is performed shortly after the RCS has been depressurized below the safety injection unblock setpoint. The redundant pressure and level alarms on each accumulator would remind the operator to close these valves, if any were inadvertently left open. Power is disconnected after the valves are closed.

During plant startup, the operator is instructed via procedures to energize and open these valves when the RCS pressure reaches the safety injection unblock setpoint. Monitor lights in conjunction with an audible alarm will alert the operator should any of these valves be left inadvertently closed once the RCS pressure increases beyond the safety injection unblock setpoint (see Dwg. NHY-503907). Once open, power is disconnected and remains off except during valve testing.

The accumulator isolation valves are not required to move during power operation or in a post-accident situation, except for valve testing. For a discussion of limiting conditions for operation and surveillance requirements of these valves, refer to the Technical Specifications.

For further discussion of the instrumentation associated with these valves, refer to Subsections 6.3.5, 7.3.1b and 7.6.4.

7. Motor-Operated Valves and Controls

Remotely operated valves for the injection mode which are under manual control (i.e., valves which normally are in their ready position and do not require a safety injection signal) have their position indicated on the control board. If a component is out of its proper position, its monitor light will indicate this on the control panel. At any time during operation, when one of these valves is not in the ready position for injection, this condition is shown visually on the board, and an audible alarm is sounded in the control room on a system basis, as part of the Bypass and Inoperable Status alarms.

The ECCS delivery lag times are given in Chapter 15. The accumulator injection time varies as the size of the assumed break varies since the RCS pressure drop will vary proportionately to the break size.

Inadvertent mis-positioning of a motor-operated valve due to a malfunction in the control circuitry in conjunction with an

TABLE 6.3-2

EMERGENCY CORE COOLING SYSTEM RELIEF VALVE DATA

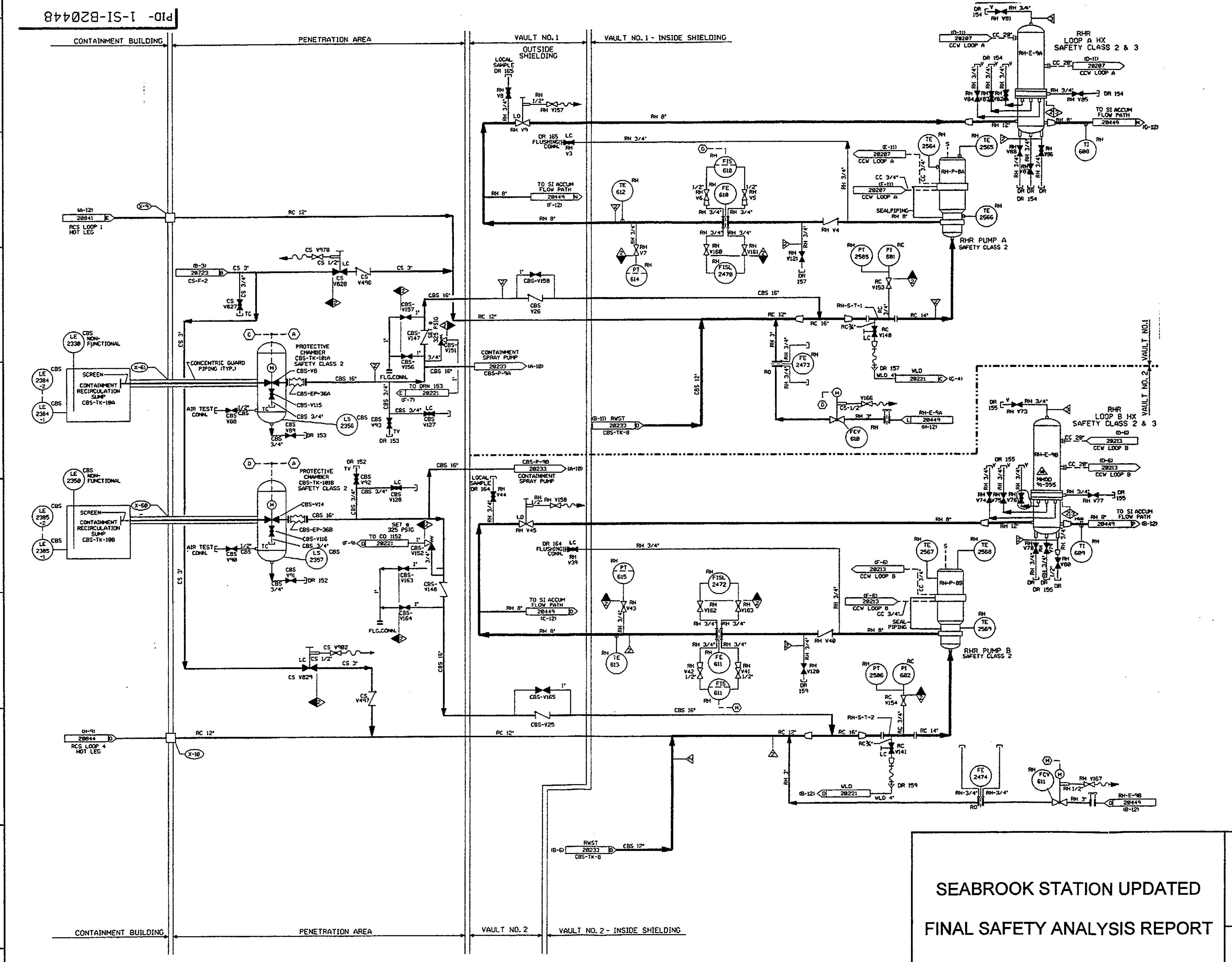
Description	Fluid Discharge	Fluid Inlet Temperature Normal (°F)	Set Pressure (psig)	Backpressure Constant (psig)	Maximum Total Backpressure (psig)	Capacity
Outside Containment N_2 supply to accumulators	N_2 gas	120	700	0	0	1500 scfm
Safety injection pump discharge	Water	120	1750	0 to 15	50	20 gpm
Residual heat removal pump safety injection line	Water	120	600	0 to 15	50	20 gpm
Safety injection pumps suction header	Water	100	220	0 to 15	50	25 gpm
Accumulator to containment	N_2 gas	120	700	0	0	1500 scfm
Inside Containment N_2 supply to accumulators (NNS-B31.1 Portion)	N_2 gas	120	800	0	50	1184 scfm

TABLE 6.3-3
(Sheet 1 of 2)

MOTOR-OPERATED ISOLATION VALVES IN THE EMERGENCY CORE COOLING SYSTEM

Location	Valve Identification	Interlocks	Automatic Features	Position Indication	Alarms
Accumulator isolation valves (See Dwg. NHY-503907)	SI-V3, -V17, -V32, -V47	"S" signal, RCS pressure > SI unblock pressure	Opens on "S" signal closed, MCC power available, and RCS pressure > SI unblock pressure	MCB	Yes-out of position
Safety injection pump suction from RWST	CBS-V47, -V51 CBS-V49, -V53	None	None	MCB	Yes-out of position
RHR suction from RWST (See Dwg. NHY-503255)	CBS-V2, -V5	Cannot be opened manually by control switch unless sump valve closed and RHR recir. valve closed	Opens on "S" signal	MCB	Yes-out of position
RHR discharge to safety injection/charging	RH-V35, -V36	Cannot be opened unless safety injection pump miniflow isolated and RHR suction valve from RCS closed	None	MCB	Yes-out of position*
Safety injection hot leg injection (See Dwg. NHY-503909)	SI-V77, -V102	None	None	MCB	Yes-out of position*
RHR hot leg injection (See Dwg. NHY-503769)	RH-V32, -V70	None	None	MCB	Yes-out of position*
Containment sump isolation valve (See Dwg. NHY-503252)	CBS-V8, -V14	Cannot be opened in normal operation unless RHR suction valves from RCS closed	Opens on RWST low-low-1 with "S" signal	MCB	Yes-out of position*
CVCS suction from RWST (See Dwg. NHY-503335)	LCV-112 D&E	"S" signal and CVC tank low level	Opens on "S" signal and CVS tank low level	MCB	Yes-out of position
CVCS normal suction (See Dwg. NHY-503341)	LCV-112 B&C	"S" signal and low level in CVC tank	Closes on "S" signal or low level in CVC tank if CVCS suction valves from RWST open	MCB	Yes-out of position
Safety injection pump to cold leg (See Dwg. NHY-503909)	SI-V114	None	None	MCB	Yes-out of position*

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20445 THRU 20450.
   PORTIONS OF RH & CBS SYSTEM DUPLICATED ON THIS DRAWING ARE DETAILED ON RH-20662, RH-20663 & CBS-20233 DUE TO DUAL SYSTEM FUNCTIONS.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX SI, UNLESS NOTED OTHERWISE.
3. DELETED.
4. DELETED.
5. DELETED.
6. INSTRUMENTS.
   A INTERLOCK FROM CBS-LT-930, LT-931, LT-932 & LT-933 ON DWG. 20233.
   C INTERLOCK FROM RC-V22 & RC-V23.
   D INTERLOCK FROM RC-V87 & RC-V88.
   G INTERLOCK BETWEEN RH-FIS-611 AND RH-FCV-611, THIS DRAWING.
   H INTERLOCK BETWEEN RH-FIS-611 AND RH-FCV-611, THIS DRAWING.
7. DELETED
8. DELETED.
9. DELETED.
10. DELETED.
11. △ INDICATES REVISION LEVEL.

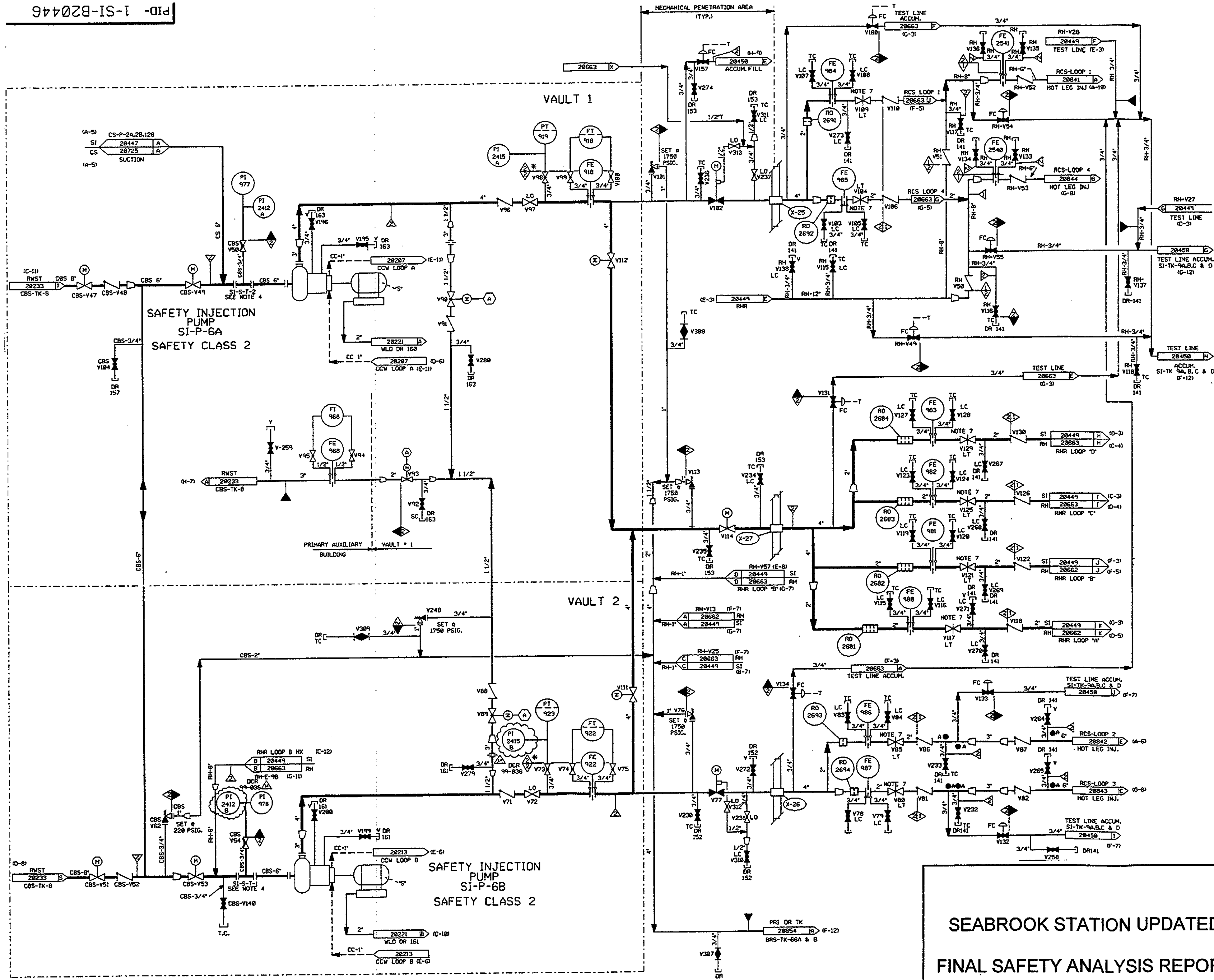
PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	SAFETY INJECTION SYSTEM LOW HEAD INJECTION DETAIL (SHT. 1 OF 2)
	1-SI-B20448 / FIGURE 6.3-2 SH 1

PID- 1-SI-B20446

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20445, 20447 THRU 20250. PORTIONS OF RH SYSTEM DUPLICATED ON THIS DRAWING ARE DETAILED ON RH-20663 DUE TO DUAL SYSTEM FUNCTIONS.
2. ALL PIPES, VALVES, EQUIPMENT, COMPONENTS, AND INSTRUMENTS HAVE SYSTEM PREFIX SI UNLESS NOTED OTHERWISE.
3. A = VALVE INTERLOCK FROM RH-V35, RH-V36 (DWG. 20449).
4. PERMANENT SPOOL PIECE INSTALLATION SHOWN. TEMPORARY STRAINER IS USED IN PLACE OF SPOOL PIECE DURING INITIAL FLUSHING OPERATION. STRAINER TO BE REMOVED BEFORE PLANT START UP.
5. VENT, DRAIN, TEST CONN'S AND ORIFICE TAP PIPING CODE BREAKS ARE AT THE DOWNSTREAM END OF THE OUTER ISOLATION VALVE PER 1-NHY-805111 UNLESS NOTED OTHERWISE.
6. DELETED
7. ADJUST VALVES IN FIELD TO LIMIT PUMP RUN OUT THEN LOCK.
8. MINIMUM FLOW ORIFICE SUPPLIED WITH PUMP.
9. △ -INDICATES REVISION LEVEL.
10. ● -INDICATES 3/8" FLOW RESTRICTOR INSTALLED, SEE APPLICABLE DETAIL ON DWG RC-20848.

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	SAFETY INJECTION SYSTEM INTERMEDIATE HEAD INJECTION SYSTEM DETAIL
	1-SI-B20446 / FIGURE 6.3-4

i. Communications (Subsection 9.5.2)

j. Normal and emergency lighting (Subsection 9.5.3)

k. Toilet facilities

l. Medical supplies

m. Kitchen area.

6.4.2.1 Definition of Control Room Envelope

The control room occupies the entire 75'-0" level of the Control Building, and includes the main control room area, computer room, Technical Support Center, office, conference room and library, emergency storage room, HVAC equipment room, kitchen and sanitary facilities, as shown in Figure 1.2-32. All controls, equipment and materials to which the control room operator would require access during an emergency are contained within this envelope, except for the makeup air intakes' manual isolation valves which are located on elevation 51'-6" of the Diesel Generator Building.

6.4.2.2 Ventilation System Design

Details of the control room complex ventilation and filtration systems are described in Subsections 9.4.1 and 6.5.1. An air flow diagram of the Control Room Ventilation System which identifies equipment, ducting, dampers, instrumentation and air flow rates for both normal and emergency modes is shown in Figures 9.4-1 through 9.4-3 and 9.4-25; major components and their major design parameters are included in Tables 9.4-1 and 6.5-6.

Two remote air intakes (east and west) are provided to furnish makeup air to the control room complex. The locations were selected to preclude both intakes from being susceptible to accident-generated airborne radioactivity or toxic gases at the same time. The east makeup air intake is located approximately 380 feet northeast from the center of the Unit 1 containment structure. The west intake is located approximately 500 feet southwest of the Unit 1 containment structure (see Figure 1.2-1).

The east air intake consists of a vertical 12-inch diameter carbon-steel pipe terminating in a tee-section. Both openings are protected by ½-inch square stainless steel wire mesh welded to the inside diameter of the respective opening. Protection of the intake against tornado missiles is provided by a reinforced concrete slab (see Figure 6.4-1). The west air intake consists of a vertical 12-inch diameter carbon-steel pipe terminating in a 180° bend. The opening is protected by ½-inch square screen and security grating welded to the inside diameter. A portion of the west intake pipe above grade adjacent to the cooling tower wall and approximately a 2-foot vertical section just below grade is not protected against tornado missiles. The unprotected

12-inch pipe, the vertical section above and below grade, and the 180° bend (see Figure 6.4-2) have an equivalent length of about 15 feet. This low effective target area results in a low mean value probability, calculated in the range of 2×10^{-9} to 3×10^{-7} per year, for tornado missile impact. Smoke monitoring equipment for the east intake, which alarms in the control room, is installed in a vault beneath the slab and within the fence enclosed area. Radiation monitoring equipment for the east air intake, which alarms in the control room, is located within the pipe in the Diesel Generator Building. Radiation and smoke monitoring equipment for the west intake, which also alarms in the control room, is located within the pipe in the Diesel Generator Building. The Diesel Generator Building is located within the protected area security fence, and access to the building is controlled by security doors which are part of the station access control system. Environmental conditions within the east intake vault is maintained by convection heaters and a sump pump.

The makeup air is transported via heavy wall carbon steel pipes from the remote air intakes to the control room HVAC equipment room (see Figures 9.4-1 through 9.4-3). The pipe enters the control room complex through the floor of one of the redundant filter units. The makeup air enters a tee located in a compartment of the filter unit upstream of the various filter components. The makeup air divides so that a portion of the air discharges through an isolation damper then enters this compartment, while the remainder of the air enters the branch leg of the tee to the interconnecting ductwork of the redundant filter unit. The filter units and connecting ductwork are located within the control room envelope.

During normal operations, makeup air is drawn from both remote intakes and delivered to the control room complex by one of the two redundant normal makeup air fans. The normal makeup air fans and associated discharge dampers are located outside the control room envelope on the 50'-0" elevation of the Diesel Generator Building. The air passes through medium efficiency prefilter(s) and electric heater(s) in both emergency filter units prior to discharging through an orifice into the control room HVAC equipment room. The prefilters remove dust and other airborne particulates and the heaters operate continuously to maintain the carbon filter relative humidity at or below 70 percent thereby optimizing carbon adsorber efficiency and life.

Under emergency conditions, makeup air is drawn from both remote air intakes and delivered to the control room complex by two fully redundant emergency filtration system fans. One hundred percent of the makeup air passes through the prefilter and heater and a HEPA-Carbon-HEPA filter configuration in either or both emergency filter units prior to discharging into the control room HVAC equipment room. In addition, approximately 2 percent of the total control room complex recirculation air flow, (i.e., including the air conditioning system flow rate) is drawn through the HEPA-Carbon-HEPA filter configuration in either or both emergency filter units. The HEPA filter(s) and carbon adsorber(s) are designed to remove radioactive airborne particulates and iodines (see Subsection 6.5.1 for filter design specifications). Under emergency conditions, the normal makeup air fans are automatically tripped off and their associated discharge dampers closed. The makeup air is transported to the control room via piping and backdraft dampers configured in parallel which bypass the normal makeup air fans and dampers. The backdraft dampers preclude short cycling of air during normal operations.

The exhaust air and supply air registers are adequately separated to preclude recycling stale air and other noxious gases. The outside makeup air intake and the point of discharge for the control room exhaust are also adequately separated to preclude recycling stale air and other noxious gases.

During normal operations, the modulating damper in the exhaust ductwork controls the amount of air being exhausted thereby maintaining a positive pressure within the control room complex. The damper is under the control of three static pressure sensing devices. The first pressure sensing point for the complex is in the HVAC equipment room, which is slightly lower in pressure than the remainder of the control room complex. The HVAC equipment room is maintained at least $\frac{1}{8}$" w.g. above the outside atmospheric pressure, the second sensing point, and at least $\frac{1}{8}$" w.g. above the cable spreading room, the third pressure sensing point.

Under emergency conditions, the exhaust system isolates by automatic trip of the exhaust fan and closure of the modulating damper and redundant isolation damper. The emergency makeup air is adequate to maintain the complex at a pressure at least 1/8" w.g. greater than the outside atmospheric and cable spreading room pressures. Air is exhausted from the complex by exfiltration.

The following system components are powered or controlled from the Emergency Electrical Distribution System, to ensure operating power during all modes of operation:

- The normal makeup air fans and associated discharge dampers
- Emergency makeup air fans and associated discharge dampers
- Filter system air heaters
- Radiation Monitoring Instrumentation System
- Exhaust system isolation dampers (Isolation Control System only).

The normal makeup air fans are electrically "cross-trained" with their associated discharge dampers, that is, the damper configured immediately downstream of the Train A fan is powered from the Train B vital bus and is controlled by the Train B control system. The damper configured immediately downstream of the Train B fan is powered from the Train A vital bus and is controlled by the Train A control system. This design ensures isolation of the normal makeup air system (i.e., trip of each makeup air and/or closure of its associated discharge damper) under emergency conditions regardless of any single active failure.

All automatic system dampers are pneumatically actuated and are designed to fail in the safe position (emergency mode configuration) on loss of instrument air.

Controls for habitability system components are located in the control room complex.

The normal makeup air fans and discharge dampers are controlled from the main control board (MCB). The components are manually actuated. Detection of high radiation in either remote intake will automatically isolate the normal makeup air subsystem. Actuation of the emergency makeup air and filtration subsystem fans will automatically isolate the normal makeup air subsystem. Failure of a vital bus or loss of instrument air will also isolate the system.

The emergency makeup air and filtration subsystem fans and discharge dampers are automatically actuated upon detection of high radiation in either remote intake or upon generation of a safety injection 'S' signal. The filtration subsystem can also be normally actuated from the MCB.

The exhaust subsystem fan and discharge dampers are controlled from the MCB. The subsystem will automatically isolate upon loss of control room pressurization, detection of high radiation in either remote intake, or actuation of the filtration subsystem fans.

The safety-related active components of the system are designed to seismic Category I requirements, and satisfy the design criteria of IEEE Standard 279 and other industry standards for electrical equipment, as defined in Subsection 8.1.4. No single failure of any of the active components will degrade the system's performance, as shown in Table 6.4-1.

The safety-related passive components (i.e., pressure boundary) of the makeup air system, ducts and filters, are also designed to seismic Category I requirements.

All safety-related active and passive components of the system are contained in missile- protected buildings, are underground, or in the case of some piping, consist of such a small exposed area that the possibility of being struck by a tornado missile is negligibly small. Specifically, the mean value probability of a missile impacting the unprotected portion of the west air intake is in the range of 2×10^{-9} to 3×10^{-7} per year. No internally generated missiles which could impair the system's ability to perform its safety-related functions are credible.

The system is designed to meet the intent of 10 CFR 50, Appendix A, General Design Criterion (GDC) 19, NUREG-0800 (Standard Review Plan) Section 6.4 and Subsection 6.5-1, and Regulatory Guides 1.52, 1.78, and 1.95.

6.4.2.3 Leak Tightness

The only openings in the control room envelope boundaries are the sealed cable penetrations, two personnel accesses, the exhaust air duct with isolation dampers, and building construction joints. The primary personnel access way is a double-door configuration. The total complex outleakage with the exhaust subsystem isolated is calculated to be 165 cfm at a pressure of (+)$\frac{1}{8}$" w.g.

6.4.2.4 Interaction with Other Zones and Pressure-Containing Equipment

a. Interaction with Other Zones

There are no potential adverse interactions between the control room complex ventilaition zones and adjacent zones that could transfer toxic or radioactive contaminants into the control room. No ducts or ventilation piping from any other zone penetrates the control room envelope.

Only the remote air intakes, makeup air piping and associated isolation valves, and normal makeup air fans and discharge dampers are located outside the control room envelope. This portion of the system is heavy carbon steel piping with primarily welded construction to minimize infiltration. The emergency makeup air and filtration subsystem fans are located within the control room envelope and downstream of their associated filter components. This configuration maintains the filter units at a negative pressure precluding makeup air from bypassing the filter components under emergency conditions.

The exhaust and static pressure control subsystem functions to maintain a positive pressure in the control room with respect to the outside and cable spreading room during normal operations. The pneumatically operated modulating damper in the exhaust ductwork controls the amount of air being exhausted and, thereby, maintains a positive pressure within the control room complex. The damper is under the control of three static pressure sensing devices. The pressure sensing point for the complex is in the HVAC equipment room, which is slightly lower in pressure than the remainder of the control room envelope. The mechanical equipment room is kept at least $\frac{1}{8}$" w.g. above the outside atmospheric pressure, the second pressure sensing point, and at least $\frac{1}{8}$" w.g. above the cable spreading room at all times, the third pressure sensing point.

Under emergency conditions, the exhaust duct and associated control room envelope penetration are isolated by redundant dampers configured in series.

During normal operation, 1000 cfm of makeup air will be delivered to the control room complex. Approximately 145 cfm will be exfiltrated and the remaining 855 cfm will be exhausted. Under emergency conditions, approximately 600 cfm of makeup air will be delivered to the control room complex all of which will be exfiltrated.

An SSE will not provide an exfiltration path for control room complex air that will negate the ability to maintain a positive pressure since the Diesel Generator Building and Control Building are seismic Category I structures. The intake air equipment and welded piping to the control room complex are also seismic Category I.

b. Pressure-Containing Equipment

The pressure-containing equipment in the Control Room Complex consists of refrigerant lines, Computer Room Halon system, fire extinguishers, and self-contained breathing apparatus.

The Computer Room Air Conditioning Refrigerant System is nonsafety-related, nonseismic Category I and contains 10 pounds of Refrigerant 22.

The Uniform Mechanical Code, 1976 Edition, Section 1505 allows 22 pounds of Refrigerant 22 per 1,000 cubic feet of occupied space. The control room complex envelope is 246,000 cubic feet. The refrigerant charge of 10 pounds in the Computer Room Air Conditioning System is well below the allowable 22 pounds per 1,000 cubic feet.

The computer room Halon 1301 system is capable of a total concentration of 5 percent of the computer room volume of 13,340 cubic feet. The release of the entire volume of Halon into the control room complex envelope would result in a concentration of 0.30 percent. The National Fire Protection Association Code, Section 12A, lists 5 to 7 percent concentration as producing minimal, if any, central nervous system effects for exposures of approximately five minutes' duration.

Fire extinguishers of various types (Halon, dry chemical, and CO_2) are installed in the Control Room complex. All of the fire extinguishers are Underwriters Laboratories listed and/or Factory Mutual approved.

If all of the Halon fire extinguishers were discharged in the Control Room, the total amount of Halon released would be less than the total amount of Halon that would be released by the Computer Room Halon system. The release of the Computer Room Halon system will not adversely affect Control Room habitability. Therefore, the release of the lesser amount of Halon contained within the fire extinguishers would also not affect the Control Room complex habitability.

6.4.2.5 Shielding Design

The design basis loss-of-coolant accident (LOCA) establishes the shielding requirements for the Control Building. The control room shielding design is discussed in Subsection 12.3.2, and is evaluated from design bases LOCA source terms and doses which are presented in Subsection 15.6.5.4.

The external walls and roof of the control room are 2-foot thick reinforced concrete. These shield thicknesses in conjunction with the habitability systems will limit the integrated dose to the operators to less than 5 rem whole body and 30 rem thyroid for a duration of 90 days after a LOCA, in conformance with General Design Criterion 19 of Appendix A of 10 CFR, Part 50.

6.4.3 System Operational Procedures

6.4.3.1 Normal Mode

a. Normal Makeup Air Subsystem

During normal plant operation, the control room normal makeup air subsystem is aligned to deliver approximately 1000 cfm of outside air from both remote intakes (500 cfm per intake). With one normal makeup air fan operating and its associated discharge damper open, the intake isolation valves are positioned to allow equal amounts of air to be drawn from the east and west intakes. The normal makeup air flows through the prefilter and heater for each emergency filter unit and discharges through an orifice into the HVAC equipment room. The heater for each unit operates continuously to limit humidity to less than or equal to 70 percent. The prefilters are periodically replaced when particulate buildup causes the differential pressure across the filters to increase to a predetermined value.

In the event normal makeup air fails or is isolated for reasons other than those delineated in Subsections 6.4.3.2 and 6.4.3.3, appropriate operator action will be taken to re-establish makeup air. If makeup air is lost because of fan failure, the redundant normal makeup air fan and its discharge damper will be manually actuated. If makeup air is lost because of a vital bus outage or failure, or a loss of instrument air supply to the dampers, the emergency makeup air and filtration subsystem will be manually actuated.

b. Emergency Makeup Air and Filtration Subsystem

During normal plant operation, the emergency makeup air and filtration subsystem fans are idle and their associated discharge dampers are closed. Normal makeup air flows through each filter unit's prefilter and heater as discussed above. In the event this subsystem must be manually actuated during normal operation, it functions similarly to emergency mode operation.

c. Exhaust and Static Pressure Control Subsystem

During normal plant operation, the control room exhaust fan is operating and its discharge control damper modulates to maintain the control room complex at a pressure of at least (+)1/8" w.g. with respect to the outside and adjacent areas. The redundant exhaust isolation damper remains fully open.

d. Cooling/Recirculation Subsystem

The non-safety related control room air conditioning subsystem will normally operate. However, the safety related trains may be placed in operation during normal plant operation. In the event of a malfunction in the non-safety related subsystem, one of two 100% capacity safety-related trains of control room air conditioning will be placed in service manually. Following a loss of offsite power, with the non-safety related subsection de-energized, one of the two redundant safety-related trains will automatically start via the emergency diesel generator load sequencer. Subsection 9.4.1 provides a more detailed description of this subsystem and its operation.

The unit heaters are not required to maintain the operation of the control room. Redundant unit heaters are not provided. If the unit heaters should fail during operation of the control room in the wintertime, the space temperature may drop below the normal comfort temperature. However, heat loads generated internally by electrical equipment would preclude excessively low temperatures.

6.4.3.2 Emergency Mode

a. Normal Makeup Air Subsystem

Following an accident, when high radiation is detected in either remote air intake or when the emergency makeup air and filtration subsystem fans are actuated, the normal makeup air fans automatically trip off and their associated discharge dampers automatically close. The control systems for these fans and dampers are "cross-trained," that is, the discharge damper associated with the Train A fan is controlled by the Train B control loop and vice versa. This configuration ensures isolation of the normal makeup air subsystem by fan trip and/or damper closure regardless of any single active failure.

Detection of smoke in either remote intake is alarmed only. Operator action is required to initiate the filter recirc. mode. Operations may, at their discretion, manually isolate the smoke-contaminated intake and re-establish makeup air from the unaffected intake to the control room complex via the emergency makeup air and filtration subsystem (see Subsection 6.4.3.3).

running and the control/isolation damper modulates to maintain a positive control room pressure. This modulating feature is controlled by a differential pressure control loop. This control system senses pressure in the Control Room HVAC Equipment Room, Cable Spreading Room and the outside atmosphere. The damper is modulated automatically to maintain the HVAC Room at greater than or equal to $\frac{1}{8}$" w.g. positive pressure with respect to atmosphere and the Cable Spreading Room. The redundant exhaust isolation damper (CBA-DP-1058) is fully open during normal plant conditions. Loss of normal makeup air and/or loss of control room pressurization will close CBA-DP-28 and trip CBA-FN-15. Interlocks are provided so that isolation dampers CBA-DP-28, -1058 and exhaust fan CBA-FN-15 are isolated whenever a high radiation signal is present or fans 16A or 16B are running.

Status indication for CBA-FN-15, CBA-DP-28, and CBA-DP-1058 is provided on the MCB. Indication of differential pressure between the control room HVAC room and outside atmosphere is provided in the HVAC room. This differential pressure, as well as the differential pressure between the Cable Spreading Room and the HVAC room, is recorded on the station computer.

The following alarms are provided at the MCB:

- Low control room/outside atmosphere differential pressure
- Low control room/cable spreading room differential pressure
- CBA-FN-15 tripped.

6.4.6.4 Cooling/Recirculation Subsystem

All principal components of the safety-related chilled water system (namely water chillers, chilled water pumps and air conditioning unit fans and dampers), except the chiller condenser exhaust fans, are controlled from the MCB. The chiller condenser exhaust fans are controlled from the control room air conditioning panel located in the control room air conditioning equipment room on elevation 75' within the control room pressure envelope. Additional instrumentation and control features are discussed in Subsection 9.4.1.

The following alarms are provided at the MCB:

- Control room high and low temperature
- Computer room high and low temperature
- Safety-related chiller trouble
- Condenser exhaust fan trip
- Condenser exhaust fan bypassed

6.4.7 References

1. N. Irving Sax, "Dangerous Properties of Industrial Chemicals," 5th Edition, Van Nostrand Reinhold, 1979.

2. "Handbook of Compressed Gases," Compressed Gas Association, Inc., Van Nostrand Reinhold, 1966.

3. "Hydrogen Chloride," Safety Data Sheet SD-39, Manufacturing Chemists Association, Washington, D.C., Quoted in Reference 2; also extracts supplied by personal communication with Miss Mott of AICE.

4. "Hazardous Materials Transportation Update, Seabrook Station," 1988, Engineering Calculation SBC-296, Yankee Atomic Electric Company, Bolton, MA.

5. Yankee Atomic Electric Company, "Seabrook Station Offsite Hazardous Chemical Analysis Update," YAEC-1660, Bolton, MA, 1988.

6. Yankee Atomic Electric Company, "Seabrook Station Onsite Hazardous Chemical Evaluation Update," YAEC-1690, Bolton, MA, 1989.

TABLE 6.4-1
(Sheet 1 of 3)

CONTROL ROOM COMPLEX SAFETY-RELATED VENTILATION SYSTEM MALFUNCTION ANALYSIS

Component	Malfunction	Comments
Normal Makeup Air Fan	Fan trips	Two 100 percent capacity fans are provided for normal makeup air and control room pressurization requirements. The redundant fan may be manually actuated to supply the makeup air. Each fan is powered from a separate emergency bus. Loss of air flow is alarmed. Fan status lights are provided on the main control board.
Normal Makeup Air Discharge Damper	Damper fails to open, or fails to close	If the damper fails to open, the redundant damper may be opened and its associated fan actuated to provide the makeup air. If both normal, makeup air trains are unavailable because of a vital bus outage, the emergency makeup air and filtration subsystem may be utilized. If the damper fails to close, the crosstrain control scheme design ensures that the associated fan trips to ensure isolation. Both normal makeup air dampers are provided with manual handwheel override actuators. Indicating lights on the main control board monitor all damper positions via limit switches on the damper linkage.
Emergency Makeup Air Fan	Fan fails to actuate on high intake radiation, 'S' signal, or manual actuation	Two 100 percent capacity fans are provided with a fully redundant filter unit associated with each fan. Both fans are automatically actuated on high radiation or an 'S' signal. If one fan fails on manual actuation, the redundant fan may be manually actuated. Flow indication and alarms are provided for each filter/fan train. Each fan is powered from separate emergency buses. Fan status lights are provided on the main control board.

TABLE 6.4-1
(Sheet 2 of 3)

CONTROL ROOM COMPLEX SAFETY-RELATED VENTILATION SYSTEM MALFUNCTION ANALYSIS

Component	Malfunction	Comments
Emergency Makeup Air Discharge Damper	Damper fails to open on high intake radiation, 'S' signal or manual actuation	Both dampers are automatically opened on high radiation or an 'S' signal. If one damper fails to open on manual actuation, the redundant damper and associated fan may be actuated. Indicating lights on the main control board monitor all damper positions via limit switches on the damper linkage.
Water (35-40% ethylene glycol by volume) Chiller	Water Chiller trips	Two 100% capacity chillers, each supplied from a separate emergency bus, are provided. Secure the tripped train and place the redundant train in service.
Chilled Water Pump	Chilled Water Pump trips	Four 100% capacity pumps, two per train with each train supplied from a separate emergency bus, are provided. The diverse pump can be manually aligned and started to provide chilled water flow or the redundant train can be placed into service.
Air Handling Unit (including cooling coils)	Air Handling Unit trips	Two 100% capacity air handling units, each supplied from a separate emergency bus, are provided. The redundant train may be started to supply conditioned air to the control room.
Air Handling Unit Discharge Damper	Fails to open	Each of the two 100% capacity units is provided with its own discharge damper. When the damper fails to open, the air handling unit and the corresponding chiller and the chilled water pump will all trip. The redundant train may be started. Indicating lights on the air handling unit control panel monitor all damper positions via limit switches on the damper linkage.
	Fails to close	The damper may be manually positioned through the use of the handwheel on the actuator.

TABLE 6.4-1
(Sheet 3 of 3)

CONTROL ROOM COMPLEX SAFETY-RELATED VENTILATION SYSTEM MALFUNCTION ANALYSIS

Component	Malfunction	Comments
		Indicating lights on the air handling unit control panel monitor all damper positions via limit switches on the damper linkage.
3-Way Temperature Control Valve (TCV)	TCV fails	Each train has its own TCV, and it is designed to fail-safe to "full flow to the air handling unit cooling coils". The failed train may be secured and the redundant train may be started.
Chiller Condenser Exhaust Fan	Exhaust fan fails	Two 100% capacity exhaust fans, each supplied from a separate emergency bus, are provided. The redundant train may be started.
Exhaust Fan Control Damper	Damper fails to close on actuation of emergency makeup air subsystem and/or a high radiation signal	Two redundant isolation dampers are configured in series. Both dampers close upon actuation of the emergency makeup air fans or generation of a remote intake high radiation signal ensuring isolation of the exhaust subsystem. Indicating lights on the main control board monitor all damper positions via limit switches on the damper actuator.

b. Fuel Storage Building Emergency Air Cleaning System

The operation of the Fuel Storage Building (FSB) Emergency Air Cleaning System is controlled and monitored in the plant control room (see Drawing NHY-503543). FSB supply air is controlled through DP 13A/B (see Drawing NHY-503541). During fuel handling operations, the FSB is maintained at a slightly negative pressure with respect to the atmosphere through the FSB exhaust system (see Drawing NHY-503542). FSB high and low differential pressure is alarmed at the MCB. Also, the FSB temperature and relative humidity are monitored, and high deviations are alarmed at the MCB. Each filter train is monitored for differential pressure and high deviations are alarmed at the MCB. Each cleaning unit is provided with a temperature switch which alarms high temperature on the MCB. Temperature switches for automatic control of cleaning unit heaters are also provided. Independent low-flow instrumentation is provided to alarm at MCB the discharge air flow through each of the redundant air cleanup filter units (see Drawing NHY-506452). In addition, the differential pressure across each individual filter unit is indicated locally. Local temperature indicators are also provided both upstream and downstream of the carbon filters.

c. Control Room Emergency Makeup Air and Filtration Subsystem

The emergency makeup air and filtration subsystem fans and discharge dampers are also controlled from the MCB. In the auto mode, CBA-DP-27A and 27B will open upon receipt of a remote intake high radiation signal or a safety injection ("S") signal. Opening of these dampers will automatically start associated fans CBA-FN-16A and 16B. A high radiation signal or starting fans 16A and 16B will trip CBA-FN-27A, 27B, and 15 and isolate dampers CBA-DP-53A, 53B, 28 and 1058. Each filter air heater operates continuously by cycling on and off. Low and High temperature controls are provided. Status lights are provided at the MCB for the fans and discharge dampers. Status lamps are also provided for the fans in the Accident Monitoring Instrumentation arrays.

Differential pressure indication across each filter component is provided locally. High differential pressure across each filter unit generates a VAS alarm at the MCB. The temperature for each filter unit is indicated locally. High temperature generates a VAS alarm. Relative humidity for each filter unit is recorded on the station computer. High and low air flow generate VAS alarms. Two carbon monoxide detectors per filter (one each at the inlet and outlet of the carbon adsorber banks) provide for early fire detection. These devices monitor the filtration systems and generate an alarm in the control room.

6.5.1.6 Materials

Listed in Tables 6.5-4, 6.5-5 and 6.5-6 by commercial name, quantity and chemical composition are the materials used in or on the filter systems. Extremes in temperature or radiation that could promote radiolytic or pyrolytic decomposition of materials are not applicable to the Containment Enclosure Building or the fuel storage for normal or postulated accident conditions.

6.5.2 Containment Spray System

This section presents a detailed description of the fission product removal function of the Containment Building Spray (CBS) System.

6.5.2.1 Design Bases

The spray system provides for iodine removal to limit the consequences of a LOCA to the limits of 10 CFR 100 by providing a rapid reduction in containment elemental iodine concentration following a LOCA. This greatly reduces the amount of iodine that can leak from the primary containment into the containment enclosure structure. The combination of iodine removal by the sprays and the containment enclosure exhaust filters limits the offsite radiological consequences resulting from the design basis accident. These consequences and the system performance are discussed in Subsection 15.6.5.

Table 15.6-17, prepared in accordance with Regulatory Guide 1.4, lists the fission products that the CBS system is designed to remove during a LOCA. A discussion of the extent to which credit is taken for fission product removal by the CBS system is included in Subsection 15.6.5.

The system uses the principles of hydrodynamic equilibrium to provide a reliable means for buffering the spray solution with an iodine removal agent. The spray rings are designed to provide a high degree of spray coverage to maximize the spray effectiveness for iodine removal.

6.5.2.2 System Design (for Fission Product Removal)

Iodine removal inside the containment following a LOCA is accomplished by 3010 gpm of spray (assuming only one train available) with a boric acid-sodium hydroxide spray solution. The design details of the spray system are presented in Subsection 6.2.2.

The spray system initially takes suction from the refueling water storage tank (RWST) and continues to add a fresh spray additive solution into the containment until the inventory of the RWST is depleted. Upon a low level in the RWST, the suctions of the spray pumps are automatically transferred to the containment sump. The time of transfer is a function of the number of emergency core cooling pumps and containment spray pumps in operation. At design flow rates and minimum pumps in operation (3575 gpm injection and 3010 gpm spray) the transfer to the recirculation mode takes place in 53.7 minutes, assuming a 10-second delay for injection and a 62-second delay in spray flow. With maximum safeguards in operation (9800 gpm injection and 6600 gpm spray), the transfer to the containment sump occurs at approximately 21.9 minutes, assuming a 10-second delay for injection and a 62-second delay in spray flow. The spray pumps remain in operation for as long as is necessary to control the containment pressure within the required limits. The maximum delay in delivery of the NaOH solution to the spray nozzles is 2.47 minutes.

The chemical additive for the spray is stored in the spray additive tank (SAT) located adjacent to the RWST. When a containment spray actuation signal occurs, two valves in parallel lines provide redundant flow paths for supplying the chemical additive to the RWST and spray pump suction. The chemical additive is stored as a 19-21 weight percent NaOH solution. The chemical spray additive flows by gravity into a mixing chamber in the RWST. The design is such that the pH of the solution leaving the mixing chamber is averaging between 8.8 and 9.2. The total amount of chemical supplied will result in a containment sump liquid having a pH range between 8.7 and 9.2 during the recirculation phase.

The spray is delivered to the containment through 198 SPRACO 1713A nozzles per flow train, each having a flow rate of 15.2 gpm at containment design pressure and 40 psi differential operating pressure. The nozzles produce a drop size spectrum with a conservatively estimated volumetric drop diameter of 1000 microns.

The nozzles are spaced to provide a uniform spray pattern across the containment cross section. The nozzles of the redundant spray trains are uniformly spaced between the nozzles of the other spray train, so that either subsystem will provide uniform coverage. The SPRACO 1713A nozzles have been used extensively in other nuclear plants and various iodine removal experiments.

The location of the spray nozzles in the dome of the containment is shown on Figures 6.2-76 and 6.2-77. The flow weighted average fall height of the spray drops is a minimum of 134 ft. for headers 1 and 2. For headers 3 and 4, the average fall height is 144 ft. The operating floor is at an elevation of 25 ft.

The following is a tabulation of important spray iodine removal parameters:

Spray fall height	134 ft.
Total containment free volume	2.715×10^6 ft^3
Sprayed containment free volume	2.310×10^6 ft^3

Containment pressure, design	52.0 psig
Containment temperature, design	296°F
Spray flow rate per train design/ minimum	3010/2808 gpm
Spray solution pH, minimum	8.9
average	8.9 to 10.3
Spray additive flow rate	
Maximum safety injection	240 gpm
Minimum safety injection	150 gpm

These parameters are chosen to minimize credit taken for iodine removal by the spray system and provide a conservative basis for calculating offsite doses.

Two headers are provided for each 3010 gpm train. Headers No. 1 and 3 are supplied by one of the CBS pumps while Headers No. 2 and 4 are supplied by the other pump.

Header No. 1 has 118 nozzles spaced approximately 3° apart. Fifty-nine of these nozzles are oriented vertically downward. The remaining 59 are oriented at 45° on the inside of the header pointing inward and downward.

Header No. 3, which is the companion header to header No. 1, has 80 nozzles spaced approximately 4° 30′ apart. Forty of these nozzles are at a 45° angle pointing downward toward the containment wall, 16 are at a 45° angle pointing downward and inward, 20 nozzles point directly downward and 4 point horizontally toward the containment vertical centerline.

Header No. 2 contains 162 nozzles spaced approximately 2° 15′ apart. Eighty are mounted at a 45° angle pointing downward toward the containment wall, 41 point directly downward and 41 are at a 45° angle pointing inward.

Header No. 4, which is the companion header to header No. 2, contains 36 nozzles spaced approximately 10° apart. Eighteen are mounted at a 45° angle pointing downward toward the containment wall. The remaining 18 point directly downward.

Figures 6.2-84 through 6.2-86 show the spray drop distribution, accumulated number percentage, and accumulated volume percentage for the SPRACO No. 1713A nozzle. These distributions are the basis for evaluation of iodine removal effectiveness.

Approximately 85 percent of the containment net free volume is covered by spray. Figures 6.2-80 through 6.2-83 show plan and elevation views of the expected spray patterns.

Table 6.2-80 lists the regions of sprayed volumes in the containment and the volume of each sprayed region. It can be seen from Figures 6.2-80 and 6.2-81 that more than 95 percent of the cross section of the containment at the operating floor (elevation 25′) is covered by spray.

6.5.2.3 Design Evaluation

The Containment Spray System has been evaluated by both conservative and realistic models. Subsection 15.6.5 presents the parameters used in both the conservative and realistic analyses and the results of these analyses.

The initial spray solution will be boric acid buffered with sodium hydroxide to a pH of approximately 8.8 to 9.2. Sump pH after recirculation commences will be above 8.5 to assure high iodine partition factors and prevent re-evolution of iodine from the spray solution.

Using the method described by Parsly (Reference 1) a fresh spray drop is calculated to be less than 10 percent saturated during its fall from the spray nozzles. The removal of iodine should therefore be controlled by gas phase resistance. Steam condensation and drop coalescence have been discussed by Pasedag and Gallagher (Reference 2) and by Parsly (Reference 1). Pasedag shows that the total reduction in drop mass transfer surface area over a 100-foot fall due to these phenomena is only about 10 percent. Parsly recommends for calculational purposes that a mean drop diameter of 25 percent larger than the actual mean be used to conservatively compensate for coalescence and size distribution. The calculations in Subsection 15.6.5 assume a mean drop diameter of 1250 microns, which conservatively estimates the iodine removal half-life.

Table 6.2-80 lists the sprayed regions in the containment. The spray headers are designed to directly spray approximately 85 percent of the containment free volume. In addition the unsprayed areas are designed to allow, as much as possible, the free exchange of air with the sprayed area to prevent the pocketing of fission products.

Due to the relatively low surface to volume ratio in the reactor containment it is not expected that wall effects will play a large part in the removal of iodine by spray solution. No wall effects are included in the calculation in Subsection 15.6.5.

The mathematical models used in calculating iodine removal by spray are presented in Subsection 15.6.5. These models assume that the spray removal function is effective throughout the containment sprayed volume and that the effectiveness is constant. Credit for mixing between the containment sprayed and unsprayed volumes is described in Subsection 15.6.5, in conjunction with the discussion of the multi-compartment spray model used to calculate iodine removal rate constants.

6.5.2.4 Tests and Inspections

Subsection 6.2.2.4 discusses the tests and inspections performed to verify the functional capability of the spray system and components, including active valves, pumps, and the spray nozzles, to deliver the required flow for containment heat removal.

Demonstration of those drawdown characteristics of the SAT and RWST which provide flow at the proper pH for effective iodine removal is also discussed in Subsection 6.2.2.4.

The spray additives are sampled on a periodic basis to verify their continued state of readiness.

6.5.2.5 Instrumentation

The system is provided with instrumentation and control to allow the operator to monitor the status and operation of the Containment Spray System from the control room to allow the automatic or manual initiation of the injection mode of operation. The instrumentation details are presented in Subsection 6.2.2.5.

The SAT is steam heated, with the steam flow controlled by a temperature controller, which modulates a pneumatic-operated flow control valve. Low temperature of the tank is alarmed at the main control board (MCB). The tank level is indicated at the MCB. Refueling water storage tank instrumentation is described in Subsection 6.3.5.

Each of the SAT discharge valves may be opened either manually by the operator or automatically upon receipt of the containment spray signal.

The valve positions are indicated at the MCB, above the control switches and also as a part of a separate status monitoring indication system.

6.5.2.6 Materials

The spray solution is stored in the spray additive tank (SAT), and the RWST holds a boron solution. Both tanks are constructed of stainless steel which has been shown to be resistant to chemical attack by the respective stored solutions of 2700 to 2900 ppm boron (as boric acid) and 19 to 21 percent by weight of sodium hydroxide. The spray solution is not susceptible to radiolytic or pyrolytic decomposition under the conditions anticipated in the post-accident environment. The corrosion properties of the spray solution are discussed in detail in Subsection 6.2.2.2.

6.5.3 Fission Product Control Systems

This section provides a discussion of the operation of all fission product control systems following the design basis loss-of-coolant accident (LOCA).

TABLE 6.5-1
(Sheet 1 of 3)

COMPLIANCE OF CONTAINMENT ENCLOSURE AIR CLEANING UNITS TO REGULATORY GUIDE 1.52, REV. 2, MARCH 1978

Regulatory Guide Section	Applicability to this System	Comment Index	Regulatory Guide Section	Applicability to this System	Comment Index
C.1.a	Yes	Note 1	C.3.f	Yes	---
C.1.b	Yes	---	C.3.g	Yes	Note 11
C.1.c	Yes	---	C.3.h	Yes	---
C.1.d	Yes	---	C.3.i	Yes	Note 17
C.1.e	Yes	---	C.3.j	Yes	Note 12
C.2.a	Yes	Note 2	C.3.k	Yes	---
C.2.b	Yes	---	C.3.l	Yes	Note 13
C.2.c	Yes	---	C.3.m	Yes	---
C.2.d	Yes	Note 3	C.3.n	No	Note 5
C.2.e	Yes	---	C.3.o	Yes	---
C.2.f	Yes	---	C.3.p	Yes	---
C.2.g	Yes	Note 15	C.4.a	Yes	Note 14
C.2.h	Yes	Note 7	C.4.b	Yes	---
C.2.i	Yes	---	C.4.c	Yes	---
C.2.j	Yes	---	C.4.d	Yes	---
C.2.k	No	Note 4	C.4.e	Yes	---
C.2.l	Yes	---	C.5.a	Yes	Note 16
C.3.a	Yes	Note 8	C.5.b	Yes	Note 16
C.3.b	No	Note 9	C.5.c	Yes	Note 16
C.3.c	No	Note 2	C.5.d	Yes	Note 16
C.3.d	Yes	Note 6	C.6.a	Yes	Note 16
C.3.e	Yes	Note 10			

TABLE 6.5-1
(Sheet 2 of 3)

COMPLIANCE OF CONTAINMENT ENCLOSURE AIR CLEANING UNITS TO REGULATORY GUIDE 1.52, REV. 2, MARCH 1978

NOTES:

1. The design basis LOCA and rupture of a CRDM housing are the postulated design basis accidents.

2. Demisters will also serve as prefilters. No other prefilters are provided. Regulatory Guide 1.52, Revision 2, Section C.2, permits the use of demisters as prefilters for HEPA filters in an ESF System. The demister element consists of a 5½" thick, 0.006" diameter, 340 stainless steel mesh, and fiberous glass fill. The "dry" efficiency of the demister is approximately 45 percent when tested in accordance with NBS Dust Spot Test. This efficiency is similar to that of a prefilter. The demister will therefore serve a dual purpose, that of a demister and a prefilter; and there is no need to consider HEPA filter particulate loading without a prefilter for this application.

   Flow instrumentation is provided in common ductwork downstream of containment enclosure emergency exhaust filter fans. One channel of volumetric flow indication is indicated and alarmed at the MCB as well as indicated locally. Secondary flow indication is available using filter train differential pressure and fan status indication.

3. No significant pressure surges to this system are foreseen; thus, no special protective devices are needed.

4. There are no outdoor air intakes that could affect the operation of the system.

5. The system is located in the containment enclosure, the area served. Therefore, any leakage will eventually be re-routed through the cleanup system before being expelled to the atmosphere.

6. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. HEPA filters are designed and qualified to MIL-F-51068, MIL-F-51069 and UL-586. There is no need to withstand iodine removal sprays.

7. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Regulatory Guide 1.89 - IEEE Std. 323, Regulatory Guide 1.30 - IEEE Std. 336, Regulatory Guide 1.100 - IEEE Std. 344, Regulatory Guide 1.118 - IEEE Std. 338, Regulatory Guide 1.32 - IEEE 308.

8. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Demisters are qualified to MSAR 71-45.

TABLE 6.5-1
(Sheet 3 of 3)

COMPLIANCE OF CONTAINMENT ENCLOSURE AIR CLEANING UNITS TO REGULATORY GUIDE 1.52, REV. 2, MARCH 1978

9. Heaters are not required for this unit. The ambient design conditions for the site are 88°F dry bulb with a maximum relative humidity of 74.4 percent. Section 5.5 of ANSI 509-1976 states that "approximately 70% RH" is required upstream of the moisture separator. Only 33 percent of the total air supplied to the containment enclosure area is outside air. The remainder is recirculated by the containment enclosure cooling units as explained in Section 9.4.6. The cooling units will maintain the space temperature at or below 153°F at the outside design conditions. Therefore, since no moisture is added to the Supply Air System, the relative humidity will not exceed 50 percent; which is less than the 70 percent RH required by ANSI N509.

10. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Filter and adsorber mounting frames are constructed and designed in accordance with Section 4.3 of ERDA 76-21.

11. Equipment was purchased prior to issuance or Regulatory Guide 1.52, Rev. 2. Filter housings, floors and doors are constructed in accordance with ERDA 76-21.

12. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Tray type adsorber cells are designed, constructed and tested in accordance with AACC CS-8T.

13. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. The system fan, its mounting, and the ductwork connections are designed, constructed and tested in accordance with the requirements of Regulatory Guide 1.52, Rev. 1 and ANSI N509-1980 and ANSI N510-1980.

14. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Accessibility of components and maintenance are compatible with the intent of Regulatory Guide 1.52, Rev. 1.

15. Pertinent pressure drops (high delta-P across filter unit) are indicated, alarmed, and recorded at MCB.

16. In-place inspection and testing is performed in accordance with ANSI N510-11980.

17. Original charcoal was tested in accordance with ANSI N509-1976. All testing of replacement charcoal and future required periodic testing of charcoal will be in accordance with ASTM D3803-1989.

TABLE 6.5-2
(Sheet 1 of 3)

COMPLIANCE OF FUEL STORAGE BUILDING AIR CLEANING UNITS TO REGULATORY GUIDE 1.52, REV. 2, MARCH 1978

Regulatory Guide Section	Applicability to this System	Comment Index	Regulatory Guide Section	Applicability to this System	Comment Index
C.1.a	Yes	Note 1	C.3.f	Yes	---
C.1.b	Yes	---	C.3.g	Yes	Note 11
C.1.c	Yes	---	C.3.h	Yes	---
C.1.d	Yes	---	C.3.i	Yes	Note 16
C.1.e	Yes	---	C.3.j	Yes	Note 12
C.2.a	Yes	Note 2	C.3.k	Yes	---
C.2.b	Yes	---	C.3.l	Yes	Note 13
C.2.c	Yes	---	C.3.m	Yes	---
C.2.d	Yes	Note 3	C.3.n	No	Note 5
C.2.e	Yes	---	C.3.o	Yes	---
C.2.f	Yes	---	C.3.p	Yes	---
C.2.g	Yes	---	C.4.a	Yes	Note 14
C.2.h	Yes	Note 7	C.4.b	Yes	---
C.2.i	Yes	---	C.4.c	Yes	---
C.2.j	Yes	---	C.4.d	Yes	---
C.2.k	No	Note 4	C.4.e	Yes	Note 15
C.2.l	Yes	---	C.5.a	Yes	Note 15
C.3.a	Yes	Note 8	C.5.b	Yes	Note 15
C.3.b	Yes	Note 9	C.5.c	Yes	Note 15
C.3.c	No	Note 2	C.5.d	Yes	Note 15
C.3.d	Yes	Note 6	C.6.a	Yes	Note 15
C.3.e	Yes	Note 10			

TABLE 6.5-2
(Sheet 2 of 3)

COMPLIANCE OF FUEL STORAGE BUILDING AIR CLEANING UNITS TO REGULATORY GUIDE 1.52, REV. 2, MARCH 1978

NOTES:

1. A fuel handling accident is the postulated design basis accident.

2. Demisters will also serve as prefilters. No other prefilters are provided. Regulatory Guide 1.52, Revision 2, Section C.2, permits the use of demisters as prefilters for HEPA filters in an ESF System. The demister element consists of a 5½" thick, 0.006" diameter, 340 stainless steel mesh, and fiberous glass fill. The "dry" efficiency of the demister is approximately 45 percent when tested in accordance with NBS Dust Spot Test. This efficiency is similar to that of a prefilter. The demister will therefore serve a dual purpose, that of a demister and a prefilter; and there is no need to consider HEPA filter particulate loading without a prefilter for this application.

3. No significant pressure surges to this system are foreseen; thus, no special protective devices are needed.

4. There are no outdoor air intakes that could affect the operation of the system.

5. The system is located in the Fuel Storage Building, the area served. Therefore, any leakage will eventually be re-routed through the cleanup system before being expelled to the atmosphere.

6. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. HEPA filters are designed and qualified to MIL-F-51068, MIL-F-51069 and UL-586. There is no need to withstand iodine removal sprays.

7. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Regulatory Guide 1.89 -IEEE Std. 323, Regulatory Guide 1.30 - IEEE Std. 336, Regulatory Guide 1.100 - IEEE Std. 344, Regulatory Guide 1.118 - IEEE Std. 338, Regulatory Guide 1.32 - IEEE 308.

8. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Demisters are qualified to MSAR 71-45.

9. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Heaters are constructed to industry standards and are seismically qualified.

10. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Filter and adsorber mounting frames are constructed and designed in accordance with Section 4.3 of ERDA 76-21.

TABLE 6.5-2
(Sheet 3 of 3)

COMPLIANCE OF FUEL STORAGE BUILDING AIR CLEANING UNITS TO REGULATORY GUIDE 1.52, REV.2, MARCH 1978

11. Equipment was purchased prior to issuance or Regulatory Guide 1.52, Rev. 2. Filter housings, floors and doors are constructed in accordance with ERDA 76-21.

12. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Tray type adsorber cells are designed, constructed and tested in accordance with AACC CS-8T.

13. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. The system fan, its mounting, and the ductwork connections are designed, constructed and tested in accordance with the requirements of Regulatory Guide 1.52, Rev. 1 and ANSI N509 and N510-1980.

14. Equipment was purchased prior to issuance of Regulatory Guide 1.52, Rev. 2. Accessibility of components and maintenance are compatible with the intent of Regulatory Guide 1.52, Rev. 1.

15. In-place inspection and testing is performed in accordance with ANSI N510-1980.

16. Original charcoal was tested in accordance with ANSI N509-1976. All testing of replacement charcoal and future required periodic testing of charcoal will be in accordance with ASTM D3803-1989.

TABLE 6.5-3
(Sheet 1 of 4)

COMPLIANCE OF CONTROL ROOM EMERGENCY FILTRATION SUBSYSTEM TO REGULATORY GUIDE 1.52, REV. 2, MARCH 1978

REGULATORY GUIDE SECTION	APPLICABILITY TO THIS SYSTEM	COMMENT INDEX
C.1.a	Yes	-------
C.1.b	Yes	Note 1
C.1.c	Yes	-------
C.1.d	Yes	-------
C.1.e	Yes	-------
C.2.a	Yes	Note 2
C.2.b	Yes	-------
C.2.c	Yes	-------
C.2.d	No	Note 3
C.2.e	Yes	-------
C.2.f	Yes	-------
C.2.g	Yes	Note 4
C.2.h	Yes	-------
C.2.i	Yes	-------
C.2.j	Yes	-------
C.2.k	Yes	-------
C.2.l	Yes	Note 8
C.3.a	No	Note 2
C.3.b	Yes	Note 5
C.3.c	Yes	Note 5
C.3.d	Yes	Note 5
C.3.e	Yes	Note 9
C.3.f	Yes	-------
C.3.g	Yes	Note 10
C.3.h	Yes	-------
C.3.i	Yes	Note 11
C.3.j	Yes	Note 12
C.3.k	Yes	Note 6
C.3.l	Yes	Note 17
C.3.m	Yes	-------
C.3.n	Yes	
C.3.o	Yes	--------
C.3.p	Yes	Note 18
C.4.a	Yes	Note 16
C.4.b	Yes	Note 13
C.4.c	Yes	Note 14
C.4.d	Yes	Note 7
C.4.e	Yes	-------
C.5.a	Yes	Notes 7, 15
C.5.b	Yes	Notes 7, 15
C.5.c	Yes	Notes 7, 15
C.5.d	Yes	Notes 7, 15
C.6.a	Yes	Notes 7, 15

TABLE 6.5-3
(Sheet 2 of 4)

COMPLIANCE OF CONTROL ROOM EMERGENCY FILTRATION SUBSYSTEM TO REGULATORY GUIDE 1.52, REV. 2, MARCH 1978

NOTE 1: The control room emergency exhaust unit is located in environmental zone CB-3 at Elevation 75' where the total of 40-year integrated radiation level is less than or equal to 1.0×10^3 rads. The other system components upstream and downstream of the unit are also located in "mild" radiation environmental zones.

NOTE 2: Demisters are not included in the filter design. The filter components for each redundant train include prefilters, an electric air heater, upstream HEPAs, an activated carbon adsorber bank, and downstream HEPAs. The piping which processes outside makeup air to the filters includes long vertical runs which will remove droplets entrained in the air stream.

NOTE 3: No accident which will cause pressure surges is postulated for the area in which the filters are located.

NOTE 4: The flow rate for each redundant filter train is indicated in the control room. High and low flow are alarmed in the control room. The pressure drop across the combined internal components for each filter train is indicated in the control room with high differential pressure generating an alarm. Pressure drops across each individual filter component are indicated locally in the control room HVAC equipment room. Additional details on system instrumentation are provided in Subsection 6.5.1.5c.

NOTE 5: The prefilters, air heaters, and HEPA filters were designed, constructed, and tested per ANSI N509-1980.

NOTE 6: The filter system design includes a low-flow air bleed system. A 4-inch diameter pipe with a manual isolation valve is provided to cross-connect the two redundant filter units at a point downstream of the carbon adsorber sections. During single train operation, this alignment will provide approximately 50 cfm of air flow through the carbon adsorber section of the inactive train. The configuration ensures that this low-flow cooling air is low humidity recirculation air. This satisfies the 70 percent maximum relative humidity criteria given a single active failure.

NOTE 7: System operability verification and surveillance testing will be performed in accordance with plant Technical Specifications.

TABLE 6.5-3
(Sheet 3 of 4)

COMPLIANCE OF CONTROL ROOM EMERGENCY FILTRATION SUBSYSTEM
TO REGULATORY GUIDE 1.52, REV. 2, MARCH 1978

NOTE 8: The atmosphere cleanup system housings and ductwork have been designed to exhibit on test a maximum, total leakage rate as defined in Section 4.12 of ANSI N509-1980. Leak tests are performed in accordance with Section 6 of ANSI N510-1980.

NOTE 9: The filter and adsorber mounting frames for Train A, except for the prefilter mounting frame, were designed and constructed prior to the issuance of this Regulatory Guide and meet the intent of ANSI N509-1980. The filter and adsorber mounting frames for Train B along with the prefilter mounting frame for Train A were designed and constructed per ANSI N509-1980.

NOTE 10: The filtration unit for Train A including floor/drains, etc., were designed and constructed prior to issuance of this Regulatory Guide and meet the intent of ANSI N509-1980. The Train B filtration unit was designed and constructed to ANSI N-509-1980.

NOTE 11: Carbon has been qualified to ANSI N509-1980*. The Train B adsorber cell is a four-inch deep bed with a minimum residence time of 0.25 seconds. The design iodine removal efficiency is consistent with that of a two-inch deep bed (95%).

NOTE 12: The adsorber cells for Train A have been designed and constructed per ANSI N509-1976 and meet the intent of ANSI N509-1980. The adsorber cells for Train A have been tested per ANSI N510-1980. The adsorber cell for Train B has been designed, constructed, and tested per ANSI N509-1980.

NOTE 13: Train B replaceable components are designed for removal from outside the filter unit.

NOTE 14: Meets requirements of Section 4.11 of ANSI N509-1980.

NOTE 15: All in-place testing/inspection is per ANSI N510-1980 requirements, with acceptance criteria of ANSI N509-1980 as applicable.

*Information is historical and applies to the original charcoal installed. All testing of replacement charcoal and future required periodic testing of charcoal will be in accordance with ASTM D3803-1989.

TABLE 6.5-3
(Sheet 4 of 4)

COMPLIANCE OF CONTROL ROOM EMERGENCY FILTRATION SUBSYSTEM TO REGULATORY GUIDE 1.52, REV. 2, MARCH 1978

NOTE 16: The system layout for Train A was performed prior to the issuance of this Regulatory Guide, however, it meets the requirements of Section 4.7 of ANSI N509-1980 and the intent of Subsection 2.3.8 of ERDA 76-21. Train B meets the requirements of ANSI N509-1980 and ERDA 76-21.

NOTE 17: The system fan motor, mountings and ductwork connections for Train A were procured prior to issuance of this Regulatory Guide. However, they are designed and constructed to meet the intent of Sections 5.7 and 5.8 of ANSI N509-1976. They are field tested/inspected per Section 8 of ANSI N510-1980, with the acceptance criteria of ANSI N509-1980. Train B has been designed, constructed, and tested per ANSI N509-1980.

NOTE 18: The system dampers were procured prior to the assurance of this Regulatory Guide. However, they are designed, constructed, and tested per the intent of Section 5.9 of ANSI N509-1976 and ANSI N510-1975. They are field tested/inspected as a part of ductwork per ANSI N510-1980 requirements, with the acceptance criteria of ANSI N509-1980.

TABLE 6.5-4
(Sheet 1 of 3)

CONTAINMENT ENCLOSURE EMERGENCY AIR CLEANING SYSTEM MATERIALS

	Component	Parameter
1)	Moisture Separator (2)	
	Filtration efficiency at design air flow (2000 cfm)	99% entrained moisture
	Water capacity at design air flow (2000 cfm)	3 lb/min entrained water
	Case Material	Type 304 Stainless Steel
	Media Material	Type 304 Stainless Steel & Fiberglass
2)	HEPA Filter (4)	
	Efficiency at rated flow (2000 cfm), 20% and 120% rated flow	99.97% at 0.3 microns of DOP
	Dust holding capacity	4 lbs.
	Case Material	409 Stainless Steel
	Media Material	Fiberglass
	Separator Material	None
3)	Carbon Adsorber	
	a. Lot Requirements (See Note 2)	Efficiency
	Low Temperature Ambient Pressure Methyl Iodide at 95% RH and 25°C	99%
	Low Temperature Ambient Pressure Elemental Iodine at 95% RH and 30°C	99.9%
	High Temperature Ambient Pressure Methyl Iodide at 95% RH and 80°C	99%

TABLE 6.5-4
(Sheet 2 of 3)

CONTAINMENT ENCLOSURE EMERGENCY AIR CLEANING SYSTEM MATERIALS

Component	Parameter
Methyl Iodine in Containment at 95% RH 130°C, 3.7 Atm., 1 Hour Load and 4 Hour Post-Sweep.	98%
b. Batch Requirements (See Note 2)	
Low Temperature Ambient Pressure Methyl Iodide at 95% RH and 30°C	97%
High Temperature Ambient Pressure Methyl Iodide at 95% RH and 80°C Except Pre and Post Sweep at 25°C	99%
Elemental Iodide Retention at 180°C	99.9% Loading 99.5% Retentivity
Media Carbon	Activated Coconut Shell
Impregnating Material	KI_3
Ignition Temperature (ASTM D3466)	330°C
Density (ASTM D2854)	0.38g/cc (min)
Hardness (ASTM D3802)	97%
Mesh Size (ASTM D2862)	5% Maximum Retention on 8 90-100% thru 8 on 16 (8x12 Mesh 40-60%) (12x16 Mesh 40-60%) 5% maximum thru 16; 1% Maximum thru 18
Depth of carbon bed	4 inches
Total weight of carbon	804 lbs
Carbon Bed Envelope Material	Type 304 Stainless Steel
4) Filter Mounting Frames	Type 304 Stainless Steel

TABLE 6.5-4
(Sheet 3 of 3)

CONTAINMENT ENCLOSURE EMERGENCY AIR CLEANING SYSTEM MATERIALS

Component	Parameter
5) Filter System Housing	Epoxy Coated Carbon Steel
6) Ductwork	Galvanized Steel
7) Fan	Carbon Steel

Note 1: Refer to Chapter 15 Appendix B for filter efficiencies assumed for design basis accidents.

Note 2: Testing information is historical and applies to the original charcoal installed. All testing of replacement charcoal and future required periodic testing of charcoal will be in accordance with ASTM D3803-1989.

TABLE 6.5-5
(Sheet 1 of 3)

FUEL STORAGE BUILDING EMERGENCY AIR CLEANING SYSTEM MATERIALS

	Component	Parameter
1)	Moisture Separator (12)	
	Filtration efficiency at design air flow (17,000 cfm)	99% entrained moisture
	Water capacity at design air flow (17,000 cfm)	24 lb/min entrained water
	Case Material	Type 304 Stainless Steel
	Media Material	Type 304 Stainless Steel & Fiberglass
2)	Medium Efficiency Filter (12)	
	Filtration efficiency at design air flow (17,000 cfm)	80%
	Dust Holding Capacity	340 grams
	Case Material	409 Stainless Steel
	Media Material	Fiberglass
	Separator Material	None
3)	HEPA Filters (24)	
	Efficiency at rated flow (17,000 cfm), 20% and 120%	99.97% at 0.3 microns of DOP rated flow
	Dust Holding Capacity	4 lbs
	Case Material	409 Stainless Steel
	Media Material	Fiberglass
	Separator Material	None

TABLE 6.5-5
(Sheet 2 of 3)

FUEL STORAGE BUILDING EMERGENCY AIR CLEANING SYSTEM MATERIALS

Component	Parameter
4) Carbon Adsorber	
a. Lot Requirements (See Note 2)	Efficiency
Low Temperature Ambient Pressure Methyl Iodide at 95% RH and 25°C	99%
Low Temperature Ambient Pressure Elemental Iodine at 95% RH and 30°C	99.9%
High Temperature Ambient Pressure Methyl Iodine at 95% RH and 80°C	99%
b. Batch Requirements (See Note 2)	
Low Temperature Ambient Pressure Methyl Iodide at 95% RH and 30°C	97%
High Temperature Ambient Pressure Methyl Iodide at 95% RH and 80°C Except Pre and Post Sweep at 25°C	99%
Elemental Iodine Retention at 180°C	99.9% Loading 99.5% Retentivity
Media Carbon	Activated Coconut Shell
Impregnating Material	KI_3
Ignition Temperature (ASTM D3466)	330°C
Density (ASTM D2854)	0.38 g/cc (min)
Hardness (ASTM D3802)	97%

TABLE 6.5-5
(Sheet 3 of 3)

FUEL STORAGE BUILDING EMERGENCY AIR CLEANING SYSTEM MATERIALS

	Component	Parameter
	Mesh Size (ASTM D2862)	5% Maximum Retention on 8 90-100% thru 8 on 16 (8x12 Mesh 40-60%) (12x16 Mesh 40-60%) 5% Maximum thru 16; 1% Maximum thru 18
	Depth of carbon bed	4 inches
	Total weight of carbon	6500 lbs
	Carbon bed envelope material	Type 304 Stainless Steel
5)	Filter Mounting Frames	Type 304 Stainless Steel
6)	Filter System Housing	Epoxy Coated Carbon Steel
7)	Ductwork	Galvanized Steel
8)	Fan	Carbon Steel

Note 1: Refer to Chapter 15 Appendix B for filter efficiencies assumed for design basis accidents.

Note 2: Testing information is historical and applies to the original charcoal installed. All testing of replacement charcoal and future required periodic testing of charcoal will be in accordance with ASTM D3803-1989.

TABLE 6.5-6
(Sheet 1 of 3)

CONTROL ROOM EMERGENCY MAKEUP AIR AND FILTRATION SUBSYSTEM MATERIALS

TRAIN A FILTER

Component	Parameter
1) Medium Efficiency Filter (1)	
Filtration Efficiency at Design Air Flow (1100 cfm)	80%
Dust Holding Capacity	340 grams
Case Material	409 Stainless Steel
Media Material	Fiberglass
Separator Material	None
2) HEPA Filters (4)	
Filtration Efficiency at Design Air Flow (1100 cfm), 20% and 120% Rated Flow	99.97% at 0.3 Microns of DOP
Dust Holding Capacity	4 lbs.
Case Material	409 Stainless Steel
Media Material	Fiberglass
Separator	None
3) Carbon Adsorber (7 trays)	
a. Lot Requirement (See Note 1)	
Methyl Iodide, 80°C, 95% RH (ASTM D3803)	1% Penetration, Maximum
Molecular Iodine, 30°C 95% RH (ASTM D3803)	0.1 Penetration, Maximum

TABLE 6.5-6
(Sheet 2 of 3)

CONTROL ROOM EMERGENCY MAKEUP AIR AND FILTRATION SUBSYSTEM MATERIALS

TRAIN A FILTER

Component	Parameter
b. Batch Requirement (See Note 1)	
Molecular Iodine, 180°C (ASTM D3803)	99.5 Retentivity, Minimum
Methyl Iodine, 30°C, 95% RH (ASTM D3803)	3% Penetration, Maximum
Media Carbon	Activated Coconut Shell
Impregnating Material	Iodine Salts & Tertiary Amines
Ignition Temperature (ASTM D3466)	330°C Minimum
Density (ASTM D2854)	0.38 g/cc Minimum
Hardness (ASTM D3802)	92 Minimum
Mesh Size (ASTM D2862)	Retained on #6 Sieve: 0.1% Maximum Retained on #8 Sieve: 5.0% Maximum Through #8, on #12 Sieve: 60% Maximum Through #12, on #16 Sieve: 40% Minimum Through #16 Sieve: 5.0% Maximum Through #18 Sieve: 1.0% Maximum
Depth of Carbon Bed	2 Inches (tray type-6 trays, 1 test tray)
Total Weight of Carbon	300 lbs. (not including test canisters)
Carbon Tray Envelope Material	Type 304 Stainless Steel

TABLE 6.5-6
(Sheet 3 of 3)

CONTROL ROOM EMERGENCY MAKEUP AIR AND
FILTRATION SUBSYSTEM MATERIALS

TRAIN A FILTER

Component	Parameter
4) Filter Mounting Frames	Type 304 Stainless Steel
5) Filter System Housing	Epoxy Coated Carbon Steel
6) Ductwork/Piping	Galvanized Steel/Carbon Steel
7) Fan	Carbon Steel Housing, Aluminum Blades and Hub

TRAIN B FILTER

Same as Train A filters with the following exceptions:

1) HEPA Filters (2)	
2) Carbon Adsorber	
Depth of Carbon Bed	4 Inches
Total Weight of Carbon	390 lbs. (not including test canisters)
3) Filter System Housing	Type 304 Stainless Steel
4) Fan	Carbon Steel

Note 1: Testing information is historical and applies to the original charcoal installed. All testing of replacement charcoal and future required periodic testing of charcoal will be in accordance with ASTM D3803-1989.

TABLE 6.5-7

VOLUMES OF CONTAINMENT ENCLOSURE AREAS

AREA	VOLUME (ft^3)
Containment Enclosure Annulus	524,344
Electrical Penetration Areas	84,035
Mechanical Penetration Areas	70,320
RHR and SI Equipment Vaults	102,816
Containment Enclosure Equipment Area	92,568
Charging Pump Areas	12,000

6.6 IN-SERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS

The in-service inspection program for Class 2 and 3 components is defined in the Technical Specifications. In general, this program meets all the requirements of the ASME Code, Section XI, Edition and Addenda as required in 10 CFR 50.55a. The references to ASME Section XI contained in paragraphs 6.6.6 are from the 1995 Edition through the 1996 Addenda.

6.6.1 Components Subject to Examination

All Class 2 components that do not meet the exemption requirements of IWC-1220, except the containment spray piping beyond the last downstream valve, will be examined in accordance with the requirements of IWC-2000, using the methods listed in Table IWC-2500-1.

Containment spray piping beyond the last downstream valve is exempted from in-service inspection for the following reasons:

a. These lines are normally empty and are required to function only in the unlikely event of a major LOCA or a main steam line break.

b. These lines are not subject to fluctuating stresses, so that propagation of cracks due to defects in the pipe is unlikely.

c. Spray piping welds are examined by radiography during fabrication, so that large defects that could cause catastrophic failure during system operation will be detected and repaired during fabrication and installation of the piping.

d. This piping contains hundreds of open nozzles so that splits or cracks which do not directly cause catastrophic failure would not have a marked effect on the function of the system.

e. System pressure is low, so that pressure stresses in the piping would be unlikely to cause failure in the presence of small defects.

All Class 3 components shall be examined per the requirements of Subsection IWD.

Safety classifications of all components are presented in Subsection 3.2.

6.6.2 Accessibility

The design and arrangement of Class 2 and Class 3 components provides adequate clearances to conduct the required examinations at the Code-required inspection interval. In general, access has been provided for examination of welds by proper layout of piping and equipment. Where surface or volumetric examination is required, insulation has been designed to be quickly and easily removed and replaced. Inaccessible Class 3 lines have been provided with taps to permit pressure decay tests in lieu of visual examination, as permitted in IWD-2000. Provisions for pump tests, as required by IWP-2000, have been made by incorporating suction and discharge pressure-measuring devices or flow measurement orifices adjacent to each safety class pump. Clearances around the pumps have been maintained to permit access for rotation speed and vibration measurements. Provisions for valve tests, as required by IWV-2000, have been incorporated by including valve position indicators in the design, as required, and by maintaining clearances for access to the valves.

6.6.3 Examination Techniques and Procedures

In general, where volumetric examination is required to be performed, manual ultrasonic techniques are employed rather than radiography. Where equipment configuration, layout, or other considerations, make it undesirable to perform a manual ultrasonic examination, an automatic technique is employed. Radiography is employed if ultrasonic examination fails to give meaningful results. For components of the reactor coolant pressure boundary, see Subsection 5.2.4.

Where surface examination is required, carbon steel components are examined by the magnetic particle method, and stainless steel components by the dye penetrant method.

The steam generator tubing is examined by the eddy current method (see Subsection 5.4.2.5).

All examination techniques are qualified to Section V or Section XI of the Code, using qualified personnel.

6.6.4 Inspection Intervals

The inspection schedule for Class 2 and Class 3 components is in accordance with subarticles IWC-2400 and IWD-2400, and is included in the Technical Specifications.

6.6.5 Examination Categories and Requirements

The examination categories and requirements for Class 2 components are in agreement with Section XI, Article IWC-2000.

For Class 3 components, the requirements of Section XI, Article IWD-2000, are complied with, as applicable.

6.6.6 Evaluation of Examination Results

a. Evaluation of Class 2 examination results will comply with the requirements of IWC-3000 of Section XI. In general, indications detected during in-service examination that exceed the acceptance standards of IWC-3000 will require repair in accordance with IWA-4000 of Section XI.

b. Repair procedures for Class 2 components comply with the requirements of IWA-4000 of Section XI.

c. Evaluation of Class 3 components is consistent with IWD-3000 of Section XI which also applies the rules of IWB-3000. Visual examinations of Class 3 components consist of evaluations for leakage and integrity of structural attachments. Defects will be evaluated per IWB-3000 and repairs performed in accordance with the requirements of IWA-4000 of Section XI.

   In general, evaluation of leakage in Class 3 piping will be consistent with the intent of IWB-3000, since defects resulting in perceptible leakage are not acceptable.

d. Defects in Class 3 pressure boundary components will be removed or reduced to acceptable size by grinding, cutting, or drilling. If the defect and the repair do not encroach on the minimum wall, then no repair welding will be required. Repaired surfaces will be smoothly blended into the surrounding material with no discontinuities. If the defect or its removal encroaches upon the minimum wall thickness, the area will be built up to the minimum wall thickness by welding, and the surfaces smoothly blended into the surrounding surface. If the built-up material thickness of carbon steel components that have been repaired by welding is greater than ¾", post-weld heat treatment is required.

6.6.7 System Pressure Tests

The system pressure testing program for Class 2 components complies with the requirements of IWC-5000 of Section XI.

The system pressure testing program for Class 3 components complies with the requirements of IWD-5000 of Section XI.

6.6.8 Augmented In-Service Inspection to Protect Against Postulated Piping Failures

As stated in Section 3.6(B).2.1.a.4, for main steam and feedwater piping penetrating containment, no breaks were postulated between the first pipe whip restraint inside the containment and the five-degree restraint outside containment. To protect against postulated piping failures, this piping is subject to augmented inservice inspection as defined in the inservice inspection program required by the Station Technical Specifications.

The augmented inspection consists of examination of essentially 100% of the longitudinal and circumferential piping welds within the defined boundaries during each inspection interval. The augmented lines are:

MS-4000-02-30"	FW-4606-03-18"
MS-4000-41-30"	FW-4606-04-16"
MS-4001-02-30"	FW-4607-03-18"
MS-4001-41-30"	FW-4607-04-16"
MS-4002-02-30"	FW-4608-03-18"
MS-4002-37-30"	FW-4608-04-16"
MS-4003-02-30"	FW-4609-03-18"
MS-4003-37-30"	FW-4609-04-16"

normal) flow control valves for the A and C steam generator will be powered by the A train with B and D steam generators' valves powered by the B train. Backup flow control valves will be powered from the opposite emergency power train. These valves can be controlled from either the main control board or the remote shutdown panel using safety-grade controls.

The five valves in the EFW pump discharge header are furnished with gear operators so that a concern for power diversity is not applicable.

The design and operation of the EFW system has been reviewed regarding the occurrence of hydraulic instabilities, characterized as water hammer. The EFW system is connected to the main feedwater system through stop-check valves outside the containment. The flow regulating valves in each EFW line are normally open, and are sized to pass the required flow under accident conditions. The only action required to establish EFW flow is to start the pumps. One pump has sufficient capacity to furnish the required flow to the Nuclear Steam Supply System.

An analysis of the EFW system has established that its function and performance are not affected by the common causes for loss of flow resulting in water hammer, such as pump trip, or rapid valve closure. A pressure transient in the main feedwater system resulting from a pipe break, pump trip, and/or valve closure would be dissipated before flow is established in lines to each SG. A trip of one pump will not affect the capability of the other pump to provide flow to the intact SGs. The only automatic valve closure in the EFW system would occur in the line to a faulted SG. During operation of the EFW system, the plant operators can initiate any changes in flow to each SG, as required.

The EFW pumps' supply and recirculation line piping runs from nozzles on the Condensate Storage Tank (CST) to the EFW pumphouse. The CST nozzles and adjacent piping are protected by a seismic Category I structure which is part of the CST enclosure and tornado-missile shield. The piping is routed underground and runs below grade into the EFW pumphouse, also a seismic Category I structure.

The EFW pump recirculation line to the CST, is designed to ASME Code Section III, Class 3 seismic Category I requirements. Valves FW-V346 and FW-V347 are administratively opened for EFW pump-surveillance testing and as required to ensure minimum EFW pump flow during system operation.

The water lines to the oil cooler are designed to seismic Category I requirements. Water lines from the oil cooler are designed to ASME Code Section III, Class 3, seismic Category I requirements. The breakdown orifice in the line to the oil cooler limits the flow to 2-3 gpm. This flow was considered in sizing the pump capacity. In the unlikely event of pipe failure, this flow will easily be handled by the pump room floor drains.

An accident analysis for this system in conjunction with the loss of the Main Feedwater System is provided in Chapter 15. A failure analysis of the

Emergency Feedwater System following a feedwater pipe break is provided in Table 6.8-2.

6.8.4 Tests and Inspections

Prior to initial plant startup, the Emergency Feedwater System is hydrostatically tested in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 3, and pre-operationally tested as described in Chapter 14.

Periodic testing in accordance with Technical Specifications will be performed during normal plant operation. During periodic surveillance testing of the EFW pumps, manual valve alignments will be required. Only one EFW pump will be tested at a time. Because each EFW pump is capable of providing 100 percent of required flow, full system flow requirements will be available at all times. Automatic indication of EFW pump inoperable status is provided as discussed in subsection 7.1.2.6.

6.8.5 Instrumentation Requirements and Controls

The Emergency Feedwater System will be actuated automatically on loss of offsite power, low-low level in any of the steam generators, safety injection signals or an ATWS mitigation system actuation signal. The engineered safety feature actuation system details are presented in Section 7.3. The ATWS mitigation system is discussed in Section 7.6. Manual controls for the turbine-driven pump steam supply valves are located at the main control board (MCB), as well as at the remote safe shutdown (RSS) panel. For the motor-driven pump, the controls are located at the MCB and in the switchgear room. The suction and discharge pressures of both pumps are indicated locally. Pump discharge pressure and CST level indication are provided on the MCB. Low suction pressures are alarmed at the MCB.

Flow indications for all four individual emergency feedwater lines are provided. Safety grade flow orifice instrumentation readouts are displayed at the MCB. The instruments are powered from the safety grade inverters - A and C steam generators on the Train A inverter, and B and D steam generators on the Train B inverter. These instrumentation channels meet or exceed the requirements for Design Category 2 instrumentation as provided in Subsection 7.5.5. The design details of the accident monitoring instrumentation are presented in Table 7.5-1. Flow venturi are also provided in each emergency feedwater line. Two of the four flow venturi instrumentation readouts are displayed at a RSS panel, and the remaining two flow venturi instrumentation readouts are displayed at a second RSS panel.

A sustained high-flow condition in any of the lines is indicative of a line break. A break isolation is incorporated so that the affected line will be isolated by automatically closing the motor-operated valves on high flow signals from redundant flow instrumentation. Break isolation is required to conserve CST inventory and to provide the required flow to the intact steam generators. High flow alarms are also provided to alert the operator to this

REV. 07

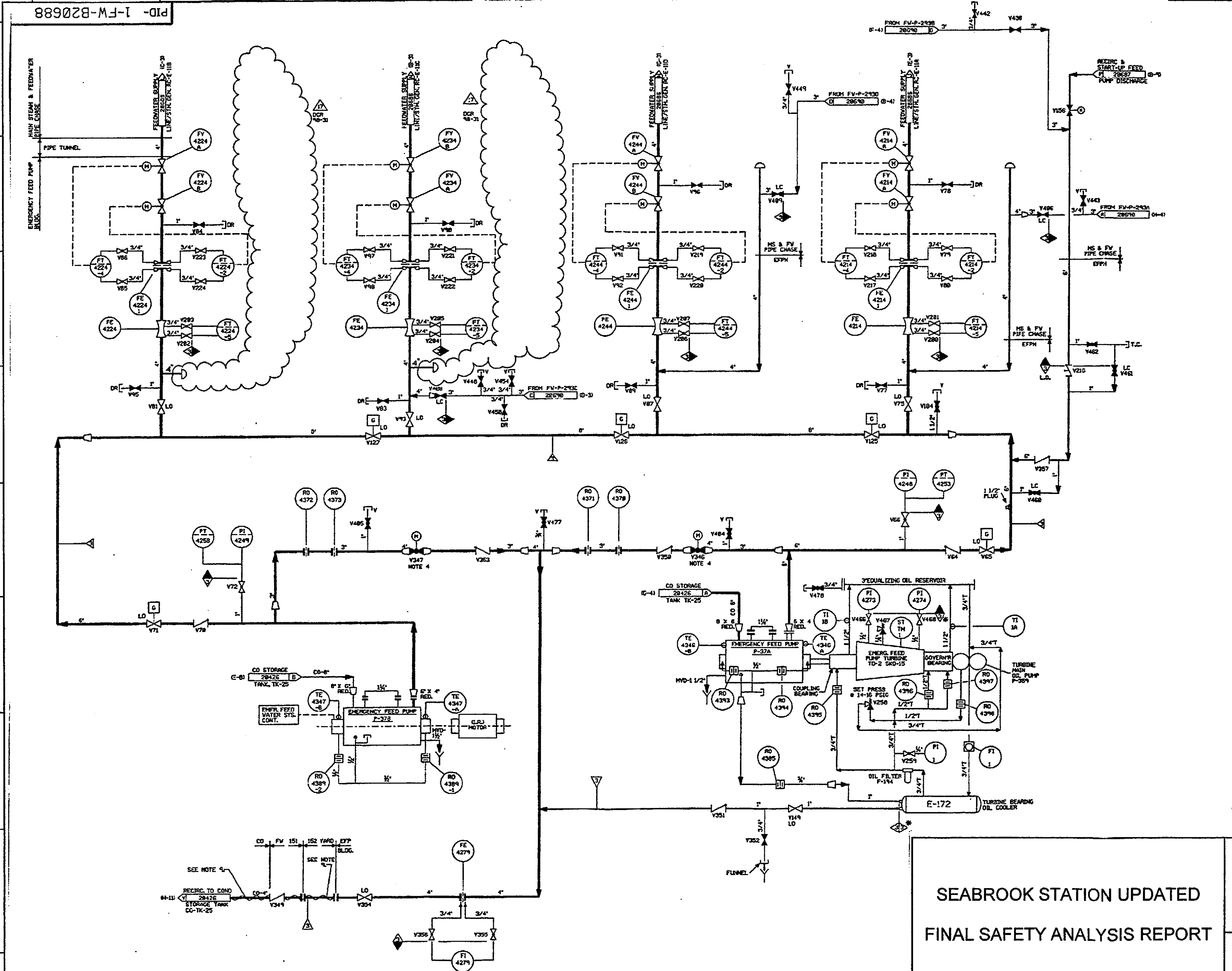
FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID-LEGEND 1 AND P&ID-LEGEND 2.

NOTES:

1. PIPING, VALVES, EQUIPMENT & INSTRUMENT TAG NUMBERS ARE PREFIXED BY DESIGNATION OF "FW" UNLESS NOTED OTHERWISE.
2. WORK THIS DWG WITH 20684 THRU 20687, & 20689.
3. VENTS & DRAINS CODE BREAKS ARE AT THE DOWNSTREAM END OF THE OUTER ISOLATION VALVE, PER 1-NHY-805111 UNLESS NOTED OTHERWISE.
4. FW-V346, V347 ARE NORMALLY CLOSED MOTOR OPERATED GLOBE VALVES WHICH ARE OPERABLE FROM THE MAIN CONTROL BOARD & REMOTE SAFE SHUTDOWN PANEL.
5. △ INDICATES REVISION LEVEL
6. ALL PIPING AND COMPONENTS ASSOCIATED WITH P-37A ARE ASME III CL. 3, ALL PIPING AND COMPONENTS ASSOCIATED WITH TD-2 ARE ANS-3.
7. DELETED
8. DELETED
9. THIS HEAT TRACING IS NONFUNCTIONAL

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED



9763-F-202876	12	FEEDWATER SYSTEM P & I DIAGRAM
REF. DRAWING NO.	REV	TITLE

SEABROOK STATION UPDATED

FINAL SAFETY ANALYSIS REPORT

EMERGENCY FEEDWATER SYSTEM DETAILS

1-FW-B20688

FIGURE 6.8-2

Seabrook Station

North Atlantic

Updated Final Safety Analysis Report

Revision 7

CHAPTER 7

INSTRUMENTATION AND CONTROLS

7.1 INTRODUCTION

This chapter presents the various plant instrumentation and control systems by relating the functional performance requirements, design bases, system descriptions, design evaluations, and test and inspections for each. The information provided in this chapter emphasizes those instruments and associated equipment which constitute the protection system as defined in IEEE Standard 279-1971 "IEEE Standard: Criteria for Protection System for Nuclear Power Generating Stations."

The primary purpose of the instrumentation and control systems is to provide automatic protection and exercise proper control against unsafe and improper reactor operation during steady-state and transient power operations (ANS Conditions I, II, III) and to provide initiating signals to mitigate the consequences of faulted condition (ANS Condition IV). ANS conditions are discussed in Chapter 15. Consequently, the information presented in this chapter emphasizes those instrumentation and control systems which are central to assuring that the reactor can be operated to produce power in a manner that ensures no undue risk to the health and safety of the public.

It is shown that the applicable criteria and codes, such as General Design Criteria and IEEE Standards, which are concerned with the safe generation of nuclear power are met by these instrumentation and control systems. (See Table 7.1-1 for a listing of applicable criteria.)

Review of Section 8.3, Onsite Power Systems, serves as necessary and sufficient background for evaluating the electrical integrity of plant instrumentation systems. Figures 8.3-2, 8.3-3 and 8.3-4 provide an overview of the distribution system with emphasis on vital and nonvital instrument buses and electrical separation divisions.

Definitions

Terminology used in this chapter is based on the definitions given in IEEE Standard 279-1971. In addition, the following definitions apply:

a. Degree of Redundancy - The difference between the number of channels monitoring a variable and the number of channels which, when tripped, will cause an automatic system trip.

b. Minimum Degree of Redundancy - The degree of redundancy below which operation is prohibited, or otherwise restricted by the Technical Specifications.

c. Cold Shutdown Condition - When the reactor is sub-critical by at least 1 percent $\Delta k/k$ and T_{avg} is ≤200°F. T_{avg} is defined as the average temperature across the reactor vessel, as measured by the hot and cold leg temperature detectors.

d. Hot Shutdown Condition - When the reactor is sub-critical, by an amount greater than or equal to the margin as specified in the applicable Technical Specification, and T_{avg} is greater than or equal to the temperature as specified in the applicable Technical Specification.

e. Phase A Containment Isolation - Closure of all nonessential process lines which penetrate Containment, initiated by the safety injection signal.

f. Phase B Containment Isolation - Closure of remaining process lines, initiated by containment Hi-3 pressure signal (process lines do not include Engineered Safety Features lines).

g. Single Failure - Any single event within the protection system which results in a loss of proper protective action at the system level when required. Single failure includes single credible malfunctions or events that cause a number of consequential component, module or channel failures.

h. DNBR - (Departure from Nucleate Boiling Ratio) - The ratio of the critical heat flux (defined as the transition from nucleate boiling to film boiling) to the actual local heat flux.

i. System Response Times:

1. Reactor Trip System Response Time

   The time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

2. Engineered Safety Features Actuation System Response Time

   The time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

j. Reproducibility - This definition is taken from Scientific Apparatus Manufactures Association (SAMA) Standard PMC-20.1-1973, Process Measurement and Control Terminology: "The closeness of agreement among repeated measurements of the output for the same value of input, under normal operating conditions over a period of time, 7.1.1.3 Instrumentation and Control System Designs

All systems discussed in Chapter 7 have definitive functional requirements developed on the basis of the Westinghouse NSSS design. Figures 7.2-1 through 7.2-15 define scope interface. Regardless of the supplier, the functional requirements necessary to assure plant safety and proper control are clearly delineated.

7.1.1.4 Plant Comparison

System functions for all systems discussed in Chapter 7 are similar to those discussed in the comparisons provided in Section 1.3.

7.1.2 Identification of Safety Criteria

Subsection 7.1.2.1 presents design bases for the systems given in Subsection 7.1.1.1. Design bases for nonsafety-related systems are provided in the sections which describe the systems. Conservative considerations for instrument errors are included in the accident analyses presented in Chapter 15. Functional requirements, developed on the basis of the results of the accident analyses, which have utilized conservative assumptions and parameters, are used in designing these systems, and a preoperational testing program verifies the adequacy of the design. Accuracies are discussed in Sections 7.2, 7.3 and 7.5.

The documents listed in Table 7.1-1 were considered in the design of the systems given in Subsection 7.1.1. In general, the scope of these documents is given in the document itself. This determines the systems or parts of systems to which the document is applicable. A discussion of compliance with each document for systems in its scope is provided in the referenced sections given in Table 7.1-1 for each criterion. Because some documents were issued after design and testing had been completed, the equipment documentation may not meet the format requirements of some standards. Justification for any exceptions taken to each document for systems in its scope is provided in the referenced sections.

7.1.2.1 Design Bases

a. Reactor Trip System

The Reactor Trip System acts to limit the consequences of Condition II events (faults of moderate frequency), such as loss of feedwater flow, by at most, a shutdown of the reactor and turbine, with the plant capable of returning to operation after corrective action. The Reactor Trip System features impose a limiting boundary region to plant operation which ensures that the reactor safety limits are not exceeded during Condition II, III and IV events, and that these events can be accommodated without developing into more severe conditions. Reactor trip setpoints are given in the Technical Specifications.

The design requirements for the Reactor Trip System are derived by analyses of plant operating and fault conditions where automatic rapid control rod insertion is necessary to prevent or limit core or reactor coolant boundary damage. The design bases addressed in IEEE Standard 279-1971 are discussed in Subsection 7.2.1. The design limits for the Reactor Trip System are:

1. Minimum DNBR shall not be less than the safety analysis limit value as a result of any anticipated transient or malfunction (Condition II faults).

2. Power density shall not exceed the rated linear power density for Condition II faults. See Chapter 4 for fuel design limits.

3. The stress limit of the Reactor Coolant System for the various conditions shall be as specified in Chapter 5.

4. Release of radioactive material shall not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius as a result of any Condition III fault (10 CFR 20, Standard for Protection Against Radiation).

5. For any Condition IV fault, release of radioactive material shall not result in an undue risk to public health and safety (10 CFR 100, Reactor Site Criteria).

b. Engineered Safety Features Actuation System

The Engineered Safety Features Actuation System acts to limit the consequences of Condition II and III events (infrequent faults such as primary coolant spillage from a small rupture which exceeds normal charging system makeup and requires actuation of the Safety Injection System). The Engineered Safety Features Actuation System acts to mitigate Condition IV events (limiting faults, which include the potential for significant release of radioactive material).

The design bases for the Engineered Safety Features Actuation System are derived from the design bases given in Chapter 6 for the Engineered Safety Features. Design bases requirements of IEEE Standard 279-1971 are addressed in Subsection 7.3.1.2. General design requirements implemented are given below.

1. Automatic Actuation Requirements

The primary functional requirement of the Engineered Safety Features Actuation System is to receive input signals (information) from the various on-going processes within the reactor plant and Containment and to automatically provide, as output, timely and effective signals to actuate the various components and subsystems comprising the Engineered Safety

3. Low pressurizer pressure

All of the above sets of signals are redundant and physically separated and meet the requirements of IEEE Standard 279-1971.

The seismic and environmental qualification for protection system sensors and channels is discussed in Sections 3.10 and 3.11, respectively.

i. Bistable Trip Setpoints

Three values applicable to reactor trip and engineered safety features actuation are specified:

1. Safety limit

2. Allowable value

3. Nominal value.

The safety limit is the value assumed in the accident analysis and is the least conservative value.

The reactor trip setpoint limits specified in the Technical Specifications are the nominal values at which the reactor trips are set for each parameter. The setpoints have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences, and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents.

The methodology to derive the nominal trip setpoints is based upon statistically combining all of the uncertainties in the channels and applying this total uncertainty with margin in the conservative direction. Inherent to the determination of the trip setpoints is the determination of the magnitudes of the channel uncertainties. Sensors and other instrumentation used in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, allowable values for the reactor trip setpoints have also been specified in the Technical Specifications. Operation with setpoints less conservative than the nominal trip setpoint, but within the allowable value, is acceptable since an allowance has been made in the selection of that setpoint to accommodate this error without exceeding the value used in the safety analysis. A further discussion on setpoints is found in Subsection 7.2.2.2a.

Range selection for the instrumentation covers the expected range of the process variable being monitored consistent with its application. The design of the Reactor Protection and Engineered Safety Features Systems is such that the bistable trip setpoints do not require process transmitters to operate within 3 percent of the high and low end of their calibrated span or range. Functional requirements established for every channel in the Reactor Protection and Engineered Safety Features Systems stipulate the maximum allowable errors on accuracy, linearity, and reproducibility. The protection channels have the capability for, and are tested to ascertain that the characteristics throughout the entire span in all aspects are acceptable and meet functional requirement specifications. As a result, no protection channel operates normally within 3 percent of the limits of its specified span. The specific functional requirements for response time, setpoint, and operating span are based on the results and evaluation of safety studies carried out using data pertinent to the plant. Emphasis is placed on establishing adequate performance requirements under both normal and faulted conditions. This includes consideration of process transmitters margins such that even under a highly improbable situation of full power operation at the limits of the operating map (as defined by the high and low pressure reactor trip, ΔT overpower and overtemperature trip lines (DNB protection) and the steam generator safety valve pressure setpoint), adequate instrument response is available to ensure plant safety.

Setpoints for safety-related BOP bistable instruments are determined using the same methodology which Westinghouse used for the protection system setpoints. This methodology complies with the methodology outlined in Regulatory Guide 1.105 (Rev 1), as supplemented by the information presented in ISA Standard S67.04 (Draft F).

This methodology is applied to the determination of setpoints for all safety-related (Class 1E) bistable instruments, as distinct from the wording of Regulatory Guide 1.105, "instruments in systems important to safety." This distinction is beneficial to the safety of Seabrook Station because it provides a tangible and controllable distinction between those setpoints which require special attention and those which do not, while still insuring adequate safety consistent with the definition of "Class 1E" as stated in IEEE Std. 308.

Error allowances used in setpoint determination are supported by qualification testing, consistent with the postulated service conditions for the required protective function. Time limits for redundant instrumentation channels and actuation trains and incorporates physical and electrical separation to prevent faults in one channel from degrading any other protection channel.

(2) Separation recommendations for redundant instrumentation racks are not the same as those given in Regulatory Guide 1.75, Revision 2, for the control boards because of different functional requirements. Main control boards contain redundant circuits that are required to be physically separated from each other. However, since there are no redundant circuits which share a single compartment of an NSSS protection instrumentation rack and since these redundant protection instrumentation racks are physically separated from each other, the physical separation specified for the main control board does not apply.

However, redundant, isolated control signal cables leaving the protection racks are brought into close proximity elsewhere in the plant, such as the control board. It could be postulated that electrical faults, or interference, at these locations might be propagated into all redundant racks and degrade protection circuits because of the close proximity of protection and control wiring within each rack. Regulatory Guide 1.75 (Regulatory Position C.4) and IEEE Standard 384-1974 (Section 4.5(3)) provide the option to demonstrate by tests that the absence of physical separation could not significantly reduce the availability of Class 1E circuits.

Westinghouse test programs have demonstrated that Class 1E protection systems (nuclear instrumentation system, solid-state protection system and 7300 process control system) are not degraded by non-Class 1E circuits sharing the same enclosure. Conformance to the requirements of IEEE Standard 279-1971 and Regulatory Guide 1.75 has been established and accepted by the NRC based on the following which is applicable to these systems at the Seabrook site.

Tests conducted on the as-built designs of the nuclear instrumentation system and solid-state protection system were reported and accepted by the NRC in support of the Diablo Canyon application (Docket Nos. 50-275 and 50-323) [See Reference 5].

Westinghouse considers these programs as applicable to all plants, including Seabrook. Westinghouse tests on the 7300 Process Control System were covered in a report entitled, "Westinghouse 7300 Series Process Control System Noise Tests," subsequently reissued as Reference 2. In a letter dated April 20, 1977 (Reference 3) the NRC accepted the report in which the applicability of the Seabrook plant is established.

(3) The physical separation criteria for instrument cabinets within Westinghouse NSSS scope meet the recommendations contained in Section 5.7 of IEEE Standard 384-1974.

(b) The physical separation criteria for redundant safety-related sensing lines meet the recommendations contained in Regulatory Guide 1.151, with the following comments.

Redundant safety-related instrument sensing lines are not routed in the same area where they would be subject to external forces such as those due to jet impingement or pipe whip caused by an accident.

All components located in seismic areas are reviewed to ensure that they will not produce seismically-generated missiles that could damage safety-related components. The maximum feasible physical separation is maintained in areas where the redundant safety-related sensing lines are not subject to external forces caused by an accident.

b. Specific Systems

Independence is maintained throughout the system, extending from the sensor through to the devices actuated by the protective function. Physical separation or barriers are used to achieve separation of redundant transmitters. Separation of wiring is achieved using separate wireways, cable trays, conduit runs and containment penetrations for each redundant protection channel set. Redundant analog equipment is separated by locating modules in different protection rack sets. Each redundant channel set is energized from a separate AC power feed.

There are four separate process analog sets. Separation of redundant analog channels begins at the process sensors and is maintained in the field wiring, containment penetrations and analog protection cabinets to the redundant trains in the logic racks. Redundant analog channels are separated by locating modules in different cabinets. Nonprotection system outputs are through qualified isolators and are appropriately separated.

The separation criteria are discussed in Subsection 7.1.2.2a.

The covered wireways, formed from solid or punched sheet steel, and the conduits above or below each board-mounted device, comply with the separation criteria required by IEEE Standards. The MCB wiring is color-coded NEC-type SIS copper conductor with flame-retardant insulation, rated and qualified per IPCEA, UL, or IEEE standards. Nonmetallic components, such as terminal blocks, wire cleats, cable ties, receptacles, indicating light lenses, nameplates, etc., are furnished of materials meeting the nonflammability requirements of UL standards.

The separation criteria within the MCB are given in Table 8.3-10 and were based on analysis and testing as permitted by Subsection 4.3.3 of IEEE 420. This analysis and testing are documented in References 1 and 2 of Subsection 8.3.4. Each component is clearly identified with a distinctively colored permanent tag. Colored nameplates are employed on the exterior surfaces of the MCB to identify the component's function (see Subsection 7.1.2.3).

c. Fire Protection

For electrical equipment within the NSSS scope of supply, Westinghouse specifies noncombustible or fire retardant material and conducts vendor-supplied specification reviews of this equipment, which includes assurance that materials will not be used which may ignite or explode from an electrical spark, flame, or from heating, or will independently support combustion. These reviews also include assurance of conservative current-carrying capacities of all instrument cabinet wiring, which precludes electrical fires resulting from excessive overcurrent (I^2R) losses. For example, wiring used for instrument cabinet construction has teflon or tefzel insulation and is adequately sized, based on current carrying capacities set forth by the National Electric Code. In addition, fire retardant paint is used on protection rack or cabinet construction to retard fire or heat propagation from rack to rack. Braided sheathed material is noncombustible.

Subsections 8.3.1 and 8.3.2 describe design aspects used for BOP electrical equipment in the prevention of fires in cable system, including separation between redundant trains and voltage levels, cable material selection and cable sizing.

Details of the Plant Fire Protection System are provided in Subsection 9.5.1.

7.1.2.3 Physical Identification of Safety-Related Equipment

There are four separate protection sets identifiable with process equipment associated with the Reactor Trip and Engineered Safeguards Actuation Systems. A protection set may be comprised of more than a single process equipment cabinet. The color coding of each process equipment rack instrument nameplate coincides with the color code established for the protection set of which it is a part. At the logic racks, the protection set color coding for redundant channels is clearly maintained until the channel loses its identity in the redundant logic trains. The color-coded nameplates described below provide identification of equipment associated with protective functions and their channel set association:

Protection Set	Color Coding
I	RED
II	WHITE
III	BLUE
IV	YELLOW

The safety-related instrumentation and control system equipment throughout the plant is identified with colored tags or nameplates. The colored tags are consistent with the color coding of cables as defined in Subsection 8.3.1.3, and with the protection set nameplates as identified above.

	Equipment Nameplate Color
Train A or Channel 1	Red
Train B or Channel 2	White
Channel 3	Blue
Channel 4	Yellow
Accident Monitoring Instrumentation	Orange (see Section 7.5 for additional clarification)
Nonsafety-Related	Black
Remote Safe Shutdown	Purple

The equipment nameplate colors described above represent the color assigned to identify each separation group. In the original nameplate design, the nameplate background color was used to identify the separation group. As a result of labeling improvements, including the addition of bar codes, a such markings is to facilitate cable routing identification. Positive permanent identification of cables is made at all terminations. There are also identification nameplates on the input panels of the Solid-State Logic Protection System.

Instrument sensing lines comply with the identification and color coding requirements of Regulatory Guide 1.151, with the following exceptions and as described in Section 1.8:

a. Instrument sensing lines for a nuclear safety-related instrument are not color coded to identify its channel.

b. Instrument sensing lines are tagged with unique line numbers at the instrument side of the root valve, on the process side of the instrument shut-off valve, and on each side of an obstacle when the tubing runs through the obstacle, such as a wall or floor. Redundant safety-related lines are installed using engineered design packages to ensure that the separation criteria are met.

7.1.2.4 Conformance to Criteria

A listing of applicable criteria and the Updated FSAR sections where conformance is discussed is given in Table 7.1-1.

7.1.2.5 Conformance to Regulatory Guide 1.22

Periodic testing of the Reactor Trip and Engineered Safety Features Actuation Systems, as described in Subsections 7.2.2 and 7.2.3, complies with Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions," as described in this chapter and in Section 1.8.

Where the ability of a system to respond to a bona fide accident signal is intentionally bypassed to perform a test during reactor operation, each bypass condition is automatically indicated to the reactor operator in the main control room for the train in test. Test circuitry does not allow two trains to be tested at the same time so that extension of the bypass condition to the redundant system is prevented.

The actuation logic for the Reactor Trip and Engineered Safety Features Actuation System is tested as described in Sections 7.2 and 7.3. As recommended by Regulatory Guide 1.22, where actuated equipment is not tested during reactor operation it has been determined that:

a. There is no practicable system that would permit operation of the equipment without adversely affecting the safety or operability of the plant;

b. The probability that the protection system will fail to initiate the operation of the equipment is, and can be maintained, acceptably low without testing the equipment during reactor operation; and

c. The equipment can routinely be tested when the reactor is shutdown.

The list of equipment that cannot be tested at full power so as not to damage equipment or upset plant operation is as follows:

a. Manual actuation switches (reactor trip containment isolation phase A, containment spray actuation, and safety injection actuation)

b. Turbine trips (initiation and detection)

c. Main steam line isolation valves (close)

d. Main feedwater isolation valves (close)

e. Feedwater control valves (close)

f. Main feedwater pump trip solenoids

g. Reactor coolant pump component cooling water isolation valves (close)

h. Reactor coolant pump seal water return valves (close)

i. Charging header to cold leg isolation valves

j. Accumulator isolation valves (open).

k. Charging and letdown isolation valves (close)

l. Letdown heat exchanger component cooling water outlet isolation valve (close)

m. Reactor coolant pump trip (underfrequency)

n. Chemical and Volume Control System TK-1 outlet isolation valves (close)

o. Refueling Water Storage Tank TK-8 to charging isolation valves (open)

The justifications for not testing the above 15 items at full power are discussed below.

a. <u>Manual Actuation Switches</u>

These would cause initiation of their protection system function at power causing equipment damage or plant upset. It should be noted that the reactor trip function that is derived from the automatic safety injection signal is tested at power as follows:

The analog signals, from which the automatic safety injection signal is derived, are tested at power in the same manner as the other analog signals as described in Subsection 7.2.2.2c. The processing of these signals in the Solid-State Protection System (SSPS) wherein their channel orientation converts to a logic train orientation is tested at power by the built-in semi-automatic test provisions of the SSPS. The reactor trip breakers are tested at power as discussed in Subsection 7.2.2.2c.

b. Turbine

There is no practicable system design that would permit actual tripping of the turbine without adversely affecting the operability of the plant. The reactor trip functions that receive input from the turbine trip signal are tested prior to startup:

c. Closing the Main Steam Line Isolation Valves

Main steam isolation valves are routinely tested by full closure during refueling outages. Testing of the main steam isolation valves by full closure at power is not practical. As the plant power is increased, the coolant average temperature is programmed to increase. If the valves are closed under these elevated temperature conditions, the steam pressure transient would unnecessarily operate the steam generator relief valves and possibly the steam generator safety valves. The steam pressure transient produced would cause shrinkage in the steam generator level, which would cause the reactor to trip on low-low steam generator water level. Testing during operation will decrease the operating life of the valve.

Based on the above identified problems incurred with periodic testing by full closure of the main steam isolation valves at power and since (1) no practical system design will permit full closure of these valves without adversely affecting the safety or operability of the plant, (2) the probability that the protection system will fail to initiate the activated equipment is acceptably low due to testing up to final actuation, including partial stroke exercising, and (3) these valves will be routinely tested during refueling outages, the proposed resolution meets the guidelines of Section D.4 of Regulatory Guide 1.22.

d. Closing the Feedwater Isolation Valve

The feedwater isolation valves are routinely tested during refueling outages. Periodic testing of these feedwater isolation valves by closing them completely, or partially, at power would induce steam generator water level transients and oscillations which would trip the reactor. These transients conditions would be caused by system will fail to close these valves is acceptably low due to testing up to final actuation.

l. Letdown Heat Exchanger Component Cooling Water Outlet Isolation Valve (Close)

The letdown heat exchanger component cooling water outlet isolation valve is routinely tested during refueling outages. Closure of this valve causes thermal cycling which may cause damage to plant equipment. The probability that the protection system will fail to close this valve is acceptably low due to testing up to final actuation.

m. Reactor Coolant Pump Trip (Underfrequency)

There is no practical system design that would permit reactor coolant pump trip on underfrequency without adversely affecting the safety and operability of the plant. This function is provided to assure that in the event of decaying grid frequency and subsequent loss of power, the pumps have enough inertia to supply coolant flow on coastdown to cool the core. Refer to Subsection 7.2.1.1(4), Reactor Coolant Low Flow Trips, for additional discussion of this function.

n. Chemical and Volume Control System TK-1 Outlet Isolation Valves (Close)

Chemical and Volume Control System TK-1 outlet isolation valves are routinely tested during refueling outages. Closure of these valves during power operation may damage plant equipment due to pressure or temperature swings on the seal flow to the reactor coolant pumps or momentary loss of seal cooling flow. The probability that the protection system will fail to close these valves is acceptably low due to testing up to the final actuation device.

o. Refueling Water Storage Tank TK-8 to Charging Pump Isolation Valves (Open)

Refueling Water Storage Tank TK-8 to charging pump isolation valves are routinely tested during refueling outages. Opening of these valves during power operation may damage plant equipment due to pressure or temperature swings on the seal flow to the reactor coolant pumps or momentary loss of seal cooling flow. The probability that the protection system will fail to open these valves is acceptably low due to testing up the final actuation device.

7.1.2.6 Conformance to Regulatory Guide 1.47

The bypass indication system which does not perform functions essential to public health and safety during an accident is designed to meet paragraph 4.13 of IEEE Standard 279-1971 and the intent of Regulatory Guide 1.47.

Automatic indication of the bypassed and inoperable status of safety systems is provided on the Video Alarm System (VAS). The VAS continuously monitors the status of selected components. When a bypass or inoperable condition is detected which results in either redundant train being inoperable, an audible alarm is sounded and an incoming alarm at the system (train) level is displayed on the VAS. The VAS also provides an overall system level alarm if both of the redundant trains are bypassed or made inoperable. When the bypass or inoperable condition affects only one channel of the protection system then only a channel level (not a system level) VAS alarm will be provided. The VAS alarm will be provided for the first condition if multiple, administratively controlled, conditions exist concurrently as part of a procedure which results in the bypass of a safety function. Once activated, the automatically initiated system level indication will remain on until the actuating condition is cleared and the VAS reset.

In addition to the automatic VAS display, manual bypassed and inoperable status indication is provided on the main control board for those systems whose complexity increases the possibility of having frequent inoperable conditions that are not monitored by the automatic system. Indication is provided at the system level on a per train basis. Activation of an indicator on the bypassed and inoperable status panel is performed through manual actuation of its corresponding pushbutton. An exception to this is the system level indication associated with the diesel generator. These indicators are initiated automatically. The bypassed and inoperable status pushbuttons are also monitored by the VAS.

The VAS and the manual bypass and inoperable status indicators will automatically indicate the dependent auxiliary and safety systems that are made inoperable by a bypassed or inoperable safety system.

The design of the bypass indication system allows testing during normal operation.

The following rules are used to develop the system design, which satisfies Regulatory Guide 1.47:

vendor testing, in situ tests in operating plants with appropriately similar design, or by suitable type-testing. The Nuclear Instrumentation System detectors are excluded since they exhibit response time characteristics such that delays attributable to them are negligible in the overall channel response time required for safety.

The measurement of response time at the time intervals in the Technical Specification provides assurance that the protective and engineered safety features action function associated with each channel is completed within the time limit assumed in the accident analyses.

b. The periodic time interval discussed in IEEE Standard 338-1975, and specified in the plant Technical Specifications, is conservatively selected to ensure that equipment associated with protection functions has not drifted beyond its minimum performance requirements. If any protection channel appears to be marginal, or requires more frequent adjustments due to plant condition changes, the time interval will be decreased to accommodate the situation until the marginal performance is resolved.

c. The test interval discussed in IEEE Standard 338-1975 is developed primarily on past operating experience and modified, if necessary, to assure that system and subsystem protection is reliably provided. Analytic methods for determining reliability are not used to determine test interval.

d. Nonroutine tests, such as operational tests performed when one of the redundant channels is inoperable, may require the use of temporary jumpers, lifted leads, or other circuit modifications.

e. See Subsection 7.2.2.2c(11) for a discussion of exceptions to the single failure criterion while a channel is bypassed for maintenance or testing.

Based on the scope definition give in IEEE Standard 338-1975, no other systems described in Chapter 7 are required to comply with this standard.

For detailed discussions regarding the testing of the non-NSSS ESF and 1E power systems, refer to Sections 7.3 and 8.3, respectively.

7.1.2.12 Conformance to Regulatory Guide 1.151

The recommendations of ISA Standard S67.02, 1980, as endorsed by Regulatory Guide 1.151, have been followed for the design and installation of safety-related instrument sensing lines, with the exceptions and clarifications listed below. See Subsections 1.8, 7.1.2.2, 7.1.2.1, 7.1.2.3 and 7.7.2 for discussions of specific sections.

1. The instrumentation defined as Category 1 in UFSAR Section 7.5 is the only instrumentation considered to be required to monitor safety-related systems.

2. In clarification of paragraph 5.2.2 (2) of ISA S67.02, where instrument tubing penetrates a shield wall, measures have been taken to reduce potential personnel exposure for radiation "streaming" from radioactive sources unless the radiation from piping nearby

would be the larger source of exposure. These measures have included:

a. Locating penetration high enough to eliminate a concern from a radiation protection standpoint

b. Locating some penetrations to avoid a direct streaming path from the source of radiation

c. When the above two methods were not used, apply radiation absorbing penetration sealant.

3. The sensing lines from safety-related HVAC ductwork are designed to the same safety class as the ductwork, and are installed to the requirements of ANSI B31.1 Seismic Category I.

4. The sealed sensing lines for containment pressure, wide range reactor coolant pressure and the Reactor Vessel Level Indication System (RVLIS) are Safety Class 2 and installed to requirements of ANSI B31.1, Seismic Category I, rather than ASME Class 2 Seismic Category I, as recommended by Regulatory Position C.2.b or ISA S67.02, Section 4.1. The sealed, fluid-filled, instrumentation systems are in accordance with the standard Westinghouse design.

   The containment penetration sleeve is part of the BOP scope and is ASME Class 2.

5. Common instrument taps are used for redundant sensors for pressurizer pressure and RCS flow (high pressure tap only). This is in conformance with the standard Westinghouse design.

6. An evaluation has been performed of those instrument lines which were downgraded in accordance with the provisions of this regulatory guide from ASME Class 2 or 3 to ANSI B31.1. This evaluation was done to determine if the failure of any of these lines would affect the safety function of the associated system. Where a passive failure of the instrument line would adversely affect the safety function of the system, an inspection of the line has been done to equivalent quality assurance requirements of ANS Safety Class 2 lines. Also, the lines have been installed to Seismic Category I criteria. Hence, a passive failure of one of these lines is not postulated to occur.

7. Commercial grade dedication may be used, where applicable, instead of the requirements of ISA-S67.02 Sections 4.2.2 and 8.2 for components installed in accordance with ANSI B31.1 Seismic Category I.

8. See Subsection 7.2.2.2c(11) for a discussion of exceptions to the single failure criterion while a channel is bypassed, i.e., removed from service, for maintenance or testing.

7.1.3 References

1. Gangloff, W. C. and Loftus, W. D., "An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients," WCAP-7706-L, July 1971 (Proprietary) and WCAP-7706, July 1971. (Nonproprietary).

2. Marasco, F. W. and Siroky, R. M., "Westinghouse 7300 Series Process Control System Noise Tests," WCAP-8892-A, June 1977.

3. Letter dated April 20, 1977 from R. L. Tedesco (NRC) to C. Eicheldinger (Westinghouse).

4. Katz, D.N., "Solid State Logic Protection System Description" WCAP-7488-L, January 1971 (Proprietary)

5. "Westinghouse Protection Systems Noise Tests," WCAP-12358, October 1975.

diversity has been evaluated for a wide variety of postulated accidents (see Chapter 15).

Table 7.2-1 provides a list of reactor trips which are described below.

1. <u>Nuclear Overpower Trips</u>

   The specific trip functions generated are as follows:

   (a) <u>Power Range High Neutron Flux Trip</u>

   The power range high neutron flux trip circuit trips the reactor when two of the four power range channels exceed the trip setpoint.

   There are two bistables, each with its own trip setting used for a high and a low range trip setting. The high trip setting provides protection during normal power operation and is always active. The low trip setting, that provides protection during startup, can be manually bypassed when two out of the four power range channels read above approximately 10 percent power (P-10). Three out of the four channels below 10 percent automatically reinstates the trip function. Refer to Table 7.2-2 for a listing of all protection system interlocks.

   (b) <u>Intermediate Range High Neutron Flux Trip</u>

   The intermediate range high neutron flux trip circuit trips the reactor when one out of the two intermediate range channels exceeds the trip setpoint. This trip, which provides protection during reactor startup, can be manually blocked if two out of four power range channels are above approximately 10 percent power (P-10). Three out of the four power range channels below this value automatically reinstate the intermediate range high neutron flux trip. The intermediate range channels (including detectors) are separate from the power range channels. The intermediate range channels can be individually bypassed at the nuclear instrumentation racks to permit channel testing during plant shutdown or prior to startup. This bypass action is annunciated on the control board.

   (c) <u>Source Range High Neutron Flux Trip</u>

   The source range high neutron flux trip circuit trips the reactor when one of the two source range channels exceeds the trip setpoint. This trip, which provides protection during reactor startup and plant shutdown, can by manually bypassed when one of the two intermediate range channels reads above the P-6 setpoint value and is automatically reinstated when both intermediate range channels decrease below the P-6 setpoint value. This trip function can also be reinstated below P-10 by an administrative action requiring manual actuation of two control board-mounted switches. Each switch will reinstate the trip function in one of the two protection logic trains. The source range trip point is set between the P-6 setpoint (source range cutoff power level) and the maximum source range power level. The channels can be individually bypassed at the nuclear instrumentation racks to permit channel testing during plant shutdown or prior to startup. This bypass action is annunciated on the control board.

(d) Power Range High Positive Neutron Flux Rate Trip

This circuit trips the reactor when a sudden abnormal increase in nuclear power occurs in two out of four power range channels. This trip provides DNB protection against rod ejection accidents of low worth from mid-power and is always active.

(e) Power Range High Negative Neutron Flux Rate Trip

This circuit trips the reactor when a sudden abnormal decrease in nuclear power occurs in two out of four power range channels. This trip provides protection when negative reactivity insertion is detected by the power range negative neutron flux rate trip circuitry due to dropped rods and is always active.

Figure 7.2-3 shows the logic for all of the nuclear overpower and rate trips. (See Reference 2 for additional information.)

2. Core Thermal Overpower Trips

The specific trip functions generated are as follows:

(a) Overtemperature ΔT Trip

This trip protects the core against low DNBR and trips the reactor on coincidence as listed in Table 7.2-1 with one set of temperature measurements per loop. The setpoint for this trip is continuously calculated by analog circuitry for each loop by solving the following equation:

τ_6 = Time constant utilized in the measured T_{avg} lag compensator

K_6 = A constant which compensates for the change in density flow and heat capacity of the water with temperature

T = Average temperature °F

T" = Indicated T_{avg} at rated thermal power

S = Laplace transform operator, sec^{-1}

$f_2(\Delta I)$ = A function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers, with gains to be selected based on measured instrument response during plant startup tests.

A separate long ion chamber unit supplies the flux signal for each Over power ΔT trip channel.

Increases in ΔI beyond a pre-defined deadband result in a decrease in the trip setpoint. Refer to Figure 7.2-16. The source of temperature and flux information is identical to that of the Overtemperature ΔT trip and the resultant ΔT setpoint is compared to the same ΔT. Figure 7.2-5 shows the logic for this trip function.

3. <u>Reactor Coolant System Pressurizer Pressure and Water Level Trips</u>

The specific trip functions generated are as follows:

(a) <u>Pressurizer Low Pressure Trip</u>

The purpose of this trip is to protect against low pressure which could lead to DNB. The parameter being sensed is reactor coolant pressure as measured in the pressurizer. Above P-7, the reactor is tripped when the pressurizer pressure measurements (compensated for rate of change) fall below preset limits. This trip is blocked below P-7. The trip logic and interlocks are given in Table 7.2-1.

The trip logic is shown on Figure 7.2-6.

(b) Pressurizer High Pressure Trip

The purpose of this trip is to protect the Reactor Coolant System against system overpressure.

The same sensors and transmitters used for the pressurizer low pressure trip are used for the high pressure trip except that separate bistables are used for the trip. These bistables trip when uncompensated pressurizer pressure signals exceed preset limits on coincidence as listed in Table 7.2-1. There are no interlocks or permissives associated with this trip function.

The logic for this trip is shown on Figure 7.2-6.

(c) Pressurizer High Water Level Trip

This trip is provided as a backup to the high pressurizer pressure trip and serves to prevent water relief through the pressurizer safety valves. This trip is blocked below P-7. The coincidence logic and interlocks of pressurizer high water level signals are given in Table 7.2-1.

The trip logic for this function is shown on Figure 7.2-6.

4. Reactor Coolant System Low Flow Trips

These trips protect the core from DNB in the event of a loss-of-coolant flow situation. Figure 7.2-5 shows the logic for these trips. The means of sensing the loss-of-coolant flow are as follows:

(a) Low Reactor Coolant Flow

The parameter sensed is reactor coolant flow. Three differential pressure measurements at a piping elbow tap in each reactor coolant loop are used for the flow measurement. The basic function of this device is to provide information as to whether or not a reduction in flow has occurred. An output signal from two out of three bistables in a loop would indicate a low flow in that loop.

The coincidence logic and interlocks are given in Table 7.2-1.

(b) Reactor Coolant Pump Undervoltage Trip

This trip is required in order to protect against low flow which can result from loss of voltage to at least one reactor coolant pump motor on each bus (e.g., from plant blackout or reactor coolant pump breakers opening).

There is one undervoltage sensing relay connected for each pump at the motor side of each reactor coolant pump breaker. These relays provide an output signal when the motor voltage goes below a preset level. Signals from these relays are time-delayed to prevent spurious trips caused by short-term voltage perturbations. Channel response time includes consideration of the bus voltage decay time due to generated Electro-Motive Force (EMF) from motors connected to the bus as the motors coast down. The coincidence logic and interlocks are given in Table 7.2-1.

(c) Reactor Coolant Pump Underfrequency Trip

This trip protects against low flow resulting from pump underfrequency, such as a major power grid frequency disturbance. The function of this trip is to trip the reactor in event of a decaying bus frequency condition. Subsection 8.2.3 provides additional details for the maximum credible frequency decay rate.

There is one underfrequency sensing relay for each reactor coolant pump motor. Underfrequency signals from two motors (one from each bus), time delayed to prevent spurious trips caused by short-term frequency perturbations, will trip the reactor if the power level is above P-7. (For an evaluation of under frequency transients, see Chapter 15.)

5. Steam Generator Trip

The specific trip function generated is the low-low steam generator water level trip.

This trip protects the reactor from loss of heat sink. This trip is actuated on two out of four low-low water level signals occurring in any steam generator.

The level channels, which have condensate pots common to both steam flow transmitters and level transmitters, are lag compensated to prevent spurious trips caused by short-term pressure waves which are generated during rapid closure of the turbine control valves.

The logic is shown on Figure 7.2-7 (see Reference 1 for additional information).

6. Reactor Trip on a Turbine Trip (anticipatory)

The reactor trip on a turbine trip is actuated by two out of three logic from emergency trip fluid pressure signals or by all closed signals from the turbine steam stop valves. A turbine trip causes a direct reactor trip above P-9. The reactor on turbine trip provides additional protection and conservatism beyond that required for the health and safety of the public. This trip is included as part of good engineering practice and prudent design. No credit is taken in any of the safety analyses (Chapter 15) for this trip. The turbine provides anticipatory trips to the Reactor Protection System from contacts which change position when the turbine stop valves close or when the turbine emergency trip fluid pressure goes below its setpoint.One of the design bases considered in the protection system is the possibility of an earthquake. With respect to these contacts, their functioning is unrelated to a seismic event in that they are anticipatory to other diverse parameters which cause reactor trip. The contacts are closed during plant operation and open to cause reactor trip when the turbine is tripped. No power is provided to the protection system from the contacts; they merely serve to interrupt power to cause reactor trip. This design functions in a de-energize-to-trip fashion to cause a plant trip if power is interrupted in the trip circuitry. This ensures that the protection system will in no way be degraded by this anticipatory trip because seismic design considerations do not form part of the design bases for anticipatory trip sensors. (The reactor protection system cabinets which receive the inputs from the anticipatory trip sensors are, of course, seismically qualified as discussed in Section 3.10.) The SSPS input circuits in nonseismic structures are routed in conduit to maintain train separation and to prevent the application of fault voltages greater than the maximum credible fault voltages. The electrical and physical independence of the connecting cabling conforms to Regulatory Guide 1.75. The anticipatory trips thus meet IEEE 279-1971, including redundancy, separation, single failure, etc. Seismic qualification of the contacts sensors is not required.

The logic for this trip is shown on Figure 7.2-15.

undervoltage, reactor coolant pump underfrequency, pressurizer low pressure, or pressurizer high water level. See Figures 7.2-5 and 7.2-6 for permissive applications. The low power signal is derived from three out of four power range neutron flux signals below the setpoint in coincidence with two out of two turbine impulse chamber pressure signals below the setpoint (low plant load). See Figures 7.2-4 and 7.2-15 for the derivation of P-7.

The P-8 interlock blocks a reactor trip when the plant is below approximately 50 percent of full power, on a low reactor coolant flow in any one loop. The block action (absence of the P-8 interlock signal) occurs when three out of four neutron flux power range signals are below the setpoint. Thus, below the P-8 setpoint, the reactor will be allowed to operate with one inactive loop and trip will not occur until two loops are indicating low flow. See Figure 7.2-4 for derivation of P-8, and Figure 7.2-5 for applicable logic.

The P-9 interlock blocks a reactor trip when the plant is below approximately 20 percent of full power on a turbine trip. The turbine trip is sensed by either: (1) all turbine main stop valves closed or (2) two out of three low trip fluid pressure. The block action (absence of P-9 interlock signal) occurs when three of four neutron flux power range signals are below the setpoint. Thus, below the P-9 setpoint, the reactor will be allowed to operate with the turbine tripped. See Figure 7.2-4 for the derivation of P-9 and 7.2-15 for applicable logic.

See Table 7.2-2 for the list of protection system blocks.

d. <u>Coolant Temperature Sensor Arrangement</u>

The hot and cold leg temperature signals required for input to the protection and control functions are obtained using thermowell-mounted RTDs installed in each reactor coolant loop.

The hot leg temperature measurement in each loop is accomplished using three fast-response dual-element narrow-range RTDs mounted in thermowells. Two of the three thermowells of each loop are located within the existing hot leg scoops. A third thermowell is mounted in the RCS process stream. On loops A, B, and D, this thermowell is mounted in an independent boss offset approximately 30° from the unused scoop location. On loop C the boss has been positioned approximately 12 inches upstream of the scoop location at approximately 105°. The two reused flow scoops have been modified by machining a flow hole in the end of the scoop to facilitate the flow of water through the leading edge of the scoop, past the thermowell and back into the pipe flow stream.

The temperature measured by each of the three thermowell-mounted RTDs, of each loop, is different due to hot leg temperature streaming. This temperature varies as a function of thermal power, core configuration and core age. Therefore, these signals are electronically averaged to generate a hot leg average temperature. If the active element of the dual-element RTD should fail, the second spare element will be wired into the processing cabinets. If the entire RTD has failed (i.e., the spare element is also inoperable), operation with two RTDs is acceptable. Provisions have been incorporated into the process electronics to allow for operation with only two RTDs in service. The two operable RTD measurements can be biased to compensate for the loss of the third.

The cold leg temperature measurement in each loop is accomplished by one fast-response dual-element narrow-range RTD. The original cold leg RTD bypass penetration nozzle has been modified to accept a thermowell. Temperature streaming in the cold leg is minimal due to the mixing action of the reactor coolant pump. Therefore, only a single temperature measurement is required in each cold leg.

e. <u>Pressurizer Water Level Reference Leg Arrangement</u>

The design of the pressurizer water level instrumentation employs the usual tank level arrangement using differential pressure between an upper and a lower tap on a column of water. A reference leg connected to the upper tap is kept full of water by condensation of steam at the top of the leg.

f. <u>Process Monitoring</u>

Process monitoring is performed by two instrumentation systems: the Process Instrumentation System and the Nuclear Instrumentation System.

Process Instrumentation System includes those <u>analog and trip actuating</u> devices (and their interconnection into systems) which measure voltage, frequency, valve position, temperature, pressure, fluid flow, and fluid level as in tanks or vessels. "Process" instrumentation specifically excludes nuclear and radiation measurements. The process instrumentation includes the field transmitters or process sensors, power supplies, indicators, recorders, alarm actuating devices, controllers, signal conditioning devices, etc., which are necessary for day-to-day operation of the Nuclear Steam Supply System as well as bistables for monitoring the plant and providing initiation of protective functions upon approach to unsafe plant conditions.

The primary function of nuclear instrumentation is to protect the reactor by monitoring the neutron flux and generating appropriate trips and alarms for various phases of reactor operating and shutdown conditions. It also provides a secondary control function and indicates reactor status during startup and power operation. The Nuclear Instrumentation System uses information from three separate types of instrumentation channels to provide three discrete protection levels. Each range of instrumentation (source, intermediate, and power) provides the necessary overpower reactor trip protection required during operation in that range. The overlap of instrument ranges provides reliable continuous protection beginning with source level through the intermediate and low power level. As the reactor power increases, the overpower protection level is increased by administrative procedures after satisfactory higher range instrumentation operation is obtained. Automatic reset to more restrictive trip protection is provided when reducing power.

Various types of neutron detectors, with appropriate solid-state electronic circuitry, are used to monitor the leakage neutron flux from a completely shutdown condition to 120 percent of full power. The power range channels are capable of recording overpower excursions up to 200 percent of full power. The neutron flux covers a wide range between these extremes. Therefore, monitoring with several ranges of instrumentation is necessary.

The lowest range ("source" range) covers six decades of leakage neutron flux. The lowest observed count rate depends on the strength of the neutron sources in the core and the core multiplication associated with the shutdown reactivity. This is generally greater than two counts per second. The next range ("intermediate" range) covers eight decades. Detectors and instrumentation are chosen to provide overlap between the higher portion of the source range and the lower portion of the power range. The highest range of instrumentation ("power" range) covers approximately two decades of the total instrumentation range. This is a linear range that overlaps with the higher portion of the intermediate range.

The system described above provides control room indication and recording of signals proportional to reactor neutron flux during core loading, shutdown, startup and power operation, as well as during subsequent refueling. Startup-rate indication for the source and intermediate range channels is provided at the control board. Reactor trip, rod stop, control and alarm signals are transmitted to the Reactor Control and Protection System for automatic plant control. Equipment failures and test status information are annunciated in the control room.

Trip actuating devices combine the process measuring, conditioning, and output functions into one device such as a pressure switch, limit switch, control switch, electrical relay, etc. Analog channel refers to signal processing equipment that provides interface with process measuring sensors; signal processing; limit checking; and output functions in one piece of equipment such as the Process Protection and Nuclear Instrumentation System cabinets.

See References 1 and 2 for additional background information on the process and nuclear instrumentation.

g. Solid-State Protection System

The logic portion consists of the Solid-State Protection System (SSPS). The SSPS takes binary inputs (voltage/no voltage) from the process and nuclear instrument channels corresponding to conditions (normal/abnormal) of plant parameters. The SSPS combines these signals in the required logic combination and generates a trip signal (no voltage) to the under voltage-trip attachment and shunt trip auxiliary relay coils (shunt trip coils are actually energized to trip, see Subsection 7.2.1.1) of the reactor trip circuit breakers when the necessary combination of signals occur. The system also provides annunciator, status light and computer input signals which indicate the condition of bistable input signals, partial trip and full trip functions and the status of the various blocking, permissive and actuation functions. In addition, the system includes means for semi-automatic testing of the logic circuits. See Reference 3 for additional background information.

h. Isolation Amplifiers

In certain applications, Westinghouse considers it advantageous to employ control signals derived from individual protection channels through isolation amplifiers contained in the protection channel, as permitted by IEEE Standard 279-1971.

In all of these cases, analog signals derived from protection channels for nonprotective functions are obtained through isolation amplifiers located in the analog racks. By definition, nonprotective functions include those signals used for control, remote process indication, and computer monitoring. Refer to Subsection 7.1.2.2a for discussion of electrical separation of control and protection functions. (See Reference 5 for isolation amplifier qualification-type tests.)

i. Energy Supply and Environmental Variations

The energy supply for the Reactor Trip System including the voltage and frequency variations, is described in Section 7.6 and Chapter 8. The environmental variations, throughout which the system will perform, are given in Section 3.11 and Chapter 8.

j. Setpoints

The setpoints that require trip action are given in the Technical Specifications. A detailed discussion on setpoints is found in Subsection 7.1.2.1i.

4. Missiles (see Section 3.5)

5. Flood (see Sections 2.4 and 3.4)

6. Wind and tornadoes (see Section 3.3)

7. Lightning (see Section 2.3).

The Reactor Trip System fulfills the requirements of IEEE Standard 279-1971 to provide automatic protection and to provide initiating signals to mitigate the consequences of faulted conditions. The Reactor Trip System relies upon provisions made by the owner and operator of the plant to provide protection against destruction of the system from fires, explosions, missiles, floods, wind, lightning and tornadoes (see each item above).

f. Minimum Performance Requirements

1. Reactor Trip System Response Times

Reactor Trip System response time is defined in Section 7.1. Typical maximum allowable time delays in generating the reactor trip signal are tabulated in Table 7.2-3. See Subsection 7.1.2.11 for a discussion of periodic response time verification capabilities.

2. Reactor Trip Accuracies

Accuracy is defined in Section 7.1. Reactor trip accuracies are tabulated in Table 7.2-3. An additional discussion on accuracy is found in Subsection 7.1.2.1i.

3. Protection System Ranges

Typical Protection System ranges are tabulated in Table 7.2-3. Range selection for the instrumentation covers the expected range of the process variable being monitored during power operation. Limiting setpoints are at least 3 percent from the end of the instrument span.

7.2.2 Analyses

7.2.2.1 Failure Mode and Effects Analyses

An analysis of the Reactor Trip System has been performed. Results of this study and a fault tree analysis are presented in Reference 4.

7.2.2.2 Evaluation of Design Limits

While most setpoints used in the Reactor Protection System are fixed, there are variable setpoints, most notably the Overtemperature ΔT and Overpower ΔT setpoints. All setpoints in the Reactor Trip System have been selected on the basis of engineering design or safety studies. The capability of the Reactor Trip System to prevent loss of integrity of the fuel cladding and/or reactor coolant system pressure boundary during Condition II, III, and IV transients is demonstrated in Chapter 15. These accident analyses are carried out using those setpoints determined from results of the engineering design studies. Setpoint limits are presented in the Technical Specifications. A discussion of the intent for each of the various reactor trips and the accident analyses (where appropriate) which use this trip is presented below. It should be noted that the selection of trip setpoints provide for margin before protection action is actually required to allow for uncertainties and instrument errors. The design meets the requirements of Criteria 10 and 20 of the 1971 GDC.

a. Trip Setpoint Discussion

It has been pointed out previously that below a DNBR equal to the safety analysis limit value there is likely to be significant local fuel cladding failure. The DNBR existing at any point in the core for a given core design can be determined as a function of the core inlet temperature, power output, operating pressure and flow. Consequently, core safety limits in terms of a DNBR equal to the safety analysis limit value for the hot channel can be developed as a function of core ΔT, T_{avg} and pressure for a specified flow as illustrated by the solid lines in Figure 15.0-1. Also shown as solid lines in Figure 15.0-1 are the loci of conditions equivalent to 118 percent of power as a function of ΔT and T_{avg} representing the overpower (kW/ft) limit on the fuel. The dashed lines indicate the maximum permissible setpoint (ΔT) as a function of T_{avg} and pressure for the overtemperature and overpower reactor trip. Actual setpoint constants in the equation representing the dashed lines are as given in the Technical Specifications.

These values are conservative to allow for instrument errors. The design meets the requirements of Criteria 10, 15, 20 and 29 of the 1971 GDC.

DNBR is not a directly measurable quantity; however, the process variables that determine DNBR are sensed and evaluated. Small isolated changes in various process variables may not individually result in violation of a core safety limit; whereas the combined variation, over sufficient time, may cause the overpower or overtemperature safety limit to be exceeded. The design concept of the Reactor Trip System recognizes this situation by providing reactor trips associated with individual process variables in addition to the overpower/overtemperature safety limit trips.

Process variable trips prevent reactor operation whenever a change in the monitored value is such that a core or system safety limit is in danger of being exceeded should operation continue. Basically, the high pressure, low pressure and Overpower/Overtemperature ΔT trips provide sufficient protection for slow transients as opposed to such trips as low flow or high flux which will trip the reactor for rapid changes in flow or flux, respectively, that would result in fuel damage before actuation of the slower responding ΔT trips could be affected.

Therefore, the Reactor Trip System has been designed to provide protection for fuel cladding and reactor coolant system pressure boundary integrity where: (1) a rapid change in a single variable or factor will quickly result in exceeding a core or a system safety limit, and (2) a slow change in one or more variables will have an integrated effect which will cause safety limits to be exceeded. Overall, the Reactor Trip System offers diverse and comprehensive protection against fuel cladding failure and/or loss of reactor coolant system integrity for Condition II,III, and IV accidents. This is demonstrated by Table 7.2-4 which lists the various trips of the Reactor Trip System, the appropriate accident in the safety analyses in which the trip could be utilized and the corresponding limiting safety system setting Technical Specification.

It should be noted that the Reactor Trip System automatically provides core protection during nonstandard operating configuration, i.e., operation with a loop out of service. Although operating with a loop out of service over an extended time is considered to be an unlikely event and is not allowed by the plant operating license, no protection system setpoints would need to be reset. This is because the nominal value of the power (P-8) interlock setpoint restricts the power so that DNB ratios less than the safety analysis limit value will not be realized during any Condition II transients occurring during this mode of operation. This restricted power is considerably below the boundary of permissible values as defined by the core safety limits for operation with a loop out of service. Thus, the P-8 interlock acts essentially as a high nuclear power reactor trip when operating with one loop not in service.

The design meets the requirements of Criterion 21 of the 1971 GDC.

Preoperational testing is performed on reactor trip system components and systems to determine equipment readiness for startup. This testing serves as a further evaluation of the system design.

Analyses of the results of Conditions I, II, III and IV events, including considerations of instrumentation installed to mitigate their consequences, are presented in Chapter 15. The instrumentation installed to mitigate the consequences of load rejection and turbine trip is given in Section 7.7.

b. Reactor Coolant Flow Measurement

The elbow taps used on each loop in the primary coolant system are instrument devices that indicate 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 has occurred. The correlation between flow and elbow tap signal is given by the following equation:

$$\frac{\Delta P}{\Delta P_o} = \left(\frac{w}{w_o}\right)^2$$

where ΔP_o is the pressure differential at the reference flow w_o, and ΔP is the pressure differential at the corresponding flow, w. The full flow reference point is established during initial plant startup. The low flow trip is then established by extrapolating along the correlation curve. The expected absolute accuracy of the channel is within ±10 percent of full flow and field results have shown the repeatability of the trip point to be within ±1 percent.

c. Evaluation of Compliance to Applicable Codes and Standards

The Reactor Trip System meets the criteria of the General Design Criteria as indicated. The Reactor Trip System meets the requirements of Section 4 of IEEE Standard 279-1971, as indicated below.

1. General Functional Requirement

The protection system automatically initiates appropriate protective action whenever a condition monitored by the system reaches a preset level. Functional performance requirements are given in Subsection 7.2.1.1a. Subsection 7.2.1.2d presents a discussion of limits, margins and levels; Subsection 7.2.1.2e discusses unusual (abnormal) events; and Subsection 7.2.1.2f presents minimum performance requirements.

2. Single Failure Criterion

The protection system is designed to provide two, three, or four instrumentation channels for each protective function and two logic train circuits. These redundant channels and trains are electrically isolated and physically separated. Thus, any single failure within a channel or train will not prevent protective action at the system level when required. Loss of input power, the most likely mode of failure, to a channel or logic train will result in a signal calling for a trip. This design meets the requirements of Criterion 23 of the 1971 GDC. To prevent the occurrences of common mode failures, such additional measures as functional diversity, physical separation, and testing as well as administrative control during design, production, installation and operation are employed, as discussed in Reference 4. The design meets the requirements of Criteria 21 and 22 of the 1971 GDC.

3. Quality of Components and Modules

For a discussion on the quality of the components and modules used in the Reactor Trip System, refer to Chapter 17. The quality assurance applied conforms to Criterion 1 of the 1971 GDC.

4. Equipment Qualification

For a discussion of the type tests and/or analyses made to verify the performance requirements, refer to Section 3.11. The tests results demonstrate that the design meets the requirements of Criterion 4 of the 1971 GDC.

5. Channel Integrity

Protection system channels required to operate in accident conditions maintain necessary functional capability under extremes of condition relating to environment, energy supply, malfunction, and accidents. The energy supply for the Reactor Trip System is described in Chapter 8. The environmental variations throughout which the system will perform are given in Section 3.11.

6. Independence

Channel independence is carried throughout the system, extending from the sensor through to the devices actuating the protective function. Physical separation is used to achieve separation of redundant transmitters. Separation of wiring is achieved using separate wireways, cable trays, conduit runs and containment penetrations for each redundant channel. Redundant analog equipment is separated by locating modules in different protection cabinets. Each redundant protection channel set is energized from a separate AC power feed. This design meets the requirements of Criterion 21 of the 1971 GDC.

Two reactor trip breakers, which are actuated by two separate logic matrices, interrupt power to the control rod mechanisms. The breaker main contacts are connected in series with the power supply so that opening either breaker interrupts power to all control rod drive mechanisms, permitting the rods to free fall into the core. See Figure 7.1-1.

The design philosophy is to make maximum use of a wide variety of measurements. The protection system continuously monitors numerous diverse protection variable. Generally, two or more diverse protection functions would terminate reactor operation before intolerable consequences could occur. This design meets the requirements of Criterion 22 of the 1971 GDC.

7. Control and Protection System Interaction

The protection system is designed to be independent of the control system. In certain applications the control signals and other nonprotective functions are derived from individual protective channels through isolation amplifiers. The isolation amplifiers are classified as part of the protection system and are located in the analog protective racks. Nonprotective functions include those signals used for control, remote process indication, and computer monitoring. The isolation amplifiers are designed so that a short circuit, open circuit, or the application of credible fault voltages from within the cabinets on the isolated output portion of the circuit (i.e., the nonprotective side of the circuit) will not affect the input (protective) side of the circuit. The signals obtained through the isolation amplifiers are never returned to the protective racks. This design meets the requirement of Criterion 24 of the 1971 GDC and paragraph 4.7 of IEEE Standard 279-1971.

The results of applying various malfunction conditions on the output portion of the isolation amplifiers show that no significant disturbance to the isolation amplifier input signal occurred.

See Subsection 7.2.2.2c(11) for a discussion of exceptions to the single failure criterion while a channel is bypassed for maintenance or testing and Subsection 7.2.2.3 for a discussion of specific control and protection interactions.

8. Derivation of System Inputs

To the extent feasible and practical, protection system inputs are derived from signals which are direct measures of the desired variables. Variables monitored for the various reactor trips are listed in Subsection 7.2.1.2b.

available. Channel checks are discussed in Technical Specification 3/4.3 and Table 4.3-1 of the Technical Specifications.

10. Capability For Testing

The Reactor Trip System is capable of being tested during power operation. Where only parts of the system are tested at any one time, the design is capable of providing the necessary overlap between the parts to assure complete system operation. The testing capabilities are in conformance with Regulatory Guide 1.22 as discussed in Subsection 7.1.2.5.

The protection system is designed to permit periodic testing of the process monitoring portion of the Reactor Trip System during reactor power operation without initiating a protective action unless a trip condition actually exists. This is because of the coincidence logic required for reactor trip. These tests may be performed at any plant power from cold shutdown to full power. Before starting any of these tests with the plant at power, all redundant reactor trip channels associated with the function to be tested must be in the normal (untripped) mode in order to avoid spurious trips. Setpoints are referenced in the Standard Instrument Schedule and Technical Specifications. Bypass Test Instrumentation (BTI) features have been added to selected Process and Nuclear Instrumentation System channels to permit periodic testing with the monitoring portion of the channel in bypass rather than in trip. This eliminates the potential for a reactor trip caused by spurious actuation of a redundant channel when a channel is in trip for testing.

(a) Analog Channel Operational Tests

Analog Channel Operational Tests (ACOTs) are performed on the Process and Nuclear Instrumentation Systems between channel calibrations to detect process monitoring failures which are not detectable by channel checks.

Process Instrumentation Channel Operational Tests

The analog channels of the Process Instrumentation System are tested by individually introducing simulated input signals into the analog channels and observing the tripping of the appropriate output bistables. When a channel with a normally energized SSPS input relay is tested in trip, the output to the logic circuitry is interrupted by a test switch which, when thrown, de-energizes the associated logic input and inserts a proving lamp in the bistable output. BTI is connected downstream of the channel test card, thus enabling the bypass of the entire analog channel for surveillance testing or maintenance.

If BTI is not used, interruption of the bistable output to the logic circuitry for any cause (test, maintenance, or removal from service) will cause that portion of the logic circuitry to be actuated (partial trip) accompanied by a partial trip alarm and channel status light actuation in the control room. If BTI is used, the logic circuitry will not be actuated. Each channel contains those switches, test points, etc., necessary to test the channel. See References 1 and 2 for additional non-BTI background information.

The following Process Instrumentation System analog channel periodic tests are performed:

(1) T_{avg} and ΔT protection channel testing

(2) Pressurizer pressure protection channel testing

(3) Pressurizer water level protection channel testing

(4) Steam generator water level protection channel testing

(5) Reactor coolant low flow protection channels testing.

Nuclear Instrumentation Analog Channels Operational Tests

The power range analog channels of the Nuclear Instrumentation System are tested either by superimposing a test signal on the actual detector signal being received by the channel at the time of testing or by injecting a test signal in place of the actual detector signal. The output of the bistable is not placed in a tripped condition prior to testing. Also, since the power range channel logic is two out of four, bypass of this reactor trip function is not required but has been provided to minimize the potential for spurious trips. BTI is connected to the power, intermediate, and source range channel output, thus enabling the bypass of the entire analog channel for surveillance testing or maintenance.

resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The channel calibration may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.

(d) <u>Solid-State Protection System Logic Testing</u>

The Reactor Trip System logic trains are part of the Solid-State Protection System (SSPS) and are designed to be capable of complete testing at power. After the individual process monitoring channel testing is complete, the logic matrices are tested from the Train A and Train B logic rack test panels. During this test, all of the logic inputs are actuated automatically in all combinations of trip and nontrip logic. Trip logic is not maintained sufficiently long enough to permit opening of the reactor trip breakers. The reactor trip undervoltage coils are "pulsed" in order to check continuity. During logic testing of one train, the other train can initiate any required protective functions. Annunciation is provided in the control room to indicate when a train is in test (train output bypassed) and when a reactor trip breaker is bypassed. Logic testing can be performed in less than 30 minutes.

This design complies with the testing requirements of IEEE Standard 279-1971 and IEEE Standard 338-1975 discussed in Subsection 7.1.2.11.

The permissive and block interlocks associated with the Reactor Trip System and Engineered Safety Features Actuation System are given on Tables 7.2-2 and 7.3-2 and designated protection or "P" interlocks. As a part of the protection system, these interlocks are designed to meet the testing requirements of IEEE Standard 279-1971 and 338-1975.

Testing of all protection system interlocks is provided by the logic testing and semi-automatic testing capabilities of the Solid-State Protection System. In the Solid-State Protection System, the undervoltage trip attachment and shunt trip auxiliary relay coils (Reactor Trip) and master relays (Engineered Safeguards Features Actuation) are pulsed for all combinations of trip or actuation logic with and without the interlock signals. For example, reactor trip on low flow (2 out of 4 loops showing 2 out of 3 low flow) is tested to verify operability of the trip above P-7 and nontrip below P-7 (see Figure 7.2-5). Interlock testing may be performed at power.

Testing of the logic trains of the Reactor Trip System includes a logic matrix check and may include a check of the input relays. The following sequence is used to test the system:

(1) Check of Input Relays

When the process instrumentation system and nuclear instrumentation system channels are tested in trip, each channel bistable is placed in a trip mode causing one input relay in Train A and one in Train B to de-energize. A refueling frequency has been justified for the testing of normally energized input relays (Reference 7). All input relays are tested at refueling intervals. A contact of each relay is connected to a universal logic printed circuit card. This card performs both the reactor trip and monitoring functions. Each reactor trip input relay contact causes a status lamp on the control board to operate and provides an input to the Video Alarm System (VAS). Either the Train A or Train B input relay operation will light the status lamp and provide the VAS alarm.

Each train contains a multiplexing test switch. During a process or nuclear instrumentation system test, this switch (in either train) is in the A + B position. The A + B position alternately allows information to be transmitted from the two trains to the control board. A steady status lamp and annunciator indicates that input relays in both trains have been de-energized. A flashing lamp means that the input relays into the two trains did not both de-energize.

Trip actuating device or associated auxiliary relay contact inputs to the SSPS, such as reactor coolant pump bus underfrequency relays, operate input relays which are tested by operating the remote contacts as described above and using the same type of indications as those provided for bistable input relays.

Actuation of the input relays provides the overlap between the testing of the SSPS and the testing of those systems supplying the inputs to the SSPS. Test indications are status lamps and VAS alarms on the control board. Inputs to the SSPS are checked one channel at a time, leaving the other channels in service. For example, a function that trips the reactor when two out of four channels trip becomes a one-out-of-three trip when one channel is placed in the trip mode or reverts to two-out-of-three when the channel is tested in bypass. Both trains of the SSPS remain in service during this portion of the test.

(2) Check of Logic Matrices

Logic matrices are checked one train at a time. Input relays are not operated during this portion of the test. Reactor trips from the train being tested are inhibited with the use of the input error inhibit switch on the semi-automatic test panel in the train. At the completion of the logic matrix tests, the input error is verified removed and returned to normal by the performance of continuity checks of the input inhibit error circuit.

The logic test scheme uses pulse techniques to check the coincidence logic. All possible trip and nontrip combinations are checked. Pulses from the tester are applied to the inputs of the universal logic card at the same terminals that connect to the input relay contacts. Thus there is an overlap between the input relay check and the logic matrix check. Pulses are fed back from the reactor trip breaker undervoltage trip attachment and shunt trip auxiliary relay coils to the tester. The pulses are of such short duration that the reactor trip breaker undervoltage coil armature cannot respond mechanically.

Test indications that are provided are an annunciator in the control room indicating that reactor trips from the train have been blocked and that the train is being tested, and green and red lamps on the semi-automatic tester to indicate a good or bad logic matrix test. Protection capability provided during this portion of the test is from the train not being tested.

The testing capability meets the requirements of Criterion 21 of the 1971 GDC.

(e) Testing of Reactor Trip Breakers

Normally, reactor trip breakers 52/RTA and 52/RTB are in service, and bypass breakers 52/BYA and 52/BYB are withdrawn (out of service). In testing the protection logic, pulse techniques are used to avoid tripping the reactor trip breakers thereby eliminating the need to bypass them during this testing. The following procedure describes the method used for testing the trip breakers:

(1) With bypass breaker 52/BYA racked out, manually close and trip it to verify its operation.

(2) Rack in and close 52/BYA. Manually trip 52/RTA through a protection system logic matrix while at the same time operating the "Auto Shunt Trip Block" push button on the automatic shunt trip panel. This verifies operation of the Undervoltage Trip Attachment (UVTA) when the breaker trips. After reclosing RTA, trip it again by operation of the "Auto Shunt Trip Test" push- button on the automatic shunt trip panel. This is to verify tripping of the breaker through the shunt trip device.

(3) Reset 52/RTA.

(4) Trip and rack out 52/BYA.

(5) Repeat above steps to test trip breaker 52/RTB using bypass breaker 52/BYB.

If an event requiring a reactor trips occurs during the testing of the reactor trip breaker, the bypass breaker 52/BYA would receive an undervoltage trip signal from the B train SSPS. 52/BYA also receives a shunt trip signal from the A train manual trip switch output. Similarly the bypass breaker 52/BYB would receive an undervoltage trip signal from the A train SSPS and a shunt trip signal from the B train manual trip switch output.

Auxiliary contacts of the bypass breakers are connected into the general warning alarm logic of their respective trains so that if either train is placed in test or if an water level. Any slow drift in the water level signal will permit the operator to respond to the level alarms and take corrective action.

Automatic protection is provided in case the spurious high level reduces feedwater flow sufficiently to cause low-low level in the steam generator. Automatic protection is also provided in case the spurious low level signal increases feedwater flow sufficiently to cause high level in the steam generator. A turbine trip and feedwater isolation would occur on two-out-of-four high-high steam generator water level in any loop.

7.2.2.4 Additional Postulated Accidents

Loss of plant instrument air or loss of component cooling water is discussed in Subsection 7.3.2.3. Load rejection and turbine trip are discussed in further detail in Section 7.7.

The control interlocks, called rod stops, that are provided to prevent abnormal power conditions which could result from excessive control rod withdrawal are discussed in Subsection 7.7.1.4a and listed on Table 7.7-1. Excessively high power operation (which is prevented by blocking of automatic rod withdrawal), if allowed to continue, might lead to a safety limit (as given in the Technical Specification) being reached. Before such a limit is reached, protection will be available from the Reactor Trip System. At the power levels of the rod block setpoints, safety limits have not been reached; therefore, these rod withdrawal stops do not come under the scope of safety-related systems, and are considered as control systems.

7.2.3 Tests and Inspections

The Reactor Trip System meets the testing requirements of IEEE Standard 338-1975, as discussed in Subsection 7.1.2.11. The testability of the system is discussed in Subsection 7.2.2.2c. The test intervals are specified in the Technical Specifications. All active devices will be tested at the operational test frequency unless a lower frequency is justified. Passive devices will be checked at the same frequency where practicable (Reference 6). References 7 and 8 document the operational test frequency and justify channel calibration frequency for testing of the normally energized SSPS input and RCP UV/UF time delay relays for channels which are tested in bypass. All components will be tested at the channel calibration frequency. Written test procedures and documentation, conforming to the requirements of IEEE Standard 338-1975, will be available for audit by responsible personnel. Periodic testing complies with Regulatory Guide 1.22, as discussed in Subsections 7.1.2.5 and 7.2.2.2c.

All active components can be tested at the operational test frequency. A lower testing frequency may be justified if adequate reliability is assured. Passive components will be tested at the operational test frequency where practicable.

7.2.4 References

1. Reid, J. B., "Process Instrumentation for Westinghouse Nuclear Steam Supply Systems," WCAP-7913, January 1973. (Addition background information only)

2. Lipchak, J. B.,"Nuclear Instrumentation System," WCAP-8255, January 1974. (Additional background information only)

3. Katz, D. N., "Solid State Logic Protection System Description," WCAP-7488-L, January 1971 (Proprietary) and WCAP-7672, June 1971 (Nonproprietary). (Additional background information only)

4. Gangloff, W. C. and Loftus, W. D., "An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients," WCAP-7706-L, February 1971 (Proprietary) and WCAP-7706, July 1971 (Nonproprietary).

5. Marasco, F. W. and Siroky, R. M., "Westinghouse 7300 Series Process Control Noise Tests," WCAP-8892-A, June 1977.

6. "Overlap Testing Requirements," NAESCo Engineering Evaluation 95-22, September 1985.

7. "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," WCAP-10271-P-A, May 1986; Supplement 1, May, 1986; and Supplement 2, Revision 1, June 1990.

8. "Risk/Reliability Evaluation of SSPS Input Relays and Timers in the Bypass Test Scheme," NAESCo Engineering Evaluation 95-20, August 1995.

9. "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 36 to Facility Operating License No. NPF-86," April 10, 1995.

TABLE 7.2-3
(Sheet 1 of 2)

REACTOR TRIP SYSTEM INSTRUMENTATION

	Reactor Trip Signal	Typical Range	Typical Trip Accuracy	Typical Time Response (sec)
1.	Power range high neutron flux	1 to 120 percent full power	±6.0 percent of span	0.5
2.	Intermediate range high neutron flux	8 decades of neutron flux overlapping source range by 2 decades	±9.8 percent of span ±1 percent of span from 10^{-4} to 50 percent full power (1)	N/A
3.	Source range high neutron flux	6 decades of neutron flux (1 to 10^6 counts/sec)	±11.5 percent of span (1)	N/A
4.	Power range high positive neutron flux rate	+15 percent of full power	±1.4 percent of span (1)	0.65
5.	Power range high negative neutron flux rate	-15 percent of full power	±1.4 percent of span (1)	0.5
6.	Overtemperature ΔT:	T_H 530 to 650°F T_C 510 to 630°F T_{AV} 530 to 630°F P_{PRZR} 1600 to 2500 psig $F_1(\Delta I)$ -60 to +60 ΔT Setpoint 0 to 100°F	±5.1 percent of span	6.0
7.	Overpower ΔT	T_H 530 to 650°F T_C 510 to 630°F T_{AV} 530 to 630°F ΔT Setpoint 0 to 100°F $F_2(\Delta I)$ -60 to +60	±5.5 percent of span	6.0

(1) Reproducibility (see definitions in Section 7.1)

TABLE 7.2-3
(Sheet 2 of 2)

	Reactor Trip Signal	Typical Range	Typical Trip Accuracy	Typical Time Response (sec)
8.	Pressurizer low pressure	1600 to 2500 psig	±2.14 percent of span (compensated signal)	2.0
9.	Pressurizer high pressure	1600 to 2500 psig	±2.14 percent of span (noncompensated signal)	2.0
10.	Pressurizer high water level	Entire cylindrical portion of pressurizer (distance between taps)	±4.59 percent of span	N/A
11.	Low reactor coolant flow	0 to 120 percent rated flow	±2.6 percent of span within range of 70 percent to 100 percent of full flow (1)	1.0
12.	Reactor coolant pump bus undervoltage	61 to 87 percent nominal bus voltage	±10.5 percent of span	1.5
13.	Reactor coolant pump underfrequency	44 to 61 Hz	±1.0 percent of span	0.6
14.	Low-low steam generator water level	± ~ 6 ft from nominal full load water level	±13.2 percent of span	2.0
15.	Turbine Trip	150 to 3000 psig	±3.3 percent of span	N/A

(1) Reproducibility (see definitions in Section 7.1)

TABLE 7.2-4
(Sheet 1 of 6)

REACTOR TRIP CORRELATION

Trip[a]	Accident[b]	Tech Spec[c]
1. Power Range High Neutron Flux Trip (Low Setpoint)	1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition (15.4.1)	2.2.1 Table 2.2-1
	2. Excessive Heat Removal Due to Feedwater System Malfunctions (15.1.2)	
	3. Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (15.4.8)	
2. Power Range High Neutron Flux Trip (High Setpoint)	1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition (15.4.1)	2.2.1 Table 2.2-1
	2. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.4.2)	
	3. Excessive Heat Removal Due to Feedwater System Malfunctions (15.1.2)	
	4. Excessive Load Increase Incident (15.1.3)	
	5. Accidental Depressurization of the Main Steam System (15.1.4)	
	6. Major Secondary System Pipe Ruptures (15.1.5)	
	7. Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (15.4.8)	
3. Intermediate Range High Neutron Flux Trip	1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition (15.4.1)	See Note d 2.2.1 Table 2.2-1

TABLE 7.2-4
(Sheet 2 of 6)

REACTOR TRIP CORRELATION

Trip[a]	Accident[b]	Tech Spec[c]
4. Source Range High Neutron Flux Trip	1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition (15.4.1)	See Note d 2.2.1 Table 2.2-1
5. Power Range High Positive Neutron Flux Rate Trip	1. Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (15.4.8)	2.2.1 Table 2.2-1
	2. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition (15.4.1)	
6. Power Range High Negative Flux Rate Trip	1. Rod Cluster Control Assembly Misalignment (15.4.3)	2.2.1 Table 2.2-1
7. Overtemperature ΔT Trip	1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.4.2)	2.2.1 Table 2.2-1
	2. Uncontrolled Boron Dilution (15.4.6)	
	3. Loss of External Electrical Load and/or Turbine Trip (15.2.2 and 15.2.3)	
	4. Excessive Heat Removal Due to Feedwater System Malfunctions (15.1.2)	
	5. Excessive Load Increase Incident (15.1.3)	
	6. Accidental Depressurization of the Reactor Coolant System (15.6.1)	
	7. Accidental Depressurization of the Main Steam System (15.1.4)	
	8. Steam Generator Tube Rupture (15.6.3)	

TABLE 7.2-4
(Sheet 3 of 6)

REACTOR TRIP CORRELATION

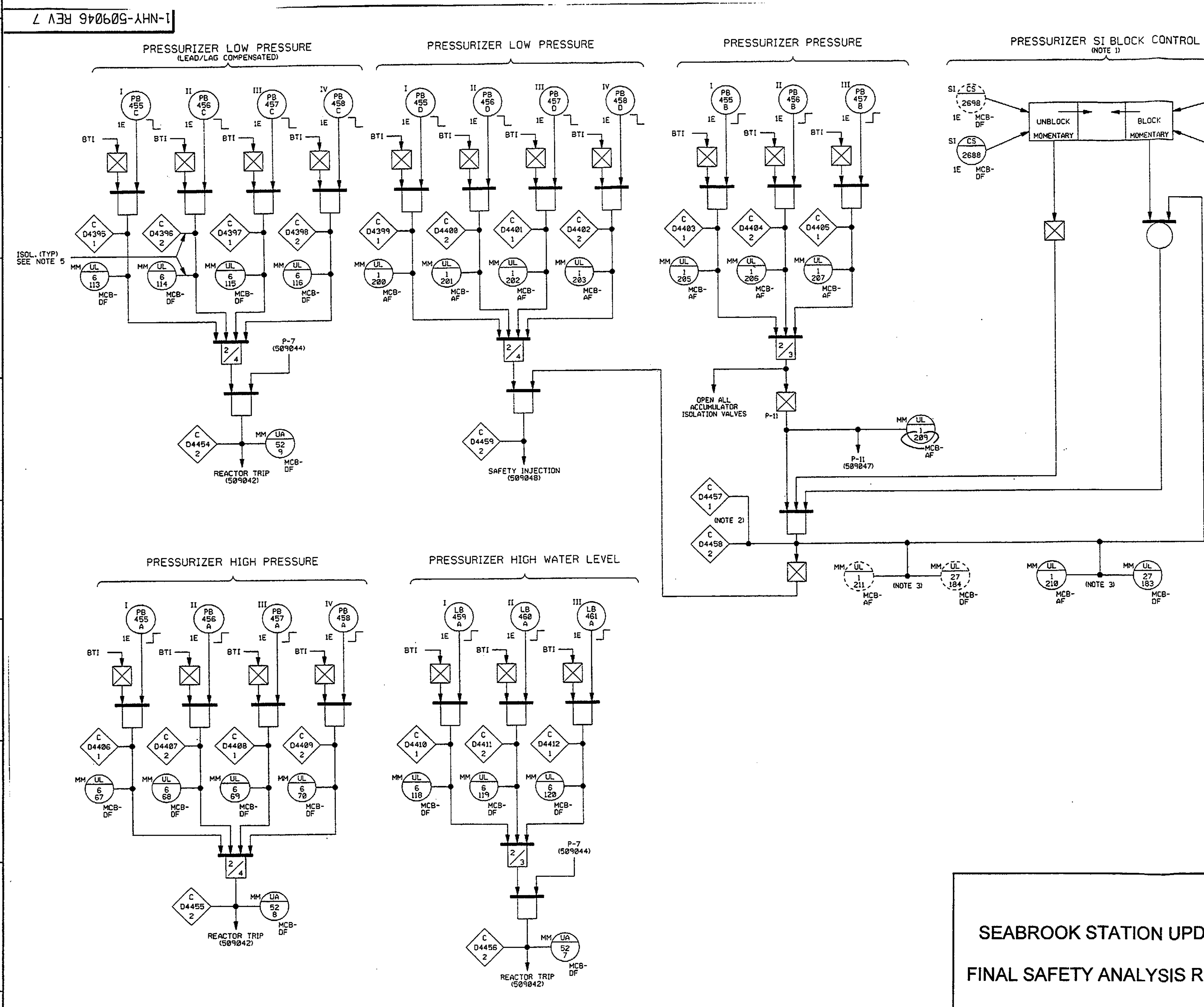
Trip[a]	Accident[b]	Tech Spec[c]
	9. Feedwater System Pipe Break (15.2.8)	
8. Overpower ΔT Trip	1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.4.2)	2.2.1 Table 2.2-1
	2. Excessive Heat Removal Due to Feedwater System Malfunctions (15.1.2)	
	3. Excessive Load Increase Incident (15.1.3)	
	4. Accidental Depressurization of the Main Steam System (15.1.4)	
	5. Major Secondary System Pipe Ruptures (15.1.5)	
	6. Rod Cluster Control Assembly Misoperation (15.4.3)	
9. Pressurizer Low Pressure Trip	1. Accidental Depressurization of the Reactor Coolant System (15.6.1)	2.2.1 Table 2.2-1
	2. Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary (15.6.5)	
	3. Major Reactor Coolant System Pipe Ruptures (LOCA) (15.6.5)	
	4. Steam Generator Tube Rupture (15.6.3)	
	5. Inadvertent Operation of Emergency Core Cooling System during Power Operation (15.5.1)	

TABLE 7.2-4
(Sheet 3a of 6)

REACTOR TRIP CORRELATION

10. Pressurizer High Pressure Trip	1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.4.2)	2.2.1 Table 2.2-1

REV. 07



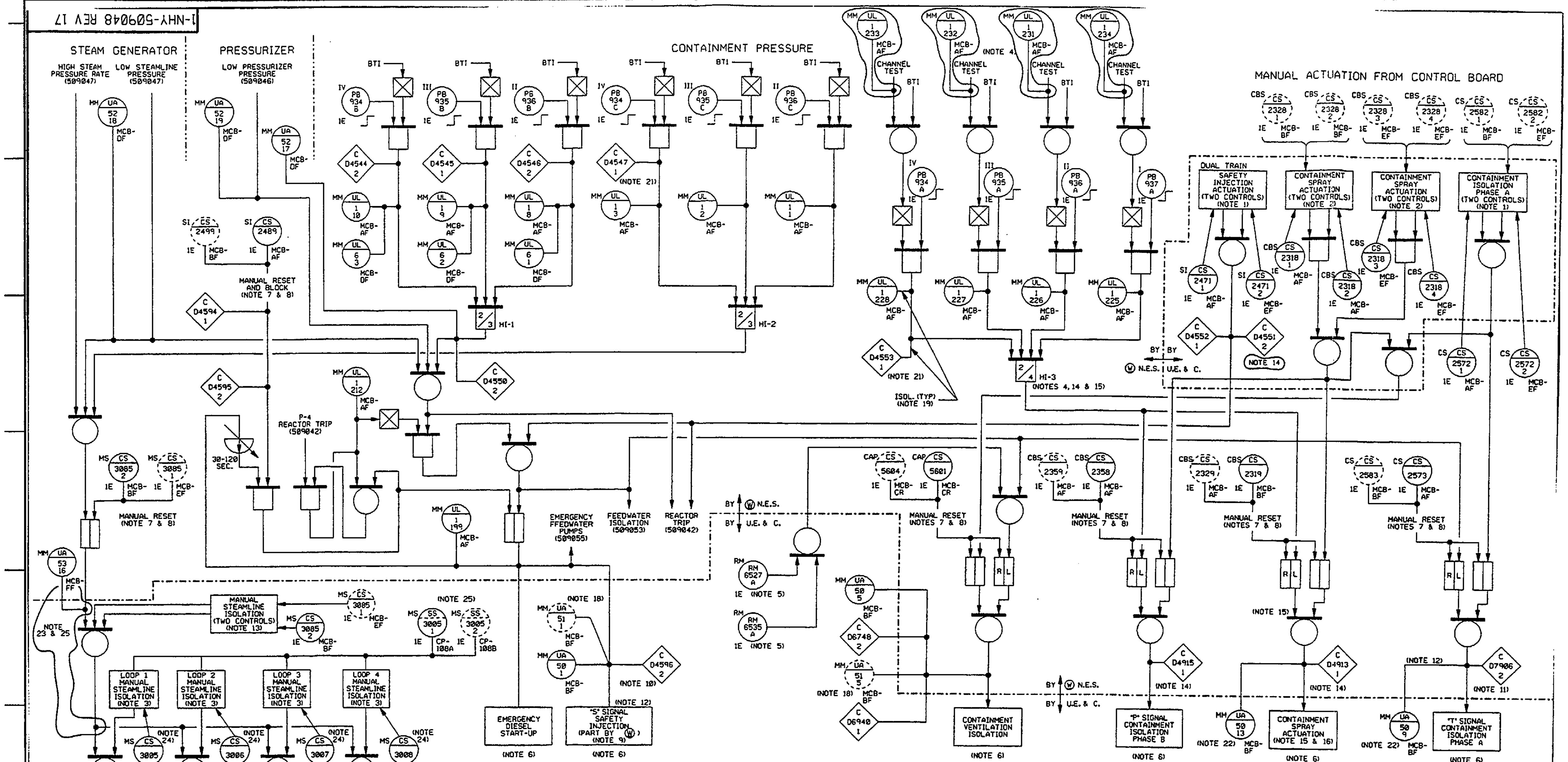
NOTES:

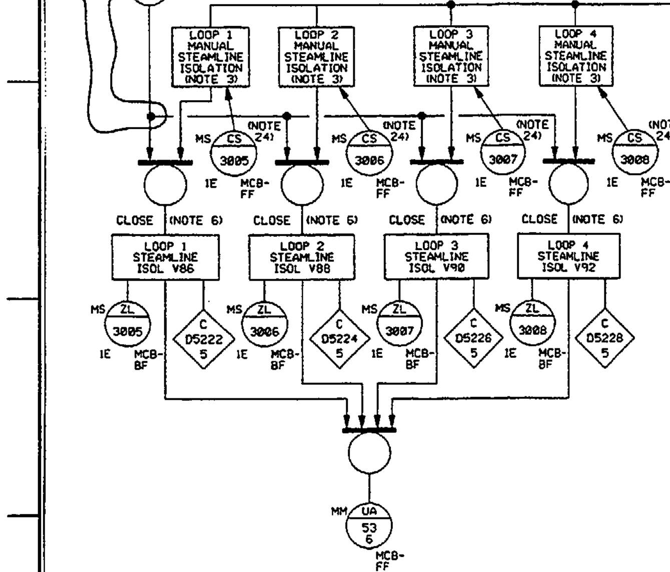
1. THE REDUNDANT MANUAL BLOCK CONTROLS CONSIST OF TWO CONTROLS ON THE CONTROL BOARD. ONE FOR EACH TRAIN. TRAIN "B" SWITCH SHOWN AS.
2. TWO COMPUTER INPUTS ARE CONNECTED TO THIS CIRCUIT, INDIVIDUAL FOR EACH TRAIN.
3. TWO PERMISSIVE STATUS LIGHTS ARE CONNECTED TO THIS CIRCUIT. INDIVIDUAL FOR EACH TRAIN. TRAIN "B" IS SHOWN AS.
4. SYSTEM PREFIX IS "RC" UNLESS OTHERWISE NOTED.
5. ISOLATION IS PROVIDED FOR ALL COMPUTER INPUTS AND MONITORING LIGHTS.

NUCLEAR-SAFETY-RELATED

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RC PRZR TRIP SINGALS W FUNCTIONAL DIAGRAMS
	NHY-509046 / FIGURE 7.2-6

REV. 07





NOTES:

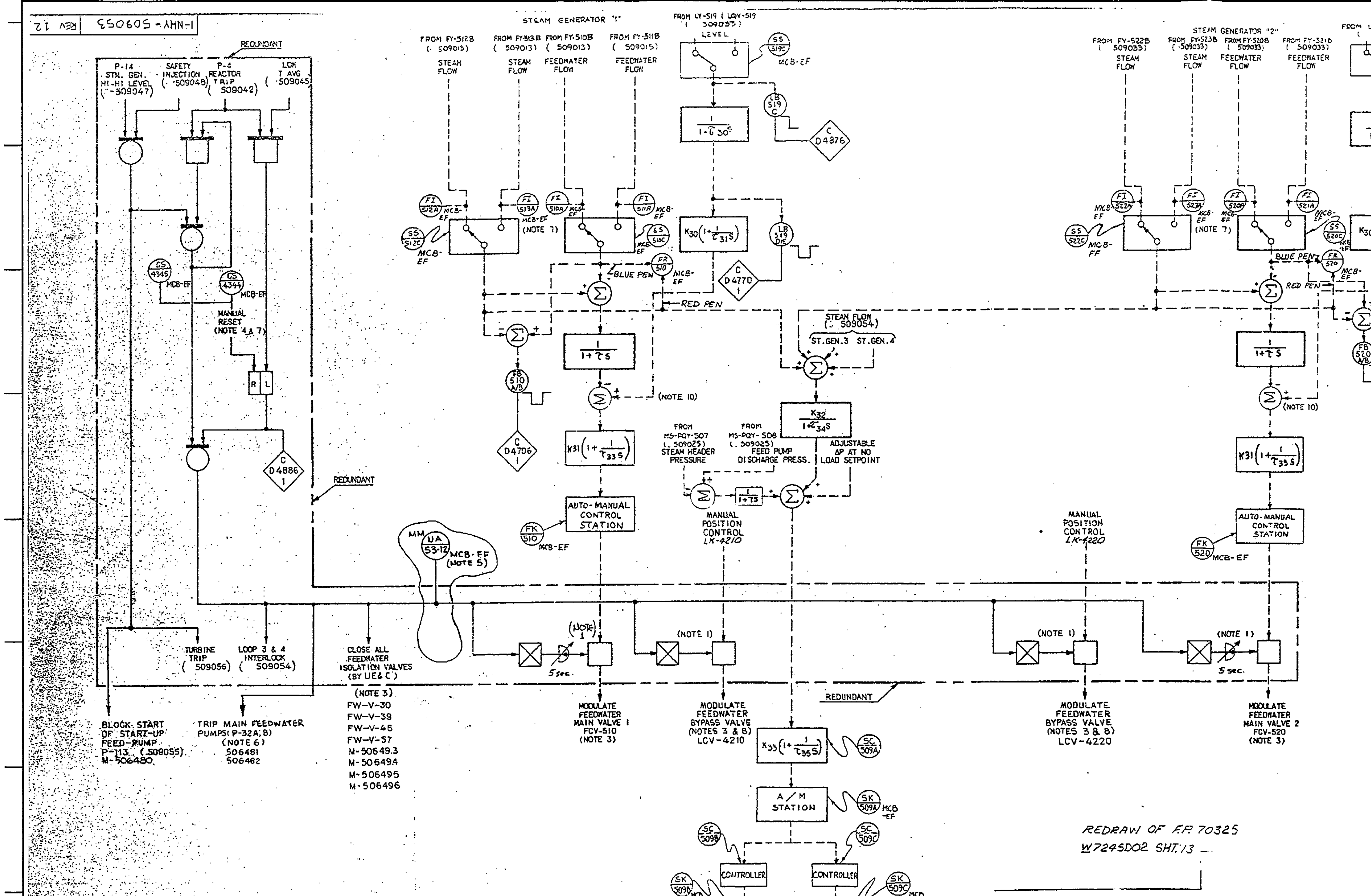
1. TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, OPERATING EITHER CONTROL WILL ACTUATE. TRAIN "B" SHOWN AS.
2. THE MANUAL SPRAY ACTUATION CONSISTS OF FOUR MOMENTARY CONTROLS. ACTUATION WILL OCCUR ONLY IF TWO ASSOCIATED CONTROLS ARE OPERATED SIMULTANEOUSLY. TRAIN "B" SHOWN AS.
3. ONE MOMENTARY CONTROL PER LOOP ON THE CONTROL BOARD.
4. CONTAINMENT PRESSURE BISTABLES FOR SPRAY ACTUATION ARE ENERGIZE-TO-ACTUATE (OTHER BISTABLES ARE DE-ENERGIZE-TO-ACTUATE).
5. RM-RM-6535A/6527A IS THE TRAIN "A" CONTAINMENT RAD. MONITOR. RM-RM-6535B/6527B IS FOR TRAIN "B".
6. COMPONENTS ARE ALL INDIVIDUALLY SEALED IN (LATCHED), SO THAT LOSS OF THE ACTUATION SIGNAL WILL NOT CAUSE THESE COMPONENTS TO RETURN TO THE CONDITION HELD PRIOR TO THE ADVENT OF THE ACTUATION SIGNAL.
7. THE REDUNDANT MANUAL RESET CONSISTS OF TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN. TRAIN "B" SHOWN AS.
8. SUPPLIED BY UE&C.
9. SAFETY INJECTION SEQUENCE REQUIREMENTS (IF SEQUENCING IS NECESSARY) ARE SPECIFIED BY WESTINGHOUSE N.E.S.
10. COMPUTER INPUT D4596 ORIGINATES FROM WESTINGHOUSE RELAY K615A.
11. COMPUTER INPUT D7906 ORIGINATES FROM WESTINGHOUSE RELAY K629A.
12. THE ACTUATION MAY BE DELAYED AND SEQUENCED IF THE EMERGENCY DIESEL POWER CAPABILITY IS LESS THAN THE TOTAL LOAD WITH ALL SYSTEMS STARTING. THE TIME DELAY(S), IF USED MAY NOT EXCEED THE MAXIMUM STARTING TIME REQUIREMENTS FOR EACH SYSTEM.
13. TWO CONTROLS ON THE CONTROL BOARD. OPERATING EITHER CONTROL WILL ACTUATE.
14. SOME ENGINEERED SAFEGUARDS FUNCTIONS ARE NOT WITHIN THE FUNCTIONAL DESIGN SCOPE OF WESTINGHOUSE NUCLEAR ENERGY SYSTEMS BUT ONLY SHOWN ON THIS SHEET AS THE FUNCTIONS ARE BUILT IN THE WESTINGHOUSE SUPPLIED EQUIPMENT.
15. THE 2 OUT OF 4 COINCIDENCE, MEMORY AND "OR" LOGIC ARE DUPLICATED WITHIN EACH TRAIN. SEPARATE OUTPUT RELAYS ARE ALSO PROVIDED IN EACH TRAIN. TO MINIMIZE FALSE CONTAINMENT SPRAY, ONE OUTPUT RELAY SHOULD START THE PUMPS WHILE ANOTHER SHOULD OPEN THE SYSTEM VALVES.
16. THE CONTAINMENT SPRAY SYSTEM SHOULD BE INDEPENDENT OF THE SAFETY INJECTION SYSTEM. IF THE CONDITIONS OF NOTE 12 APPLY, THE SEQUENCE INTERLOCK SHOULD BE SUCH THAT SPRAY WILL START WITHIN THE REQUIRED TIME INDEPENDENT OF THE SAFETY INJECTION SIGNAL STATUS.
17. SYSTEM PREFIX IS "SI" UNLESS OTHERWISE NOTED.
18. TRAIN B ANNUNCIATORS ARE SHOWN AS.
19. ISOLATION IS PROVIDED FOR COMPUTER INPUTS AND MONITORING LIGHTS.
20. SEE C-509041 FOR GENERAL NOTES & SYMBOL LEGEND.
21. COMPUTER INPUTS D4547 AND D4553 ARE ACTUATED WHEN ANY OF THE REDUNDANT BISTABLES TRIP.
22. THIS ANNUNCIATOR IS WIRED TO PROVIDE BOTH TRAIN A & B ALARM.
23. BY-PASS VALVES V204, V205, V206 AND V207 AND DRAIN VALVES V44, V45, V46 AND V47 (NOT SHOWN) ARE ACTUATED FROM TRAIN "A" MAIN STEAM ISOLATION. SEE LOGIC DIAGRAM 1-NHY-503669 & 503695 FOR OPERATION OF THESE VALVES.
24. SLOW MSIV CLOSURE.
25. FAST MSIV CLOSURE.

NUCLEAR-SAFETY-RELATED

REFERENCE DRAWINGS:
509022
503667 (MS)
503668 (MS)
503915 (SI)
503256 (CBS)
503392 (CS)
503221 (CAP)
503298 (COP)
503341 (CS)

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	SAFEGUARD ACTUATION SIGNALS W FUNCTIONAL DIAGRAMS
	NHY-509048 | FIGURE 7.2-8

REV. 07



NOTES:

1. ANALOG GATE CONSISTS OF TWO SOLENOID VENT VALVES IN SERIES TO REDUNDANTLY INTERLOCK THE AIR LINE BETWEEN EACH VALVE DIAPHRAGM AND ITS ASSOCIATED POSITIONER. THE SOLENOID VALVES ARE DE-ENERGIZE TO VENT, CAUSING THE FEEDWATER VALVE TO CLOSE IN FIVE SECONDS. EITHER OF THE TWO REDUNDANT BLOCK SIGNALS WILL CLOSE THE ASSOCIATED VALVES INDEPENDENT OF THE OTHER SIGNAL.
2. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT, EXCEPT WHERE INDICATED "REDUNDANT".
3. OPEN/SHUT INDICATION FOR EACH FEEDWATER VALVE IN CONTROL ROOM.
4. THE REDUNDANT MANUAL RESET CONSISTS OF TWO MOMENTARY CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
5. TRAIN "A" ONLY
6. TRAIN "A" TRIPS PUMP A. TRAIN "B" TRIPS PUMP B.
7. THESE CONTROLS ON THE CONTROL BOARD ARE SUPPLIED BY UE&C
8. SUPPLIED BY UE&C
9. SYSTEM PREFIX IS "FW" UNLESS OTHERWISE NOTED
10. SUMMING JUNCTION SHOWN FOR FUNCTIONAL PURPOSE ONLY (NOT AN ACTUAL DEVICE)

REFERENCE DRAWINGS:

M-503551 F.P. 70001 SH 33,25
M-504151 509025
M-503598 509033
M-506485
M-506486
M-506487
M-506488
M-506482
M-506481

REV	DATE	DRWN	CHKD	CE	LDE	DESCRIPTION
12	4-20-00	MRJ	MBB	JPS	CMA	INCORP. MMOD 96-619 DCN-00
11	7/6/95	MRB	JWD		CMA	INCORP DCR 94-039 CA 3

SEABROOK STATION UPDATED

FINAL SAFETY ANALYSIS REPORT

FW CONTROL & ISOLATION W FUNCTIONAL DIAGRAMS

NHY-509053

FIGURE 7.2-13 SH 1

7.3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM

In addition to the requirements for a reactor trip for anticipated abnormal transients, the facility is provided with adequate instrumentation and controls to sense accident situations and initiate the operation of necessary Engineered Safety Features (ESF). The occurrence of a limiting fault, such as a loss-of-coolant accident or a steam line break, requires a reactor trip plus actuation of one or more of the Engineered Safety Features to prevent or mitigate damage to the core and reactor coolant system components, and ensure containment integrity.

To accomplish these design objectives, the Engineered Safety Features System has proper and timely initiating signals which are to be supplied by the sensors, transmitters and logic components making up the various instrumentation channels of the Engineered Safety Features Actuation System.

7.3.1 Description

The Engineered Safety Features Actuation System uses selected plant parameters, determines whether or not predetermined safety limits are being exceeded and, if they are, combines the signals into logic matrices sensitive to combinations indicative of primary or secondary system boundary ruptures (Class III or IV faults). Once the required logic combination is completed, the system sends actuation signals to the appropriate engineered safety features components. The Engineered Safety Features Actuation System meets the requirements of Criteria 13, 20, and 38 of the 1971 General Design Criteria (GDC).

7.3.1.1 System Description

The Engineered Safety Features Actuation System is a functionally defined system described in this section. The equipment which provides the actuation functions identified in Subsection 7.3.1.1a is listed below and discussed in this section. For additional background information refer to References 1, 2 and 3 and Sections 11.5 and 12.3.4.

- Process Instrumentation System (Reference 1)
- Radiation Monitoring System (Sections 11.5 and 12.3.4)
- Solid-State Protection System (Reference 2)
- Engineered Safety Features Test Cabinet (Reference 3)
- Manual Actuation Circuits

The Engineered Safety Features Actuation System consists of sensors, connected to signal processing circuitry consisting of two to four redundant channels that monitor various plant parameters, and digital circuitry consisting of two redundant logic trains which receive inputs from the signal processing channels to complete the logic needed to actuate the Engineered Safety Features.

Each of the two logic trains is capable of actuating the Engineered Safety Features equipment required. The intent is that any single failure within the Engineered Safety Features Actuation System shall not prevent system action when required.

The redundant concept is applied to both the monitoring and logic portions of the system. Separation of redundant monitoring channels begins at the process sensors and is maintained in the field wiring, containment vessel penetrations and electronics terminating at the redundant safeguards logic racks. The design meets the requirements of Criteria 20, 21, 22, 23 and 24 of the 1971 GDC.

The variables are sensed by the monitoring portion as discussed in Reference 1 and in Sections 7.2, 11.5 and 12.3.4. The outputs from the monitoring portion are combined into actuation logic as shown on Figures 7.2-5 through 7.2-8. Table 7.3-1 gives additional information pertaining to logic and function.

The interlocks associated with the Engineered Safety Features Actuation System are outlined in Table 7.3-2. These interlocks satisfy the functional requirements discussed in Subsection 7.1.2.

Manual actuation from the control board of containment isolation Phase A is provided by operation of either one of the redundant momentary containment isolation Phase A controls. Each manual actuation switch actuates the equipment in its respective train. In a similar manner, manual actuation of containment spray and containment isolation Phase B is initiated from the control board, except that two switches must be operated for each train. Also on the control board, are manual actuation switches for safety injection. Each of these controls are dual train switches, which will actuate both Train A and Train B equipment.

Manual controls are also provided to switch from the injection to the recirculation phase after a loss-of-coolant accident.

The transfer from the injection to the recirculation phase is initiated automatically and completed manually by operator action from the main control board. Protection logic is provided to automatically open the two recirculation sump isolation valves on a 2/4 RWST Lo-Lo level in conjunction with initiation of an "S" signal (see Figure 7.6-4, sheets 1 and 2). This aligns the two RHR pumps to take suction from the containment sump and to deliver directly to the RCS without stopping the RHR pumps. The charging/safety injection pumps will continue to take suction from the RWST, during the above automatic action, until manual operator action is initiated to align these pumps in series with the RHR pumps. The Safety Injection System will be aligned for cold leg recirculation after the completion of the required manual actions.

Refer to Subsections 6.3.2.8 and 7.6.5 for additional discussions of recirculation initiation.

a. Function Initiation

The specific functions which rely on the Engineered Safety Features Actuation System for initiation are:

1. A reactor trip, provided one has not already been generated the by Reactor Trip System (see Figure 7.2-2).

2. Opening of cold leg injection isolation valves for injection of borated water by charging and safety injection pumps into the cold legs of the Reactor Coolant System.

3. Start of charging pumps, safety injection pumps, residual heat removal pumps and actuation of associated valving to provide emergency makeup water to the cold legs of the Reactor Coolant System following a loss-of-coolant accident.

4. The start of the containment enclosure emergency exhaust filter fans and the filtration system designed to function following a LOCA to maintain a negative pressure within the enclosure by exhausting air and in-leakage to the plant unit vent (see Drawing NHY-503515).

5. Start of motor-driven and turbine-driven emergency feedwater pumps, and actuation of associated valves to provide secondary heat removal (see Figure 7.2-14).

6. Phase A containment isolation, whose function is to prevent fission product release (isolation of all lines not essential to reactor protection). See Figure 7.2-8.

7. Steam line isolation to prevent the continuous, uncontrolled blowdown of more than one steam generator and thereby uncontrolled reactor coolant system cooldown (see Figure 7.2-8).

8. Isolation of the nonnuclear parts of the Component Cooling Water and Service Water Systems (see Drawings NHY-503273 and NHY-503965).

9. Automatic opening of the isolation valves in the cooling water discharge lines of the residual heat removal heat exchangers (see Drawing NHY-503271).

10. Main feedwater line isolation as required to prevent or mitigate the effect of excessive cooldown (see Figure 7.2-13, Sheet 1).

11. Start of the emergency diesels to assure backup supply of power to emergency and supporting systems components (see Figure 7.2-8).

12. The start of emergency cleanup filter system, designed to clean up a portion of recirculated air and meet control room occupancy requirements following a loss-of-coolant accident (see Drawing NHY-503242).

13. Containment spray actuation which performs the following functions:

    (a) Initiates containment spray to reduce containment pressure and temperature following a loss-of-coolant or steam line break accident inside of Containment (see Figure 7.2-8).

    (b) Initiation of Phase B containment isolation which isolates the Containment following a loss of reactor coolant accident, or a steam or feedwater line break within containment to limit radioactive releases. (Phase B isolation together with Phase A isolation results in isolation of all but Engineered Safety Features lines penetrating the Containment.) See Figure 7.2-8.

    (c) Opens the isolation valves in the cooling water discharge lines for the containment spray heat exchangers (see Drawing NHY-503271).

14. Switchover of the RHR and containment spray system pump suction from the RWST to the containment sump (see Figure 7.6-4, sheets 1 and 2).

15. Start of containment structure recirculating filter fans (see Drawing NHY-503204).

16. Containment Ventilation Isolation (see Figure 7.2-8).

b. <u>Process Monitoring</u>

Process monitoring is performed by the Process Instrumentation and Radiation Data Management Systems. The monitoring sensors and signal processing circuits for the Engineered Safety Features Actuation System are covered in Reference 1 and Sections 11.5 and 12.3.4. The parameters to be measured including pressures, flows, tank and vessel water levels, radiation levels, and temperatures, as well as the measurement and signal transmission considerations, are discussed. These latter considerations include the sensors, transmitters, orifices and flow elements, resistance temperature detectors, as well as automatic calculations, signal conditioning and location and mounting of the devices.

The sensors monitoring the primary system are located as shown on the piping flow diagrams in Chapter 5, Reactor Coolant System. The secondary system sensor locations are shown on the steam system flow diagrams given in Chapter 10.

Containment pressure is sensed by four physically separated differential pressure transmitters connected to containment atmosphere by a filled and sealed hydraulic transmission system. These transmitters are mounted on seismically qualified supports outside of Containment. The distance from penetration to transmitter is kept to a minimum, and separation is maintained. This arrangement, together with the pressure sensors external to the Containment, forms a double barrier and conforms to GDC 56 and Regulatory Guide 1.141, as well as the intent of Regulatory Guide 1.11.

Radiation sensors are located as shown in Sections 11.5 and 12.3.4.

c. Logic Circuitry

The Engineered Safety Features Actuation System logic and actuation functions are performed by the Solid-State Protection System (SSPS) which is discussed in detail in Reference 2. The description includes the considerations and provisions for physical and electrical separation as well as details of the circuitry. Reference 2 also covers certain aspects of online test provisions, provisions for test points, considerations for the instrument power source and considerations for accomplishing physical separation. The outputs from the analog channels are combined into actuation logic as shown on Figure 7.2-5 (T_{avg}), 7.2-6 (Pressurizer Pressure), 7.2-7 (Steam Line Pressure and Steam Line Pressure Rate), 7.2-8 (Engineered Safety Features Actuation), and 7.2-14 (Emergency Feedwater Pumps Startup).

To facilitate Engineered Safety Features Actuation testing, four cabinets (two per train) are provided which enable operation, to the maximum practical extent, of safety features loads on a group by group basis until actuation of all devices has been checked. Final actuation testing is discussed in detail in Subsection 7.3.2.

d. <u>Final Actuation Circuitry</u>

The outputs of the Solid-State Protection System (the slave relays) are energized to actuate, as are most final actuators and actuated devices. These devices are listed as follows:

1. Safety injection system pump and valve actuators. See Chapter 6 for flow diagrams and additional information.

2. Containment isolation (Phase A - "T" signal isolates all nonessential process lines on receipt of safety injection signal; Phase B - "P" signal isolates remaining process lines (which do not include Engineered Safety Features lines) on receipt of 2/4 Hi-3 containment pressure signal). For further information, see Subsection 6.2.4.

3. Component cooling water system and service water system valve actuators (see Subsection 9.2.2)

4. Emergency feed pumps start (see Subsection 6.8.5)

5. Diesel start (see Section 8.3)

6. Feedwater isolation (see Section 10.4)

7. Containment air purge isolation valves (see Section 6.2)

8. Steam line isolation valve actuators (see Section 10.3)

9. Containment spray pump and valve actuators (see Section 6.2)

10. Control room emergency cleanup filter system fans and dampers (see Subsection 9.4.1)

11. Containment recirculation fans and dampers (see Subsection 9.4.5)

12. Containment enclosure emergency cleanup filter system fans and dampers (see Subsection 9.4.6).

If an accident is assumed to occur coincident with a loss of offsite power, the Engineered Safety Features loads must be sequenced onto the diesel generators to prevent overloading the diesel generator. This sequence is discussed in Section 8.3. The design meets the requirements of Criterion 35 of the 1971 GDC.

e. <u>Support Systems</u>

The following systems are required for support of the Engineered Safety Features:

Typical accuracies required in generating the required actuation signals for steam line break protection are:

(a)	Steam line pressure	±12.5 percent of span
(b)	Steam line pressure rate	±1.8 percent of span
(c)	T_{avg}	±5.0 percent of span
(d)	Containment pressure signal	±2.5 percent of span
(e)	Refueling water storage tank level	±2.75 percent of span

3. Range of Sensed Variables

Ranges of sensed variables to be accommodated until conclusion of protective action is assured.

Typical ranges required in generating the required actuation signals for loss-of-coolant protection are:

(a)	Pressurizer pressure	1600 to 2500 psig
(b)	Containment pressure	0 to 60 psig
(c)	Refueling water storage tank level	0 to 140 inches (suppressed range)

Typical ranges needed to generate the required actuation signals for steam line break protection are:

(a)	T_{avg}	530 to 630°F
(b)	Steam line pressure	0 to 1300 psig
(c)	Containment pressure	0 to 60 psig
(d)	Refueling water storage tank level	0 to 140 inches (suppressed range)

7.3.2 Analysis

7.3.2.1 Failure Mode Effects Analysis

Failure mode and effects analyses have been performed, Reference 5, on ESF systems equipment within the Westinghouse scope of supply. The Seabrook ESF systems, although not identical, have been designed to equivalent safety design criteria.

7.3.2.2 Compliance with Standards and Design Criteria

Discussion of the General Design Criteria (GDC) is provided in various sections of Chapter 7 where a particular GDC is applicable. Applicable GDCs include Criteria 13, 20, 21, 22, 23, 24, 35, 37, 38, 40, 43 and 46 of the 1971 GDC. Compliance with certain IEEE Standards is presented in Subsections 7.1.2.6, 7.1.2.7, 7.1.2.10 and 7.1.2.11. Compliance with Regulatory Guide 1.22 is discussed in Subsection 7.1.2.5. The discussion given below shows the Engineered Safety Features Actuation System complies with IEEE Standard 279-1971, Reference 4.

a. Single Failure Criteria

The discussion presented in Subsection 7.2.2.2c is applicable to the Engineered Safety Features Actuation System, with the following exception.

In the Engineered Safety Features Actuation System, a loss of instrument power will call for actuation of Engineered Safety Features equipment controlled by the specific bistable that lost power (containment pressure Hi-3 and RWST level low excepted). The actuated equipment that does not fail to the actuated condition must have power to comply. The power supply for the protection systems is discussed in Chapter 8. For containment pressure Hi-3 and RWST level low, the final bistables are energized to trip to avoid spurious actuation. In addition, manual containment spray requires simultaneous actuation of two manual controls. This is considered acceptable because spray actuation on High-3 containment pressure signal provides automatic initiation of the system via protection channels. Moreover, two sets (two switches per set) of containment spray manual initiation switches are provided to meet the requirements of IEEE Standard 279-1971. Also it is possible for all Engineered Safety Features equipment (valves, pumps, etc.) to be individually manually actuated from the control board. Hence, a third mode of containment spray initiation is available. The design meets the requirements of Criteria 21 and 23 of the 1971 GDC.

b. Equipment Qualification

Equipment qualifications are discussed in Sections 3.10 and 3.11.

c. Channel Independence

The discussion presented in Subsection 7.2.2.2c is applicable. The Engineered Safety Features Actuation System slave relay outputs from the solid-state logic protection cabinets are redundant, and the actuations associated with each train are energized up to and including the final actuators by the separate AC power supplies which power the logic trains.

d. Control and Protection System Interaction

The discussions presented in Subsection 7.2.2.2c are applicable.

e. Capability for Sensor Checks and Equipment Test and Calibration

The discussions of system testability in Subsection 7.2.2.2c are applicable to the sensors, signal processing, and logic trains of the Engineered Safety Features Actuation System.

The following discussions cover those areas in which the testing provisions differ from those for the Reactor Trip System.

1. Testing of Engineered Safety Features Actuation Systems

The Engineered Safety Features Actuation Systems are tested to provide assurance that the systems will operate as designed and will be available to function properly in the unlikely event of an accident. The testing program meets the requirements of Criteria 21, 37, 40 and 43 of the 1971 GDC and Regulatory Guide 1.22 as discussed in Subsection 3.1.4 and 7.1.2.5. The tests described in Subsection 7.2.2.2c and further discussed in Subsection 6.3.4 meet the requirements on testing of the Emergency Core Cooling System as stated in GDC 37 except for the operation of those components that will cause an actual safety injection or are not compatible with plant operation. The test, as described, demonstrates the performance of the full operational sequence that brings the system into operation, the transfer between normal and emergency power sources and the operation of associated cooling water systems. The safety injection and residual heat removal pumps are started and operated and their performance verified in a separate test discussed in Subsection 6.3.4. When the pump tests are considered in conjunction with the emergency core cooling system test, the requirements of GDC 37 on testing of the Emergency Core Cooling System are met as closely as possible without causing an actual safety injection.

Testing as described in Subsections 6.3.4 and 7.2.2.2c provides complete periodic testability during reactor operation of all logic and components associated with the Emergency Core Cooling System. This design meets the requirements of Regulatory Guide 1.22 as discussed in the above sections. The program is as follows:

(a) Prior to initial plant operations, Engineered Safety Features System tests will be conducted.

(b) Subsequent to initial startup, Engineered Safety Features System tests will be conducted during each regularly scheduled refueling outage.

(c) During online operation of the reactor, all of the Engineered Safety Features Actuation System monitoring and logic circuitry will be fully tested. All active components will be tested at the operational test frequency. A lower testing frequency may be justified if adequate reliability is assured. Passive components will be tested at the operational test frequency where practicable. All components will be tested at the channel calibration frequency. In addition, essentially all of the Engineered Safety Features actuated equipment, with the exceptions listed in Section 7.1.2.5, will be fully tested. The remaining actuated equipment whose operation is not compatible with continued online plant operation will be checked by means of continuity check of associated testable actuation devices or overlapping testing.

(d) During normal operation, the operability of testable final actuation devices of the Engineered Safety Features Systems will be tested by manual initiation from the control room.

2. <u>Performance Test Acceptability Standard for the "S" (Safety Injection Signal) and for the "P" (the Automatic Demand Signal for Containment Spray Actuation) Actuation Signals Generation</u>

The basis for Engineered Safety Features Actuation Systems acceptability will be the successful completion of overlapping tests (see Figure 7.3-1).

Channel checks of process indications verify operability of the sensors and the associated signal processing equipment. Channel operational tests performed with the channel in trip verify the operability of the channels from the signal processing equipment input through to and including the logic input relays.

Bypass Test Instrumentation (BTI) permits the testing of the Process Instrumentation System channels in bypass instead of in trip. Channel operational tests performed with the Process Instrumentation channel in bypass verify the operability of the channel from its input to the output of the signal processing equipment. Input relays for functions tested in bypass, except for the input relays associated with containment pressure Hi-3 and RWST level low, are tested at the refueling frequency.

The input relays associated with the containment pressure Hi-3 function are tested during the solid-state logic testing. The input relays associated with RWST level low are tested during the operational test.

Solid-state logic testing also checks the digital signal path from the input to the logic matrices to the inputs to the slave relays. Logic testing includes continuity tests on the coils of the output slave relays. Final actuator testing operates the output slave relays and verifies operability of those devices which require Engineered Safety Features Actuation and which can be tested without causing plant upset. A continuity check is performed on the testable actuation devices of the untestable devices. Operation of the final devices is confirmed by control board indication and visual observation that the appropriate pump breakers close and automatic valves have completed their travel.

The basis for acceptability for the Engineered Safety Features interlocks is control board indication of proper receipt of the signal upon introducing the required input at the appropriate setpoint.

3. <u>Frequency of Performance of Engineered Safety Features Actuation Tests</u>

Test frequencies are specified in the Technical Specifications. References 6 and 7 document the operational test frequency and justify channel calibration frequency testing of the SSPS input relays.

4. <u>Engineered Safety Features Actuation Test Description</u>

The following sections describe the testing circuitry and procedures for the online portion of the testing program. The guidelines used in developing the circuitry and procedures are:

(a) The test procedures must not involve the potential for damage to any plant equipment.

(b) The test procedures must minimize the potential for accidental tripping.

(c) The provisions for online testing must minimize complication of Engineered Safety Features Actuation circuits so that their reliability is not degraded.

(d) The active components are tested at the operational test frequency unless the channel calibration frequency is justified. The passive components are tested at the operational test frequency where practicable.

(e) All active and passive components required for each protective function will be tested at the channel calibration frequency.

5. <u>Description of Initiation Circuitry</u>

Several systems which provide the specific functions listed in Subsection 7.3.1.1a compose the Engineered Safety Features Actuation Systems, the majority of which may be initiated by different process conditions and be reset independently of each other.

The remaining functions are initiated by a common signal (safety injection) which in turn may be generated by different process conditions.

In addition, operation of all other vital auxiliary support systems, such as component cooling and service water, is initiated by the safety injection signal.

The output of each of the initiation circuits consists of a master relay which drives slave relays for contact multiplication as required. The logic, master, and slave relays are mounted in the Solid-State Protection System cabinets designated Train A, and Train B, respectively, for the redundant counterparts. The master and slave relay circuits operate various pump and fan circuit breakers or starters, motor-operated valve contactors, solenoid-operated valves, emergency generator starting, etc.

6. <u>Process Monitoring Testing</u>

Process Monitoring testing is identical to that used for reactor trip circuitry and is described in Subsection 7.2.2.2c.

Exceptions to this are containment pressure Hi-3 and Refueling Water Storage Tank (RWST) level low, which are energized to actuate channels. A test point for the containment pressure Hi-3 channels is provided to permit continuity testing of comparator trip switch BTI relay contacts, wiring between the comparator and the Solid-State Protection System (SSPS) input relay, and the input relay coil during operational testing. The RWST level low input relays are tested as part of the operational test.

7. <u>Solid-State Logic Testing</u>

Except for containment pressure Hi-3 channels, logic testing is the same as that discussed in Subsection 7.2.2.2c and 7.2.3. The containment pressure Hi-3 channels have special test switches which are used to energize the input relays as part of the logic test. During logic testing of one train, the other train can initiate the required engineered safety features function. For additional details, see Reference 2.

8. <u>Actuator Testing</u>

At this point, testing of the initiation circuits through operation of the master relay and its contacts to the coils of the slave relays, with the exception of the normally energized SSPS input relays for channels tested in bypass, has been accomplished. The Engineered Safety Features Actuation System (ESFAS) logic slave relays in the solid-state protection system output cabinets are subjected to coil continuity tests by the output relay tester in the SSPS cabinets. Slave relays (K601, K602, etc.) do not operate because of reduced voltage applied to their coils by the mode selector switch (TEST/OPERATE). A multiple position master relay selector switch selects the master relays and corresponding slave relays to which the coil continuity test voltage applied. The master relay selector switch is returned to "OFF" before the mode selector switch is placed back in the "OPERATE" mode. However, failure to do so will not result in defeat of the protective function. The ESFAS slave relays are activated during testing by the online test cabinet, so that overlap testing is maintained.

The engineered safety features actuation system final actuation device or actuated equipment testing is performed from the engineered safeguards test cabinets. These cabinets are located near the solid-state logic protection system equipment. There is one set of test cabinets provided for each of the two protection Trains A and B. Each set of cabinets contains individual test switches necessary to actuate the slave relays. To prevent accidental actuation, test switches are of the type that must be rotated and then depressed to operate the slave relays.

Assignments of contacts of the slave relays for actuation of various final devices or actuators have been made so that groups of devices or actuated equipment, can be operated individually during plant operation without causing plant upset or equipment damage. In the unlikely event that a safety injection signal is initiated during the test of the final device that is actuated by this test, the device will already be in its safeguards position.

During this last procedure, close communication between the main control room operator and the tester at the test cabinet is required. Prior to the energizing of a slave relay, the operator in the main control room assures that plant conditions will permit operation of the equipment that will be actuated by the relay. After the tester has energized the slave relay, the main control room operator observes that all equipment has operated as indicated by appropriate indicating lamps, monitor lamps and annunciators on the control board and records all operations. He then resets all devices and prepares for operation of the next slave-relay actuated equipment.

By means of the procedure outlined above, all engineered safety features devices actuated by engineered safety features actuation systems initiation circuits, with the exceptions noted in Subsection 7.1.2.5 under a discussion of Regulatory Guide 1.22 are operated by the automatic circuitry.

9. Actuator Blocking and Continuity Test Circuits

Those few final actuation devices that cannot be designed to be actuated during plant operation (discussed in Subsection 7.1.2.5) have been assigned to slave relays for which additional test circuitry has been provided to individually block actuation to a final device upon operation of the associated slave relay during testing. Except for the main steam isolation valve (MSIV) operation of these slave relays, including contact operations, and continuity of the electrical circuits associated with the final devices' control are checked in lieu of actual operation. The circuits provide for monitoring of the slave relay contacts, the devices' control circuit cabling, control voltage and the devices' actuation solenoids. The MSIVs are controlled by a solid-state logic that is not compatible with the standard SSPS test circuitry. The MSIVs are blocked from actuation during slave relay testing by a special SSPS test circuit that sends a signal to the MSIV logic cabinets to block the final MSIV logic gate. A light on the MSIV logic test panel on the MCB is illuminated to indicate that the MSIV closing logic is blocked. Operation of the slave relay will test all the MSIV logic up to the final gate and illuminate a light on the MSIV logic test panel to indicate a satisfactory test. After slave relay testing the SSPS block signal is removed, returning the MSIV logic cabinets to their normal condition. The final logic gate, MSIV logic cabinet output relay and MSIV actuating solenoid are tested by partial stroke exercising of the MSIVs. This overlapping test of the

MSIV logic with MSIV exercising provides a complete test of the MSIVs.

Interlocking prevents blocking the output from more than one output relay in a protection train at a time. Interlocking between trains is also provided to prevent continuity testing in both trains simultaneously; therefore, the redundant device associated with the protection train not under test will be available in the event protection action is required. If an accident occurs during testing, the automatic actuation circuitry will override testing as noted above. One exception to this is that if the accident occurs while testing a slave relay whose output must be blocked, those few final actuation devices associated with this slave relay will not be actuated; however, the redundant devices in the other train would be operational and would perform the required safety function. Actuation devices to be blocked are identified in Subsections 7.1.2.5a through j.

The continuity test circuits for these components that cannot be actuated on line are verified by proving lights on the safeguards test cabinet.

The charging and letdown isolation valves (described in Subsection 7.1.2.5k) are blocked by administrative controls. If an accident occurs while testing, the redundant equipment in the other train would be operational and would perform the required safety function.

The letdown heat exchanger component cooling water outlet isolation valve (described in Subsection 7.1.2.5l) is blocked by administrative controls. If an accident occurs while testing, the valve will be immediately closed by manual action, removing the nonessential heat load from the Component Cooling Water System.

The Chemical and Volume Control System TK-1 outlet isolation valves (described in Subsection 7.1.2.5n) are blocked by administrative controls. If an accident occurs while testing, the redundant equipment in the other train would be operational and would perform the required safety function.

The Refueling Water Storage Tank TK-8 to charging pump isolation valves (described in Subsection 7.1.2.5o) are blocked by administrative controls. If an accident occurs while testing, the redundant equipment in the other train would be operational and would perform the required safety function. The typical schemes for blocking operation of selected protection function actuator circuits are shown in Figure 7.3-2

as details A and B. The schemes operate as explained below and are duplicated for each safeguards train.
Detail A shows the circuit for contact closure for protection function actuation. Under normal plant operation, and equipment not under test, the test lamps "DS*" for the various circuits will be energized. Typical circuit path will be through the normally closed test relay contact "K8*" and through test lamp connections 1 to 3. Coils "X1" and "X2" will be capable of being energized for protection function actuation upon closure of solid-state logic output relay contacts "K*." Coil "X1" is typical for a motor control center starter coil, "X2" coil is typical for a breaker closing auxiliary coil, motor starter master coil, coil of a solenoid valve, auxiliary relay, etc. When the contacts "K8*" are opened to block energizing of coil "X1" or "X2," the white lamp is de-energized and the slave relay "K*" may be energized to perform continuity testing. To verify operability of the blocking relay in both blocking and restoring normal service, open the blocking relay contact in series with lamp terminal 1 - the test lamp should be de-energized; close the blocking relay contact in series with the lamp terminal 1 - the test lamp should now be energized, which verifies that the circuit is now in its normal, i.e., operable condition.

Detail B shows the circuit for contact opening for protection function actuation. Under normal plant operation, and equipment not under test, for 125V DC actuation devices the white test lamps "DS*" for the various circuits will be energized, and green test lamp "DS*" will be de-energized. Typical circuit path for white lamp "DS*" will be through the normally closed solid-state logic output relay contact "K*" and through test lamp connections 3 to 1. Coil "Y2" will be capable of being de-energized for protection function actuation upon opening of solid-state logic output relay contact "K*." Coil "Y2" is typical for a solenoid valve coil, auxiliary relay, etc. When the contact "K8*" is closed to block de-energizing of coil "Y2," the green test lamp is energized and the slave relay "K*" may be energized to verify operation (opening of its contacts). To verify operability of the blocking relay in both blocking and restoring normal service, close the blocking relay contact to the green lamp - the green test lamp should now be energized also; open this blocking relay contact - the green test lamp should now be de-energized, which verifies that the circuit is now in its normal, i.e., operable position.

10. <u>Time Required for Testing</u>

Analog testing can be performed at a rate of several channels per hour. Logic testing of both Trains A and B can be performed in less than 30 minutes each. Testing of actuated components (including those which can only be partially tested) is a function of control room operator availability. It requires several shifts to accomplish these tests. During this procedure automatic actuation circuitry overrides testing, except for those few devices associated with a single slave relay whose outputs must be blocked and tested only while blocked. Continuity testing associated with a blocked slave relay takes several minutes. During this time the redundant devices in the other trains are functional.

11. Summary of Online Testing Capabilities

The design described above provides capability for checking completely from the process signal to the logic cabinets and from there to the individual pump and fan circuit breakers or starter, valve contactors, pilot solenoid valves, etc., including all field cabling actually used in the circuitry called upon to operate for an accident condition. All passive and active components are tested at the refueling frequency. All active components are tested at the operational test frequency except for the normally energized SSPS input relays where refueling frequency testing has been justified (Reference 7). For those few devices whose operation could adversely affect plant or equipment operation (see Subsection 7.1.2.5), the same procedure provides for checking from the process signal to the logic rack. To check the final actuation device a continuity or overlapping blocked logic/manual exercise test of the individual control circuits is performed.

The procedures require testing at various locations.

(a) Monitoring circuit testing and verification of bistable setpoint are accomplished at the signal processing circuits. Verification of bistable relay operation is done at the main control room status lights or at the bistable output if the SSPS input relay is not tested during the operational test.

(b) Logic testing through operation of the master relays and low voltage application to slave relays is done at the SSPS logic rack test panel.

(c) Testing of pumps, fans and valves is done at the test panel located in the vicinity of the SSPS logic racks in combination with the control room operator.

(d) Continuity or overlapping blocked logic/manual exercise testing for those circuits that cannot be operated is done at the same test panel mentioned in (c) above.

12. Testing During Shutdown

Engineered Safety Features Actuation Systems tests will be performed periodically in accordance with the Technical Specifications with the Reactor Coolant System isolated from the Emergency Core Cooling System by closing the appropriate valves and other valve alignments as required to prevent unacceptable actions. A test safety injection signal will then be applied to initiate operation of active components (pumps and valves) of the Engineered Safety Features. This is in compliance with Criteria 37, 40 and 43 of the 1971 GDC.

13. Periodic Maintenance Inspections

The maintenance inspections which follow will be accomplished per applicable plant programs and procedures. The frequency will depend on the operating conditions and requirements of the reactor power plant. Typically maintenance inspections occur during preventive and corrective maintenance activities. If any degradation of equipment operation is noted, either mechanically or electrically, remedial action is taken to repair, replace, or readjust the equipment.

Typical maintenance inspections include the following:

(a) Check cleanliness of exterior and interior surfaces

(b) Check fuses for corrosion

(c) Inspect for loose or broken control knobs and burned out indicator lamps

(d) Inspect for moisture and condition of cables and wiring

(e) Mechanically check connectors and terminal boards for looseness, poor connection, dirt or corrosion

(f) Inspect the components of an assembly for signs of overheating or component deterioration

(g) Perform complete system operating check.

The balance of the requirements listed in Reference 4 (Paragraphs 4.11 through 4.22) are discussed in Subsection 7.2.2.2a. Paragraph 4.20 receives special attention in Section 7.5.

f. Manual Resets and Blocking Features

The manual reset and block control associated with safety injection actuation is provided to permit manual control of components actuated by the safety injection signal and to prevent automatic re-actuation of safety injection once it is blocked. Manual reset and block cannot be performed until sufficient time has elapsed after safety injection actuation to permit the actuated equipment to perform the required functions. Subsequent automatic safety injection actuations will be blocked if the reactor trip interlock (P-4) is present. A status light is illuminated to indicate when automatic safety injection actuation is blocked. The automatic safety injection block circuit is returned to normal when the reactor is not tripped.

Subsequent manual safety injection actuations are not blocked if the automatic signal has been cleared by operation of the automatic block circuit or clearing of the condition that resulted in an automatic actuation.

The manual reset feature associated with ESF actuation is provided in the standard design of the Westinghouse Solid-State Protection System design for two basic purposes. First, the feature permits the operator to start an interruption procedure of automatic ESF in event of false initiation of an actuate signal. Second, although ESF performance is automatic, the reset feature enables the operator to start a manual takeover of the system to handle unexpected events that can be better dealt with by operator appraisal of changing conditions following an accident.

It is most important to note that manual control of the ESF system does not occur, once actuation has begun, by just resetting the associated logic devices alone. Components will seal in (latch) or complete the protective action before reset of the actuate signal is credible so that removal of the actuate signal, in itself, will neither cancel nor prevent completion of protective action or provide the operator with manual override of the automatic system by this single action. In order to take complete control of the system to interrupt its automatic performance, the operator must take deliberate action to individually operate or realign affected equipment.

The manual reset feature associated with ESF, therefore, does not perform a bypass function. It is merely the first of several manual operations required to take control from the automatic system or interrupt its completion should such an action be considered necessary.

In the event that the operator anticipates system actuation and erroneously concludes that it is undesirable or unnecessary and imposes a standing reset condition in one train (by operating and holding the corresponding reset switch at the time the initiate signal is transmitted) the other train will automatically carry the protective action to completion. In the event that the reset condition is imposed simultaneously in both trains at the time the initiate signals are generated, the automatic sequential completion of system action is interrupted and control has been taken by the operator. Manual takeover will be maintained, even though the reset switches are released, if the original initiate signal exists. Should the initiate signal then clear and return again, automatic system actuation will repeat.

Note also that any time delays imposed on the system action are to be applied after the initiating signals are latched. Delay of actuate signals for fluid systems lineup, load sequencing, valve stroke time etc., do not provide the operator time to interrupt automatic completion, with manual reset alone, as would be the case if time delay was imposed prior to sealing of the initial actuate signal.

The manual block features associated with pressurizer and steam line safety injection signals provide the operator with the means to block initiation of safety injection during plant startup. These block features meet the requirement of Paragraph 4.12 of IEEE Standard 279-1971 in that automatic removal of the block occurs when plant conditions require the protection system to be functional.

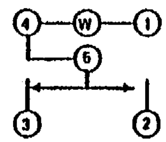
g. Manual Initiation of Protective Actions (Regulatory Guide 1.62)

There are four individual main steam stop valve momentary control switches (one per loop) mounted on the control board. Each switch when actuated, will close one of the main steam line isolation valves. In addition, there are two system level switches. Operating either switch will actuate all four main steam line isolation valves on the system level. Manual initiation of switchover to recirculation is in compliance with Section 4.17 of IEEE Standard 279-1971, with the following comment:

Manual initiation of either one of two redundant safety injection actuation main control board-mounted switches provides for actuation of the components required for reactor protection and mitigation of adverse consequences of the postulated accident, including delayed actuation of sequenced started emergency electrical loads as well as components providing switchover from the safety injection mode to the cold leg recirculation mode (see Subsection 7.6.5) following a loss of primary coolant accident. Therefore, once safety injection is initiated, those components of the Emergency Core Cooling System (see Section 6.3) that are realigned as part of the semi-automatic switchover, go to completion on low refueling storage tank water level without any manual action. Manual operation of other components or manual verification of proper position as part of emergency procedures is not precluded nor otherwise in conflict with the above described compliance to Paragraph 4.17 of IEEE Standard 279-1971 of the semi-automatic switchover circuits.

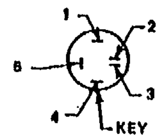
No exception to the requirements of IEEE Standard 279-1971 has been taken in the manual initiation circuit of safety injection. Although paragraph 4.17 of IEEE Standard 279-1971 requires that a single failure within common portions of the protective system shall not defeat the protective action by manual or automatic means, the standard does not specifically preclude the sharing of initiated

TEST LIGHT DS*
DEV 3



ILLUMINATED PUSHBUTTON SWITCH
WITH 28V LAMP NO. 327
(EXCEPT AS NOTED)

REAR OF PANEL



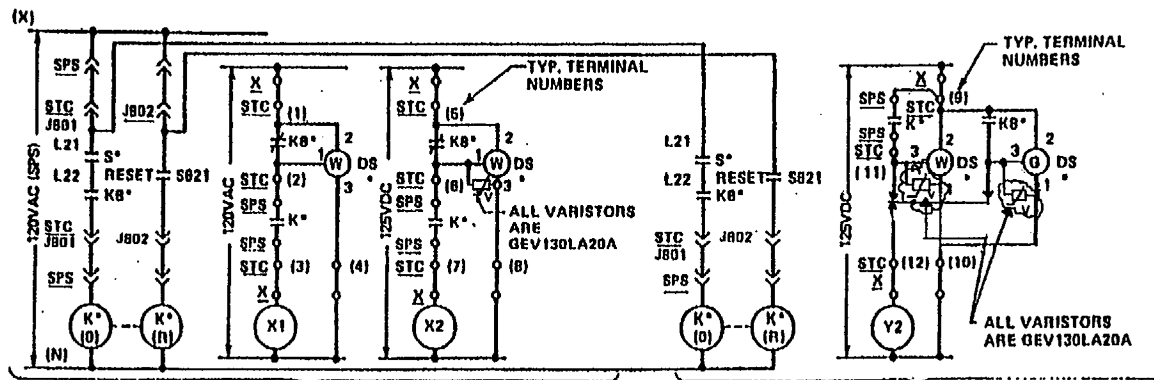
CONTACT LOCATION SCHEME

GENERAL NOTES:*

1. CIRCUITRY AND HARDWARE FOR REDUNDANT PROTECTION TRAINS "A" AND "B" TEST CABINETS ARE DUPLICATE EXCEPT AS NOTED
   A – TRAIN "A" ONLY
   B – TRAIN "B" ONLY
2. IN DETAILS A & B THE SYMBOL * REPRESENTS THE SUFFIX NUMBERS OF THE DEVICE REFERENCED.
   EXAMPLE:
   K* – SPS RELAY, K601, K602, ETC.
   K(0) – OPERATING COIL
   K(R) – RESET COIL
   S* – STC TEST SWITCH, S802, S834 ETC.
   K8* – STC RELAY, K811, K817, ETC.
   DS* – STC LIGHT, DS8009, DS8077, ETC
3. "DETAIL A" & "B" TYPE CIRCUITS ARE DETAILED ON THE SCHEMATICS, "DETAIL B" CIRCUITS WILL BE SUBSTITUTED FOR "DETAIL A" CIRCUITS WHERE REQUIRED.

LOCATION LEGEND

SPS – SOLID STATE PROTECTION SYSTEM
STC – SAFEGUARDS TEST CABINET
X – SWGR, MCC, AUXILIARY RELAY RACK, ETC.
ASC – AUXILIARY SAFEGUARDS CABINET



DETAIL A TYPICAL PROTECTION ACTUATION CIRCUIT BLOCKING SCHEMES (CONTACT CLOSURE FOR ACTUATION)

DETAIL B TYPICAL PROTECTION ACTUATION CIRCUIT BLOCKING SCHEMES (CONTACT OPENING FOR ACTUATION)

*DETAILS A AND B OF THIS FIGURE ARE NOT TO BE CONFUSED WITH ALPHA DESIGNATION OF LOGIC TRAINS A AND B

G:\Images_P\Images_P.UFSAR\732.ds4

Seabrook Station Updated Final Safety Analysis Report	Engineered Safeguards Test Cabinet Index, Notes and Ledgend	
	Rev. 07	Figure 7.3-2

TABLE 7.4-1
(Sheet 9 of 9)

EQUIPMENT REQUIRED FOR SAFE SHUTDOWN

Description	Device	RSS Control Location	Instrumentation Location MCB	CP108A	CP108B	Local
h. Sampling:						
RCS Sampling (Loop #1)	RC-FV-2832 RC-FV-2894	CP-108A CP-108A				
RCS Sampling (Loop #3)	RC-FV-2833 RC-FV-2896	CP-108B CP-108B				
RHR Local Sample Valves	RH-V-8 RH-V-44	Local Local				
i. Solid State Protection System (SSPS):						
SSPS Output Train A	MM-CP-12	Distr. Panel PP-1A				
SSPS Output Train B	MM-CP-13	Distr. Panel PP-1B				
j. Electrical Power Supply:						
Diesel Generator A	DG-1A	Local				
Diesel Generator B	DG-1B	Local				

- Reactor Trip Signals
- ESF Actuation Signals
- Certain Technical Specification Deviations
- Important Systems

The annunciators are powered from instrumentation power sources that are independent of the power sources for the VAS.

Bypassed/inoperable condition of safety systems is displayed on the VAS and on status lamp arrays on the MCB - one per train. Refer to Subsection 7.1.2.6 for a complete discussion of compliance with Regulatory Guide 1.47.

7.5.4 Accident Monitoring Instrumentation

7.5.4.1 Compliance with Regulatory Guide 1.97

Regulatory Guide 1.97, Revision 3 endorses, subject to certain clarifications, ANSI/ANS 4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Nuclear Reactors." The guidance provided in Regulatory Guide 1.97 and ANS 4.5, with certain exceptions, and NUREG-0737 has been used in selecting the Seabrook Accident Monitoring Instrumentation (AMI).

The exceptions to the guidance provided in Regulatory Guide 1.97 and ANSI/ANS 4.5 are:

a. Not all the variables recommended by Regulatory Guide 1.97, Table 3 have been included in the AMI List. Specific deviations and the associated justifications are provided in Appendix 7A.

b. Not all the AMI characteristics recommended by Regulatory Guide 1.97, Table 3 have been met. Specific deviations and the associated justifications are provided in Appendix 7A.

c. The determination of performance requirements for AMI did not follow the guidance of Regulatory Guide 1.97, Section C.2.4 in that:

   1. Required accuracy of measurement was not determined in procuring the instrumentation. Instead, the accuracy of the as-procured instrumentation was determined and then reviewed for acceptability. Further details are provided in Subsection 7.5.4.4e.5.

   2. Except for meteorological monitoring instrumentation, response characteristics (time) have not been determined for instrumentation channels that provide monitoring functions only. The response time for these channels is similar to the response time determined for ESF actuation channels since similar hardware is used. Therefore, determination of the response time for each channel is not necessary. See

Subsection 2.3.3.3a for a description of the meteorological monitoring system.

7.5.4.2 Description of Variable Types

The accident monitoring variables are classified into five types (A, B, C, D or E) according to the monitoring function they perform. A definition of each type is provided in the following subsections.

a. Type A Variables

Type A variables for Seabrook Station are those variables to be monitored that provide the primary information for the control room operators to take specific preplanned manual actions for which no automatic control is provided. These actions are required for safety systems to accomplish their safety function for design basis accident events. Actions taken as a result of equipment failures, e.g., the "Response Not Obtained" column in the Emergency Response Procedures (ERPs), are excluded.

b. Type B Variables

Type B variables provide the most direct indication to monitor the accomplishment of the critical safety functions (CSFs). CSFs are those safety functions that are essential to prevent a direct and immediate threat to the health and safety of the public. The accomplishment of these functions ensures the integrity of the physical barriers against radiation releases.

The six CSFs for Seabrook are:

1. Subcriticality
2. Containment Integrity (including radioactive effluent control)
3. Heat Sink
4. Core Cooling
5. RCS Integrity
6. RCS Inventory

c. Type C Variables

Type C variables provide the most direct indication of the potential for or the actual breach of the barriers to fission product releases. These barriers are: fuel cladding, primary coolant pressure boundary, and Containment.

d. Type D Variables

Type D variables are those variables that provide information to indicate the operation of individual safety systems and nonsafety systems used in the mitigation of design basis accidents.

physical separation of redundant channels is discussed in Sections 8.3 and 7.1.

Where failure of one accident monitoring channel results in information ambiguity (i.e., the redundant displays disagree) that could lead operators to defeat or fail to accomplish a required safety function, backup information is provided to allow the operators to deduce the actual conditions in the plant. This is accomplished by providing additional independent channels of information of the same variable (an identical channel) or by providing an independent channel to monitor a different variable that bears a known relationship to the multiple channels (a diverse channel). Information on redundant/diverse channel availability is included in the operator training program.

For systems having redundant components, single channel monitoring of the redundant parts of the system is provided. Verifying the proper functioning of one the redundant parts of the system is sufficient to monitor the accomplishment of the safety function.

3. <u>Power Source</u>

Design Category 1 instrumentation is powered from safety-related power sources. Where momentary power interruption is not tolerable, uninterruptible power sources are used.

4. <u>Availability</u>

The Design Category 1 instrumentation channels will be available prior to an accident except for testing and maintenance as provided in Paragraph 4.11 of IEEE Standard 279-1971 or as specified in the Technical Specifications.

5. <u>Quality Assurance</u>

Quality Assurance for Design Category 1 instrumentation is provided in accordance with the QA Program described in Chapter 17 of the Updated FSAR. Conformance to appropriate regulatory guides is discussed both in Chapter 17 and Section 1.8 of the Updated FSAR.

6. <u>Display and Recording</u>

<u>Indication</u>: For design Category 1 variables, continuous, redundant indication is provided. This indication meets the applicable requirements for design Category I instrumentation.

<u>Recording</u>: Recording of instrumentation readout information

is provided for at least one of the redundant channels.

Trend Indication: Where direct and immediate trend or transient information is essential for operator information or action, this information is available from multiple displays such as:

- Dedicated recorders, or
- Dedicated ratemeters, or
- CRT display (via the plant computer) available on demand, or
- Plasma displays - available on demand by use of dedicated function push buttons.

For trend display channels, at least one of the display devices meets the applicable requirements for design Category 1 instrumentation.

7. Identification

Type A, B, C, & D instrumentation displays provided for operator use during accident conditions are identified by an orange nameplate containing black lettering.

8. Interfaces

The transmission of signals to the accident monitoring equipment from protection equipment is through isolation devices which are classified as part of the protection system.

No credible failure at the output of an isolation device will prevent the associated protection channel from meeting the minimum performance requirements considered in the design bases. Examples of credible failures include short circuits, open circuits, grounds, and the application of the maximum credible AC or DC potential (140V DC or 129V AC). Refer to Updated FSAR Subsection 7.2.2.2c.7 for further discussion.

c. Design Category 2 - Design and Qualification Criteria

1. Equipment Qualification

Design Category 2 instrumentation is environmentally qualified in accordance with IEEE 323-1974 and associated daughter standards. Further details on the methods used and compliance with associated regulations and Regulatory Guides are provided in Section 3.10.

2. Power Source

Design Category 2 instrumentation is powered from highly reliable power sources, very often Class 1E. Where momentary power interruption is not tolerable, uninterruptible power sources are used.

3. Quality Assurance

Quality Assurance for Design Category 2 instrumentation is provided by United Engineers and Constructors for the design, procurement and installation phases. Their QA Program contains the measures necessary to insure that the instrumentation has been properly specified, procured and installed. This program contains the applicable elements of 10 CFR 50, Appendix B.

Quality Assurance for the testing phase is provided by the standard testing procedures of the NHY Startup and Test Department. Auditable records are available for each Design Category 2 instrument.

Quality Assurance during the operational phase is provided under the North Atlantic Operational Quality Assurance Program (OQAP). Further details are provided in Section 17.2.

4. Display and Recording

Indication

For Design Category 2 instruments, either display on demand or continuous indication is provided.

Recording

Effluent radioactivity and area radiation variables are recorded.

Trend Indication

Where direct and immediate trend or transient information is essential for operator information or action, trend indication is provided. This indication consists of either dedicated recorders or CRT displays.

5. Identification

Types A, B and C instrumentation displays provided for operator use during accident conditions are identified by an orange nameplate containing black lettering.

6. Channel Availability

Design Category 2 instrumentation channels will be available prior to an accident as provided in the plant administrative procedures.

7. Interfaces

Same as Design Category 1.

d. Design Category 3 - Design and Qualification Criteria

1. Quality Assurance

This instrumentation is of high-quality commercial grade and is selected to withstand the expected plant service environment.

2. Display and Recording

Indication

The information display can be either continuous or available on demand.

Recording

Effluent radioactivity variables and meteorological variables are recorded.

Trend Indication

Where direct and immediate trend or transient information is essential for operation information or action, trend information is provided. Trend information may be from a dedicated recorder or available on demand from the plant computer system.

e. Design and Qualification Criteria Applicable to Design Categories 1, 2, and 3

1. Range

The range of the read-outs extends over the maximum expected range of the variable being measured. Where two or more instruments are needed to cover a particular range, overlapping of the instrument spans is provided.