GE NUCLEAR ENERGY

TIER 1 DESIGN CERTIFICATION MATERIAL

PILOT ITAAC EXAMPLES

FOR

THE GE ABWR DESIGN

JANUARY 17, 1992

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1.0 INTRODUCTION

The purpose of this memorandum is to present examples of proposed Tier 1 design certification material for the GE ABWR. This version dated 1/17/1992 and is an update of the original memorandum dated 9/20/1991. It includes revisions that:

1) reflect NRC comments on the 9/20/91 version

2) present examples of generic and design acceptance criteria (DAC) inspections, tests, analyses and acceptance criteria (ITAAC).

Section 2 contains revised versions of the material for each of the nine ABWR systems chosen for pilot examples. Each pilot starts with a proposed Tier 1 design description and then provides suggested inspections, tests, analyses, and acceptance criteria (ITAAC) based on the design description contents. The scope of these pilots is based on the nuclear industry's current understanding of the guiding principles governing scope and content of Tier 1 material. As presented in the NUMARC report of the Task Force on Inspections, Tests, Analyses and Acceptance Criteria, NUMARC 90-15, December 1990. An important supplementary document is the GE memorandum "Guidelines for Preparation of Inspections, Tests, Analyses and Acceptance Criteria (ITAAC)", A.J. James, December 1991.

As a result of ongoing discussions with NRC, it has become clear that the ABWR design certification ITAAC material will eventually include entries that are not part of the system-by-system approach that forms the basis for the nine pilot examples. These non-system attics will include:

- generic/discipline ITAAC

- design acceptance criteria (DAC) ITAAC

- Plant interface ITAAC

Section 3 provides examples of the first two categories of non-system ITAAC material. It is intended these items also be considered pilots in that they are representative examples which can be used to reach agreement on the required scope and content of this type of ITAAC.

It is intended that this memorandum be distributed to interested parties for review and comment. Based on feedback from this review, GE will (if necessary) modify the scope and content guidelines and will then initiate preparation of a full set of ABWR design descriptions and ITAAC for submittal to NRC as part of the ABWR design certification application.

1.1 PILOT EXAMPLES OF TIER 1 MATERIAL - INDIVIDUAL SYSTEMS

This section provides pilot examples of Tier 1 design descriptions and their associated ITAAC. These pilots have been chosen with the objective of covering representative examples of nuclear island mechanical systems, turbine island mechanical systems, structural/civil items, electrical systems, control and instrumentation systems. The section numbering system is based on the ABWR product structure.

Figure Information

For a number of the ITAAC, simplified figures have been included to help facilitate the design description. The figures contain information that uses the following conventions:

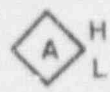
Line classification:

ASME Code Class 1 — - - — - - — - -
ASME Code Class 2 - - - - - - - - - - -
ASME Code Class 3 ————————
Non ASME code — - — - — - — -

Instrumentation:

Flow element	FE
Restricting orifice	RO
Temperature element	TE
Radiation element	RE
Level indicator	L
Pressure indicator	P

Equipment:

Gate valve	
Globe valve	
Check valve	
Valve type not specified	
Relief valve	
Open cirucit breaker	
Closed circuit breaker	
Annunciator (H=high, L=low)	

Valves are shown on the figures in their normal postition.

Valve Operators:

Motor	M
Nitrogen	N
Air	A

2.1.1 Reactor Pressure Vessel System

Design Description

The reactor pressure vessel system (RPVS) consists of (1) the reactor pressure vessel and its appurtenances, supports and insulation, and (2) the reactor internals enclosed by the vessel, excluding the core, in-core nuclear instrumentation, reactor internal pumps and control rod drives.

The reactor coolant pressure boundary (RCPB) portion of the RPVS retains integrity as a radioactive material barrier during normal operation and following abnormal operational transients and design basis accidents.

Certain reactor pressure vessel (RPV) internals support the core, flood the core during a design basis accident, and support instrumentation utilized during a design basis accident. Other RPV internals direct coolant flow, separate steam, hold material surveillance specimens and support instrumentation utilized for normal operation.

The reactor pressure vessel system provides guidance and support for the control rod drives. It also admits and distributes the sodium pentaborate from the standby liquid control system.

The RPV system restrains the control rod drive (CRD) in order to prevent the ejection of the control rod connected with the CRD in the event of a postulated failure of the reactor coolant boundary associated with the CRD housing. A restraint is also provided for the reactor internal pump (RIP) in order to prevent it from becoming a missile in case of a postulated failure of the reactor coolant boundary associated with the reactor internal pump.

Reactor Pressure Vessel, Appurtenances, Supports and Insulation

The reactor pressure vessel (RPV), as shown schematically in Figure 2.1.1, consists of a vertical, cylindrical pressure vessel of welded construction, removable top head and head closure bolting and seals. The vessel includes the cylindrical shell, flange, bottom head, reactor internal pump (RIP) casings, penetrations, brackets, nozzles, venturi shaped flow restrictors in the steam outlet nozzles, and the shroud support which includes the pump deck forming the partition between the RIP suction and discharge. The shroud support is an assembly consisting of a vertical circular plate, a horizontal annular pump deck plate and vertical stilt legs. This support carries the weight of peripheral fuel elements, neutron sources, core plate, top guide, shroud and shroud head with steam separators. It also supports laterally the fuel assemblies and the pump diffusers. The shroud support also sustains the differential pressures.

The control rod drives are mounted into the control rod drive housings. Sodium pentaborate solution from the standby liquid control system enters the vessel via one of the two high pressure core flooding lines and is distributed through the sparger connected to the line.

The control rod drive housings are inserted through and connected to the control rod drive penetrations (stub tubes) in the reactor vessel bottom head. The in-core neutron flux monitor housings are inserted through and connected to the bottom head.

A flanged nozzle is provided in the top head for bolting of the flange associated with the instrumentation for vibration test of internals.

The integral reactor vessel skirt supports the vessel on the RPV pedestal. Steel anchor bolts extend through the pedestal and secure the flange of the skirt to the pedestal. RPV stabilizers are provided in the upper portion of the RPV to resist horizontal loads. Lateral supports among the CRD housings and in-core housings are provided by restraints which at the periphery are supported off the CRD housing restraint beams.

A restraint consisting of a pair of energy absorbing rods is provided to prevent the RIP from being a missile in case of a failure in the casing weld with the bottom head penetration. The restraint is connected to lugs on the RPV bottom head and the RIP motor cover.

The reactor pressure vessel insulation is supported from the biological shield wall surrounding the vessel. Insulation for the upper head and flange is supported by a steel frame independent of the vessel and piping. Insulation access panels and insulation around penetrations are designed for ease of installation and removal for vessel inservice inspection and maintenance operation.

The reactor coolant pressure boundary portion of the RPV and appurtenances and the supports (RPV skirt, stabilizer and CRD housing/in-core housing restraints and beams) are classified as Quality Group A, Seismic Category I. The design, materials, manufacturing, fabrication, testing, examination, and inspection used in the construction of these components meet requirements of ASME Code Class 1 vessel and supports, respectively. The shroud support is classified as Quality Group C, Seismic Category I, and designed and fabricated to ASME Code Class CS, core support structures. Hydrostatic test of the RPV is performed in accordance with the requirements for ASME Code Class 1 vessels. The design pressure and temperature of the RPV are 87.9 kg/cm^{2}g and 302°C, respectively. The components are code-stamped according to their code class.

The materials used in the reactor coolant pressure boundary (RCPB) portion of the RPV and appurtenances are as listed here and these or their equivalents will be used: ASME SA-533, Type B, Class 1 (plate); SA-508, Class 3 (forging); SA-508, Class 1 (forging); SB-166, Type 600 (UNS 06600, forging); SA-182, F316L (maximum carbon 0.020%) or F316 (maximum carbon 0.020% and nitrogen from 0.060 to 0.120%, forging); and SA-540, Grade B23 or B24 (bolting).

The materials of the low alloy plates and forging used in construction of the RPV are melted to fine grain practice and are supplied in quenched and tempered condition. Vacuum degassing is performed to lower the hydrogen level and improve the cleanliness of the low-alloy steels.

Electroslag welding is not applied for structural welds. Preheat and interpass temperatures employed for welding of low alloy steel meet or exceed the values given in ASME, Section III, Appendix D. Post weld heat treatment at 593°C minimum is applied to all low-alloy steel welds.

Pressure boundary welds are given an ultrasonic examination in addition to the radiographic examination performed during fabrication. The ultrasonic examination method, including calibration, instrumentation, scanning sensitivity, and coverage, is based on the requirements imposed by ASME, Section XI, Appendix I. Acceptance standards are equivalent or more restrictive than required by ASME, Section XI.

A stainless steel weld overlay is applied to the interior of the cylindrical shell and the steam outlet nozzle. Other nozzles and the RIP motor casing do not have cladding. The bottom head is clad with Ni-Cr-Fe alloy. The RIP penetrations are clad with Ni-Cr-Fe alloy or stainless steel alternatively.

The fracture toughness tests of pressure boundary ferritic materials, weld metal and heat affected zone (HAZ) are performed in accordance with the requirements for ASME Code Class 1 vessel. Both longitudinal and transverse specimens are used to determine the minimum upper shelf energy level of the core beltline materials. Separate, unirradiated baseline specimens are used to determine the transition temperature curve of the core beltline base materials, weld metal and HAZ.

For the vessel material surveillance program, specimens are manufactured from the material actually used in the reactor beltline region and weld typical of those in the beltline region, thus representing base metal, weld material, and the weld heat-affected zone material. The plate and weld specimens are heat treated in a manner which simulates the actual heat treatment performed on the core region shell plates of the completed vessel. Each in-reactor surveillance capsule contains Charpy V-notch specimens of base metal, weld metal, and heat-affected zone material, and tensile specimens from base metal and weld metal. Brackets are welded to the vessel cladding in the core belt region for retention of the detachable holders, each of which contains a number of the specimen capsules. Neutron dosimeters and temperature monitors are located within the capsules.

Access for examinations of the installed RPV is incorporated into the design of the vessel, biological shield wall and vessel insulation.

Reactor Pressure Vessel Internals

The major reactor internal components that are included in the RPVS are:

a. Core Support Structures:

Shroud; shroud support (integral to the RPV and including the internal pump deck); core plate; top guide; fuel supports (orificed fuel supports and peripheral fuel supports); control rod guide tubes; and

b. Other Reactor Internals:

Control rods; feedwater spargers; RHR/ECCS low pressure flooding spargers; ECCS high pressure core flooding spargers and coupling; in-core guide tubes and stabilizers; core plate differential pressure lines; surveillance specimen holders; shroud head and steam separators assembly; and steam dryer assembly.

A general assembly drawing of these reactor internal components is shown in Figure 2.1.1. The core support structures locate and support the fuel assemblies, form partitions within the reactor vessel to sustain pressure differentials across the partitions, and direct the flow of the coolant water.

The shroud support, shroud, and top guide make up a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the downcomer annulus.

The core plate consists of a circular stainless steel plate with round openings and is stiffened with a rim and beam structure. The core plate provides lateral support and guidance for the control rod guide tubes, in-core flux monitor guide tubes, peripheral fuel supports and startup neutron sources. The last two items are also supported vertically by the core plate.

The top guide consists of a circular plate with square openings for fuel with a cylindrical side forming an upper shroud extension. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, less than four fuel assemblies. Holes are provided in the bottom of the support intersections to anchor the in-core instrumentation detectors and startup neutron sources.

The fuel supports are of two types. The peripheral fuel supports are located at the outer edge of the active core and are not adjacent to control rods. Each peripheral fuel support supports one peripheral fuel assembly and contains an orifice to provide coolant flow to the fuel assembly. Each orificed fuel support supports four fuel assemblies vertically upward and horizontally and contains four orifices to provide coolant flow distribution to each fuel assembly. The orificed fuel supports rest on the top of the control rod guide tubes which are

supported laterally by the core plate. The control rods pass through cruciform openings in the center of the orificed fuel support.

The control rod guide tubes located inside the vessel extend from the top of the control rod drive housings up through holes in the core plate. Each guide tube is designed as the guide for the lower end of a control rod and as the support for an orificed fuel support. This locates the four fuel assemblies surrounding the control rod. The lower end of the of the guide tube is supported by the control rod drive housing, which in turn transmits the weight of the guide tube, fuel supports, and fuel assemblies to the reactor vessel bottom head. The control rod guide tubes also contain holes, near the top of the control rod guide tube and below the core plate, for coolant flow to the orificed fuel supports.

The control rod guide tube base is provided with a device for coupling control rod drive (CRD) with it. The CRD is restrained from ejection, in the case of a stub tube weld failure, by the coupling of the CRD with the control rod guide tube base; in this event, the flange at the top of the guide tube will contact the core plate and restrain the ejection. The coupling will also prevent ejection if the housing fails at the stub tube weld; in this event, the guide tube remains supported on the intact upper housing.

The control rods are cruciform shaped neutron absorbing members that can be inserted or withdrawn from the core by the control rod drives to control reactivity and reactor power.

Each of the two feedwater lines is connected to three spargers via three RPV nozzles. The feedwater spargers, which also function as ECCS high or low pressure flooding spargers depending upon their connection to the line designated to receive high pressure or low pressure coolant flooding supply, respectively, are stainless steel headers located in the mixing plenum above the downcomer annulus. Each sparger in two halves, with a tee connected in the middle, is fitted to each feedwater nozzle with the tee. The sparger tee inlet is connected to the RPV nozzle safe end by a double thermal sleeve arrangement. Feedwater flow enters the center of the spargers and is discharged radially inward to mix the cooler feedwater with the downcomer flow from the steam separators and steam dryer before it contacts the vessel wall.

The design feature of the two residual heat removal (RHR) shutdown cooling system spargers, which also function as ECCS low pressure flooding (LPFL) spargers, is similar to that of the feedwater spargers. Two lines of RHR shutdown cooling system enter the reactor vessel through the two diagonally opposite nozzles and connect to the spargers. The sparger tee inlet is connected to the RPV nozzle safe end by a thermal sleeve arrangement.

The two ECCS high pressure core flooding (HPCF) spargers and couplings are the means for directing high pressure ECCS flow to the upper end of the core. Each of the two HPCF lines enters the reactor vessel through a diagonally opposite nozzle with a thermal sleeve arrangement. The curved sparger

including the connecting tee is located around the inside of and is supported by the cylindrical portion of the top guide. The sparger tee is connected to the thermal sleeve by the HPCF coupling.

In-core guide tubes protect the in-core flux monitoring instrumentation from flow of water in the bottom head plenum. The in-core guide tubes extend from the top of the in-core housing to the top of the core plate. The local power range monitoring (LPRM) detectors for the power range neutron monitoring (PRNM) system and the detectors for the startup range neutron monitoring (SRNM) system are inserted through the guide tubes.

Two levels of stainless steel stabilizer latticework of clamps, tie bars and spacers give lateral support and rigidity to the guide tubes. The stabilizers are connected to the shroud and shroud support.

The core plate differential pressure (DP) lines enter the reactor vessel through reactor bottom head penetrations. Four pairs of the core plate DP lines enter the head in four quadrants through four penetrations and terminate immediately above and below the core plate to sense the pressure in the region outside the bottom of the fuel assemblies and below the core plate during normal operation.

Surveillance specimen capsules, which are held in capsule holders mentioned earlier, are located at three azimuths at a common elevation in the core beltline region. The capsule holders are non-safety related internals. The capsule holders are mechanically retained by capsule holder brackets welded to the vessel cladding in order to allow their removal and reattachment.

The shroud head and steam separators assembly includes the connecting standpipes and forms the top of the core discharge mixture plenum. The steam dryer assembly removes moisture from the wet steam leaving the steam separators. The extracted moisture flows down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus. The shroud head and steam separators assembly and the steam dryer assembly are non-safety related internals.

The core support structures are classified as Quality Group C, Seismic Category I. The design, materials, manufacturing, fabrication, examination and inspection used in the construction of the core support structures meet requirements of ASME Code Class CS structures. These structures are code-stamped accordingly. Other reactor internals are designed per the guidelines of ASME Code NG-3000 and are constructed so as not to adversely affect the integrity of the core support structures as required by NG-1122.

Special controls are exercised when austenitic stainless steel is used for construction of RPV internals in order to avoid cracking during service.

Design and construction of the RPV internals assure that the internals can withstand the effects of flow induced vibration (FIV).

Inspection, Test, Analyses and Acceptance Criteria

Table 2.1.1 provides a definition of the instructions, tests, and/or analyses together with associated acceptance criteria which will be undertaken for the reactor pressure vessel system.

Table 2.1.1: REACTOR PRESSURE VESSEL SYSTEM

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections Tests, Analyses	Acceptance Criteria
1. System configuration of the reactor pressure vessel system (RPVS) as described in Section 2.1.1.1 is shown on Figure 2.1.1.	1. Visual field inspections will be conducted of the installed RPVS key components identified in Section 2.1.1 and Figure 2.1.1.	1. The installed configuration of the RPVS will be considered acceptable if it complies with Figure 2.1.1 and Section 2.1.1.
2. The reactor coolant pressure boundary (RCPB) portion of the RPV and appurtenances and their supports are classified as Quality Group A, Seismic Category I. These components are designed, fabricated, examined and hydrotested in accordance with the rules of ASME Code Class 1 vessel or component support, and are code stamped accordingly. The core support structures are Quality Group C, Seismic Category I, and are designed, fabricated and examined in accordance with the rules of ASME Code Class CS structures, and are code-stamped accordingly.	2. Inspections will be conducted of ASME Code required documents and the Code stamp on the components.	2. Existence of necessary ASME Code required documents and the code stamps on the components confirm that the components in the RCPB of the RPV and the supports, and the core support structures are designed, fabricated and examined as ASME Code Class 1 and CS respectively. This also confirms that the RPV is hydrotested per the ASME Code Class 1 requirements.
3. The RCPB of the RPVS retains its integrity under internal pressure that will be experienced during the service.	3. A hydrostatic test of the RCPB will be conducted in accordance with the ASME Code requirements.	3. The results of the hydrostatic test must conform with the requirements in the ASME Code.
4. The materials used for RCPB portion of the RPV and appurtenances are certain proven low and high alloy steels with certain additional requirements for construction, as identified in Section 2.1.1. Special controls are exercised when austenitic stainless steel is used for construction of RPV internals in order to avoid cracking during service.	4. Inspection will be conducted of the records of materials, fabrication, and examination used in construction of the RCPB and austenitic stainless steel reactor internals.	4. Records of the materials and processes must confirm that the requirements specified for the RCPB in Section 2.1.1 are satisfied and that the manufacture and fabrication of the RPV internals made of austenitic stainless steel avoid potential for cracking in service. **Validation Attributes**: The following special controls are exercised when austenitic stainless steel is used in manufacture and fabrication of

Table 2.1.1: REACTOR PRESSURE VESSEL SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		4. (Continued) RPVS internals. Where stainless steel surfaces are exposed to water at temperatures above 93°C, low carbon (0.020% maximum) or nuclear grade materials (maximum C=0.020% with nitrogen added) or CF3 type castings are used. All materials are supplied in the solution heat treated condition. Sensitization tests are applied to assure that the material is in the annealed condition. During fabrication, any heating operation (except welding) between 427°-982°C is avoided, unless followed by solution heat treatment. During welding, heat input and interpass temperature are controlled. Weld filler material used is Type 308L/316L/309L or equivalent. All weld filler materials used have a minimum of 8 FN average (ferrite number) determined on undiluted weld pads by magnetic measuring instruments. During fabrication, cold work is controlled by applying limits in hardness, bend radii and surface finish on ground surfaces. Process controls are exercised during all stages of component manufacturing, fabrication and installation to minimize contaminants. Surface contaminants are removed prior to any heating operations.
5. The ferritic materials used in RCPB portion of the RPV and appurtenances are not susceptable to brittle fracture under pressure during the service.	5. Fracture toughness tests of the ferritic base, weld and heat affected zone (HAZ) metal used in the RCPB will be conducted in accordance with the requirements for ASME Class 1 components.	5. Records of the fracture toughness data of the RCPB ferritic materials must confirm that 1) the requirements of the ASME Code are met, and 2) the reactor vessel beltline materials will not be susceptible to brittle fracture during the service.

Table 2.1.1: REACTOR PRESSURE VESSEL SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

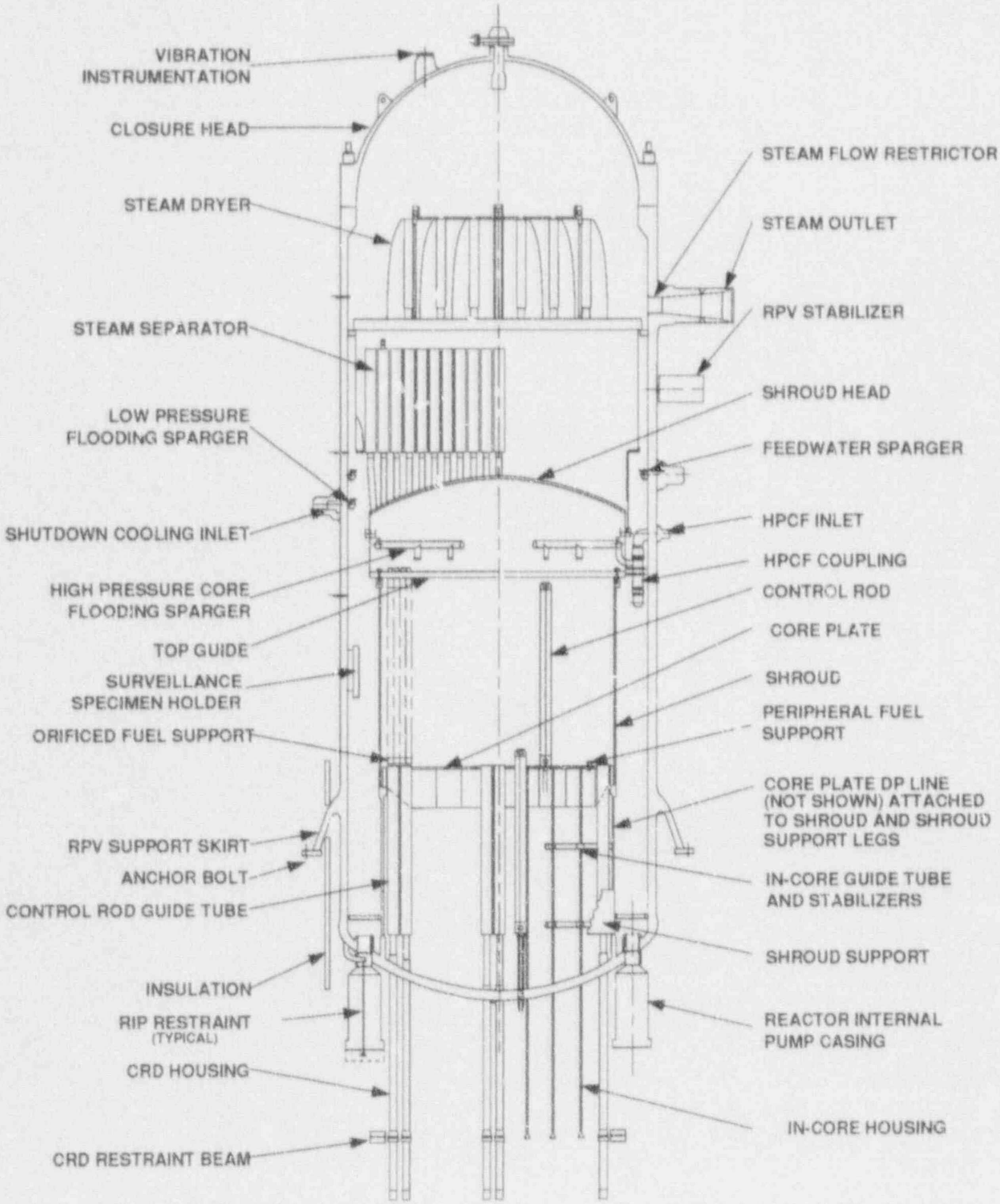
Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		5. (Continued) **Validation Attributes:** a. The minimum upper-shelf energy level for base and weld metal in reactor vessel beltline must be 10.4 kg-m. b. The predicted minimum upper-shelf energy level for base and weld metal in reactor vessel beltline at end of life must be 6.9 kg-m. c. The predicted value of adjusted reference temperature, RT_{NDT}, of base and weld metal in reactor vessel beltline at end of life must be 93°C or less.
6. Specimens for the surveillance program are selected from the vessel base metal and weld metal.	6. Inspection will be conducted of the records of the specimens selected from the reactor beltline region.	6. Records of the specimens with respect to location and orientation, types (tensile or Charpy V-notch), and quantities must meet the requirements of ASTM E-185
7. Design and construction of the RPV internals assure that the internals can withstand the effects of flow induced vibration (FIV).	7. A vibration test will be conducted of the reactor internals to verify the adequacy of the internals design, manufacture, and assembly with respect to the potential effects of FIV. The first-of-a-kind prototype internals will be flow tested by vibration instrumentation followed by inspection for damage. The internals in subsequent plants will be flow tested, but without vibration instrumentation, followed by inspection for damage.	7. Reactor vessel internals vibration is considered acceptable when results of the vibration analysis, vibration measurement testing and inspection of the internals indicate no sign of damage, loose parts, or excessive wear in the prototype test. The vibration of reactor internals in subsequent plants is considered acceptable when inspection of the internals indicate no sign of damage, loose parts, or excessive wear.

Table 2.1.1: REACTOR PRESSURE VESSEL SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. Access for examinations of the RPV is incorporated into the design of the vessel, biological shield wall and vessel insulation.	8. Visual inspection will be conducted of accessibility for examinations of the vessel and welds.	8. Provisions for access in the design of the vessel, biological shield wall, and vessel insulation shall be, in the minimum, as follows: The shield wall and vessel insulation behind the shield wall must be spaced away from the RPV outside surface. Access for the insertion of automated devices must be provided through removable insulation panels at the top of the shield wall and at access ports at reactor vessel nozzles. Access to the reactor pressure vessel welds above the top of the biological shield wall must be provided by removable insulation panels. The closure head must have removable insulation to provide access for manual ultrasonic examinations of its welds. Access to the bottom head to shell weld must be provided through openings in the RPV support pedestal and removable insulation panels around the cylindrical lower portion of the vessel. Access must be provided to partial penetration nozzle welds, i.e., CRD penetrations, instrumentation nozzles and recirculation internal pump penetration welds, for performance of the visual examinations. Access must be provided for examination of the attachment weld between the support skirt knuckle (forged integrally on the shell ring) and the RPV support skirt. Access must be provided to the balance of the support skirt for performance of visual examination.

Figure 2.1.1 REACTOR PRESSURE VESSEL SYSTEM KEY FEATURES



2.2.4 Standby Liquid Control System

The standby liquid control system (SLCS) is design to inject neutron absorbing poison using a boron solution into the reactor and thus provide back-up reactor shutdown capability independent of the normal reactivity control system based on insertion of control rods into the core. The system is capable of operation over a wide range of reactor pressure conditions up to and including the elevated pressures associated with an anticipated plant transient coupled with a failure to scram (ATWS).

The standby liquid control system (SLCS) is designed to provide the capability of bringing the reactor, at any time in a cycle, from full power and at all conditions to a subcritical condition with the reactor in the most reactive xenon-free state without control rod movement.

The SLCS consists of a boron solution storage tank, two positive displacement pumps, two motor operated injection valves which are provided in parallel for redundancy and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged through the 'B' high pressure core flooder (HPCF) subsystem sparger. Figure 2.2.4 shows major system components. Key equipment performance requirements are:

a.	Pump flow	100 gpm with both pumps running
a.	Maximum reactor pressure (for injection)	1250 psig
a.	Pumpable volume in storage tank (minimum)	6100 U.S. gal

The required volume of solution contained in the storage tank is dependent upon the solution concentration and this concentration can vary during reactor operations. A required boron solution volume/concentration relationship is used to define acceptable SLCS storage tank conditions during plant operation.

The SLCS is automatically initiated during an ATWS or can be manually initiated from the main control room. When the SLCS is automatically initiated to inject a liquid neutron absorber into the reactor, the following devices are actuated:

a. the two injection valves are opened;

b. the two storage tank discharge valves are opened;

c. the two injection pumps are started; and

d. the reactor water cleanup isolation valves are closed.

When the SLCS is manually initiated to inject a liquid neutron absorber into the reactor, the following devices are actuated by each switch:

a. one of the two injection valves is opened;

b. one of the two storage tank discharge valves is opened;

c. one of the two injection pumps is started; and

d. one of the reactor water cleanup isolation valves is closed.

The SLCS provides borated water to the reactor core to compensate for the various reactivity effects during the required conditions. These effects include xenon decay, elimination of steam voids, changing water density due to the reduction in water temperature, Doppler effect in uranium, changes in neutron leakage and changes in control rod worth as boron affects neutron migration length. To meet this objective, it is necessary to inject a quantity of boron which produces a minimum concentration of 850 ppm of natural boron in the reactor core at 70°F. To allow for potential leakage and imperfect mixing in the reactor system, an additional 25% (220) is added to the above requirement. The required concentration is achieved accounting for dilution in the RPV with normal water level and including the volume in the residual heat removal shutdown cooling piping. This quantity of boron solution is the amount which is above the pump suction shutoff level in the tank thus allowing for the portion of the tank volume which cannot be injected.

The pumps are capable of producing discharge pressure to inject the solution into the reactor when the reactor is at high pressure conditions corresponding to the system relief valve actuation.

The SLCS includes sufficient Control Room indication to allow for the necessary monitoring and control during design basis operational conditions. This includes pump discharge pressure, storage tank liquid level and temperature as well as valve open/close and pump on/off indication for those components shown on Figure 2.2.4 (with the exception of the simple check valves).

The SLCS uses a dissolved solution of sodium pentaborate as the neutron-absorbing poison. This solution is held in a storage tank which has a heater to maintain solution temperature above the saturation temperature. The heater is capable of automatic operation and automatic shutoff to maintain an acceptable solution temperature. The SLCS solution tank, a test water tank, the two positive displacement pumps, and associated valving is located in the secondary containment on the floor elevation below the operating floor. This is a Seismic Category I structure, and the SLCS equipment is protected from phenomena such as earthquakes, tornados, hurricanes and floods as well as from internal postulated accident phenomena. In this area, the SLCS is not subject to conditions such as missiles, pipe whip, and discharging fluids.

The pumps, heater, valves and controls are powered from the standby power supply or normal offsite power. The pumps and valves are powered and controlled from separate buses and circuits so that single active failure will not prevent system operation. The power supplied to one motor operated injection valve, storage tank discharge valve, and injection pump is powered from Division I, 480 VAC. The power supply to the other motor-operated injection valve, storage tank outlet valve, and injection pump is powered from Division II, 480 VAC. The power supply to the tank heaters and heater controls is connectable to a standby power source. The standby power source is Class 1E from an on-site source and is independent of the off-site power.

All components of the system which are required for injection of the neutron absorber into the reactor are classified Seismic Category I. All major mechanical components are designed to meet ASME Code requirements as shown below.

Component	ASME Code Class	Design Conditions Pressure	Temperature
Storage Tank	2	Static Head	150°F
Pump	2	1560 psig	150°F
Injection Valves	1	1560 psig	150°F
Piping Inboard of Injection Valves	1	1250 psig	575°F

Design provisions to permit system testing include a test tank and associated piping and valves. The tank can be supplied with demineralized water which can be pumped in a closed loop through either pump or injected into the reactor.

The SLCS is separated both physically and electrically from the control rod drive system.

Inspection, Test, Analyses and Acceptance Criteria

Table 2.2.4 provides a definition of the inspections, tests, and/or analyses together with associated acceptance criteria which will be undertaken for the SLCS.

Table 2.2.4: STANDBY LIQUID CONTROL SYSTEM

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The minimum average poison concentration in the reactor after operation of the SLCS shall be equal to or greater than 850 ppm.	1. Construction records, revisions and plant visual examinations will be undertaken to assess as-built parameters listed below for compatibility with SLCS design calculations. If necessary, an as-built SLCS analysis will be conducted to demonstrate the acceptance criteria is met. **Critical Parameters:** a. Storage tank pumpable volume b. RPV water inventory at 70°F c. RHR shutdown cooling system water inventory at 70°F	1. It must be shown the SLCS can achieve a poison concentration of 850 ppm or greater assuming a 25% dilution due to non-uniform mixing in the reactor and accounting for dilution in the RHR shutdown cooling systems. This concentration must be achieved under system design basis conditions. This requires that SLCS meet the following values: Storage tank pumpable volume range 6100-6800 gal. RPV water inventory $\leq 1.00 \times 10^6$ lb RHR shutdown cooling system inventory $\leq .287 \times 10^6$ lb
2. A simplified system configuration in shown in Figure 2.2.4.	2. Inspections of installation records together with plant walkdowns will be conducted to confirm that the installed equipment is in compliance with the design configuration defined in Figure 2.2.4.	2. The system configuration is in accordance with Figure 2.2.4 .

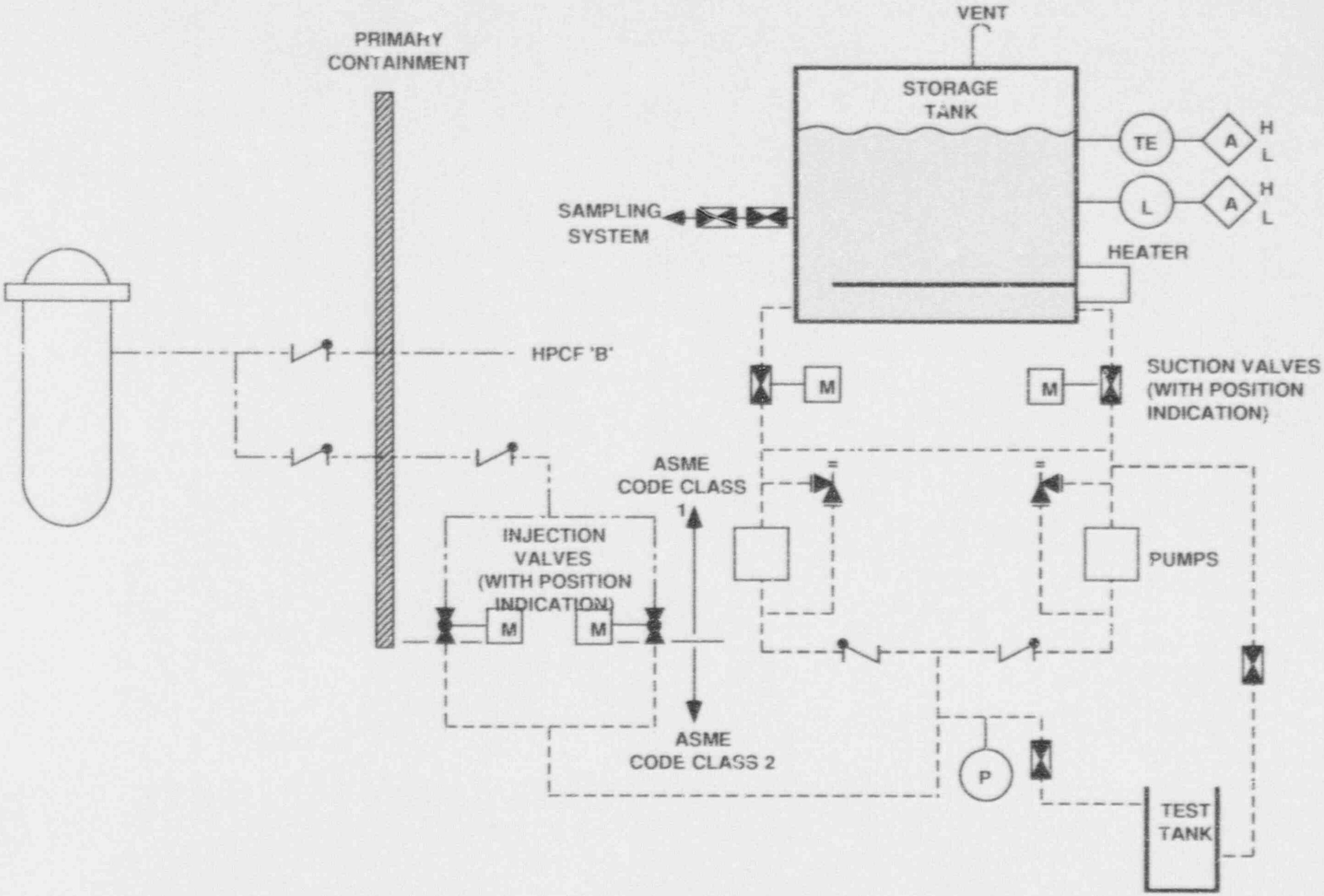
Table 2.2.4: STANDBY LIQUID CONTROL SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. SLCS shall be capable of delivering 100 gpm of solution with both pumps operating against the elevated pressure conditions which can exist in the reactor during events involving SLCS initiation.	3. System preoperation tests will be conducted to demonstrate acceptable pump and system performance. These tests will involve establishing test conditions that simulate conditions which will exist during an SLCS design basis event. To demonstrate adequate Net Positive Suction Head (NPSH), delivery of rated flow will be confirmed by tests conducted at conditions of low level and maximum temperature in the storage tank, and the water will be injected from the storage tank to the RPV.	3. It must be shown that the SLCS can automatically inject 100 gpm (both pumps running) against a reactor pressure of 1250 psig with simmulated ATWS conditions. It must also be shown that the SLCS pumps can pump the entire storage tank pumpable volume.
4. The system is designed to permit in-service functional testing of SLCS.	4. Field tests will be conducted after system installation to confirm, in-service system testing can be performed.	4. Using normally installed controls, power supplies and other auxiliaries, the system has the capability to: a. Pump tests in a closed loop on the test tank and b. Reactor pressure vessel injection tests using demineralized water from the test tank.
5. The pump, heater, valves and controls can be powered from the standby AC power supply as described in Section 2.2.4.*	5. System tests will be conducted after installation to confirm that the electrical power supply configurations are in compliance with design commitments.	5. The installed equipment can be powered from the standby AC power supply.
6. All SLCS components which are required for the injection of the neutron absorber into the reactor are classified Seismic Category I and qualified for appropriate environment for locations where installed.	6. See Generic Equipment Qualification verification activities (ITA).	6. See Generic Equipment Qualification Acceptance Criteria (AC).

* This entry may be transferred to the standby AC power ITAAC in Section 2.12.13.

Figure 2.2.4 STANDBY LIQUID CONTROL SYSTEM (STANDBY MODE)



2.2.7 Reactor Protection System

The reactor protection system (RPS) for the Advanced Boiling Water Reactor (ABWR) is a warning and trip system where initial warning and trip decisions are implemented with software logic installed in microprocessors. The primary functions of this system are to: (1) make the logic decisions related to warning and trip conditions of the individual instrument channels, and (2) make the decision for system trip (emergency reactor shutdown) based on coincidence of instrument channel trip conditions.

The RPS is classified as a safety protection system (i.e., as differing from a reactor control system or a power generation system). All functions of the RPS and the components of the system are safety-related. The RPS and the electrical equipment of the system are also classified as Safety Class 3, Seismic Category I and as IEEE electrical category Class 1E.

Basic System Parameters are:

a.	Number of independent divisions of equipment	4
b.	Minimum number of sensors per trip variable (at least one per division)	4
c.	Number of automatic trip systems (one per division)	4
d.	Automatic trip logic used for plant sensor inputs (per division)	2-out-of-4
e.	Separate automatic trip logic used for division trip outputs	2-out-of-4
f.	Number of separate manual trip systems	2
g.	Manual trip logic	2-out-of-2

The RPS consists of instrument channels, trip logics, trip actuators, manual controls and scram logic circuitry that initiates rapid insertion of control rods (scram) to shut down the reactor for situations that could result in unsafe reactor operating conditions. The RPS also establishes the required trip conditions that are appropriate for the different reactor operating modes and provides status and control signals to other systems and annunciators. The RPS related equipment includes detectors, switches, microprocessors, solid-state logic circuits, relay type contactors, relays, solid-state load drivers, lamps, displays, signal transmission routes, circuits and other equipment which are required to execute the functions of the system. To accomplish its overall function, the RPS utilizes the functions of the essential multiplexing system (EMS) and of portions of the safety system logic and control (SSLC) system.

As shown in Figure 2.2.7a, the RPS interfaces with the neutron monitoring system (NMS), the process radiation monitoring (PRRM) system, the nuclear boiler system (NBS), the control rod drive (CRD) system, the rod control and information system (RC&IS), the recirculation flow control (RFC) system, the process computer system and with other plant systems and equipment. RPS components and equipment are separated or segregated from process control system sensors, circuits and functions such as to minimize control and protection system interactions. Any necessary interlocks from the RPS to control systems are through isolation devices.

The RPS is a four division system which is designed to provide reliable single-failure proof capability to automatically or manually initiate a reactor scram while maintaining protection against unnecessary scrams resulting from single failures in the RPS. The RPS remains single-failure proof even when one entire division of channel sensors is bypassed and/or when one of the four automatic RPS trip logic systems is out-of-service. All equipment within the RPS is designed to fail into a trip initiating state or other safe state on loss of power or input signals or disconnection of portions of the system. The system also includes trip bypasses and isolated outputs for display, annunciation or performance monitoring. RPS inputs to annunciators, recorders and the computer are electrically isolated so that no malfunction of the annunciating, recording, or computing equipment can functionally disable any portion of the RPS. The RPS related equipment is divided into four redundant divisions of sensor (instrument) channels, trip logics and trip actuators, and two divisions of manual scram controls and scram logic circuitry. The automatic and manual scram initiation logic systems are independent of each other and use diverse methods and equipment to initiate a reactor scram. The RPS design is such that, once a full reactor scram has been initiated automatically or manually, this scram condition seals-in such that the intended fast insertion of all control rods into the reactor core can continue to completion. After a time delay, deliberate operator action is required to return the RPS to normal.

Figure 2.2.7b shows the RPS divisional separation aspects and the signal flow paths from sensors to scram pilot valve solenoids. Equipment within a RPS related sensor channel consists of sensors (transducers or switches), multiplexers and digital trip modules (DTMs). The sensors within each channel monitor for abnormal operating conditions and send either discrete bistable (trip/no trip) or analog signals directly to the RPS related DTM or else send analog output signals to the RPS related DTM by means of the remote multiplexer unit (RMU) within the associated division of essential multiplexing system (EMS). The RPS related bistable switch type sensors, or, in the case of analog channels, the RPS software logic, will initiate reactor trip signals within the individual sensor channels, when any one or more of the conditions listed below exist within the plant during different conditions of reactor operation, and will initiate reactor scram if coincidence logic is satisfied.

a. Turbine Stop Valves Closure (above 40% power levels) [RPS]

b. Turbine Control Valves Fast Closure (above 40% power levels) [RPS]

c. NMS monitored SRNM and APRM conditions exceed acceptable limits [NMS]

d. High Main Steam Line Radiation [PRRM System]

e. High Reactor Pressure [NBS]

f. Low Reactor Water Level (Level 3) [NBS]

g. High Drywell Pressure [NBS]

h. Main Steam Lines Isolation (MSLI) (Run mode only) [NBS]

i. Low Control Rod Drive Accumulator Charging Header Pressure [CRD]

j. Operator-initiated Manual Scram [RPS]

The system monitoring the process condition is indicated in brackets in the list above. The RPS outputs, the NMS outputs, the PRRM system outputs and the MSLI and manual scram outputs are provided directly to the RPS by hard-wired or fiber-optic signals. The NBS and the CRD system provide other sensor outputs through the EMS. Analog to digital conversion of these latter sensor output values is done by EMS equipment. The DTM in each division uses either the discrete bistable input signals, or compares the current values of the individual monitored analog variables with their trip setpoint values, and for each variable sends a separate, discrete bistable (trip/no trip) output signal to the trip logic units (TLUs) in all four divisions of trip logics. The DTMs and TLUs utilized by the RPS are microprocessor components within the SSLC system.

RPS related equipment within a RPS division of trip logic consists of manual control switches, bypass units (BPUs), trip logic units (TLUs) and output logic units (OLUs). The manual control switches and the BPUs, TLUs and OLUs are components of the RPS portions of the SSLC system. The various manual switches provide the operator means to modify the RPS trip logic for special operation, maintenance, testing and system reset. The bypass units perform bypass and interlock logic for the single division of channel sensors bypass function and for the single division TLU bypass function. The TLUs perform the automatic scram initiation logic, normally checking for two-out-of-four coincidence of trip conditions in any set of instrument channel signals coming from the four division DTMs or from isolated bistable inputs from all four divisions of NMS equipment, and outputting a trip signal if any one of the two-out-of-four coincidence checks is satisfied. TLU trip decision logic in all four RPS TLUs becomes a check for two-out-of-three coincidence of trip conditions if any one division of channel sensors has been bypassed. The OLUs perform the division trip, seal-in, reset and trip test functions. Trip signals from the OLUs within a single division are used to trip the trip actuators, which are fast response,

bistable, solid-state load drivers for automatic scram initiation, and are trip relays for air header dump (back-up scram) initiation. Load driver outputs toggled by a division OLU interconnect with load driver outputs toggled by other division OLUs into two separate arrangements which results in two-out-of-four scram logic, i.e., reactor scram will occur if load drivers associated with any two or more divisions receive trip signals.

The isolated ac load drivers are fast response time, bistable, solid-state, high current interrupting devices. The operation of the load drivers is such that a trip signal on the input side will create a high impedance, current interrupting condition on the output side. The output side of each load driver is electrically isolated from its input signal. The load driver outputs are arranged in the scram logic circuitry, between the scram pilot valves' solenoids and the solenoids ac power source, such that when in a tripped state the load drivers will cause deenergization of the scram pilot valve solenoids (scram initiation). Normally closed relay contacts are arranged in the two back-up scram logic circuits, between the air header dump valve solenoid and air header dump valve dc solenoid power source, such that when in a tripped state (coil deenergized) the relays will cause energization of the air header dump valve solenoids (air header dump initiation). Associated dc voltage relay logic is also utilized to effect scram reset permissives and scram-follow (control rod run-in) initiation.

The RPS design for the ABWR is testable for correct response and performance, in over-lapping stages, either on-line or off-line (to minimize potential of unwanted trips). Access to bypass capabilities of trip functions, instrument channels or a trip system and access to setpoints, calibration controls and test points are designed to be under administrative control.

Inspection, Test, Analyses and Acceptance Criteria

Table 2.2.7 provides a definition of the visual inspections, tests and/or analyses, together with associated acceptance criteria, which will be used by the RPS.

Table 2.2.7: REACTOR PROTECTION SYSTEM

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. RPS safety-related software, which is utilized in effecting individual sensor channel trip decisions and trip system coincidence trip decisions, has been developed and verified, the firmware implemented and validated and then integrated with hardware; all according to a formal documented plan.	1. See Generic Software Development verification activities (ITA).	1. See Generic Software Development Acceptance Criteria (AC).
2. Certain process signals utilized by the RPS are transmitted to RPS sensor channel signal processing equipment by means of four separate divisions of Essential Multiplexing System equipment.	2. See the Essential Multiplexing System verification activities (ITA).	2. See the Essential Multiplexing System Acceptance Criteria (AC).
3. Critical parameter trip setpoints are based upon values used in analyses of abnormal operational occurrences. Documented instrument setpoint methodology has been used to account for uncertainties (such as instrument inaccuracies and drift) in order to establish RPS related setpoints.	3. See Generic Setpoint Methodology verification activities (ITA).	3. See Generic Setpoint Methodology Acceptance Criteria (AC).
4. RPS equipment is designed to be protected from the effects of noise, such as electromagnetic interference (EMI), and has adequate surge withstand capability (SWC).	4. See Generic EMI/SWC Qualification verification activities (ITA).	4. See Generic EMI/SWC Qualification Acceptance Criteria (AC).
5. RPS equipment is qualified for seismic loads and appropriate environment for locations where installed.	5. See Generic Equipment Qualification verification activities (ITA).	5. See Generic Equipment Qualification Acceptance Criteria (AC).

Table 2.2.7: REACTOR PROTECTION SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. RPS components and equipment are kept separate from equipment associated with process control systems.	6. Visual field inspections and analyses of relationship of installed RPS equipment and of installed equipment of interfacing process control systems (and/or tests of interfaces) to confirm appropriate isolation methods used to satisfy separation and segregation requirements.	6. RPS equipment installation acceptable if inspections, analyses and/or tests confirm that any failure in process control systems can not prevent RPS safety functions.
7. Fail-safe failure modes result upon loss of power or disconnection of components.	7. Field tests to confirm that trip conditions and/or bypass inhibits result upon loss of power or disconnection of components.	7. Acceptable if safe state conditions result upon loss of power or disconnection of portions of the RPS.
8. Provisions exist to limit access to trip setpoints, calibration controls and test points.	8. Visual field inspections of the installed RPS equipment will be used to confirm the existence of appropriate administrative controls.	8. The RPS hardware/firmware will be considered acceptable if appropriate methods exist to enforce administrative control for access to sensitive areas.
9. The four redundant divisions of RPS equipment and the four automatic trip systems are independent from each other except in the area of the required coincidence of trip logic decisions and are both electrically and physically separated from each other. Similarly, the two manual trip systems are separate and independent of each other and of the four automatic trip systems.	9. Inspections of fabrication and installation records and construction drawings or visual field inspections of the installed RPS equipment will be used to confirm the quadruple redundancy of the RPS and the electrical and physical separation aspects of the RPS instrument channels and the four automatic trip systems as well as their diversity and independence from the two manual trip systems.	9. Installed RPS equipment will be determined to conform to the documented description of the design as depicted in Figure 2.2.7b.

Table 2.2.7: REACTOR PROTECTION SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. It is possible to conduct verifications of RPS operations, both on-line and off-line, by means of a) individual instrument channel functional tests, b) trip system functional tests and c) total system functional tests.	10. Preoperational tests will be conducted to confirm that system testing such as channel checks, channel functional tests, channel calibrations, coincident logic tests and paired control rods scram tests can be performed. These tests will involve simulation of RPS testing modes of operation. Interlocks associated with the reactor mode switch positions, and with other operational and maintenance bypasses or test switches will be tested and annunciation, display and logging functions will be confirmed.	10. The installed reactor protection system configuration, controls, power sources and installations of interfacing systems supports the RPS logic system functional testing and the operability verification of design as follows: a. Installed RPS hardware/firmware initiates trip conditions in all four RPS automatic trip systems upon coincidence of trip conditions in two or more instrument channels associated with the same trip variable(s). b. Installed system initiates full reactor trip and emergency shutdown (i.e., deenergization of both solenoids associated with all scram pilot valves) upon coincidence of trip conditions in two or more of the four RPS automatic trip systems. c. Installed system initiates trip conditions in both RPS manual trip systems if both manual trip switches are operated or if the reactor mode switch is placed in the "shutdown" position. d. Trip system (automatic and manual) trip conditions seal-in and protective actions go to completion. Trip reset (after appropriate delay for trip completion) requires deliberate Operator action.

Table 2.2.7: REACTOR PROTECTION SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		10. (Continued) e. Installed system energizes both air header dump (back-up scram) valves of the CRD hydraulic system, and initiates CRD motor run-in, concurrent only with a full scram condition. f. When not bypassed, trips result upon loss or disconnection of portions of the system. When bypassed, inappropriate trips do not result. g. Installed system provides isolated status and control signals to data logging, display and annunciator systems. h. Installed system demonstrates operational interlocks (i.e., trip inhibits or permissives) required for different conditions of reactor operation.

Table 2.2.7: REACTOR PROTECTION SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. The RPS design provides prompt protection against the onset and consequences of events or conditions that threaten the integrity of the fuel barrier.	11. Preoperational tests will be conducted to measure the RPS and supporting systems response times to: (1) monitor the variation of the selected processes; (2) detect when trip setpoints have been exceeded; and, (3) execute the subsequent protection actions when coincidence of trip conditions exist.	11. The RPS hardware/firmware response to initiate reactor scram will be considered acceptable if such response is demonstrated to be sufficient to assure that the specified acceptable fuel design limits are not exceeded. **Validation Attributes:** Total trip system response, from time when sensor input is beyond setpoint to time of scram pilot valve solenoids deenergization: - NMS APRM ≤ 0.090 sec. - Reactor pressure ≤ 0.55 sec. - Reactor water level ≤ 1.05 sec. - Turbine stop valve closure ≤ 0.060 sec. - Turbine control valve fast closure ≤ 0.080 sec. - Main steam lines isolation ≤ 0.060 sec.

Figure 2.2.7a REACTOR PROTECTION SYSTEM

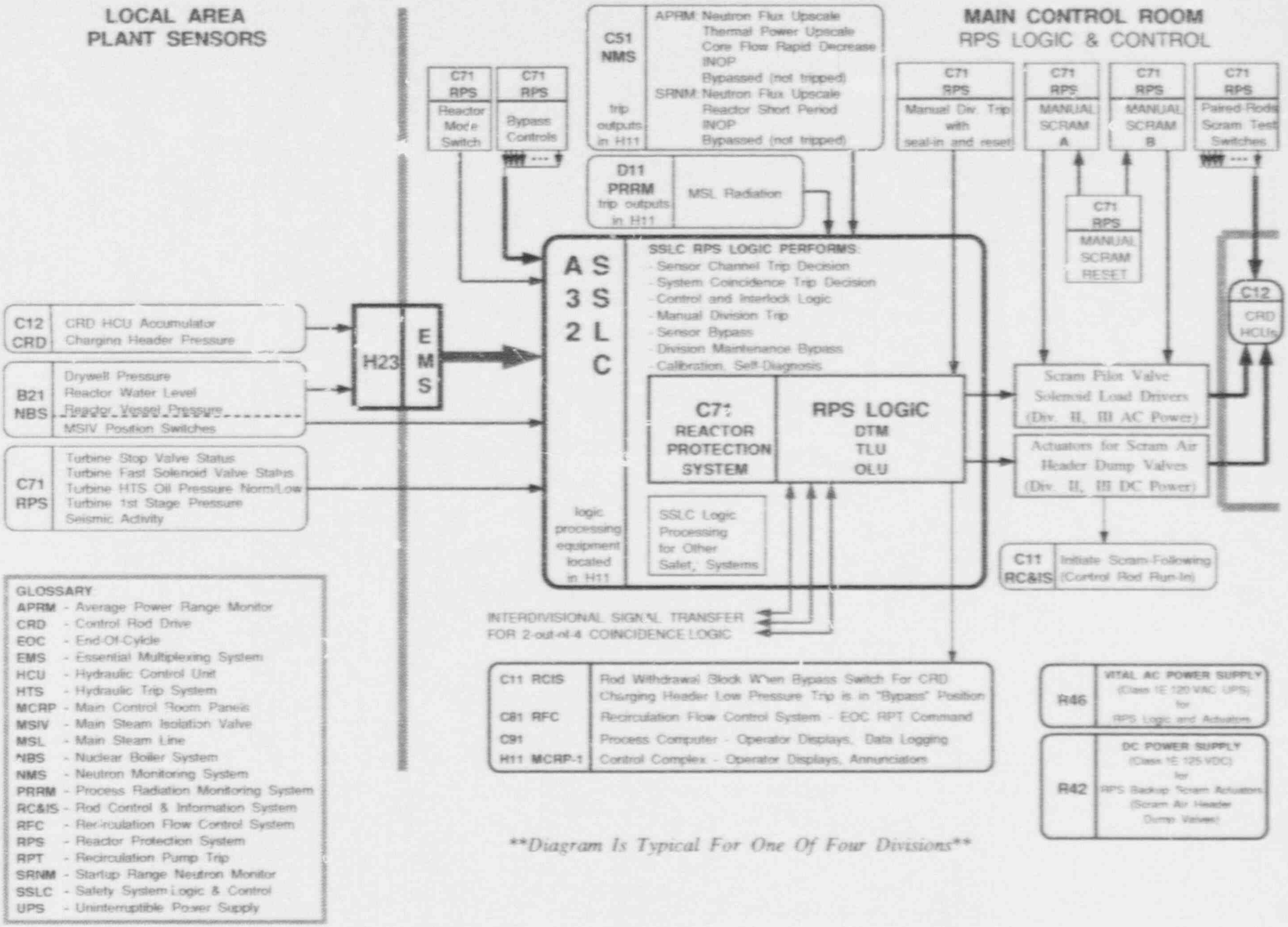
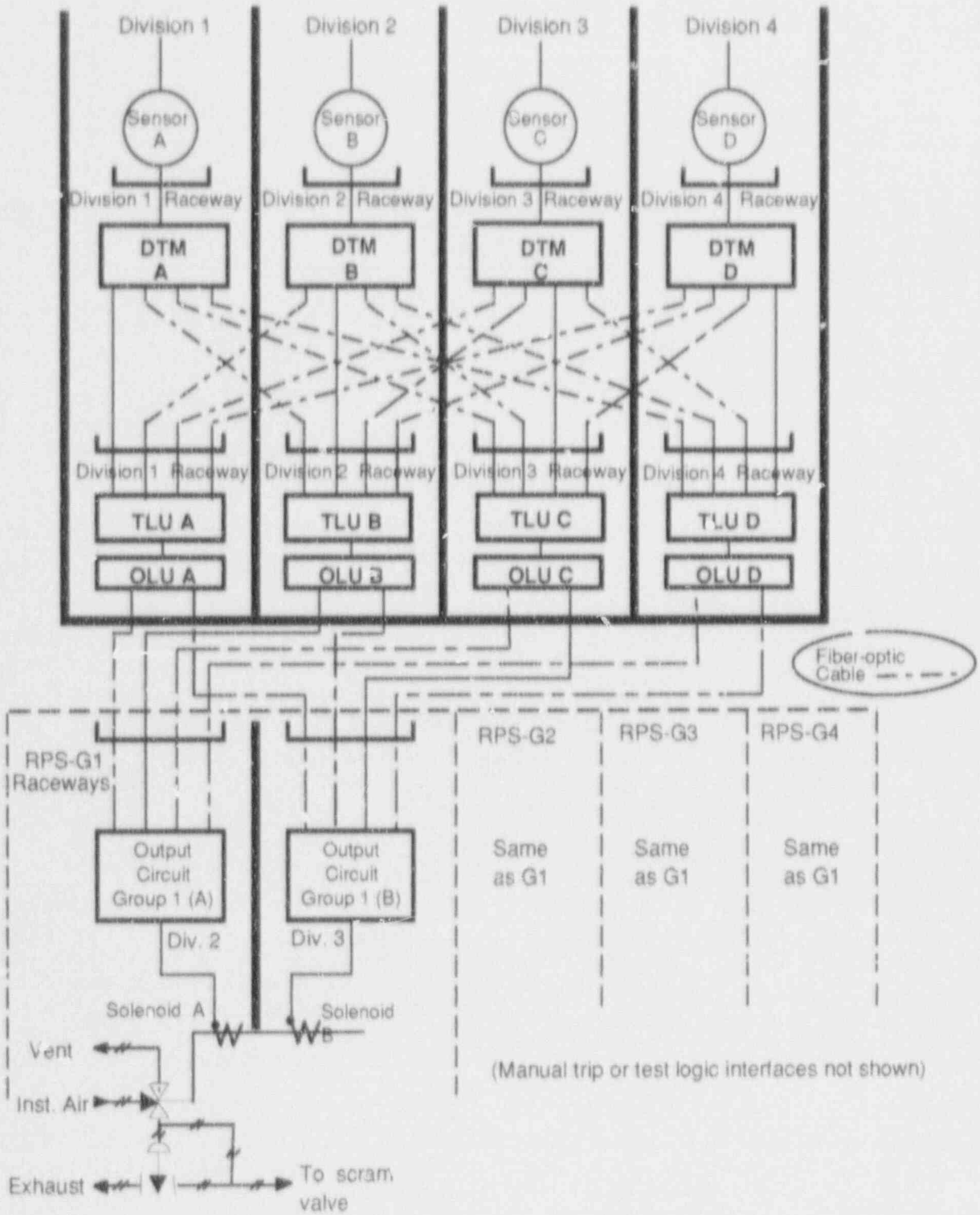


Diagram Is Typical For One Of Four Divisions

Figure 2.2.7b REACTOR PROTECTION SYSTEM



2.4.1 Residual Heat Removal (RHR) System

Design Description

The RHR system is comprised of three divisionally separate subsystems that perform a variety of functions utilizing the following six basic modes of operation: (1) shutdown cooling, (2) suppression pool cooling, (3) wetwell and drywell spray cooling, (4) low pressure core flooder (LPFL), (5) fuel pool cooling, and (6) AC independent water addition. The configuration of each loop is shown on its P&ID in Figure 2.4.1 (aligned in the standby mode). The major functions of the various modes of operation include containment heat removal, reactor decay heat removal, emergency reactor vessel level makeup and augmented fuel pool cooling. In line with its given functions, portions of the system are a part of the ECCS network and the containment cooling system. Additionally, portions of the RHR system are considered a part of the reactor coolant pressure boundary (RCPB).

The entire RHR system is designed to safety related standards although it performs some non-safety functions, i.e. those that are not taken credit for when evaluating design basis accidents. The safety related modes of operation include low pressure flooding, suppression pool cooling, wetwell spray cooling and shutdown cooling. Non-safety related modes of operation include drywell spray cooling, AC independent water addition and augmented fuel pool cooling. RHR also provides a back-up, safety-related fuel pool make-up capability. Ancillary modes of operation include minimum flow bypass and full flow testing.

The ECCS function of the RHR system is performed by the LPFL mode. Following receipt of a LOCA signal (low reactor water level or high drywell pressure) the RHR system automatically initiates and operates in the LPFL mode, in conjunction with the remainder of the ECCS network, to provide emergency makeup to the reactor vessel in order to keep the reactor core cooled such that the criteria of 10 CFR 50.46 are met. The LPFL mode is accomplished by all 3 loops of the RHR system by transferring water from the suppression pool to the RPV, via the RHR heat exchangers. The LPFL mode is the only automatically initiated mode of RHR, but may also be initiated manually. The system will also automatically revert to the LPFL mode of operation from any other test or operating mode upon receipt of a LOCA signal. Each RHR loop's RPV injection valve requires a low reactor pressure permissive signal whether being opened manually or automatically in response to a LOCA signal.

The containment heat removal function in the ABWR is performed by the containment cooling system which is comprised of the low pressure core flooder (LPFL), suppression pool cooling, and wetwell and drywell spray cooling modes of the RHR system. Following a LOCA the energy present within the reactor primary system is dumped either directly to the suppression pool via the SRVs, or indirectly via the drywell and connecting vents. Subsequently, fission product decay heat continues to add energy to the pool. The containment cooling system is designed to limit the long-term bulk temperature of the suppression

pool, and thus limit the long term peak temperatures and pressures within the wetwell and drywell regions of the containment to within their analyzed design limits, with only 2 of the 3 loops in operation (i.e. worse case single failure). The cooling requirements of the containment cooling function establish the necessary RHR heat exchanger heat removal capacity.

The LPFL mode, along with its primary function of cooling the core, also serves to cool the containment, as the heat exchanger is designed to always be in the loop. The dedicated suppression pool cooling mode is made available in each of the 3 loops of the RHR system by circulating suppression pool water through the respective RHR heat exchanger and then directly back to the suppression pool. This mode of RHR is initiated manually. The wetwell and drywell spray modes of RHR are each available in only 2 of the 3 subsystems (loops B & C). These functions are performed by drawing water from the suppression pool and delivering it to a common wetwell spray header and/or a common drywell spray header, both via the associated RHR heat exchanger(s). These containment spray modes of the RHR system are initiated manually. However, the drywell spray inlet valves can only be opened if there exists high drywell pressure and the RPV injection valves are fully closed. Wetwell and drywell sprays serve as an augmented method of containment cooling. Wetwell spray also serves to mitigate the consequences of steam bypassing the suppression pool.

The normal operational mode of the RHR system is in the shutdown cooling mode of operation which is used to remove decay heat from the reactor core. This mode provides the required safety related capability needed to achieve and maintain a cold shutdown condition, including consideration of the worst case system single failure. The RHR heat exchanger heat removal capacity requirements in this mode are bounded by containment cooling requirements. Shutdown cooling is initiated manually once the RPV has been depressurized below the system low pressure permissive. In this mode each loop takes suction from the RPV via its dedicated suction line, pumps the water through its respective heat exchanger and returns the cooled water to the RPV. Two loops (B & C) discharge water back to the RPV via dedicated spargers while the third loop (A) utilizes the vessel spargers of one of the two feedwater lines (FW-A). The heat removed in the RHR heat exchangers is transported to the ultimate heat sink via the respective division of reactor cooling water and service water. Each shutdown cooling suction valve is interlocked with that loop's suppression pool suction and discharge valves and wetwell spray valve to prevent draining of the reactor vessel to the suppression pool. Also, each shutdown cooling suction valve is interlocked with and automatically closes on low reactor water level.

The augmented fuel pool cooling mode of RHR supplements/replaces the normal fuel pool cooling system during infrequent conditions of high heat load. This mode is accomplished manually in one of two ways. When the reactor vessel head is removed, the cavity flooded and the fuel pool gates removed, the RHR system cools the fuel pool in the normal shutdown cooling mode. When the fuel pool is otherwise isolated from the reactor cavity, two loops (B & C) of the RHR system can directly cool the pool by taking suction from and discharging back to

the normal fuel pool cooling system. This connection also provides for emergency fuel pool make-up capability by supplying a safety related make-up path to the fuel pool from a safety related source, i.e. the suppression pool.

One loop (C) of the RHR system also functions in an AC independent water addition mode. This mode provides a means of cross connecting the reactor building fire protection system header to the RHR system just outside containment in the absence of the normal ECCS network and independent of the normal essential AC power distribution network.. The connection is accomplished by the manual opening of two in-series valves on the cross connection piping just upstream of its tie-in to the normal RHR piping. Fire protection system water can be directed to either the RPV or the drywell spray sparger by manual opening of the respective RHR injection valve. The fire water is supplied via the system's reactor building distribution header by either the direct diesel driven fire pump or from an external source utilizing a dedicated connection just outside the reactor building.

Each loop of RHR also has both a minimum flow mode and a full flow test mode. The minimum flow mode assures that there is pump flow sufficient to keep the pump cool by opening a minimum flow valve that directs flow back to the suppression pool anytime the pump is running and the main discharge valve is closed. Upon sensing that there is adequate flow in the pump main discharge line, the minimum flow valve is automatically closed. In the full flow test mode the system is essentially operated in the suppression pool cooling mode, drawing suction from and discharging back to the suppression pool.

The RHR system is comprised of three separate loops or subsystems, each of which includes a pump and a heat exchanger, takes suction from either the RPV or the suppression pool, and directs water back to either the RPV or the suppression pool. Two of the three loops can divert a portion of the suppression pool return flow to a common wetwell spray sparger or direct the entire flow to a common drywell spray sparger. The divisional subsystems of the RHR system are separated both mechanically and electrically as well as being physically located in different areas of the plant to address requirements pertaining to fire protection and other separation criteria. Each of the three subsystems is powered from a separate divisional power distribution bus that can be supplied from either an on-site or off-site source. Cooling water to each division of RHR equipment (heat exchanger as well as pump and motor coolers) is supplied by the respective division of the reactor cooling water (RCW) system. The RHR system also includes provisions for containment isolation and RCPB pressure isolation.

The RHR system will maintain the capability to perform its intended safety related functions either following a Safe Shutdown Earthquake or during the environmental conditions imposed by a LOCA, and in each case assuming the worst case single failure. The system will also accommodate calculated movement and thermal stresses. The system is designed so that the pumps have sufficient NPSH available in all operating modes. The system can be powered

from either normal off-site sources or by the emergency diesel generators. The RHR system is Seismic Category 1 and is housed in the Seismic Category 1 reactor building to provide protection against tornados, floods, and other natural phenomena.

The RHR pumps are motor-driven centrifugal pumps capable of supplying at least 4200 gpm at 40 psid (drywell to RPV). The pumps are ASME Code Class 2 components with a design pressure of 500 psig and a design temperature of 360 °F. The pumps are interlocked from starting without an open suction path. The RHR pumps are protected from possible pump run-out conditions in all operating modes. The RHR heat exchangers are horizontal U-tube/ shell type sized to provide a minimum effective heat removal capacity (K-coefficient) of 195 Btu/sec°F. The primary and secondary sides of the heat exchangers are ASME Code Class 2 and 3, respectively. The primary side design temperature and pressure are 500 psig and 360 °F, respectively. The secondary side design temperature and pressure are consistent with that of the RCW system. Each loop of RHR has its own jockey pump to act as a keep fill system for that loop's pump discharge piping. The jockey pumps are ASME Code Class 2.

The RHR system piping and valves are ASME Code Class 1 or 2 as shown on the P&ID (Figures 2.4.1). The design pressure and temperature of piping and valves varies across the system. For that piping attached to the RPV, from the RPV out to and including the outboard containment isolation valves, the design pressure and temperature are 1250 psig and 575 °F, respectively. For other piping open to the containment atmosphere, out to and including the outboard containment isolation valves, the design pressure and temperature are 45 psig and 219 °F, respectively. For piping and valves outside the containment isolation valves, the design pressure and temperature depends on whether it is located on the suction or discharge side of the main pump. Those portions on the suction side are rated at 200 psig and 360 °F, while those portions on the discharge side are rated at 500 psig and 360 °F, respectively. The low pressure portions of the shutdown cooling piping are protected from full reactor pressure by automatic pressure isolation valves that are interlocked with reactor pressure. High reliability of this interlock is assured by utilizing 4 separate and divisionally independent pressure sensors in a 2-out-of-4 logic. Additionally, in-series inboard and outboard containment/pressure isolation valves in each loop are powered from separate electrical divisions. Relief valves are also provided for protection from overpressure.

The RHR system includes sufficient Control Room indication to allow for the necessary monitoring and control during design basis operational conditions. This includes system flows, temperatures and pressures as well as valve open/close and pump on/off indication for those instruments and components shown on Figures 2.4.1.a, b and c, with the exception of simple check valves and overpressure relief valves (of the check valves shown only the testable check valves downstream of each loop's RPV injection valve has control room status indication).

Inspection, Test, Analyses and Acceptance Criteria

This section provides a definition of the inspections, tests and/or analyses together with associated acceptance criteria which will be undertaken for the RHR system.

Table 2.4.1: RESIDUAL HEAT REMOVAL SYSTEM

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The configuration of the RHR system is shown in Figures 2.4.1.a, b and c.	1. Inspections of the as-built RHR configuration shall be performed.	1. Actual RHR system configuration, for those components shown, conforms with Figures 2.4.1.a, b and c.
2. The RHR system operates in the LPFL mode as part of the overall ECCS network.	2. The ECCS LOCA performance analysis for assuring core cooling shall be validated by RHR system functional testing, including demonstration that the LPFL mode (of each RHR loop) is capable of automatically initiating and operating in response to a LOCA signal.	2. RHR system actuation and operation is consistent with the ECCS performance analysis as follows: a) RHR Pump Flow (at 40 psid) .....≥ 4200 gpm b) Time to Rated Flow .....≤ 36 seconds
3. The RHR system operates in the suppression pool cooling mode to limit the long term temperature and pressure of the containment under post-LOCA conditions.	3. The primary containment performance analysis for long term peak pressure and temperature shall be validated by RHR system functional testing demonstrating the required flowrate through the heat exchanger and by inspection of vendor test data and/or certifications confirming the heat exchanger's effective heat removal capability.	3. RHR heat exchanger performance is consistent with the containment cooling system analysis as follows: a) RHR Heat Exchanger effective heat removal capability (K coefficient) .....≥ 195 Btu/sec°F. b) RHR Heat Exchanger tube side flow .....≥ 4200 gpm
4. A portion of the RHR system return flow (in loops B & C) can be diverted to the wetwell spray header.	4. RHR system functional tests shall be performed to demonstrate wetwell spray flow capability.	4. RHR loops B & C each separately are capable of providing wetwell spray flow consistent with the suppression pool bypass analysis as follows: a) Wetwell spray flow.....≥ 500 gpm.
5. The RHR system operates in the shutdown cooling mode to remove reactor core decay heat and bring the reactor to cold shutdown conditions.	5. RHR system functional tests shall be performed to demonstrate operation in the shutdown cooling mode of operation.	5. RHR system (each loop) is capable of taking suction from and discharging back to the reactor pressure vessel. [Heat exchanger heat removal capability in this mode is bounded by containment cooling requirements - ITAAC # 3]

Table 2.4.1: RESIDUAL HEAT REMOVAL SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The RHR system (loops B & C) operates in the augmented fuel pool cooling mode to supply supplemental or replacement cooling to the spent fuel storage pool under abnormal conditions.	6. RHR system functional tests shall be performed to demonstrate operation in the augmented fuel pool cooling mode of operation.	6. RHR system (loops B & C) is capable of taking suction from and discharging back to the normal fuel pool cooling system. [Required cooling capability in this mode bounded by containment cooling requirements - ITAAC #3]
7. The RHR system (loop C) provides an AC independent water addition function.	7. RHR systems functional testing shall be performed to demonstrate operation in the AC independent water addition mode of operation.	7. Flow capability exists for directing water from the fire protection system to the RPV and drywell spray sparger, via the RHR system (loop C), without power being available from the essential AC distribution system.
8. The RHR system operates when powered from both normal off-site and emergency on-site sources.	8. RHR system functional tests shall be performed to demonstrate operation when supplied by either normal off-site power or the emergency diesel generator(s).	8. RHR system is capable of operating when supplied by either power source.
9. If already operating in any other mode, the RHR system automatically reverts to the LPFL mode in response to a LOCA signal.	9. Using simulated inputs, logic and functional testing shall be performed to demonstrate the RHR systems ability to automatically revert to the LPFL mode from any other mode.	9. RHR logic functions as designed to automatically reconfigure the system to the LPFL mode of operation.
10. Pressure isolation valves are provided to protect low pressure RHR piping from being subjected to excessively high reactor pressure.	10. Using simulated inputs, logic and functional testing shall be performed to demonstrate operation of automatic isolation and interlock functions of pressure isolation valves.	10. Automatic isolation and interlock features function as designed to prevent possible overpressure conditions.
11. Each RHR loop operates automatically in a minimum flow mode to protect the pump from overheating.	11. Logic and functional testing shall be performed to demonstrate operation of the minimum flow mode for each loop (including extended minimum flow operational conditions).	11. RHR system logic functions automatically to assure a pump minimum flow path exists and no deleterious affects are observed during extended operation in the minimum flow mode.

Table 2.4.1: RESIDUAL HEAT REMOVAL SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. The RHR system automatically isolates shutdown cooling suction valves to prevent draining of the reactor vessel.	12. Using simulated inputs, logic and valve functional testing shall be conducted to demonstrate operation of the shutdown cooling mode isolation function.	12. The shutdown cooling suction isolation valves automatically isolate on a low reactor water level signal.
13. RHR system valve interlocks prevent establishment of a drainage path from the reactor vessel to the suppression pool.	13. Using simulated inputs, logic and functional testing shall be conducted to demonstrate operation of interlocking between RPV suction valves and other RHR valves providing potential flow paths to the suppression pool.	13. RHR system valve interlock logic functions to prevent possible RPV drain down.
14. The drywell spray inlet valves can only be opened if there exists high drywell pressure and the RPV injection valves are fully closed.	14. Using simulated inputs, logic and functional testing shall be conducted to demonstrate operation of drywell spray permissive logic.	14. RHR drywell spray permissive logic functions to prevent drywell spray inlet valves from opening in the absence of either a high drywell pressure signal or a signal indicating RHR RPV injection valve(s) not fully closed.
15. The RHR pumps are interlocked from starting without an open suction path.	15. Logic tests shall be conducted to demonstrate that the RHR pumps will not start without an open suction path being available.	15. An RHR pump start signal is not generated in the absence of indication of an open suction path.
16. The RHR system utilizes jockey pumps (1 in each loop) to keep the pump discharge lines filled.	16. Functional tests will be performed to demonstrate the ability of the jockey pump (in each loop) to keep its respective RHR pump discharge line full while in the standby mode.	16. Each jockey pump performs its keep fill function.
17. The RHR system full flow test mode allows periodic demonstration of RHR capability during normal power operation.	17. Functional tests will be performed to demonstrate operation in the full flow test mode.	17. Each RHR subsystem demonstrates full flow functional capability while approximating actual vessel injection conditions during operation in the full flow test mode.
18. The RHR pumps have sufficient NPSH during all postulated operating conditions.	18. Actual system installation will be inspected, and appropriate measurements taken, to verify adequate pump NPSH.	18. Minimum pump NPSH available, as determined based on as-built conditions, exceeds as-procured pump requirements.

Table 2.4.1: RESIDUAL HEAT REMOVAL SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
19. RHR mechanical eq ment is built in accordance with ASME Code, Section III requirements.	19. Procurement records and actual equipment shall be inspected to verify applicable RHR system components have been manufactured per the relevant ASME uirements.	19. RHR equipment has appropriate ASME, Section III, Class 1, 2 or 3 certifications in accordance with its proper classification (as described in Section 2.4.1).
20. Control room indications are provided for certain RHR system parameters.	ections will be performed to ify presence of control room indication r the RHR system as described n 2.4.1.	20. The designated instrumentation is present in the control room.

Figure 2.4.1a RESIDUAL HEAT REMOVAL (RHR-A) SYSTEM

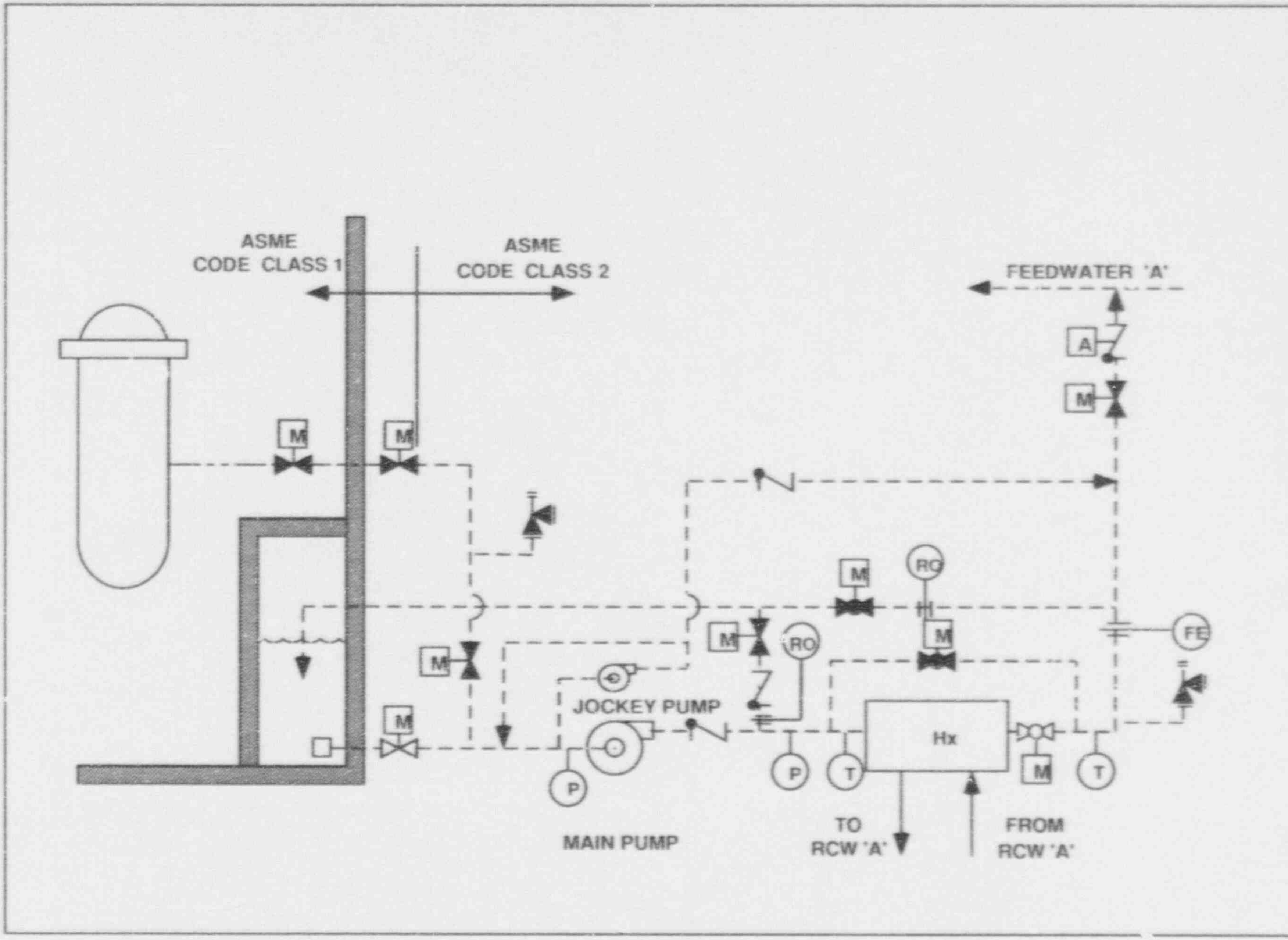


Figure 2.4.1b RESIDUAL HEAT REMOVAL (RHR-B) SYSTEM

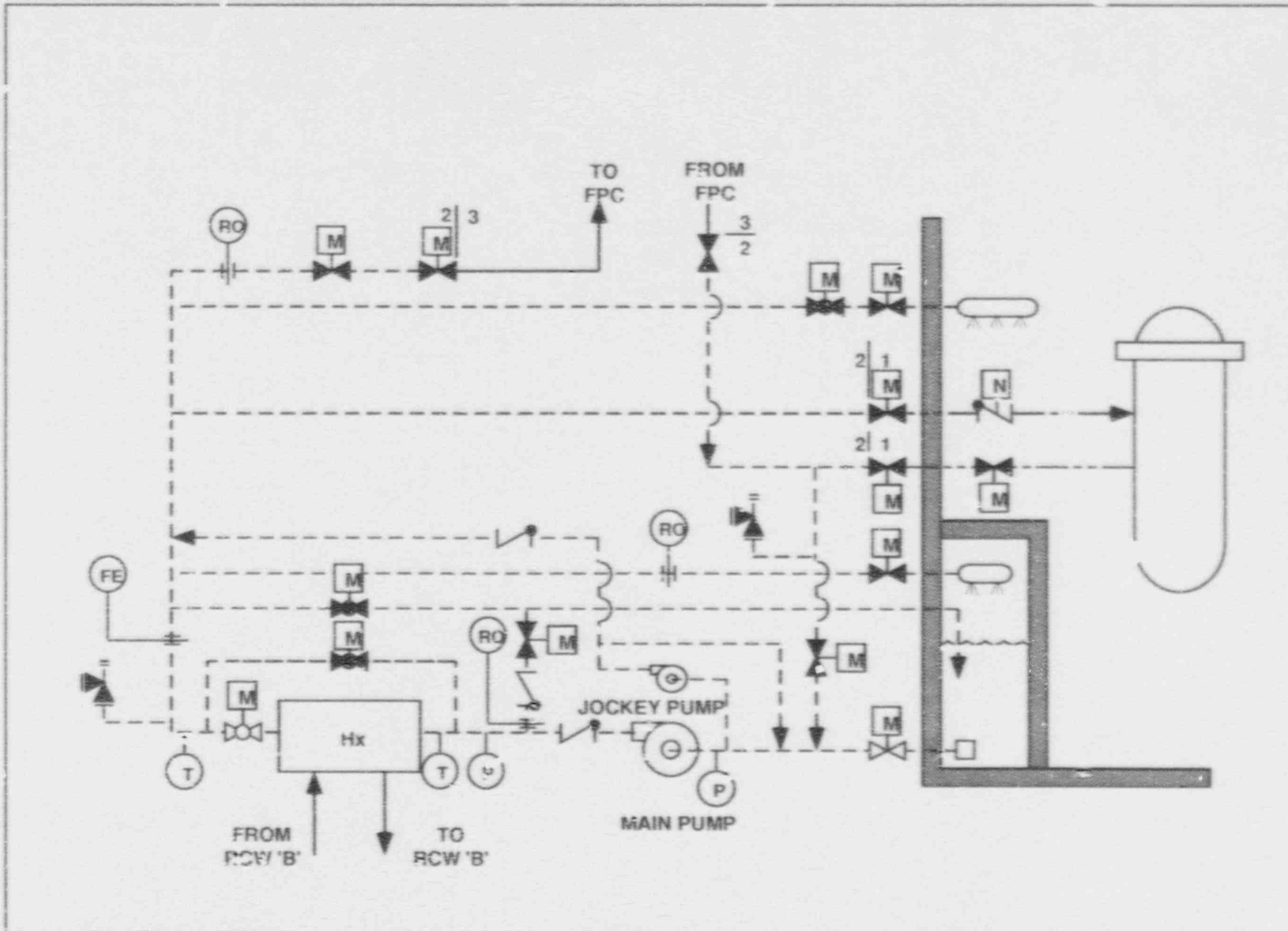
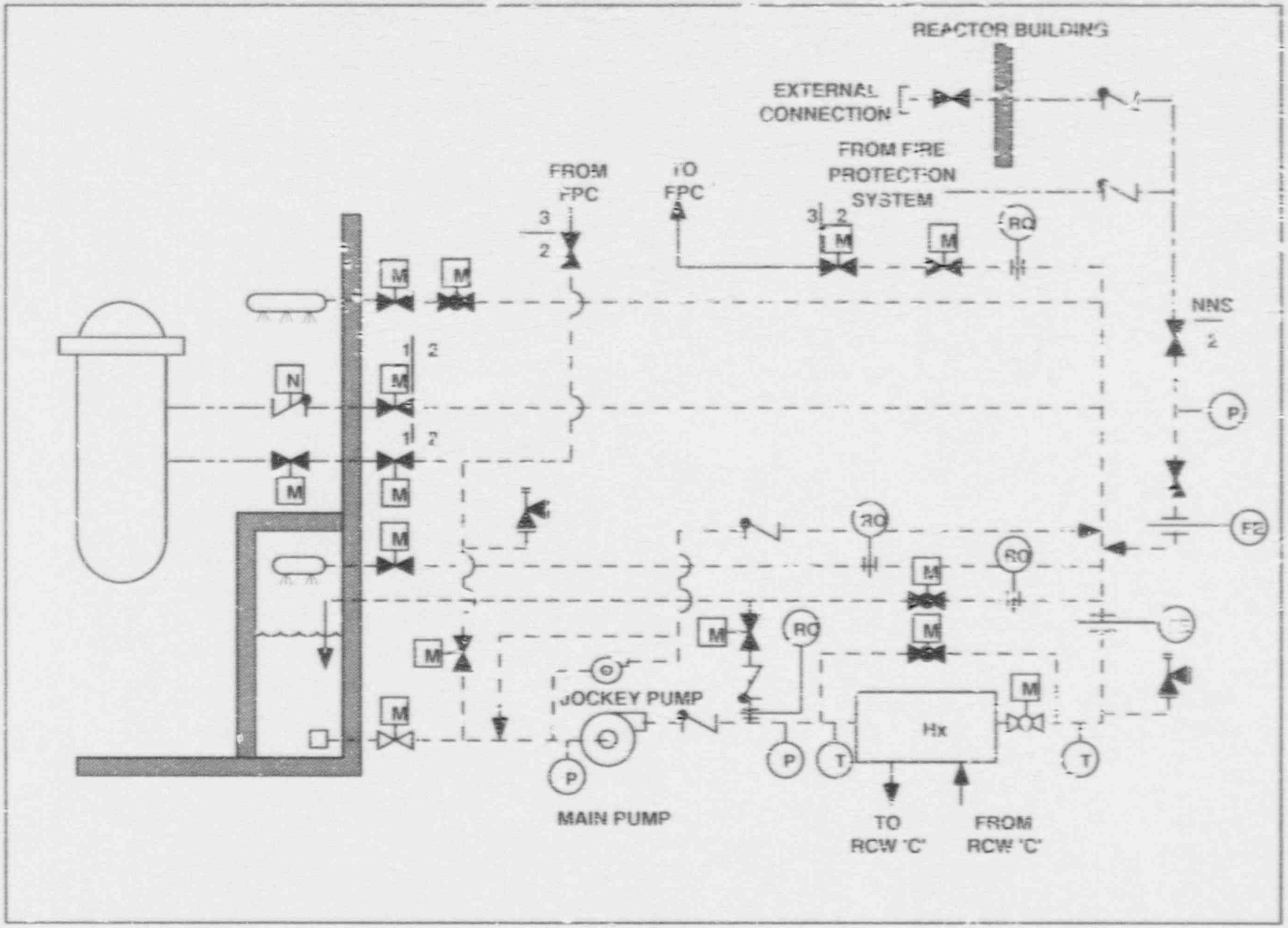


Figure 2.4.1c RESIDUAL HEAT REMOVAL (RHR-C) SYSTEM



2.6.1 Reactor Water Cleanup System

Design Description

The CUW system removes particulate and dissolved impurities from the reactor coolant by recirculating a portion of the reactor coolant through a filter-demineralizer. The CUW system is designed to process a nominal flow of 2% of rated feedwater flow. The CUW is designed for 87.9 kg/cm^{2}g and 302°C.

The CUW system removes excess coolant from the reactor system during startup, shutdown and hot standby. The excess water is directed to the main condenser, radwaste or suppression pool. The CUW system also provides processed water to the reactor head spray nozzle for RPV cooldown.

The CUW system minimizes RPV temperature gradients by maintaining circulation in the bottom head of the RPV during periods when the reactor internal pumps are unavailable.

The suction line through the PCPB contains two motor operated isolation valves which automatically close in response to signals from the leak detection system, actuation of the SLCS, and high filter-demineralizer inlet temperature.

The CUW system is classified as a nonsafety system with a major portion of the system located outside of the primary containment pressure boundary (PCPB) and automatically isolatable. System piping and components within the PCPB, including the suction piping up to and including the outboard suction isolation valve, and all containment isolation valves including interconnecting piping, are ASME Section III Seismic Category I, Quality Group A. The flow element used for CUW system leak detection meets Seismic Category I and Quality Group A requirements to maintain structural integrity during a faulted condition. All nonsafety equipment is designed as Nonseismic, Quality Group C. Low pressure piping in the filter-demineralizer area, downstream of the high pressure block valves, is designed to Quality Group D.

The CUW system is a single closed loop system that takes suction from the reactor vessel bottom head drain line or the shutdown cooling suction line connection to RHR loop "B". CUW flow passes through a regenerative heat exchanger (RHX) and two parallel nonregenerative heat exchangers (NRHX) to two pumps in parallel. The flow is discharged to two filer-demineralizers and returned, through the regenerative heat exchanger to feedwater lines "A" and "B". Each pump, NRHX and filter-demineralizer is capable of 50% system capacity operation. See Figure 2.6.1 for system arrangement.

Each filter-demineralizer vessel is installed in an individual shielded compartment with provisions for handling filter material. Inlet, outlet, vent, drain and other process valves are located outside the filter-demineralizer compartment in a separate shielded area together with the necessary piping and associated equipment.

Process equipment and controls are arranged so that normal operations are conducted at a panel from outside the vessel or valve and pump compartment shielding walls.

Penetrations through compartment walls are designed so that they preclude direct radiation shine.

A remote, manually operated valve on the return line to the feedwater lines in the steam tunnel provides long team leakage control and reverse flow isolation is provided by a check valve in the CUW piping.

Inspection, Test, Analyses and Acceptance Criteria

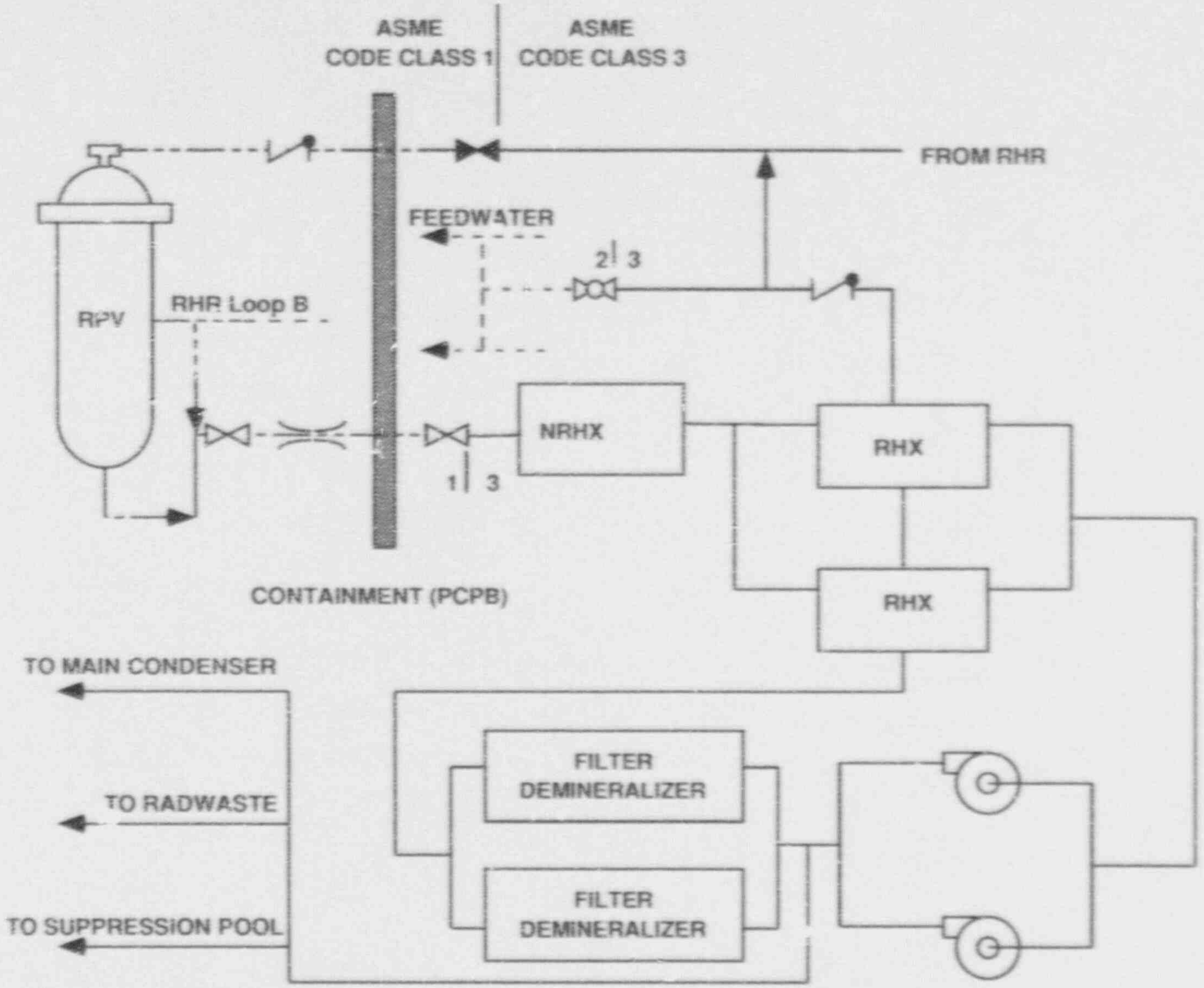
Table 2.6.1 provides a definition of the instructions, tests, and/or analyses together with associated acceptance criteria which will be undertaken for CUW.

Table 2.6.1: REACTOR WATER CLEANUP SYSTEM

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitments	Inspections, Test, Analysis	Acceptance Criteria
1. The configuration of the CUW system is shown in Figure 2.4.1.	1. Inspection of the as-built CUW configuration shall be performed.	1. Actual CUW system configuration conforms with Figure 2.6.1.
2. Suction line isolation valves automatically isolate CUW upon SLCS actuation, leak detection, and high filter-demineralizer temperature.	2. Field test will be conducted to confirm that CUW will isolate upon SLCS actuation and leak detection by applying a simulated isolation signal to the isolation logic circuit.	2. CUW isolates when SLCS is actuated or leak detection limit is sensed by closing the primary containment pressure boundary isolation valves.
3. CUW equipment is provided with shielding.	3. Inspection of the as-built CUW equipment location will be performed to show that equipment is located in shielded areas.	3. Actual location of the CUW equipment conforms with the reactor building arrangement drawings (see Figure (later)).

Figure 2.6.1 REACTOR WATER CLEANUP (CUW) SYSTEM P&ID



2.10.23 Circulating Water System

Design Description

The circulating water system (CWS) provides a continuous supply of cooling water to the main condenser to remove the heat rejected by the turbine cycle and auxiliary systems.

The CWS does not serve or support any safety function and has no safety design basis.

To prevent flooding of the turbine building, the CWS is designed to automatically isolate in the event of gross system leakage. The circulating water pumps are tripped and the pump and condenser valves are closed in the event of a system isolation signal from the condenser area high-high level switches. A condenser area high level alarm is provided in the control room.

The CWS is designed and constructed in accordance with Quality Group D specifications.

The CWS consists of the following components:

a. Intake screens located in a screen house

b. Pumps

c. Condenser water boxes

d. Piping and valves

e. Tube-side of the main condenser

f. Water-box fill and drain subsystem

Figure 2.10.23 is a simplified system diagram showing major system components.

Inspection, Test, Analyses and Acceptance Criteria

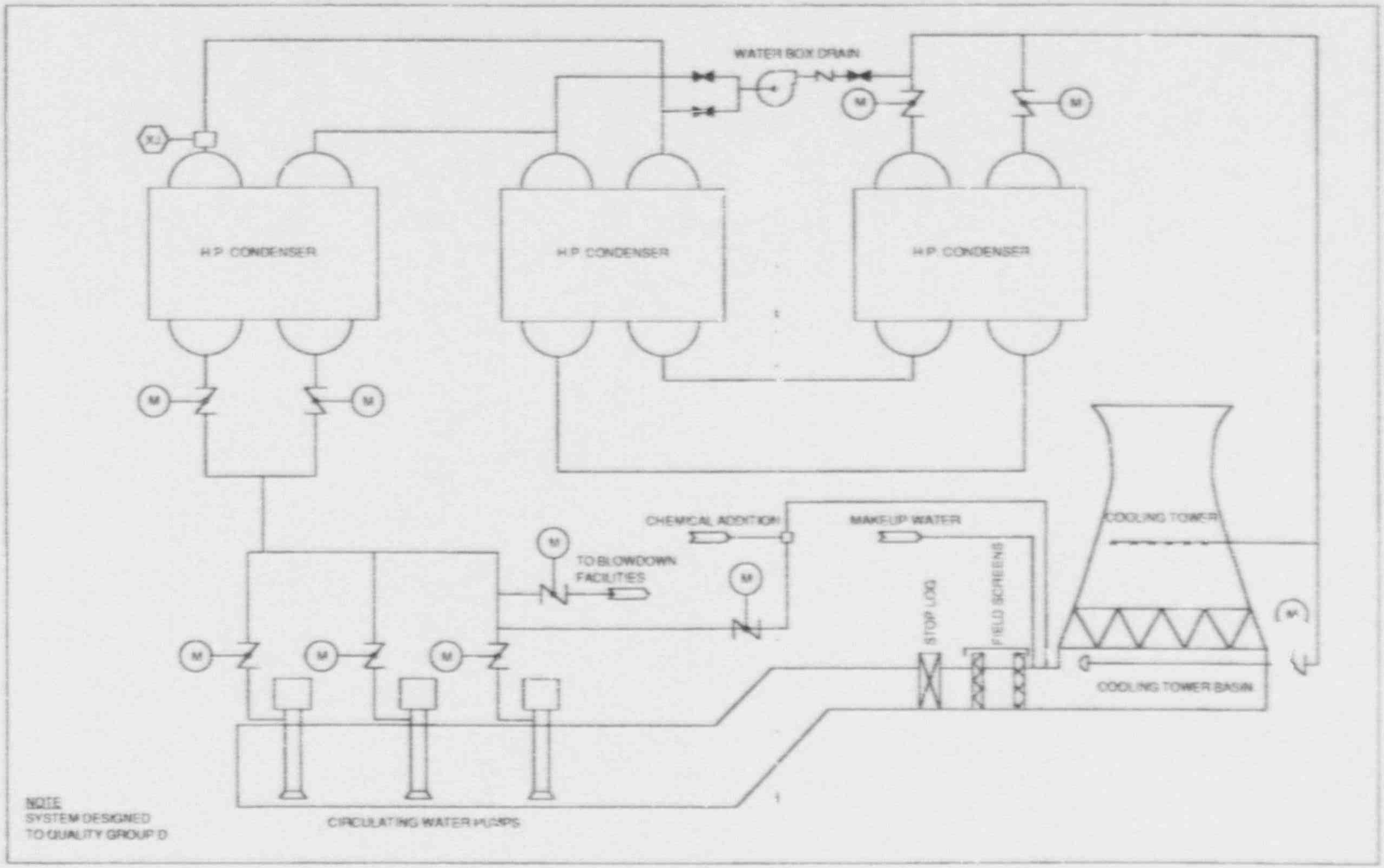
Table 2.10.23 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria which will be undertaken and for the Control Building.

Table 2.10.23: CIRCULATING WATER SYSTEM

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Flooding of the turbine building will be prevented by CWS isolation in the event of gross system leakage.	1. Visual inspection of the installed equipment coupled with the analyses of the leakage/flooding characteristics of the as-built CWS will be performed using simulated signals to verify system isolates on high level.	1. System isolates.

Figure 2.10.23 CIRCULATING WATER SYSTEM



2.11.3 Reactor Building Cooling Water System

Design Description

The reactor building cooling water RCW system distributes cooling water during various plant operating modes, as well as during shutdown, and during post-LOCA operation of the various safety systems. The system removes heat from plant auxiliaries and transfers it to the ultimate heat sink (UHS) via the reactor service water (RSW) system. The RCW removes heat from the ECCS equipment including the emergency diesel generators during a safe reactor shutdown cooling function.

The RCW system is designed to perform its required safe reactor shutdown cooling function following a postulated loss of coolant accident/loss of offsite power (LOCA/LOOP), assuming a single active failure in any mechanical or electrical RCW subsystem or RCW support system. In case of a failure which disables any one of the three RCW divisions, the other two divisions meet plant safe shutdown requirements, including a LOCA or a loss of offsite power, or both.

Redundant isolation valves are able to separate the essential portions of the RCW cooled components from the nonsafety-related RCW cooled components during a LOCA, to assure the integrity and safety functions of the safety related parts of the system. The isolation valves to the nonessential RCW system are automatically or remote-manually operated and their positions are indicated in the main control room.

Each RCW division includes two pumps which circulate RCW through the various equipment cooled by RCW and through three heat exchangers which transfers the RCW heat to the UHS via the RSW.

Each RCW division Main Control Room (MCR) instrument indication includes main loop and RHR HX flow and temperature. MCR control includes MOV's for RCW/RSW Heat Exchanger (HX) isolation, surge tank makeup control, HECW refrigerator RCW flow control, D/G RCW isolation, non essential RCW isolation, and RCW primary containment isolation.

The three RCW train configurations are shown on Figure 2.11.3. The RCW system provides three similar complete trains, A, B and C which are mechanically and electrically separated. The RCW pumps and valves for each RCW division is supplied electrical power from a different division of the ESF power system.

The RCW ASME code classifications for different portions of the system are indicated on Figure 2.11.3a-c. The safety related portions of the RCW divisions are designed to Seismic Category I and Quality Group C.

During various plant operating modes, one RCW water pump and two heat exchangers are normally operating in each division. Flow balancing provisions are included within each RCW division.

Pump design parameters are:

	RCW A/B	RCW C
Design pressure (psig)	200	200
Design temperature (°F)	158	158
Discharge flow rate (gpm/pump)	≥ 5,700	≥ 4,800
Pump total head (psig)	≥ 80	≥ 75
Heat exchanger capa ities are each:	≥ 45E^6 Btu/h	≥ 42E^6 Btu/h

Connections to a radiation monitor are provided in each division to detect radioactive contamination resulting from a tube leak in one of the RHR exchangers, fuel pool exhumers, or other exchangers.

.e RCW pumps and heat exchangers are located in the lower floors of the control building. The equipment cooled by the RCW divisions are located in the reactor building, turbine building, and radwaste building, as stated on Figure 2.11.3a-c. Tables 2.11.3.2b,c,d show which equipment receives RCW flow during various plant operating and emergency modes. The tables also indicate how many heat exchangers are in service in each mode.

During normal plant operation, RCW flows through equipment which is normally operating and requires cooling and all ECCS equipment, except RHR heat exchangers and ..F diesel generators as shown by open or closed valves in Figure 2.11.3.

If a LOC/ curs, a second RCW pump and third heat exchanger in each loop are placed in service. Automatic or remote operated isolation valves will separate the RCW for the LOCA required safety equipment from the nonsafety-related equipment, if a RCW surge tank low water level signal occurs. The primary containment RCW isolation valves automatically close if a LOCA occurs.

After a LOCA, the following sequence will be followed:

a. If the nonsafety portion of RCW is available to the instrument air/service air (IA/SA) compressors, the CRD pumps and CUW pumps, RCW flow to these nonsafety components shown on Figure 2.11.3 is maintained. Flow is automatically shutoff to other non-essential equipment after the LOCA.

b. If the operator determines after the LOCA, from essential RCW instrumentation, that the integrity of the non-safety RCW system to the above mentioned compressors and pumps has been lost he can shut the remote operated nonessential isolation valves shown in Figure 2.11.3.

If the surge tank water level reaches a low level, indicating loss of water out of the RCW system, isolation valves in the supply and return piping to the nonessential equipment will automatically close, including the compressors and pumps mentioned above.

The RCW/RSW heat exchanger design basis condition occurs during post-LOCA cooling of the containment via the RHR heat exchangers.

The RCW pumps have the flow capacity to deliver required flow to the ECCS equipment in each division and the above mentioned compressors and pumps if the isolation valves cannot be closed.

After a LOOP, the RCW pumps are automatically powered by the emergency diesel generators.

A separate surge tank is provided for each RCW division. Normal makeup water source to the surge tank is the makeup demineralized water system. For LOCA conditions, the suppression pool cleanup system (SPCU) provides a backup surge tank water supply.

Inspection, Test, Analyses and Acceptance Criteria

Table 2.11.3a provides a definition of the inspections, tests, and/or analyses together with associated acceptance criteria which will be and undertaken for the RCW.

Table 2.11.3a: REACTOR COOLING WATER (RCW) SYSTEM

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. System configuration including key components and flow paths are shown in Figure 2.11.3.	1. Inspection of construction records will be performed. Visual inspection (VI) will be performed based on Figure 2.11.3.	1. The system configuration conforms with Figure 2.11.3.
2. Three RCW trains are mechanically and electrically independent.	2 Tests and VI of the three independent trains will be conducted which will include independent and coincident operation of the three trains to demonstrate complete divisional separation.	2. Plant tests and VI confirm proper independence of three RCW divisions.
3. During various modes of operation, RCW has adequate hydraulic capability for plant auxiliaries and the primary containment required for safe shutdown following a design accident or transient. These safe shutdown requirements are satisfied with only any 2 of 3 RCW divisions operating.	3. Limited system hydrualic tests will be conducted according to available nonnuclear heat plant conditions. The tests will demonstrate a safe plant shutdown with one RCW division out of service.	3. The results confirm the RCW has the water flow capability specified by the certified design commitment, including safe shutdown operation with 1 RCW division out of service.
4. Isolation valves as shown in Figure 2.11.3 can automatically or remote manually separate the RCW for the essential equipment from the RCW for the nonessential equipment.	4. VI of the installed RCW system and RCW preoperational tests as follows will be completed. a. Remote-manual operation of the isolation valves from the main control room. b. During simulated LOCA conditions, a simulated LOCA condition will be combined with a simulated RCW surge tank water level signal to automatically close the isolation valves. c. A LOCA signal will shut RCW isolation valves which will shut off RCW flow to all nonessential equipment except the IA/SA compressors, CRD pumps and CUW pumps.	4. Isolation valves are properly located as shown in Figure 2.11.3 and are demonstrated to operate automatically or remote manually to isolate RCW for nonessential from RCW for essential equipment cooled by RCW.

Table 2.11.3a: REACTOR COOLING WATER (RCW) SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. A LOCA will result in the automatic start of the second RCW pump in each division and start flow through the third RCW/RSW Hx in each division. During LOCA/LOOP (loss of coolant accident/loss of off-site power) conditions, RCW pumps and valves are powered by the emergency diesel generators (D/G).	5. Tests simulating LOCA/LOOP conditions will be conducted for the RCW system which confirm the RCW and it's support systems will perform it's function under those conditions. Tests will be conducted for the RCW, which confirm that after the LOOP, each division of RCW pumps and valves operate with the same division of emergency D/G power and associated DC control power sources.	5. LOCA/LOOP signal successfully starts second RCW pump and initiates RCW/RSW Hx flow in each division including the following confirmations: a. Regardless of which RCW pump was operating during normal operation before the LOCA, after the LOAC/LOOP simulation occurs, the first and second RCW pump will start automatically, powered by the emergency diesel generator. b. Regardless of which two RCW/RSW Hx's were operating before the LOCA, after the LOCA/LOOP occurs, the RCW motor operated valve on the third Hx discharge will open automatically.

Table 2.11.3b: REACTOR BUILDING COOLING WATER CONSUMERS

DIVISION A

Operating Mode/ Components	Normal Operating Conditions	Shutdown at 4 hours	Shutdown at 20 hours	Hot Standby (no loss of AC)	Hot Standby (loss of AC)	Emergency (LOCA) (Suppression Pool at 97°C
RCW/RSW Heat Exchangers In Service	2	3	3	2	3	3
ESSENTIAL	Note 1					
Emergency Diesel Generator A	–	–	–	–	X	X
RHR Heat Exchanger A	–	X	X	–	X	X
FPC Heat Exchanger A	X	X	X	X	X	X
Others (essential) (Note 2)	X	X	X	X	X	X
NON-ESSENTIAL						
RWCU Heat Exchanger	X	X	X	X	X	–
Inside Drywell (Note 3)	X	X	X	X	X	–
Others (non-essential) (Note 4)	X	X	X	X	X	X

NOTES:

(1) (X) = Equipment receives RCW in this mode.
(-) = Equipment does not receive RCW in this mode.

(2) HECW refrigerator, room coolers (FPC pump, RHR, RCIC, SGTS, FCS, CAMS), RHR motor and seal coolers.

(3) Drywell (A & C) and RIP coolers.

(4) Instruments and service air coolers; RWCU pump cooler, CRD pump oil, and RIP Mg sets.

Table 2.11.3c: REACTOR BUILDING COOLING WATER CONSUMERS

DIVISION B

Operating Mode/ Components	Normal Operating Conditions	Shutdown at 4 hours	Shutdown at 20 hours	Hot Standby (no loss of AC)	Hot Standby (loss of AC)	Emergency (LOCA) (Suppression Pool at 97°C
RCW/RSW Heat Exchangers In Service	2	3	3	2	3	3
ESSENTIAL	Note 1					
Emergency Diesel Generator B	-	-	-	-	X	X
RHR Heat Exchanger B	-	X	X	-	X	X
FPC Heat Exchanger B	X	X	X	X	X	X
Others (essential)(Note 2)	X	X	X	X	X	X
NON-ESSENTIAL						
RWCU Heat Exchanger	X	X	X	X	X	-
Inside Drywell (Note 3)	X	X	X	X	X	-
Others (non-essential) (Note 4)	X	X	X	X	X	X

NOTES:

(1) (X) = Equipment receives RCW in this mode.
(-) = Equipment does not receive RCW in this mode.

(2) HECW refrigerator, room coolers (FPC pump, RHR, RCIC, SGTS, FCS, CAMS), RHR motor and seal coolers.

(3) Drywell (B) and RIP coolers.

(4) Reactor Building sampling coolers; LCW sump coolers (in drywell and reactor building), RIP MG sets and RWCU pump coolers.

Table 2.11.3d: REACTOR BUILDING COOLING WATER CONSUMERS

DIVISION C

Operating Mode/ Components	Normal Operating Conditions	Shutdown at 4 hours	Shutdown at 20 hours	Hot Standby (no loss of AC)	Hot Standby (loss of AC)	Emergency (LOCA) (Suppression Pool at 97°C
RCW/RSW Heat Exchangers	[illegible]	3	3	2	3	3
ESSENTIAL	Note 1					
Emergency Diesel Generator B	–	–	–	–	X	X
RHR Heat Exchanger B	–	X	X	–	X	X
Others (essential)(Note 2)	X	X	X	X	X	X
NON-ESSENTIAL						
Others (non-essential) (Note 4)	X	X	X	X	X	X

NOTES:

(1) (X) = Equipment receives RCW in this mode.
(-) = Equipment does not receive RCW in this mode.

(2) HECW refrigerator, room coolers motor coolers, and mechanical seal coolers for RHR and HPCF.

(3) Instrument and service air coolers, CRD pump oil cooler, radwaste components, HSCR condenser, and turbine building sampling coolers.

Figure 2.11.3a RCW DIVISION - A

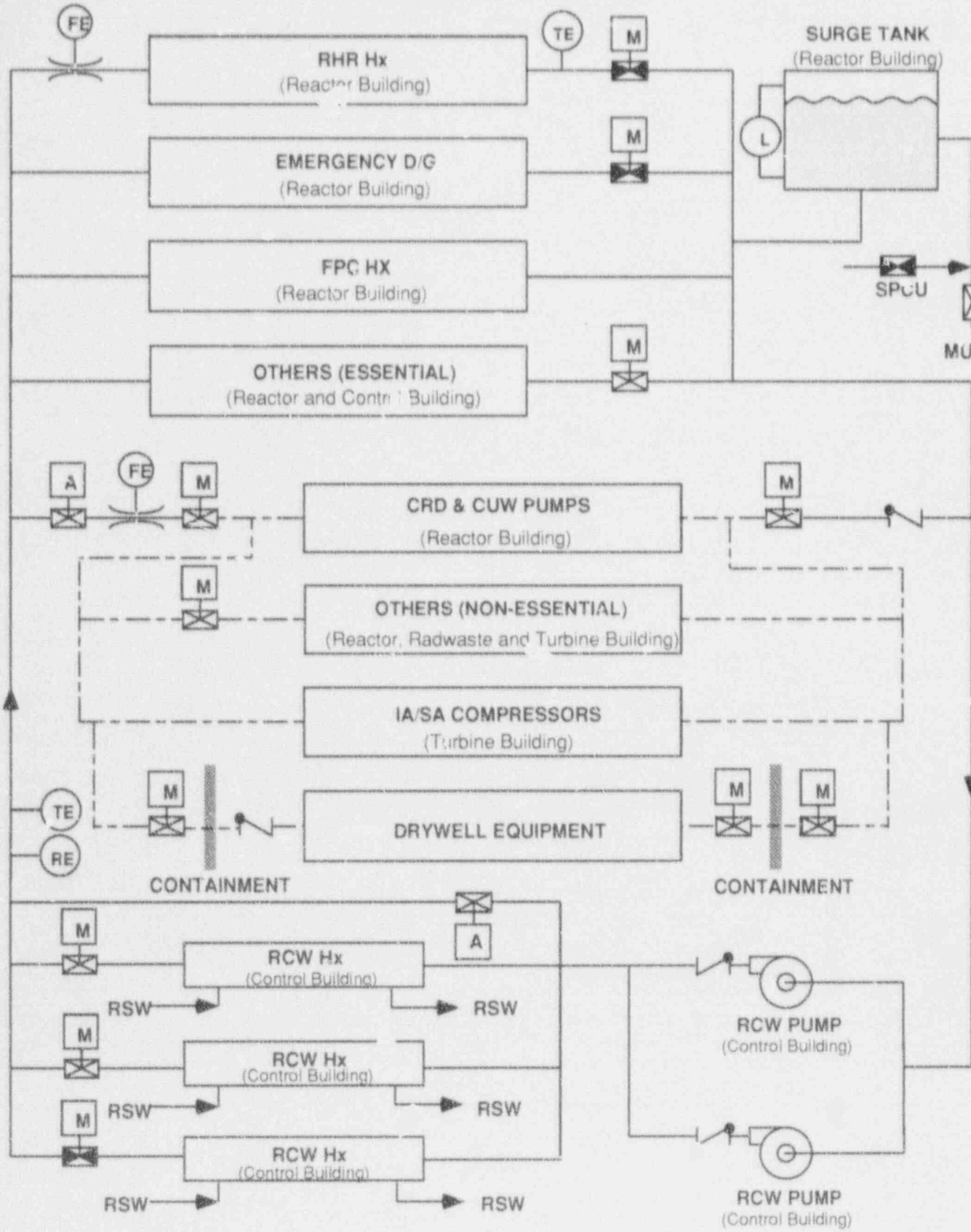


Figure 2.11.3b RCW DIVISION - B

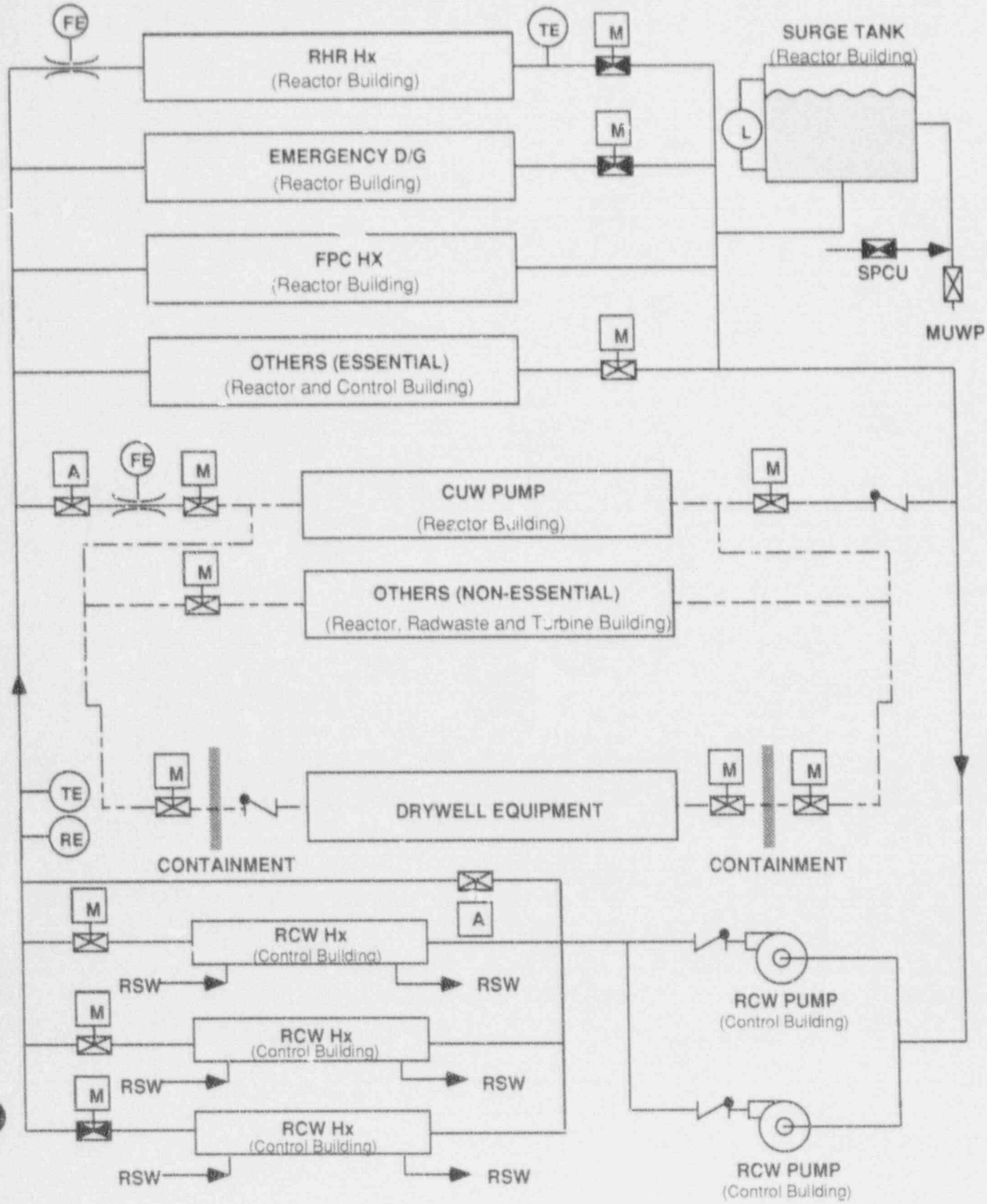
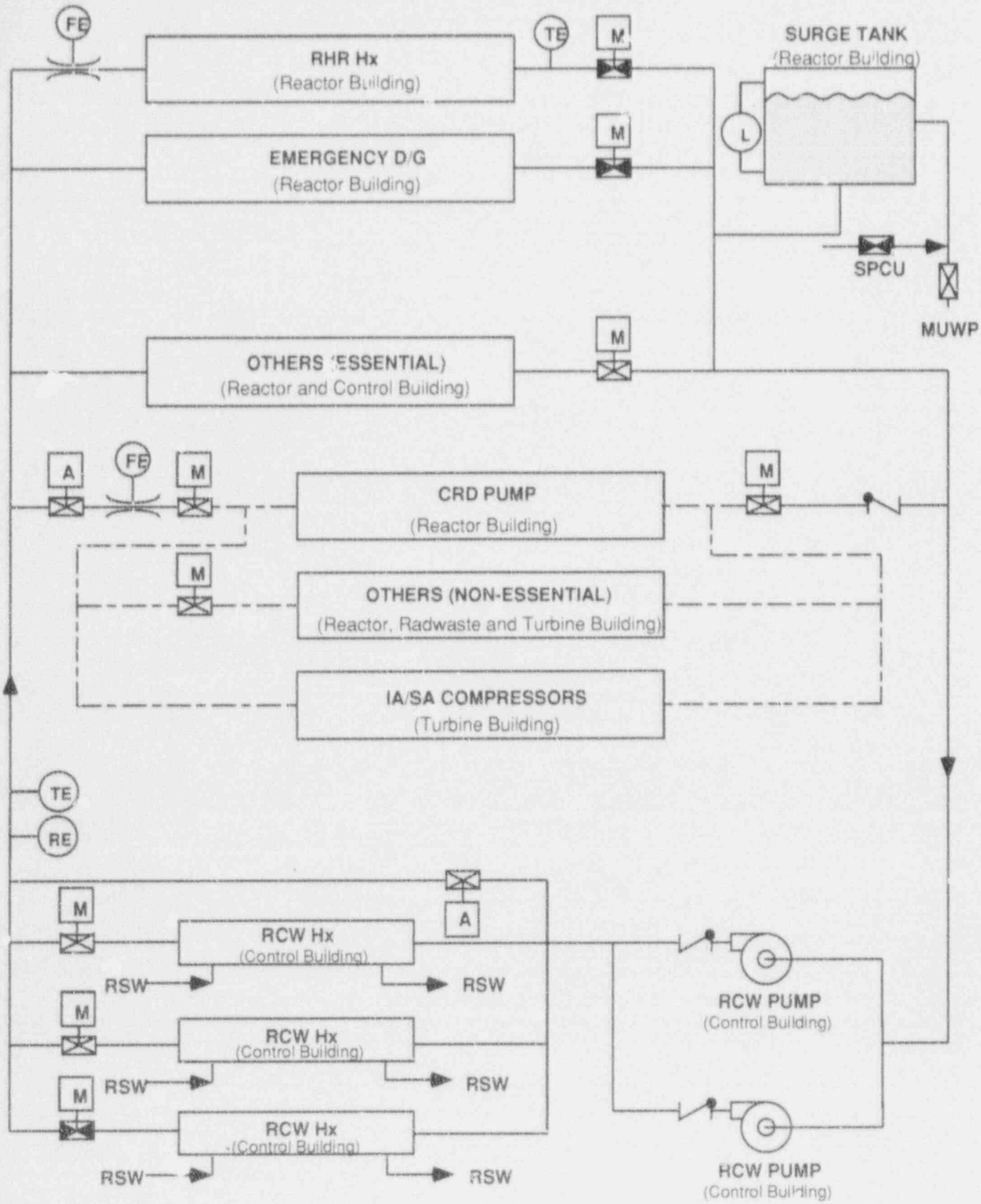


Figure 2.11.3c RCW DIVISION - C



2.12.13 Emergency Diesel Generator System (Standby AC Power Supply)

Design Description

The Class 1E diesel generators comprising the Divisions I, II, and III standby AC power supplies are designed to restore power to their respective Class 1E distribution system divisions as required to achieve safe shutdown of the plant and/or to mitigate the consequences of a loss-of-coolant accident (LOCA) in the event of a coincident loss of normal electrical power. Each of the three divisions of the AC power system has its own diesel generator.

The major loads consist of the following systems for all three divisions: Residual Heat Removal (RHR) System, Reactor Building Cooling Water (RCW) System, HVAC Emergency Cooling Water (HECW) System, and Reactor Service Water (RSW) System. In addition, Divisions II and III include the High Pressure Core Flooder (HPCF) System loads. (The Division I RCIC system is also part of the ECCS network, but is steam-driven and therefore does not present a significant load to the diesel generator.)

Each Class 1E standby power system division, including the diesel generator, its auxiliary systems, and the distribution of power to various Class 1E loads through the 6.9 kv and 480 v systems, is segregated and separated from the other divisions. No automatic interconnection is provided between the Class 1E divisions. Each diesel-generator set is operated independently of the other sets, and is connected to the utility power system by manual control only during testing or for bus transfer. A failure of any component of one diesel generator set will not jeopardize the capability of either of the two remaining diesel generator sets to perform their functions. The diesel generators are Seismic Category 1, and are located within the Reactor Building (see Figure [later]).

Each diesel generator unit is rated at 6.9 kv, 60 Hz; and is capable of automatically starting, accelerating, attaining rated frequency and voltage within 20 seconds, and supplying its loads in the sequence and timing specified in the plant design documents. In addition, each diesel generator is capable of starting, accelerating and running its largest motor at any time after the automatic loading sequence is completed, assuming that the motor had failed to start initially.

The diesel generators start automatically on loss of bus voltage. Under-voltage sensors are used to start each diesel engine in the event of a sustained drop in bus voltage below 70% of the nominal 6.9 kv rating of the bus. Low-water-level sensors and drywell high-pressure sensors in each division are also used to initiate the respective diesel start under accident conditions. However, the diesels will remain on standby (i.e., running at rated voltage and frequency, but unloaded) unless the bus under-voltage sensors trigger the need for bus transfer to the diesel supply. Manual start capability (without need of DC power) is also provided.

Each diesel is supplied by its own independent fuel tank, which is located in an area protected from natural phenomena, and which is capable of supplying fuel for at least 7 days of continuous operation.

The standby AC power supplies are designed such that testing and inspection of equipment is possible during both normal and shutdown plant conditions.

Each standby AC power supply is composed of a three-phase synchronous generator and exciter, the diesel engine, the engine auxiliaries (including the fuel tanks), and the control panels. Figure 2.12.13 shows the emergency diesel generator system interfaces, which includes the interconnections between the offsite power supplies and the Divisions I, II, and III diesel-generator standby AC power supplies.

The transfer of each Class 1E bus to its standby power supply is automatic, should this become necessary, on loss of its offsite power. After the circuit breaker connecting the bus to the preferred power supply is open, major loads are tripped from the Class 1E bus, except for the Class 1E 480 v unit substation feeders. Then the diesel-generator breaker is closed when required generator voltage and frequency are established. The large motor loads are later re-applied sequentially and automatically to the bus after closing of the diesel-generator breaker.

Each diesel generator is capable of being started or stopped manually from the main control room. Start/stop control and bus transfer control may be transferred to a local control station in the diesel generator room.

Each diesel generator, when operating other than in test mode, is independent of the preferred power supply. Additional interlocks to the LOCA and loss-of-power sensing circuits terminate parallel operation tests and cause the diesel generator to automatically revert and reset to its standby mode if either signal appears during a test. A lockout or maintenance mode removes the diesel generator from service. The inoperable status is indicated in the control room.

Devices monitor the conditions of the diesel generators, and effect action in accordance with one of the following categories: 1) Conditions to trip the diesel engine even under LOCA, 2) Conditions to trip the diesel engine except under LOCA, 3) Conditions to trip the generator breaker but not the diesel, and 4) Conditions which are only annunciated.

Inspection, Test, Analyses and Acceptance Criteria

Table 2.12.13 provides a definition of the inspections, tests, and/or analyses together with associated acceptance criteria which will be undertaken for the emergency diesel generator system.

Table 2.12.13: EMERGENCY DIESEL GENERATOR SYSTEM

Inspections, Tests, Analyses and Acceptance Criteria

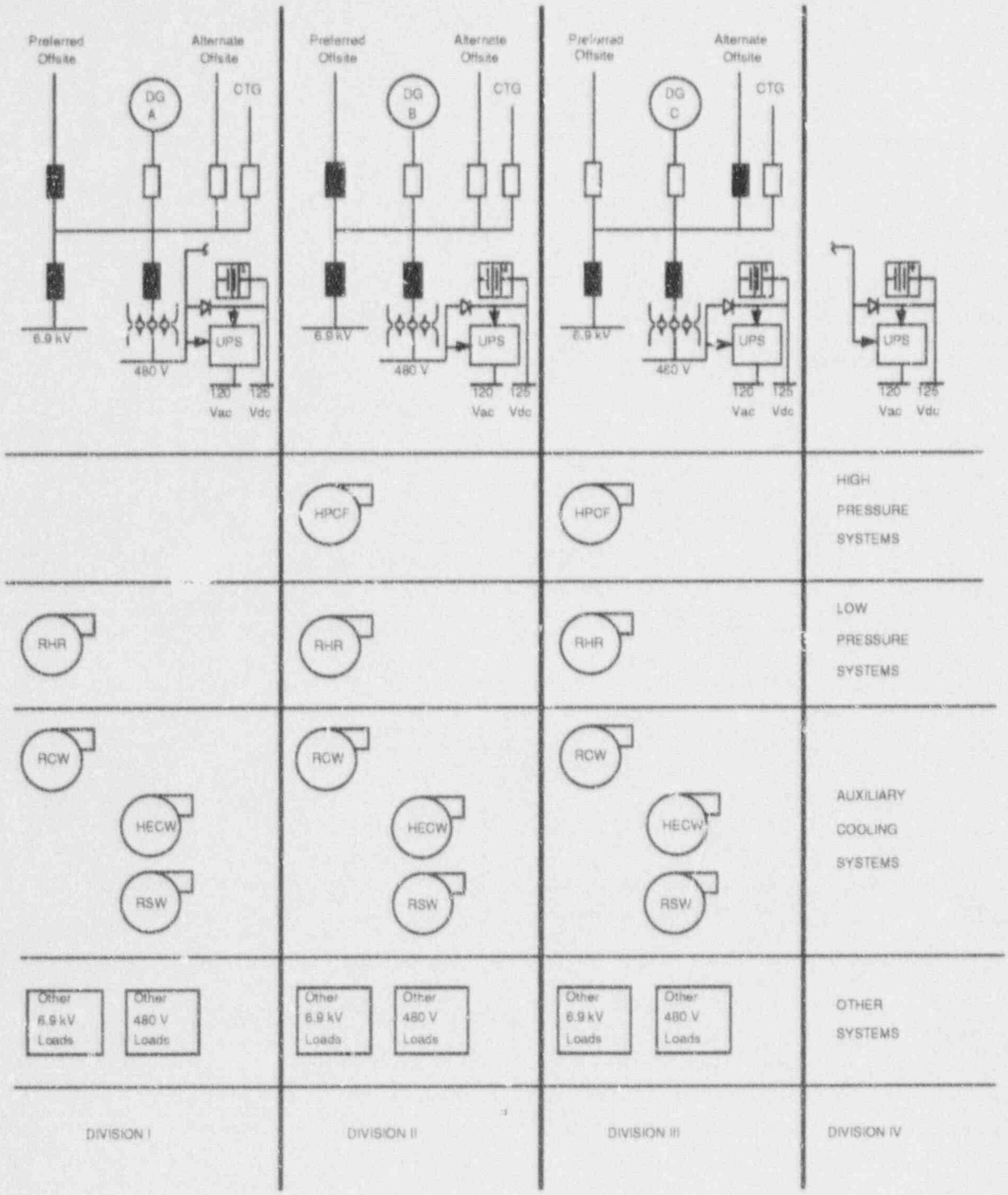
Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The three diesel generators are capable of supplying sufficient AC power to achieve safe shutdown of the plant and/or to mitigate the consequences of a LOCA in the event of a coincident loss of normal power. (See Figure 2.12.13.)	1.a Confirmatory inspection will be performed to assure the maximum design loads expected to occur for each division are within the ratings of the corresponding diesel generator.	1.a The maximum loads expected to occur for each division (according to nameplate ratings) shall not exceed 90% of the rated power output of the diesel generator (i.e., 4500 kw).
	1.b Testing will be conducted by synchronizing each diesel generator to the plant offsite power system and increasing its output power level to its fully rated load condition.	1.b Each of the three units shall produce rated power output (5000 kw) at ≥0.8 pf for a period of ≥24 hours (momentary transients excepted). Each unit will then experience full load rejection by tripping the load and verifying the unit does not trip.
2. Each diesel generator is rated at 6.9 kv, three-phase, 60 Hz; and is capable of attaining rated frequency and voltage within 20 seconds after receipt of a start signal.	2. Perform a test of each diesel generator to confirm its ability to attain rated frequency and voltage.	2. Each diesel generator attains a voltage of 6.9 kv±10%, and a frequency of 60 Hz±2% within 20 seconds after application of a start signal.
3. In the event of a loss of normal power, each diesel generator unit is capable of starting (both manually and automatically), accelerating, and supplying its loads in the proper sequence and timing specified in the plant design documents. It is also capable of recovery following trip and restart of its largest load.	3. The automatic and manual start sequences will be tested for each diesel generator unit.	3. Each of the three units starts (via each manual and automatic signal), accelerates, and properly sequences its loads in accordance with the plant sequence diagram. The load sequence begins at 20±2 seconds and ends ≤65 seconds. Following application of each load, the bus voltage will not drop more than 25% at its terminals. Frequency should be restored to within 2% of nominal, and voltage should be restored to within 10% of nominal within 60% of each load-sequence time interval. In addition, the unit's largest motor load shall be tripped and restarted after the unit has completed its sequence, and the bus voltage will recover to 6.9 kv±10% at 60±2% Hz within 10 seconds.

Table 2.12.13: EMERGENCY DIESEL GENERATOR SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Each diesel generator unit is capable of manually starting without the need for external electrical power.	4. Each unit will be tested to assure its black-start capability is functional.	4. Black-start capability is demonstrated following manual start, acceleration, and bus energization for each of the three units without assist from any external electric power.
5. Interlocks to the LOCA and loss-of-power sensing circuits terminate parallel operation tests and cause the diesel generator to automatically revert and reset to its standby mode if either signal appears during a test.	5. Interlocks for the standby AC power system will be tested.	5. While in a parallel test mode, each unit will revert and reset to its standby mode following individual application of a LOCA signal and a loss-of-power signal.
6. Devices monitor the conditions of the diesel generators, and effect action in accordance with one of the following categories: 1) Conditions to trip the diesel engine even under LOCA, 2) Conditions to trip the diesel engine except under LOCA, 3) Conditions to trip the generator breaker but not the diesel, and 4) Conditions which are only annunciated.	6. Using simulated signals, protective interlocks and annunciations will be tested to assure they perform their functions, in accordance with the four categorical conditions described.	6. Successful circuit testing will be confirmed for the individual diesel generator protective sensors according to the following: <u>Category 1 sensors:</u> Annunciations and diesel engine trip signals will be confirmed in combination with a LOCA signal. <u>Category 2 sensors:</u> Annunciations and diesel engine trip signals will be confirmed without a LOCA, but trips will be bypassed when LOCA signal is present. <u>Category 3 sensors:</u> Annunciations and generator circuit breaker trip signals will be confirmed. <u>Category 4 sensors:</u> Annunciation signals will be confirmed.

Figure 2.12.13 EMERGENCY DIESEL GENERATOR SYSTEM INTERFACES



2.15.12 Control Building

Design Description

The control building (CB) is the building that houses the main control room, control equipment and operations personnel for the Reactor and Turbine Islands. The control building is located between the reactor and turbine buildings.

In addition to the control room and operations personnel, this building houses the essential electrical, control and instrumentation equipment, essential switch gear, essential battery rooms, the CB heating and air conditioning (HVAC) equipment, reactor building component cooling water pumps and heat-exchangers, and the steam tunnel.

The general building arrangement including watertight doors and sills for doorways where needed for flood control is found in Figures 2.15.12a through 2.15.12g.

The CB is a Seismic Category I structure designed to provide missile and tornado protection.

The CB is constructed of reinforced concrete with steel truss roof. The CB has two stories above the grade level and four stories below. The building shape is rectangle. Major nominal dimensions are as follows:

Overall height above top of basemat	30.5 m
Overall planar dimensions (outside)	
0 deg-180 deg direction	24.0 m
90 deg-270 deg direcion	56.0 m
Thickness of Outer Wall	
from -8.2m TMSL to 17.15m TMSL	1.2m
from 17.15m TMSL to 22.2 m TMSL	0.6m
Thickness of Steam Tunnel	
Walls, Floors, and Ceiling	1.6m
Thickness of Walls supporting Steam Tunnel	1.6m

Note that dimensional variations shall be within the limits which will have insignificant effect on the structural design.

The CB is a shear wall structure designed to accommodate all seismic loads with its perimeter walls. Therefore, frame members such as beams or columns are designed to accommodate deformations of the wall in case of earthquake condition.

To protect against external flood damage, the following design features are provided:

a. wall thickness below flood level greater than 0.6m.

b. water stops provided in all construction joints below grade.

c. watertight doors and piping penetrations installed below flood level.

d. waterproof coating on exterior walls.

e. foundations and walls of structures below grade are designed with water stops at expansion and construction joints.

f. roofs are designed to prevent pooling of large amounts of water.

To protect against internal flood damage, the following design features are provided:

a. elevation differences and divisional separations from remainder of the CB.

b. drainage system to divert water to assigned floor and location.

c. sills for doorways as needed to provide flood control.

d. watertight doors installed below internal flood level.

e. wall thickness below internal flood level greater than 0.6m.

Inside the steam tunnel is the mainsteam piping, the mainsteam drain line, and the feedwater piping. The steam tunnel has no penetrations from the steam tunnel into the control building. Any high energy line breaks inside the steam tunnel will vent out to the turbine building. All standing water will collect in the large volumes in the lower portions of the steam tunnel at the reactor building or turbine building ends.

On Floor B1F, there are fire hose stands and reactor cooling water (RCW) piping. It is designed that any rupture of the fire hose stand will leak onto the floor and drain to the -8200 level by floor drains. Sills will be provided at doorways to prevent the entry of standing water into the control room complex. The RCW piping runs vertically in a concrete pipe chase. No flooding outside this pipe chase is possible.

On the floor where computer room located, there are fire hose stands, RCW piping, and other piping systems. A limited amount of standing water is expected upon a rupture of any of these systems. Sills will be provided at doorways to prevent water from crossing divisional boundaries. Similar arrangements and designs are also provided for other floors for floods protection.

During normal operation, the concrete surrounding the steamline tunnel provides shielding so that operator doses are below the value associated with uncontrrolled,unlimited access. The outer walls of the control building are designed to attenuate radiation from radioactive materials contained within the reactor building and from possible airborne radiation surrounding the control building following a LOCA. The walls provide shielding to limit the direct-shine exposure of control room personnel following a LOCA. Shielding for the outdoor air cleanup filters also is provided to allow temporary access to the mechanical equipment area of the control building following a LOCA, should it be required.

The control building is not a vented structure. The exposed exterior roofs and walls of the structure are designed for the required pressure drop. Tornado dampers are provided on all air intake and exhaust openings. These dampers are designed to withstand the specified negative pressure.

Inspection, Test, Analyses and Acceptance Criteria

Table 2.15.12 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria which will be undertaken for the control building.

Table 2.15.12: CONTROL BUILDING

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Control building general arrangement is shown in Figures 2.15.12a through 2.15.12g.	1. Plant walk through to check and verify requirements are met.	1. Per Figure 2.15.12a through 2.15.12.g..
2. Design features are provided to protect against design basis internal and external floods.	2. Review construction records and perform visual inspections of the flood control features.	2. For external flooding: a. Exterior wall thickness below flood level greater than 0.6m. b. Water stop c. Watertight door and piping penetrations below flood level d. Water proof coating on exterior walls e. Foundations and walls of structures below grade are designed with water stops at expansion and construction joints f. Roofs are designed to prevent pooling of large amounts of water. For internal flooding: a. Elevation differences and divisional separation of the mechanical functions from the remainder of the CB b. Drainage system to divert water to assigned floor and location c. Sills for doorways as needed to provide flood protection d. Watertight doors installed below internal flood level e. Wall thickness below internal flood level greater than 0.6m. f. Steam tunnel has no penetrations from the steam tunnel into the control building. Any high energy line or feedwater piping breaks inside the steam tunnel will vent out to the Turbine Building.

Table 2.15.12: CONTROL BUILDING (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The control building is designed to have adequate radiation shielding to protect operating personnel during operation and following a LOCA.	3. Performed dimensional inspections of the Control Building walls, ceiling, floors, and other structural features.	3. The concrete thickness for the steam tunnel wall, floor and ceiling shall be greater than 1.6m. The steam tunnel interface structure and control building wall below the steam tunnel should have a combined thickness of 1.6m.
4. The CB is designed to protect against design basis tornado and tornado missiles.	4. Review construction records and perform visual inspections of the tornado protection features.	4. For tornado a. Roof and walls above grade designed greater than 0.5m b. HVAC dampers designed for differential pressure > 1.46 psi. c. HVAC dampers have tornado missile barriers.

Figure 2.15.12a CONTROL BUILDING ELEVATION (90° - 270°)

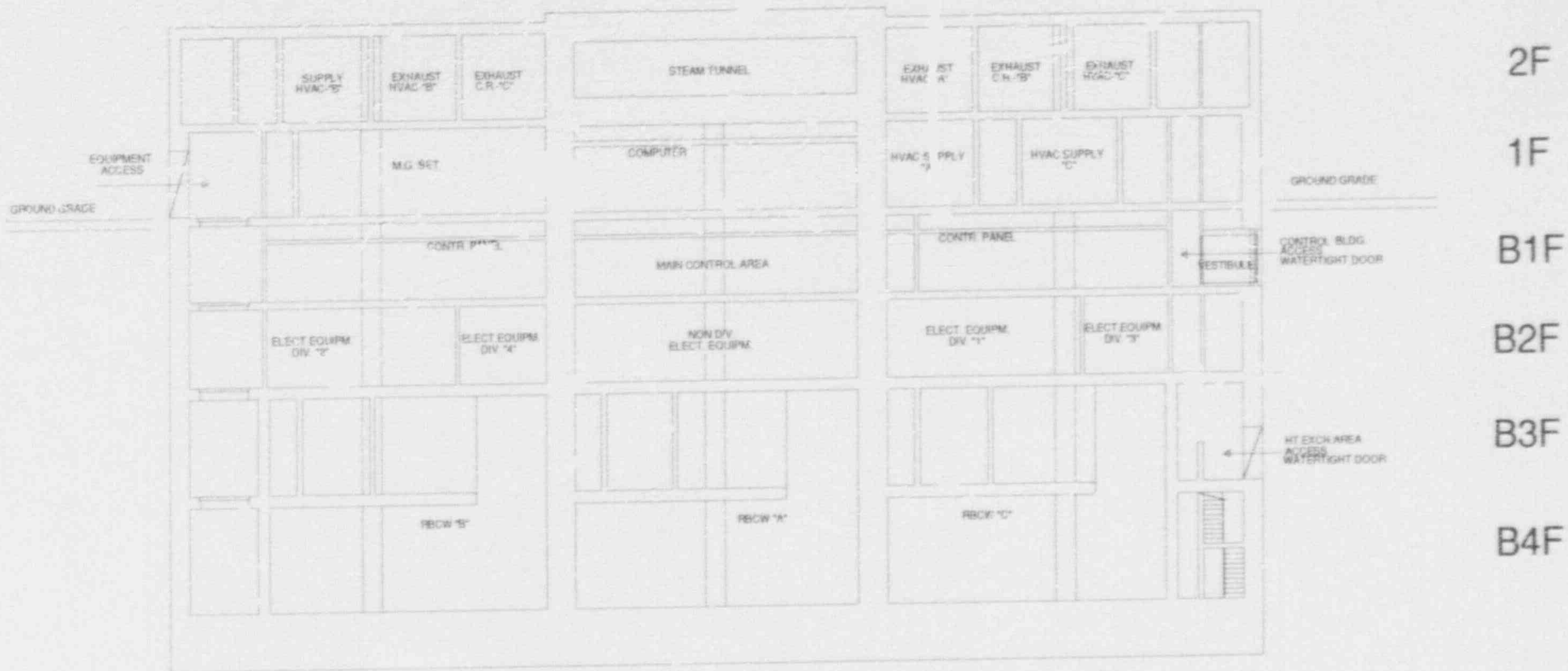


Figure 2.15.12b CONTROL BUILDING - FLOOR 2F

Figure 2.15.12c CONTROL BUIDING FLOOR 1F - GROUND GRADE

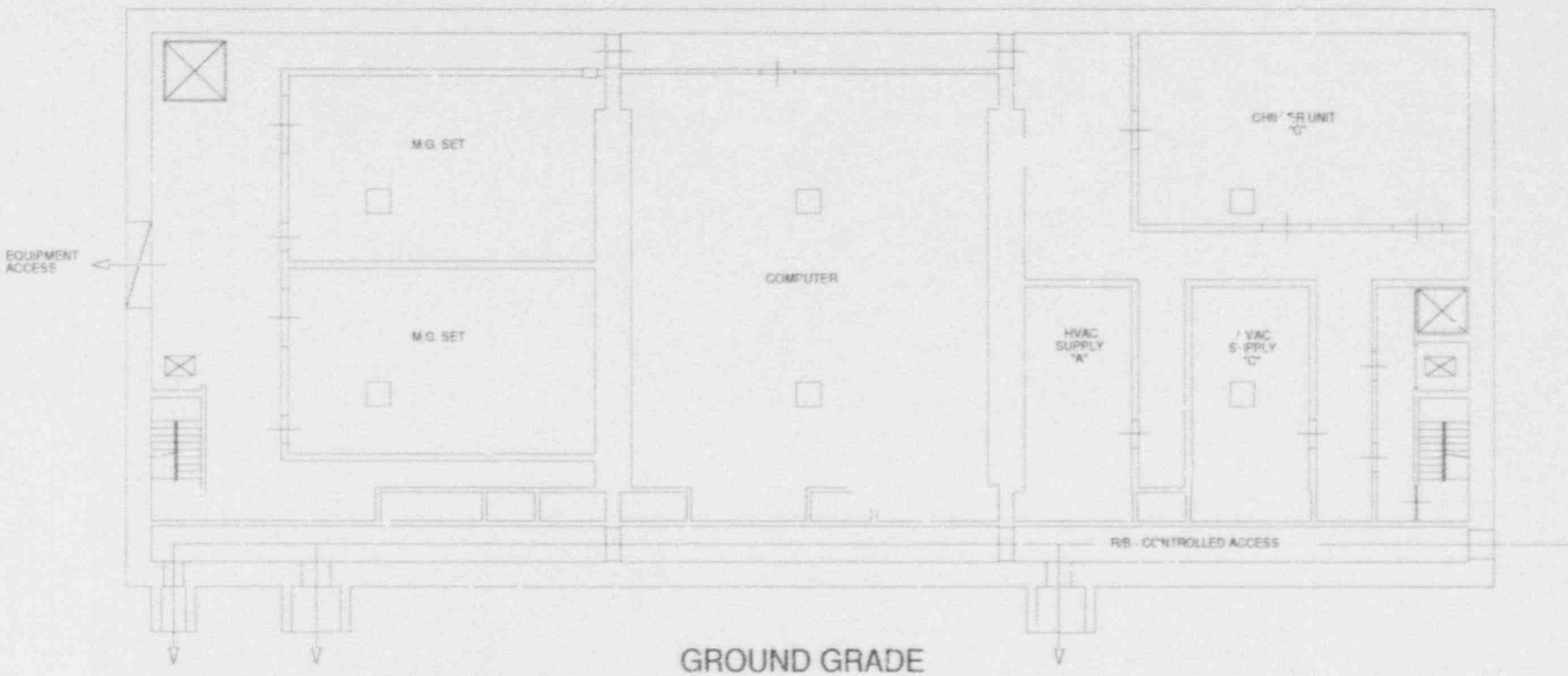


Figure 2.15.12d CONTROL BUILDING FLOOR B1F

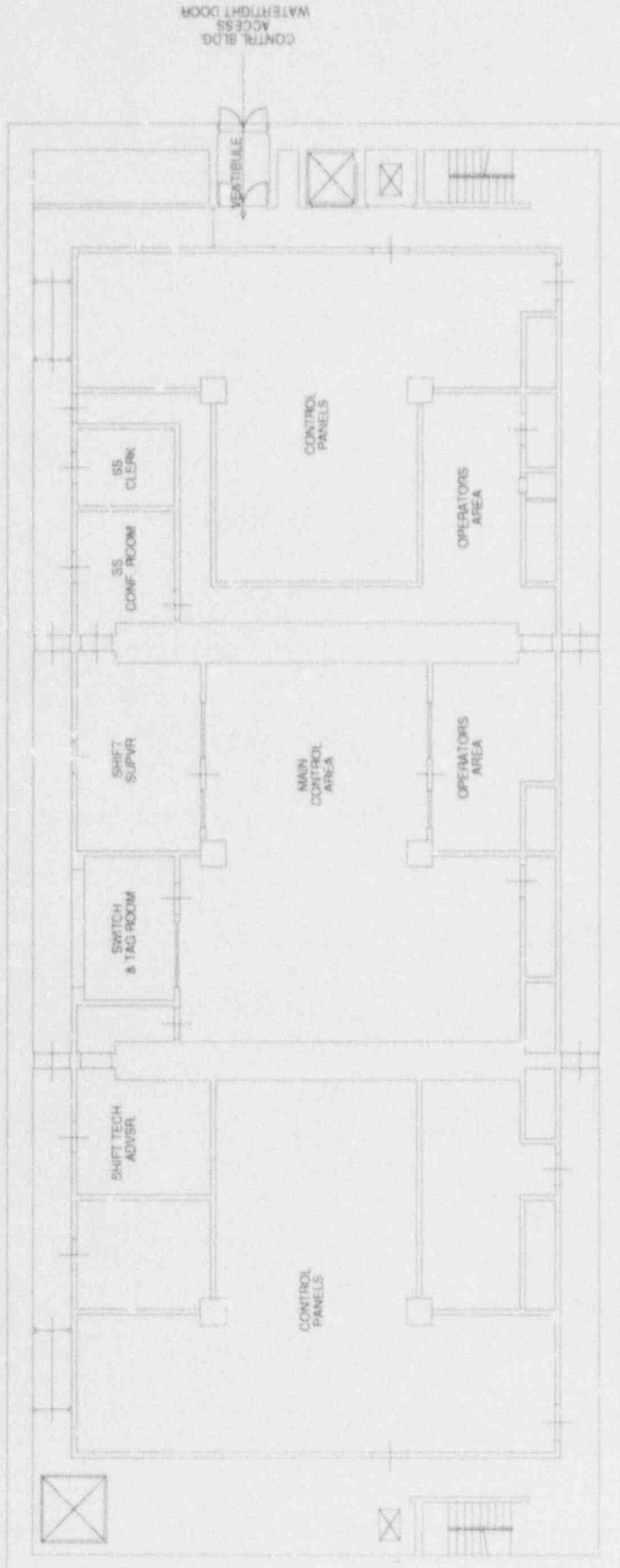


Figure 2.15.12e CONTROL BUILDING FLOOR B2F

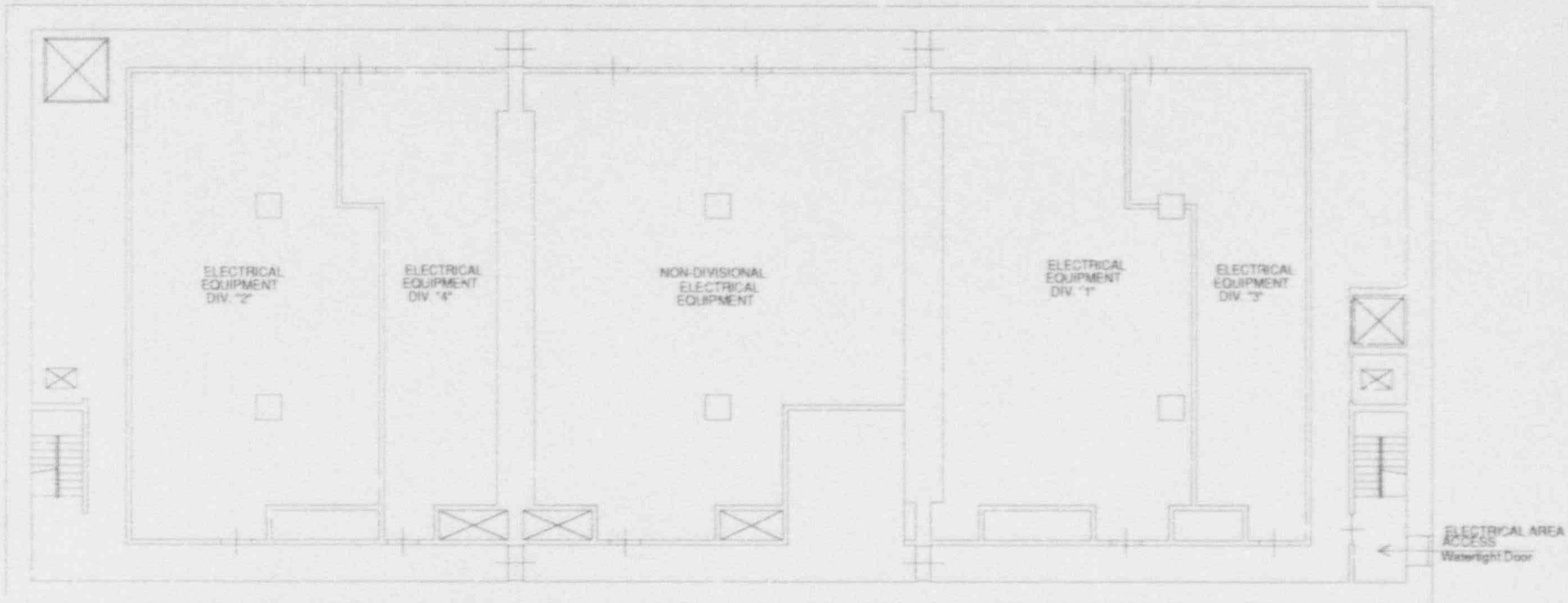


Figure 2.15.12f CONTROL BUILDING FLOOR B3F

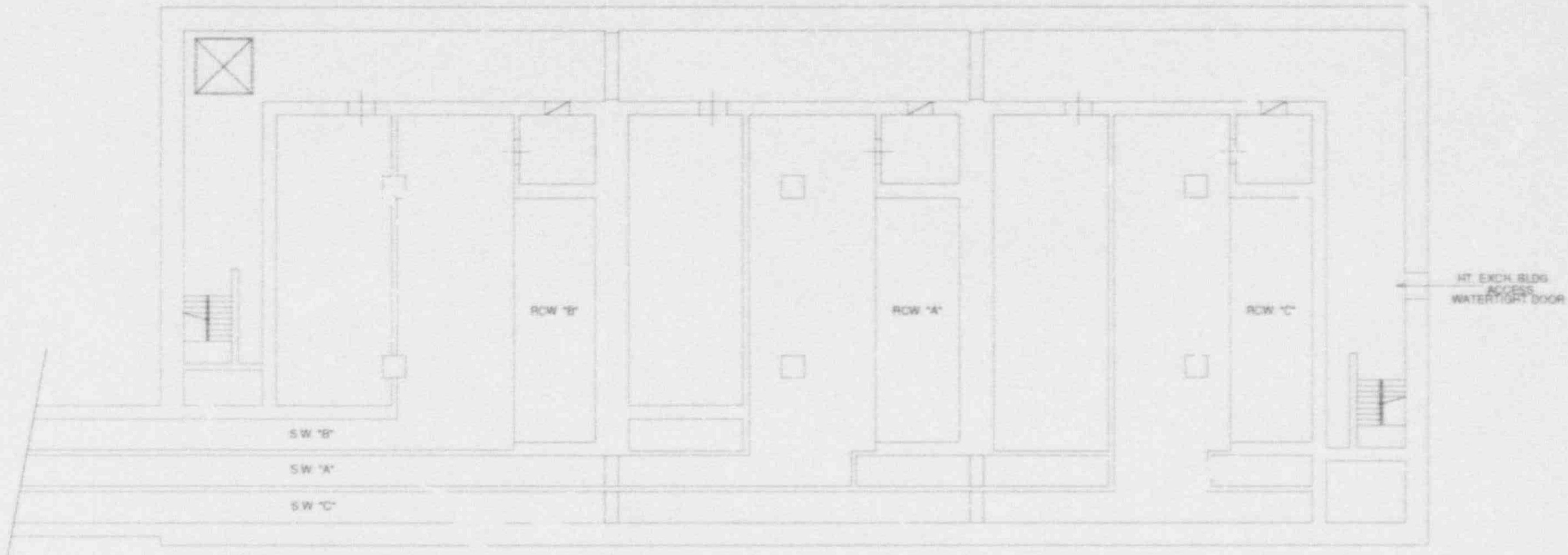
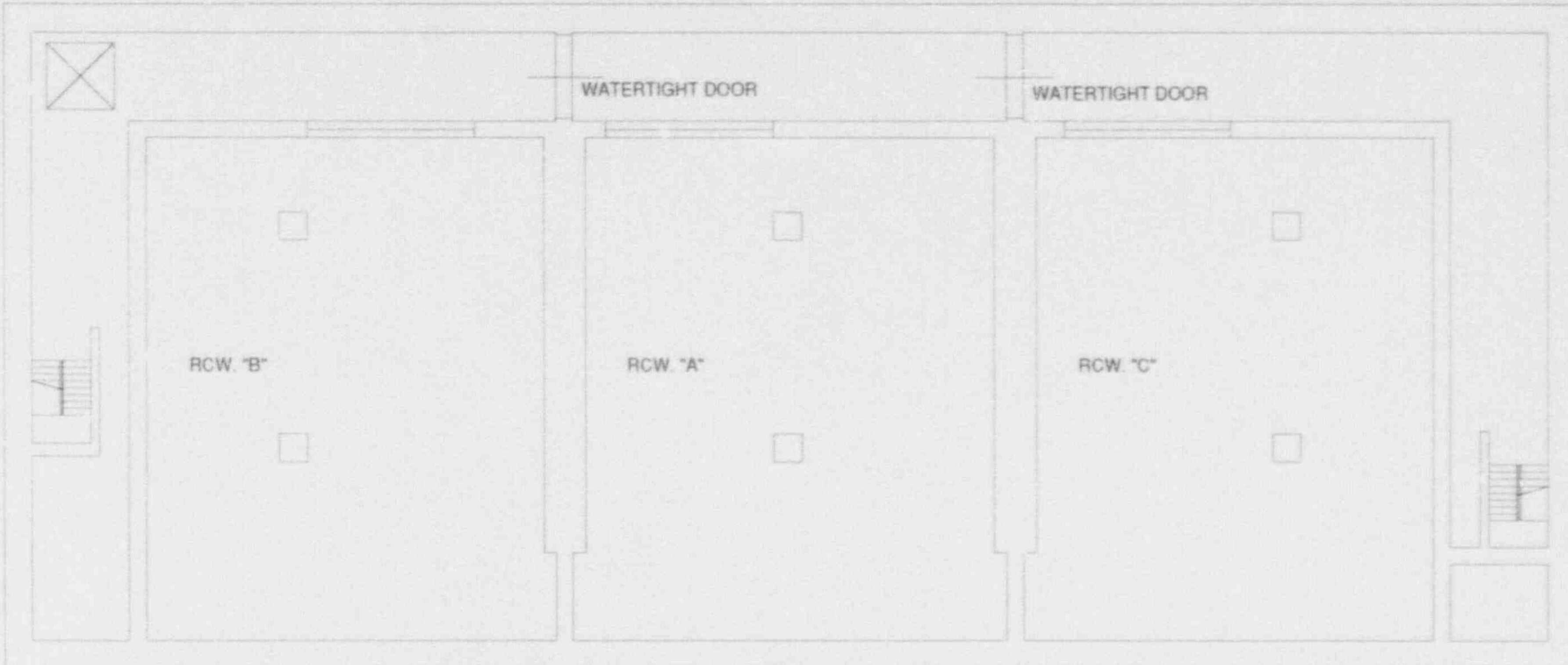


Figure 2.15.12g CONTROL BUILDING FLOOR B4F



3.0 EXAMPLES OF GENERIC AND DAC ITAAC

This section provides examples of ITAAC related material that is not within the scope of the system-by-system approach used to develop the pilot examples in Section 2.

3.1 ENVIRONMENTAL QUALIFICATION (EQ) GENERIC ITAAC

This section contains a proposed ITAAC covering environmental qualification (EQ) of safety equipment. It is based on a programmatic approach to EQ and is representative of the scope and content anticipated for other generic ITAAC entries covering similar technical issues.

Table 3.1: CONSTRUCTION-RELATED ITAAC – ENVIRONMENTAL QUALIFICATION EXAMPLE

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Mechanical and electrical equipment important to safety will be qualified for the environmental conditions that will exist up to and including the time the equipment has finished performing its safety-related function. Conditions that exist during normal, abnormal and design basis accident events will be considered in terms of their cumulative effect on equipment performance. These conditions will be considered for the time period up to the end of component refurbishment interval or end of equipment life. These conditions include number and/or duration of equipment functional and test cycles/ events; process fluid conditions (where applicable); the voltage, frequency, load, and other electrical characteristics of the equipment; the dynamic loads associated with seismic and other vibration inducing events, and the pressure, temperature, humidity, chemical and radiation environments, aging, and submergence (if any) that can affect or degrade equipment performance.	Documentation relating to EQ issues will be audited for selected equipment items important to safety. This documentation will be in the form of the equipment qualification list and the device specific qualification files. The audit will consist of a statistically valid sample of the qualification files. Timing of the audits will be phased so as to include a representative sample of all qualification activities. Sample selection will be biased towards complex equipment which utilizes materials potentially susceptible to degradation and is located in areas subject to harsh environmental conditions. In the event noncompliances with the Acceptance Criteria are discovered, the sampling process will be expanded to include an audit of the attribute[s] in question for an equal or greater sample of items. The audit will include review of specified environmental conditions, qualification methods (e.g., analyses or testing), and documentation of qualification results.	It will be confirmed that a comprehensive list of equipment important to safety has been prepared. The following information for this equipment shall be provided in a qualification file and subject to audit: a. The performance specifications under conditions existing during and following design basis accidents. For electrical items this will include the voltage, frequency, load and other electrical characteristics for which the performance specified above can be ensured. b. The environmental conditions, including temperature, pressure, humidity, radiation, chemicals and submergence at the location where the equipment must perform as specified above. This will include environmental conditions defined in 10 CFR 50.49, for electrical items and shall include consideration of synergistic effects and margins for unquantified uncertainty. c. The testing method used to qualify the equipment. Each item of equipment important to safety must be qualified by one of the following methods: 1. Testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

Table 3.1: CONSTRUCTION-RELATED ITAAC – ENVIRONMENTAL QUALIFICATION EXAMPLE

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		(Continued) 2. Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable. 3. Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable. 4. Analysis in combination with partial type test data that supports the analytical assumptions and conclusions. e. The results of the qualification have been documented to permit verification that the item of equipment important to safety: 1. Is qualified for its application; and 2. Meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life.

3.2 RADIATION PROTECTION DAC ITAAC

This section presents a proposed set of ITAAC entries addressing the issue of plant design features required for protection of workers against radiation. Numerical values of source terms are dependent upon as-built/as-provided equipment. Consequently, design details and final dose evaluations cannot be performed at the time of design certification and included in the SSAR. Closure of this issue requires:

a. Incorporating in the SSAR material which describes the basic plant features and the methods which will be used to evaluate occupational exposures when equipment details are available, and

b. establishing design acceptance criteria (DAC) ITAAC which will be used when the processes described in the SSAR are properly executed.

Tables 3.2a and 3.2b are the proposed DAC entries covering plant radiation protection. Table 3.2c is provided for information and summarizes the material that will be included in the SSAR to support resolution of this issue.

Table 3.2a: PLANT SHIELDING DESIGN

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The plant design shall provide adequate shielding to insure low radiation levels in areas normally occupied.	1. Analyses shall be performed to identify the areas of the plant normally occupied and to identify the radiation levels in each area. The analyses shall be based upon the following: a. For each area , significant sources of radiation (greater than an estimated 5% contribution) near the area will be identified and categorized by reference to the source terms found in SSAR subsection 12.2. b. For non-complex geometries, point kernel shielding codes such as QAD using nuclear properties derived from well known sources such as Vitamin C, ANSI/ANS-6.4 shall be used to model and evaluate area radiation environments. c. For complex geometries, more sophisticated two or three dimensional transport codes such at DORT or TORT will be used to model the radiation environments. d. In any calculation, a safety factor shall be applied based upon benchmark comparisons of the code and data as applied to known and measured environments.	1. The design shall identify in the SSAR all areas of the plant deemed normally occupied. The evaluation supplied will show radiation conditions to be well within (defined as 25%) of the limits as given in SSAR Subsection 12.3 for the area.

Table 3.2a: PLANT SHIELDING DESIGN (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	1. (Continued) e. The as built shield thickness shall be checked to insure that wall thickness are at least as large as the assumed analysis. The computer codes referenced above are representative only and do not designate required analytic tools.	
2. The plant design shall provide for shielded cubicles, labyrinth access, and space for temporary shielding to permit shielding of high radiation areas so that adequate maintenance space can be provided without any significant radiation exposure from adjacent equipment.	2. Using the methods identified in (1) above, analyses will be performed for areas identified as high radiation areas in the SSAR to determine the shielding for areas of low radiation.	2. The design shall be shown to reduce the radiation field in the high area to a small fraction (10% or less) of the radiation field in any adjacent low radiation areas or less than 0.06mr/hr which ever is larger.
3. The plant design shall permit access and egress to vital areas of the plant under accident conditions.	3. Radiation levels in areas with vital equipment as stipulated in SSAR section 12.3 shall be analyzed using standard design basis source terms such as given in TID 14844 and the shielding identified in (1) and (2) above.	3. The design shall provide for a radiation level of < 15mrem/hr (averaged over 30 days) for areas requiring continuous occupancy and a dose of ≤ 5 rem whole body (or its equivalent to any part of the body) for areas requiring infrequent access.

Table 3.2b: VENTILATION AND AIRBORNE MONITORING

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The plant design shall insure adequate ventilation to (a) control the flow of air from designated low radiation areas to high radiation areas, and (b) maintain the concentration of airborne radionuclide species in normally occupied areas to low concentrations.	1. For each area designated as normally occupied in SSAR section 12.3, an equilibrium air concentration calculation similar to that shown in Appendix A to SSAR section 12.3 will be made.	1. The overall plant design for air ventilation for normal and maintenance conditions will insure that all air flow is directed from areas of low radiation to areas of higher radiation. For low radiation designated areas, the radioisotopic air concentration will be found to be a small fraction (10% or less) of the air concentrations stipulated in 10CFR20.
2. For those areas of the plant designated as not normally occupied, the ventilation system shall be designed to reduce the airborne contamination to levels reasonable for maintenance activities or maintenance personnel shall be required to use protective breathing apparatus.	2. For those areas designated as not normally occupied, an evaluation will be made for airborne contamination conditions representative of maintenance activities.	2. For areas not normally occupied, the evaluation will designate the area as meeting the radioisotopic air concentration limits for use with out breathing apparatus if the evaluated air concentration is a small fraction (10% or less) of 10CFR20 criteria, or requiring protective air breathing equipment.
3. The plant design shall incorporate area radiation monitors in those areas of the plant designated as normally occupied and in which there exists a significant probability for airborne contamination.	3. Areas designated as normally occupied will be evaluated to determine if there is a significant probability for airborne contamination.	3. Area radiation monitors shall be provided if: a. Area contains a mobile source of radiation such as reactor coolant which can become rapidly airborne. b. An assessment of the equipment in the area shows a high probability (>0.1/ year) of equipment leakage which would lead to elevated airborne contamination levels Radiation monitors will have the capability to detect concentrations equivalent to 10CFR20 limits for the most restrictive particulate and/or iodine radionuclide in area designated within 10 hours.

Table 3.2c
Summary of SSAR Material Supporting Plant Radiation Protection Commitment

The following items will be added to the SSAR to support resolution of the plant radiation protection issue.

1. A table will be added to Subsection 12.3 breaking the following plant buildings down into either room or area designations.

   Reactor Building
   Control Building
   Turbine Building
   Radwaste Building

For each area, the table will specify the following:

   Location - coordinate or fire control designation
   Normal Operations designation (occupied or not)
   Normal Operations radiation designation - mr/hr
   Maintenance operations designation
      occupied
      non-occupied
      controlled - may be occupied under limited circumstance
   Maintenance operations radiation designation - mr/hr
   Accident conditions
      required access
      high radiation - referencing accident
      low radiation

2. Appendix A to Subsection 12.3

      Sample calculation for airborne contamination calculation

3. Expanded table in section 12.3 on shielding computer codes

      code
      geometry
      typical data sources

3.3 CONFIGURATION MANAGEMENT PLAN DAC ITAAC

This section contains the proposed Configuration Management Plan DAC ITAAC. The DAC ITAAC is included as a section (APPENDIX B) of the generic software ITAAC. The generic ITAAC establishes acceptance criteria for the overall software development process which includes a Software Management Plan, Configuration Management Plan and Verification and Validation (V&V) Plan. Each ABWR safety system that uses the safety-related software functions of the Safety System Logic and Control (SSLC) equipment will reference the generic software ITAAC as part of that safety system's ITAAC acceptance criteria. The ITAAC of other safety-related equipment that contains software to perform safety functions will also reference the generic software ITAAC.

The generic software ITAAC will reference the DAC ITAAC for Software Management, Configuration Management and V&V, which in turn establish design acceptance criteria that will ensure that proper controls are placed on the step-by-step software development process.

APPENDIX B is an example of a software development DAC ITAAC for the Configuration Management Plan. APPENDIX A for the Software Management Plan DAC ITAAC and APPENDIX C for the V&V DAC ITAAC will be developed later.

Table 3.3: SOFTWARE FOR PROGRAMMABLE DIGITAL COMPUTERS IN SAFETY-RELATED APPLICATIONS

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. A plan shall be developed for software used in microprocessor-based equipment that performs safety-related functions. The plan shall describe the organizational and procedural aspects of software development and shall comprise the following elements: - Software Management Plan - Configuration Management Plan - Verification and Validation (V&V) plan	1. Review: - Software Management Plan - Configuration Management Plan - Verification and Validation Plan	1. The overall development plan documents the requirements and methodology for achieving the software attributes of consistency, accuracy, error tolerance and modularity. The plan includes the methodology for assuring the software is both auditable and testable during the design, implementation and integration phases. Each element of the plan contains the following items as a minimum: a. Software Management Plan establishes standards, conventions and design processes for the design, development, and maintenance of safety-related software. The plan meets the design acceptance criteria described in Appendix A. b. Configuration Management Plan establishes a formal set of standards and procedures to provide visible status and control of software documentation. The following basic elements are addressed: 1) Unique identification of each software documentation item 2) Management of software documentation change control 3) Accounting methods to provide visibility and traceability for all changes to baseline product software 4) Verification steps required to assure proper adherence to software design requirements

Table 3.3: SOFTWARE FOR PROGRAMMABLE DIGITAL COMPUTERS IN SAFETY-RELATED APPLICATIONS (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		1. (Continued) The plan meets the design acceptance criteria described in Appendix B. c. Verification and Validation Plan establishes verification reviews and validation testing procedures with the following components: 1) Independent design verification 2) Baseline reviews 3) Testing 4) Firmware issue and validation procedure a) Unstructured testing b) Formal validation testing 5) Procedure for future revisions The plan meets the design acceptance criteria described in Appendix C.
2. The software design documentation meets the requirements of the development plan.	2. Review design documentation: - Hardware/Software System Specification - Software Requirements Specification - Software Design Specification - Hardware Requirements Specification - Hardware Design Specification	2. The documentation complies with the requirements in the Certified Design Commitments. The design documentation allows correlation of the design elements with each specific software requirement as determined by the V&V process described in Appendix C. The computer system hardware documentation identifies the hardware requirements that impact software.
3. Details of software implementation and the integration of hardware and software into the final product shall be addressed in Tier 2.	3. Tier 2 requirement	3. Tier 2 requirement

Table 3.3: SOFTWARE FOR PROGRAMMABLE DIGITAL COMPUTERS IN SAFETY-RELATED APPLICATIONS (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The assembled, final production computer system shall be exercised through static and dynamic simulations of input signals present during normal operation and design basis event conditions requiring computer system action. The validation test plan shall identify the validation tests for each safety-related, software-based system component.	4. Review formal (verified) validation test report.	4. The test report summarizes the results of the computer system validation testing and shows how the system is in compliance with the requirements. The test report identifies the validation tests for each computer system and safety system requirement. In addition, the required input signals and their values, the anticipated output signals, and the acceptance criteria are stated. The test report identifies the hardware and software used, test equipment and calibrations, simulation models used, test results, and discrepancies and corrective actions. The test plan was developed, the tests executed, and the test results evaluated by individuals who did not participate in the design or implementation phases.

Table 3.3: SOFTWARE FOR PROGRAMMABLE DIGITAL COMPUTERS IN SAFETY-RELATED APPLICATIONS

APPENDIX A: SOFTWARE MANAGEMENT PLAN DESIGN ACCEPTANCE CRITERIA

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	[LATER]	

Table 3.3: SOFTWARE FOR PROGRAMMABLE DIGITAL COMPUTERS IN SAFETY-RELATED APPLICATIONS

APPENDIX B: CONFIGURATION MANAGEMENT PLAN DESIGN ACCEPTANCE CRITERIA

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Development of software for the microprocessor-based safety systems shall be controlled according to a configuration management plan.	1. A review shall be performed of the contents of the configuration management plan.	1. A configuration management plan has been issued.
2. The configuration management plan will define the purpose and scope of the plan with emphasis on the groups to which it applies and the specific product which is to be developed. The product description shall include both executable and non-executable material	2. A review shall be performed of the contents of the configuration management plan.	2. The configuration management plan identifies each group which develops and/ or maintains software for safety systems. The plan includes both executable and non-executable portions of the design.
3. The configuration plan shall describe the organizational responsibilities. The organizational independence or dependence of the groups responsible for the software configuration management shall be specifically described. The plan shall describe a function independent of the software designers that is responsible for verifying that the software is maintained under this plan. The plan shall detail the relationships of the configuration control with the software QA, development and other groups.	3. A review shall be performed of the contents of the configuration management plan.	3. The configuration plan describes the organizational independence and responsibilities.

Table 3.3: SOFTWARE FOR PROGRAMMABLE DIGITAL COMPUTERS IN SAFETY-RELATED APPLICATIONS

APPENDIX B: CONFIGURATION MANAGEMENT PLAN DESIGN ACCEPTANCE CRITERIA (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Applicable procedures, such as standards for the designation of software versions, shall be described in the plan or specifically referenced. All software shall be identified such that the version can be verified directly, either embedded in the software if in a non-programmable/erasable format or permanently inscribed directly on the component.	4. A review shall be performed of the contents of the configuration management plan.	4. The plan describes the procedures for implementation of the plan.
5. The plan shall describe the audits and reviews that are to be performed to verify that the software is being maintained under configuration management. The plan shall describe a procedure for corrective actions if any problems are discovered.	5. A review shall be performed of the contents of the configuration management plan.	5. The plan describes audits and reviews and describes a procedure for corrective actions.
6. The configuration management of tools, techniques, and methodologies shall be specifically delineated. The plan shall address control of development methods to used (such as formal specification) and tools (such as compilers).	6. A review shall be performed of the contents of the configuration management plan.	6. The plan describes control of tools and methodologies.
7. The plan shall describe the method of records collection and retention.	7. A review shall be performed of the contents of the configuration management plan.	7. The plan describes the record storage plan.

Table 3.3: SOFTWARE FOR PROGRAMMABLE DIGITAL COMPUTERS IN SAFETY-RELATED APPLICATIONS

APPENDIX B: CONFIGURATION MANAGEMENT PLAN DESIGN ACCEPTANCE CRITERIA (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. The plan shall address control of the final user documentation and the information to be supplied. The method of informing the user of each product of known faults, failures, and changes shall be specifically described.	8. A review shall be performed of the contents of the configuration management plan.	8. The plan will identify the method by which faults, failures, and changes are identified to the affected user.
9. The configuration management plan shall be in place and approved by the implementer prior to the first concept development phases of software development.	9. A review shall be performed of the contents of the configuration management plan.	9. The configuration management plan will be approved and in place at the beginning of the project.
10. The configuration management plan shall require that the design documents (such as software requirements specifications) shall provide specific reference to the applicable configuration management plan. The plan shall define procedures for change control, including change request, evaluation, approval, and implementation.	10. A review shall be performed of the contents of the configuration management plan.	10. The plan will require that the design documents reference the configuration management plan.

Table 3.3: SOFTWARE FOR PROGRAMMABLE DIGITAL COMPUTERS IN SAFETY-RELATED APPLICATIONS

APPENDIX C: VERIFICATION AND VALIDATION PLAN DESIGN ACCEPTANCE CRITERIA

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	[LATER]	

3.4 MAN-MACHINE INTERFACE SYSTEMS (MMIS) DESIGN TEAM DAC ITAAC

This section presents a sample ITAAC entry addressing the issue of how Human Factors considerations are addressed in the implementation of ABWR design areas of human-system interface. Human Factors evaluation results are dependant upon as-built/as-provided equipment and upon details of plant operation (e.g. detailed operating procedures and personnel training) that must be developed by each individual COL applicant. Consequently, design details and final Human Factors evaluation cannot be performed at the time of design certification and included in the SSAR. Closure of this issue requires establishing design acceptance criteria (DAC) ITAAC which will be used to ensure that the implemented designs of human-system interface areas are consistent with accepted human factors engineering principles. In their final form, DAC ITAACS will be developed to address the full scope of human factors related design implementation activities. Table 3.4a is a proposed DAC ITAAC which has been developed, as an example, for the first of the series of design conformation reviews which will be done on the human-system interface design implementation activities. The DAC ITAAC presented in Table 3.4a addresses the formation of the Man-Machine Interface System (MMIS) Design Team and the procedures the MMIS Design Team will develop to direct the subsequent human factors related design implementation activities.

Table 3.4: MAN-MACHINE INTERFACE SYSTEMS (MMIS) DESIGN TEAM DESIGN ACCEPTANCE CRITERIA

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Implementation of the Man-Machine Interface Systems (i.e., control and instrumentation equipment, main control room and local control facilities) shall be directed by a dedicated MMIS Design Team. The MMIS Design Team shall have the charge of assuring that MMIS design implementations support plant personnel in the safe operation and maintenance of the plant.	1. The MMIS Design Team organizational structure, functional responsibilities, levels of authority and lines of communication will be reviewed.	The MMIS Design Team shall have authority, access to work areas and design documentation, and organizational freedom to identify problems in the implementation of the MMIS designs; initiate, recommend or provide solutions to such problems through designated channels; verify implementation of solutions; and assure that further processing, delivery, installation or use is controlled until proper disposition of a non-conformance, deficiency or unsatisfactory condition has occurred.
2. The MMIS Design Team shall be composed of multidisciplinary technical staff with technical expertise and experience in the technologies and techniques to be utilized in the MMIS designs.	2. Inspection of MMIS Design Team composition.	2. The technical discipline and experience of the MMIS Design Team shall include, as a minimum, the following: - nuclear engineering - instrumentation and control engineering - human factors engineering - reliability engineering - maintainability engineering - systems engineering - architect engineering - computer systems engineering - training development - nuclear power plant operation - safety engineering - plant procedure development

Table 3.4: MAN-MACHINE INTERFACE SYSTEMS (MMIS) DESIGN TEAM DESIGN ACCEPTANCE CRITERIA

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests Analyses	Acceptance Criteria
3. The MMIS Design Team shall establish detailed plans and procedures directing the implementation of MMIS designs.	3. Inspection of MMIS Design Team plans and procedures.	3. Plans and procedures shall address: - Development of MMIS Program Plan - Development of System Functional Requirements - Allocation of functions and Conduct of Task Analysis - Evaluation of Human Factors and Human-System Interfaces - Development of Plant Procedures - Development of Plant-Specific Emergency Operating Procedures - Development of Training Requirements - Implementation of Software Verification and Validation