GE Nuclear Energy

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Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Richard W. Borchardt, Acting Director
Standardization Project Directorate

Subject: **Submittal Supporting Accelerated ABWR Design Certification Review - Submittal of Second Phase of Revised ABWR Tier 1/ITAAC Material**

Dear Mr. Borchardt:

Enclosed are thirty-four (34) revised versions of selected ABWR Tier 1/ITAAC material for thirty (30) ABWR systems and the proposed Tier 1 entries for the Design Reliability Assurance Program and the Initial Test Program. Also enclosed are proposed Road Maps for eight (8) of the eleven (11) subjects that GE and NRC staff have agreed should receive road-map treatment. The attached table lists the items covered by this submittal. This material represents the second of four ABWR Tier 1/ITAAC submittals we have scheduled. Our letter of April 26, 1993, provided material for 23 systems; transmittals scheduled for June 4, 1993, and June 18, 1993, will complete the submittal of the entire ABWR Tier 1 document.

The enclosed material does not include revised figures for Section 2.15.12 Control Building. The Control Building documents are still being revised to reflect recent design changes agreed to by GE and the staff. Revised Control Building drawings will be submitted as soon as they are available (within the next two weeks). Entries 2.8.1, 2.8.2, 2.8.3 (Nuclear Fuel, Fuel Channel, Control Rod) were scheduled for this submittal but are not included because the Tier 1 scope and content of these aspects of the ABWR design are currently the subject of intensive GE/NRC interactions and are not yet agreed to.

The updated Tier 1 material in this and the other scheduled submittals is being prepared in parallel with closure of open items on the SSAR and may not reflect very recent GE/NRC agreements on SSAR open issues. For example, some of the systems in either this transmittal or the April 26, 1993, transmittal may not fully reflect the outcome of GE/NRC discussions on probabilistic risk assessment and severe accident insights. However, we do not view this as a major problem and plan to update the Tier 1 systems material as and when SAR changes are implemented. We anticipate this will result in some revisions being sent to you in the June time-frame. We do not anticipate any major perturbations and do not believe this change process should impede your review of the ABWR Tier 1 document.

As always, GE personnel will be happy to provide any support the NRC staff review teams feel they need to complete their review of this material.

Sincerely,

R. C. Mitchell

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AJJ5-19:c

ABWR DESIGN CERTIFICATION

ABWR Systems Addressed in the 5/21/93 Submittal to NRC

2.1.1	Reactor Pressure Vessel
2.1.3	Reactor Recirculation System
2.2.2	Control Rod Drive System
2.2.5	Neutron Monitoring System
2.3.2	Area Radiation Monitoring System
2.3.3	Containment Atmospheric Monitoring System
2.4.4	Reactor Core Isolation Cooling System
2.5.5	Refueling Equipment
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* Underlined sections –Title only, no entry for design certification.

** Section number in parentheses – Section, under which the subject is covered.

2.1 Nuclear Steam Supply

2.1.1 Reactor Pressure Vessel System

Design Description

The Reactor Pressure Vessel (RPV) System consists of (1) the RPV and its appurtenances, supports and insulation, excluding the loose parts monitoring system, and (2) the reactor internal components enclosed by the vessel, excluding the core (fuel assemblies, control rods, in-core nuclear instrumentation and neutron sources), reactor internal pumps (RIPs), and control rod drives (CRDs). The RPV System is located in the primary containment.

The reactor coolant pressure boundary (RCPB) portion of the RPV and its appurtenances (referred to in this section as the RPV pressure boundary) act as a radioactive material barrier during plant operation.

Certain reactor internals support the core, flood the core during a loss-of-coolant accident (LOCA) and support safety-related instrumentation. Other RPV internals direct coolant flow, separate steam, hold material surveillance specimens, and support instrumentation utilized for plant operation.

The RPV System provides guidance and support for the CRDs. It also distributes sodium pentaborate solution when injected from the Standby Liquid Control (SLC) System.

The RPV System restrains the CRD to prevent ejection of the control rod connected with the CRD in the event of a failure of the RCPB associated with the CRD housing weld. A restraint system is also provided for each RIP in order to prevent the RIP from becoming a missile in the event of a failure of the RCPB associated with the RIP casing weld.

The RPV System is shown on Figures 2.1.1a, 2.1.1b and 2.1.1c; key dimensions and the acceptable variations in these dimensions are presented in Table 2.1.1a. The RPV System parameters (break areas) used in LOCA analyses are identified in Table 2.1.1b. The principal design parameters for the RPV System are in Table 2.1.1c.

Reactor Pressure Vessel, Appurtenances, Supports and Insulation

The RPV, as shown in Figures 2.1.1a and 2.1.1b, is a vertical, cylindrical vessel of welded construction with removable top head and head closure bolting and seals. The main body of the installed RPV has a cylindrical shell, flange, bottom head, RIP casings, penetrations (including inserted housings), brackets, nozzles, and the shroud support, which has a pump deck forming the partition between the RIP suction and discharge. The shroud support is an assembly consisting of

a short vertical cylindrical shell, a horizontal annular pump deck plate and vertical shroud support legs.

The CRD housings are inserted through and welded to the CRD penetrations in the reactor vessel bottom head. The CRDs are mounted into the CRD housings. The in-core housings are inserted through and connected to the bottom head.

For an RPV System that requires to be instrumented for flow-induced vibration (FIV) testing, a flanged nozzle is provided in the top head for bolting of the flange associated with the test instrumentation.

The integral reactor vessel skirt supports the vessel on the Reactor Pressure Vessel Pedestal. Anchor bolts extend from the pedestal through the flange of the skirt. RPV stabilizers are provided in the upper portion of the RPV to resist horizontal loads. Lateral supports for the CRD housings and in-core housings are provided.

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

The RPV insulation is supported from the reactor shield wall surrounding the vessel. Insulation for the upper head and flange is supported by a steel frame independent of the vessel and piping.

The RPV pressure boundary and the supports (RPV skirt, stabilizer and CRD housing/in-core housing lateral supports) are classified as Seismic Category I. These components are ASME Code Class 1 vessel and supports, respectively. The shroud support and a portion of the CRD housings inside the RPV are classified as Seismic Category I and ASME Code Class CS structures.

The following ASME materials (or their equivalents) are used in the RPV pressure boundary: 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%, forging) or F316 (maximum carbon 0.020% and nitrogen from 0.060 to 0.120%, forging); and SA-540, Grade B23 or B24 (bolting).

A stainless steel weld overlay is applied to the interior of the RPV cylindrical shell and the steam outlet nozzles. Other nozzles and the RIP motor casings 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, alternatively, stainless steel.

The materials of the low alloy plates and forging used in construction of the RPV pressure boundary are melted using vacuum degassing to fine grain practice and are supplied in quenched and tempered condition.

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

The RPV 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 Code Section XI, Appendix I. Acceptance standards also meet the requirements of ASME Code Section XI.

The fracture toughness tests of the RPV pressure boundary ferritic materials, weld metal and heat-affected zone (HAZ) are performed in accordance with the requirements for ASME Code Section III, Class 1 vessel. Both longitudinal and transverse specimens are used to determine the minimum upper-shelf energy (USE) level of the core beltline materials. The minimum USE level for base and weld metal in the core beltline is initially at least 10.4 kg-m. Separate, unirradiated baseline specimens are used to determine the transition temperature curve of the core beltline base materials, weld metal, and HAZ.

For the reactor pressure vessel material surveillance program, specimens are manufactured from the same material used in the reactor beltline region and from the sample welds made to represent the welds in the beltline region, thus representing base metal, weld material, and the weld HAZ material. In-reactor surveillance capsules contain Charpy V-notch specimens of base metal, weld metal, and HAZ material, and tensile specimens from base metal and weld metal. Brackets are welded to the vessel cladding in the core beltline region for retention of the detachable surveillance specimen holders, each of which contains a number of the specimen capsules. Neutron dosimeters and temperature monitors are located within the capsules.

Reactor Pressure Vessel Internals

The major reactor internal components in the RPV System are:

(1) Core Support Structures:

Shroud, shroud support and a portion of CRD housings inside the RPV (both integral to the RPV), core plate, top guide, fuel supports (orificed

fuel supports and peripheral fuel supports), and control rod guide tubes (CRGTs). The core support structures are classified as Seismic Category I and ASME Code Class CS structures.

(2) Other Reactor Internals:

(a) Feedwater spargers, shutdown cooling (SDC) and low pressure flooding (LPFL) spargers, high pressure core flooder (HPCF) spargers and couplings, and a portion of the in-core housings inside the RPV and in-core guide tubes (ICGTs) with stabilizers. These components are classified as Seismic Category I, and safety-related.

(b) Surveillance specimen holders, shroud head and steam separators assembly and the steam dryer assembly and the steam dryer assembly. These components are classified as non-safety-related.

A general assembly of these reactor internal components is shown in Figures 2.1.1a, 2.1.1b, and 2.1.1c.

The shroud support, shroud, and top guide make up a 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 plate with round openings and is stiffened with a rim and beam structure. The core plate provides lateral support and guidance for the CRGTs, ICGTs, 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 assemblies and 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, where the sides of the openings intersect, to anchor the in-core instrumentation detectors and startup neutron sources.

The fuel supports are of two types: (1) peripheral and (2) orificed. 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 has an orifice to provide coolant flow to the fuel assembly. Each orificed fuel support holds four fuel assemblies and has four orifices to provide coolant flow distribution to each fuel assembly. The control rods pass through cruciform openings in the center of the orificed fuel supports. This locates the four fuel assemblies surrounding a control rod.

The CRGTs pass through holes in the core plate, have four holes under the core plate and rest on top of the CRD housings. Each CRGT guides the lower end of a control rod, and supports an orificed fuel support such that the orifices of the orificed fuel support align with the holes in the CRGT for coolant flow. The lower end of the CRGT is supported by the CRD housing, which, in turn, transmits the weight of the guide tube, fuel supports, and fuel assemblies to the reactor vessel bottom head.

The CRGT base is provided with a device for coupling the CRD with it. The CRD is restrained from ejection, in the case of failure of the weld between a CRD housing and CRD penetration, by the coupling of the CRD with the CRGT base; in this event, the flange at the top of the guide tube contacts the core plate and acts to restrain the ejection. The coupling will also prevent ejection if the housing fails beneath the weld; in this event, the guide tube remains supported on the intact upper housing.

There are six feedwater spargers, three for each of the two feedwater lines. Each sparger is connected to an RPV feedwater nozzle at the double thermal sleeve fitted with the safe end (straight piece) of the nozzle. Feedwater flow enters the middle of the spargers, which are located above the RPV downcomer annulus, and is discharged inward.

Two spargers are provided for two loops of the RHR System; both spargers function as SDC and LPFL spargers. Each sparger is connected to a thermal sleeve fitted with the safe end of each SDC and LPFL inlet nozzle.

Two HPCF spargers with couplings are provided for the two loops of the HPCF System to direct high pressure coolant flow to the upper end of the core during emergency core cooling. One of the HPCF spargers also distributes sodium pentaborate solution when injected from the SLC System via the connecting HPCF line. The spargers are located inside the cylindrical portion of the top guide. Each sparger is connected via an HPCF coupling to a thermal sleeve fitted with the safe end of each HPCF inlet nozzle.

The ICGTs house the in-core flux monitoring instrumentation assemblies, pass through holes in the core plate, and rest on top of the in-core housings. Two levels of stabilizer latticework give lateral support to the ICGTs. The ICGT stabilizers are connected to either the shroud or the shroud support.

In-reactor surveillance specimen capsules are held in detachable surveillance specimen holders which are retained in the core beltline region by brackets.

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.

Cobalt-base material is only used for hard surfacing of areas in HPCF coupling and CRGT base. The wrought austenitic stainless steel used for the RPV internals is limited to a maximum of 0.02% carbon content. Cold-worked stainless steel is not used in the RPV internals except for vanes in the steam dryers. Stainless steel materials are supplied in solution heat-treated condition. Furnace sensitized (heated to a temperature above 427°C) stainless steel material is not used. Electroslag welding is not applied for structural welds of stainless steel.

The RPV internals are designed to withstand the effects of FIV.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.1.1d provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Reactor Pressure Vessel System.

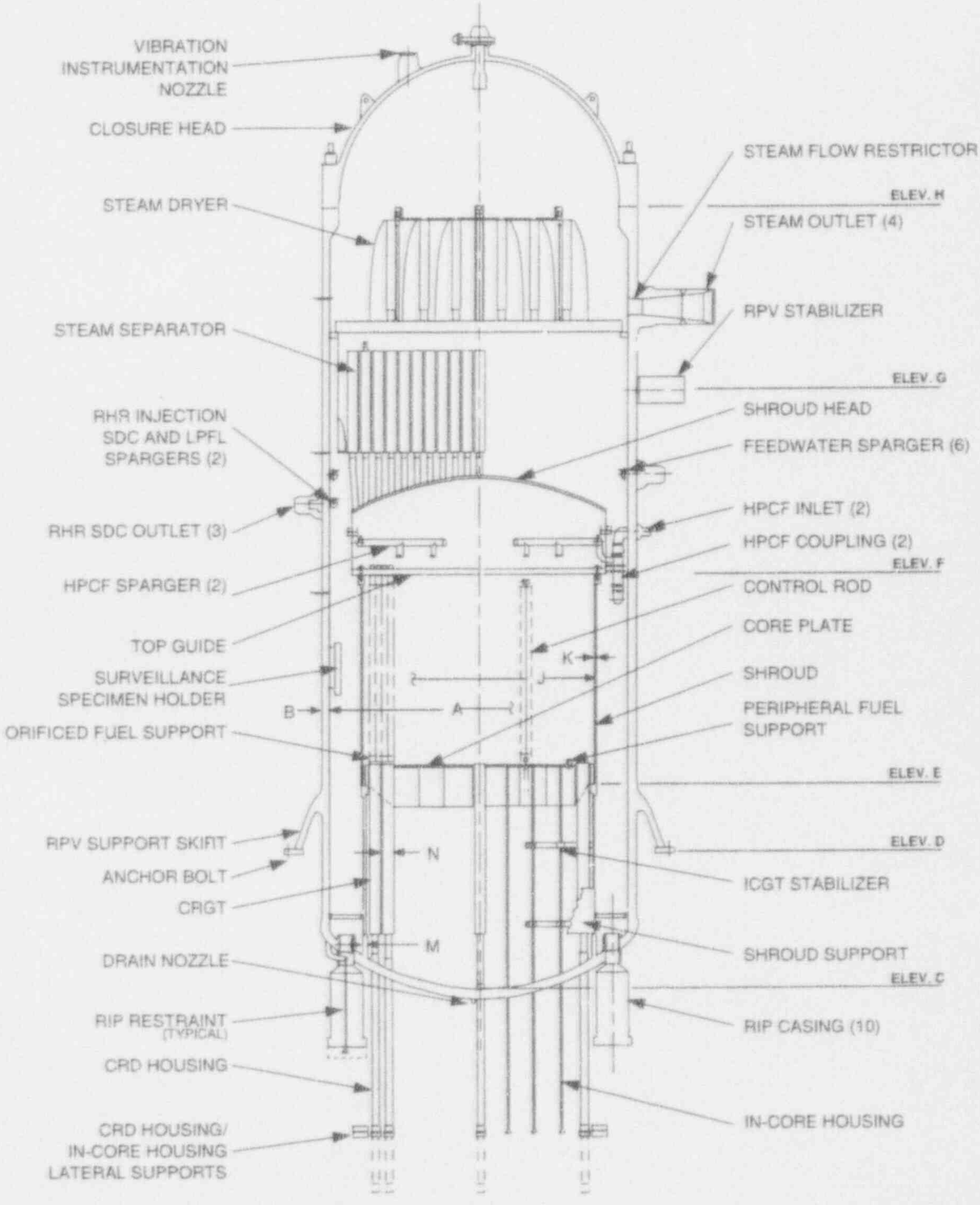


Figure 2.1.1a Reactor Pressure Vessel System Key Features

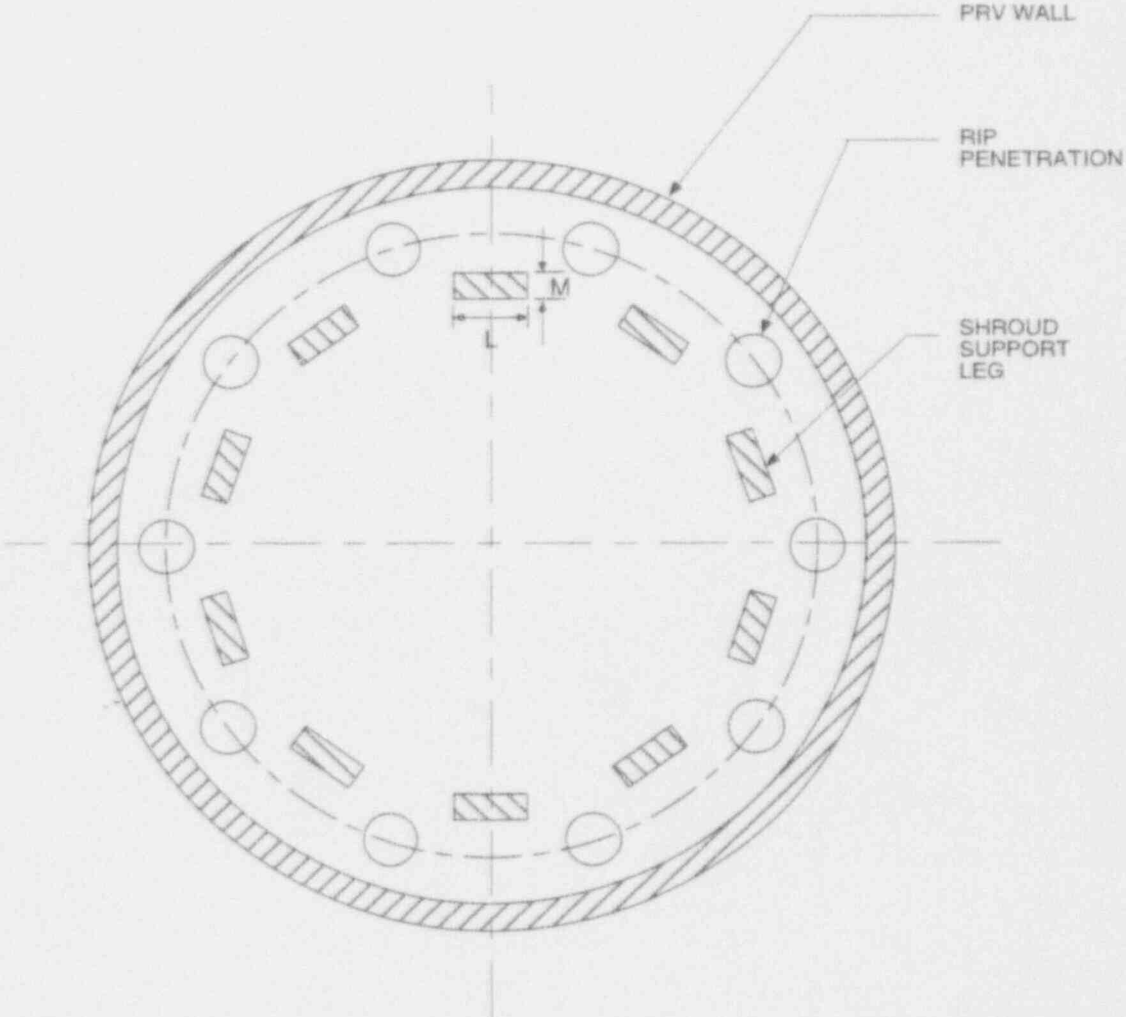


Figure 2.1.1b Pump Penetration and Shroud Support Leg Arrangement

NOTES:

1. THE ARRANGEMENT IS SHOWN FOR QUARTER CORE ONLY. ROTATIONAL SYMMETRY APPLIES. THE REACTOR INTERNALS ACCOMMODATE THE SHOWN CORE ARRANGEMENT; THE CORE IS NOT A PART OF THE RPV SYSTEM.

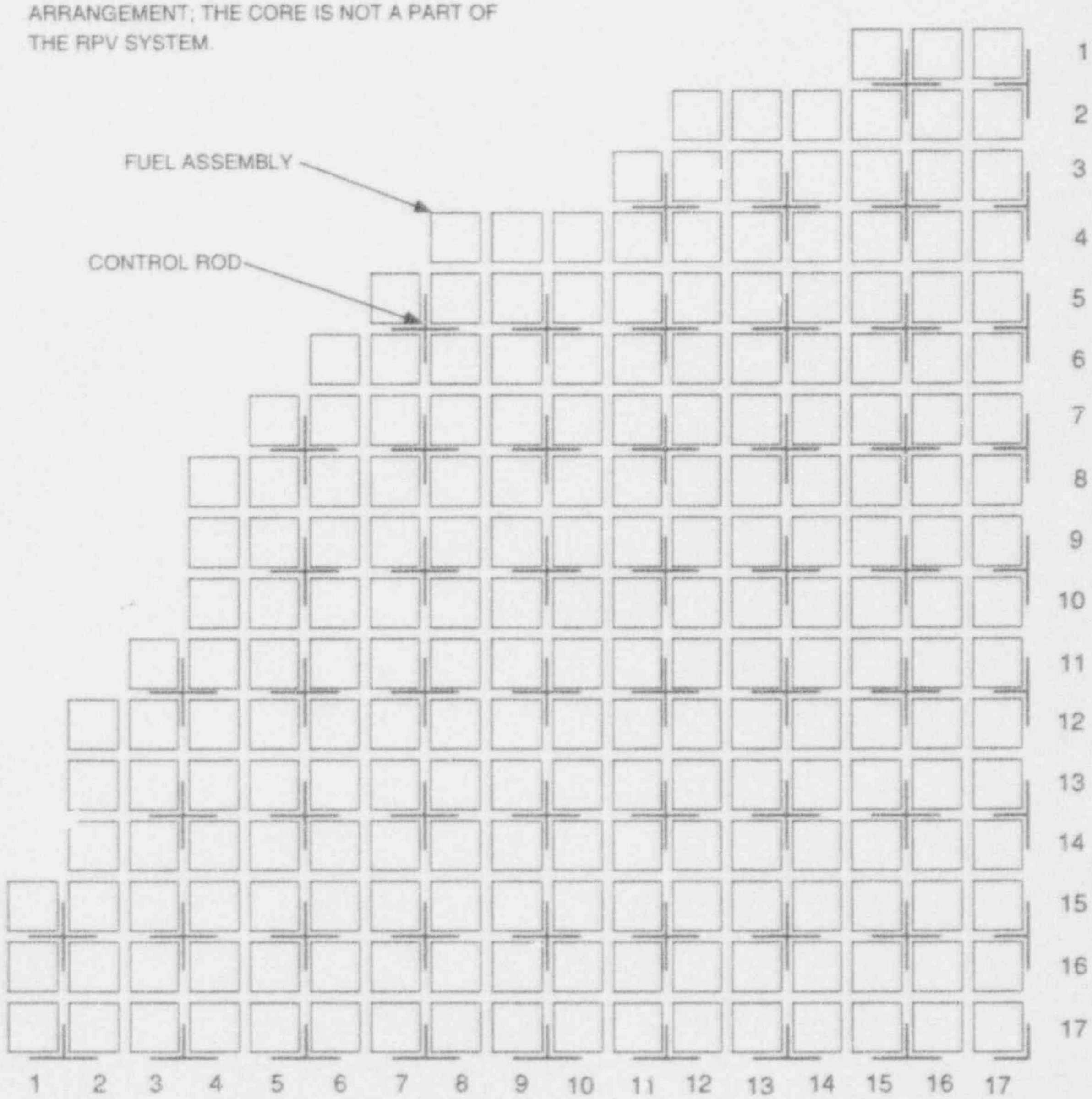


Figure 2.1.1c Core Arrangement

Table 2.1.1a Key Dimensions of RPV System Components and Acceptable Variations

Description	Dimension/ Elevation (Figure 2.1.1a)	Nominal Value (mm)	Acceptable Variation (mm)
RPV inside diameter (inside cladding)	A	7112.0	±65.0
RPV wall thickness in beltline (without cladding)	B	174.0	+20.0/−5.0
RPV bottom head inside invert, Elevation	C	0.0	Reference 0.0
RPV support skirt bottom, Elevation	D	3200.0	+65.0/−15.0
Core plate support/Top of shroud middle flange, Elevation	E	4695.2	±15.0
Top guide support/Top of shroud top flange, Elevation	F	9351.2	±20.0
RPV stabilizer connection, Elevation	G	13,766.0	±20.0
Top of RPV flange, Elevation	H	17,703.0	±65.0
Shroud outside diameter	J	5550.0	±25.0
Shroud wall thickness	K	50.8	±5.0
Shroud support legs (Fig. 2.1.1b)	LxM	662.0x153.0	±20.0 for L ±10.0 for M
Control rod guide tube outside diameter	N	273.05	±5.0

Table 2.1.1b RPV System Parameters Used in LOCA Analyses

Line	Inspection Location	Postulated Break Area, mm^2
Steamline	Flow restrictor throat diameter in a steam outlet nozzle.	98,480
Feedwater	Inside diameters of flow nozzles on the spargers of a feedwater line for the total flow area.	83,890
RHR Injection	Inside diameters of flow nozzles on an SDC and LPFL sparger for the total flow area.	20,530
High Pressure Core Flooder	Inside diameters of flow nozzles on an HPCF sparger for the total flow area.	9200
RHR Shutdown Cooling	Inside diameter of an RHR SDC outlet nozzle at the safe end weld.	79,150
Drain	Inside diameter of the bottom head hole for the drain outlet nozzle, near the inside surface of the head and below the hole chamfer.	2030

Note: The areas calculated from the inspections shall not exceed the postulated break areas by 5%.

Table 2.1.1c: Principal Design Parameters for RPV System

Description	Value
Rated core coolant flow rate (kg/hr)	52.2×10^6
RCPB design pressure (kg/cm^2g)	87.9
RCPB design temperature (°C)	302
Number of fuel assemblies	872
Number of control rods	205
Number of internal pumps	10

Table 2.1.1d Reactor Pressure Vessel System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the RPV System is as defined as Section 2.1.1.	1. Inspections of the as-built RPV System will be conducted.	1. The RPV System conforms with the basic configuration defined in Section 2.1.1.
2. The ASME Code components of the RPV System retain their pressure boundary integrity under internal pressure that will be experienced during service.	2. A hydrostatic test will be conducted on those code components of the RPV System required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME Code components of the RPV System conform with the requirements in the ASME Code, Section III.
3. The materials selection and materials testing requirements for the RPV System are as defined in Section 2.1.1.	3. Inspections of the as-built RPV System will be conducted.	3. The RPV System conforms with the materials selection and materials testing requirements defined in Section 2.1.1.
4. The fabrication process and examination process requirements for the RPV System are as defined in Section 2.1.1.	4. Inspections of the as-built RPV System will be conducted.	4. The RPV System conforms with the fabrication process and examination process requirements defined in Section 2.1.1.
5. The material surveillance commitments for the reactor pressure vessel core beltline materials are as defined in Section 2.1.1.	5. Inspections of the as-built RPV System will be conducted for implementation of the material surveillance commitments.	5. The material surveillance program for the reactor pressure vessel core beltline materials conforms with the commitments defined in Section 2.1.1.
6. The RPV internals withstand the effects of FIV.	6. A flow test and post-test inspections will be conducted on the as-built RPV internals.	6. The RPV internals have no damage or loose parts.

2.1.3 Reactor Recirculation System

Design Description

The Reactor Recirculation System (RRS) is an arrangement of 10 variable speed reactor internal pumps (RIP) with motors mounted in the bottom of the RPV. The RRS circulates coolant through the reactor core at variable flow rates. The motor cooling heat exchangers are located inside the RPV pedestal adjacent to the RIP motors. Figure 2.1.3 shows the basic system configuration and scope.

Individual RIPs and motors provide at least 6912 m^3/hr flow with a total developed head of at least 32.6 m with water at 278°C and 73.9 kg/cm^2 or less. The individual RIPs, and motors have a dry rotating inertia of not less than 19.5 kg-m^2.

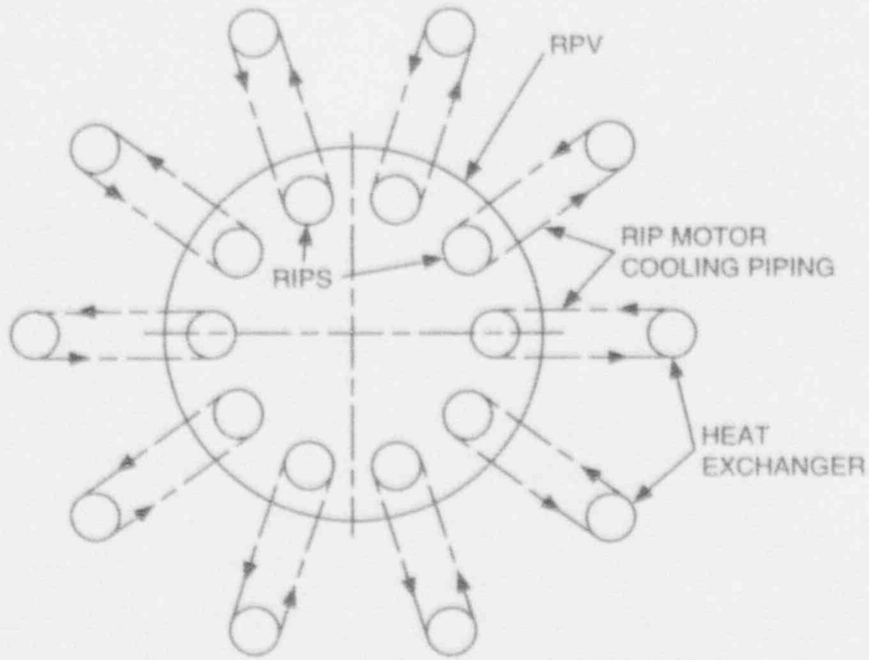
Figure 2.1.3 shows the ASME Code class for the RRS piping and components. The motor cover and its nuts and bolts are classified as safety-related, Seismic Category I, ASME Code Class 1 components. The remainder of the system is classified as non-safety-related.

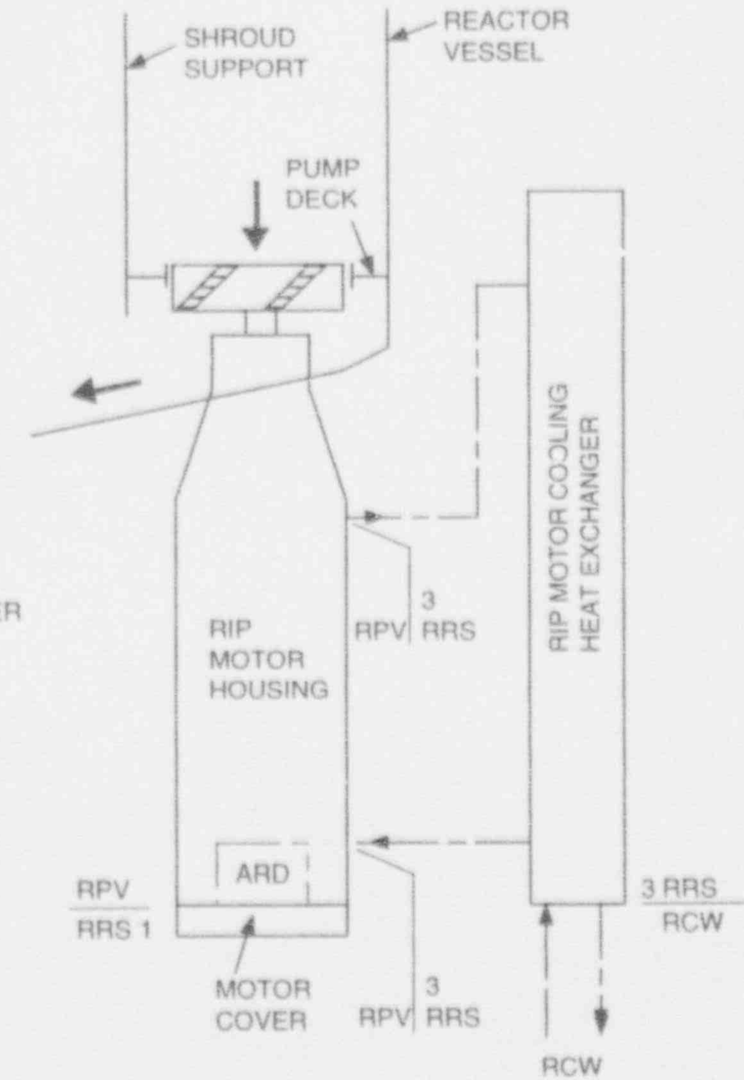
The RIP motor cooling is provided by an auxiliary impeller mounted on the bottom of the motor rotor, which circulates water through the RIP motor and it's cooling heat exchanger. The heat exchangers are cooled by the Reactor Building Cooling Water System (RCW).

Each RIP includes an anti-rotation-device (ARD) which prevents reverse RIP motor rotation by reverse flow induced torque of equal to or less than 770 kg-m when there is no motor power.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.1.3 provides a definition of the instructions, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the RRS.





NOTES:
1. THE MOTOR COVER, COVER BOLTS AND NUTS ARE ASME CODE CLASS 1 COMPONENTS.
2. RRS SCOPE IS RIP PUMP AND MOTOR, RIP MOTOR COVER, COVER BOLTS AND NUTS, RIP MOTOR COOLING HEAT EXCHANGER AND PIPING, AND ANTI-ROTATION DEVICE.

Figure 2.1.3 Reactor Recirculation System

Table 2.1.3 Reactor Recirculation System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the RRS is shown on Figure 2.1.3.	1. Inspections of the as-built system will be conducted.	1. The as-built RRS conforms with the basic configuration shown in Figure 2.1.3.
2. The ASME components of the RRS retain their pressure integrity under internal pressures that will be experienced during service.	2. A hydrostatic test will be conducted on those code components of the RRS required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME components of the RRS conform with the requirements in the ASME Code, Section III.
3. Individual RIPs and motors provide at least 6912 m^3/hr flow with a total developed head of at least 32.6 m with water at least 278°C and 73.9 kg/cm^2 or less.	3. Tests will be conducted on the individual RIP in a test facility.	3. Individual RIPs and motors provide at least 6912 m^3/hr flow with a total developed head of at least 32.6 m with water at least 278°C and 73.9 kg/cm^2 or less.
4. The individual RIPs and motors have a dry rotating inertia of $\geq$ 19.5 kg-m^2.	4. Tests will be conducted on a RIP and motor rotating assembly in a test facility.	4. RIP and motor dry rotating inertia is $\geq$ 19.5 kg-m^2.
5. Each RIP includes an ARD which prevents reverse RIP motor rotation by reverse flow induced torque of $\leq$ 770 kg-m when there is no motor power.	5. Tests will be conducted in a test facility.	5. Each ARD prevents RIP motor rotation in the reverse direction with a reverse torque of $\leq$ 770 kg-m.

2.2.2 Control Rod Drive System

Design Description

The Control Rod Drive (CRD) System controls changes in core reactivity during power operation by movement and positioning of the neutron absorbing control rods within the core in fine increments in response to control signals from the Rod Control and Information System (RCIS). The CRD System provides rapid control rod insertion (scram) in response to manual or automatic signals from the Reactor Protection System (RPS). Figure 2.2.2 shows the basic system configuration and scope.

The CRD System consists of three major elements: (1) the electro-hydraulic fine motion control rod drive (FMCRD) mechanisms, (2) the hydraulic control unit (HCU) assemblies, and (3) the control rod drive hydraulic system (CRDHS). The FMCRDs provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion (scram) for abnormal operating conditions. Simultaneous with scram, the FMCRDs also provide electric-motor driven run-in of control rods as a path to rod insertion that is diverse from the hydraulic-powered scram. The hydraulic power required for scram is provided by high pressure water stored in the individual HCUs. An HCU can scram two FMCRDs. It also provides the flow path for purge water to the associated drives during normal operation. The CRDHS supplies pressurized water for charging the HCU scram accumulators and purging to the FMCRDs.

There are 205 FMCRDs mounted in housings welded into the reactor vessel bottom head. The FMCRD has a movable hollow piston tube that is coupled at its upper end, inside the reactor vessel, to the bottom of a control rod. The FMCRD can move the control rod up or down over its entire range, by a ball nut and ball screw driven at a speed of 30 mm/sec ±10% by the electric stepper motor. In response to a scram signal, the piston inserts the control rod into the core hydraulically using stored energy in the HCU scram accumulator. The scram water is introduced into the drive through a scram inlet connection on the FMCRD housing, and is then discharged directly into the reactor vessel via clearances between FMCRD parts. The average scram times of all FMCRDs with the reactor pressure as measured at the vessel bottom below 76.3 kg/cm^{2}g are:

Percent Insertion	Time (sec)
10	≤ 0.42
40	≤ 1.00
60	≤ 1.44
100	≤ 2.80

These times are measured starting from loss of signal to the scram solenoid pilot valves in the HCUs.

The FMCRD has an electro-mechanical brake with a minimum holding torque of 5 kg-m on the motor drive shaft and a ball check valve at the point of connection with the scram inlet line.

Two redundant and separate switches in the FMCRD detect separation of the hollow piston from the ball nut.

There are 103 HCUs, each of which provides water stored in a pre-charged accumulator for scramming two FMCRDs. Figure 2.2.2 shows the major HCU components. The accumulator is connected to its associated FMCRDs by a hydraulic line that includes a scram valve held closed by pressurized control air. To cause a scram, the RPS provides a signal to de-energize the scram solenoid pilot valve (SSPV)that vents the control air from the scram valve, which then opens by spring action. Loss of either electrical power to the SSPV or loss of control air pressure causes scram.

The CRD System also provides alternate rod insertion (ARI) as a means of actuating hydraulic scram when an anticipated transient without scram (ATWS) condition exists. Following receipt of an ARI signal, solenoid valves on the scram air header open to reduce pressure in the header, allowing the HCU scram valves to open. The control rod drives then insert the control rods hydraulically.

The CRDHS has pumps, valves, filters, instrumentation, and piping to supply pressurized water for charging the HCUs and purging the FMCRDs.

The CRD System components classified as safety-related are: the HCU components required for scram; the FMCRD components required for scram; the scram inlet piping; the FMCRD reactor coolant primary pressure boundary components; the FMCRD brake and ball check valve; the internal drive housing support, the FMCRD separation switches, and; the HCU charging water header pressure instrumentation.

The CRD System components classified as Seismic Category I are: the HCU components required for scram; the FMCRD components required for scram; the scram inlet piping; the FMCRD reactor coolant primary pressure boundary components; the FMCRD brake and ball check valve; the internal drive housing support; the FMCRD separation switches, and; the HCU charging water header pressure instrumentation.

Figure 2.2.2 shows the ASME Code class for the CRD System piping and components.

The CRD System is located in the Reactor Building. The FMCRDs are mounted to the reactor vessel bottom head inside primary containment. The HCUs and CRDHS equipment are located in the secondary containment at the basemat elevation.

Each of the four divisional HCU charging header pressure sensors are powered from their respective divisional Class 1E power supply. Independence is provided between the Class 1E divisions for these sensors and also between the Class 1E divisions and non-Class 1E equipment.

For their preferred source of power, the FMCRDs are collectively powered from one Class 1E division; for their alternate source of power, they are collectively powered from one non-Class 1E Plant Investment Protection (PIP) bus.

The hydraulic portion of the CRD System which performs the scram function is physically separated from and independent of the Standby Liquid Control System.

The CRD System has the following alarms, displays, and controls in the main control room:

(1) Alarm for separation of the hollow piston from the ball-nut.

(2) Parameter displays for the instruments shown in Figure 2.2.2.

(3) Controls and status indication for the CRD pumps and flow control valves shown on Figure 2.2.2.

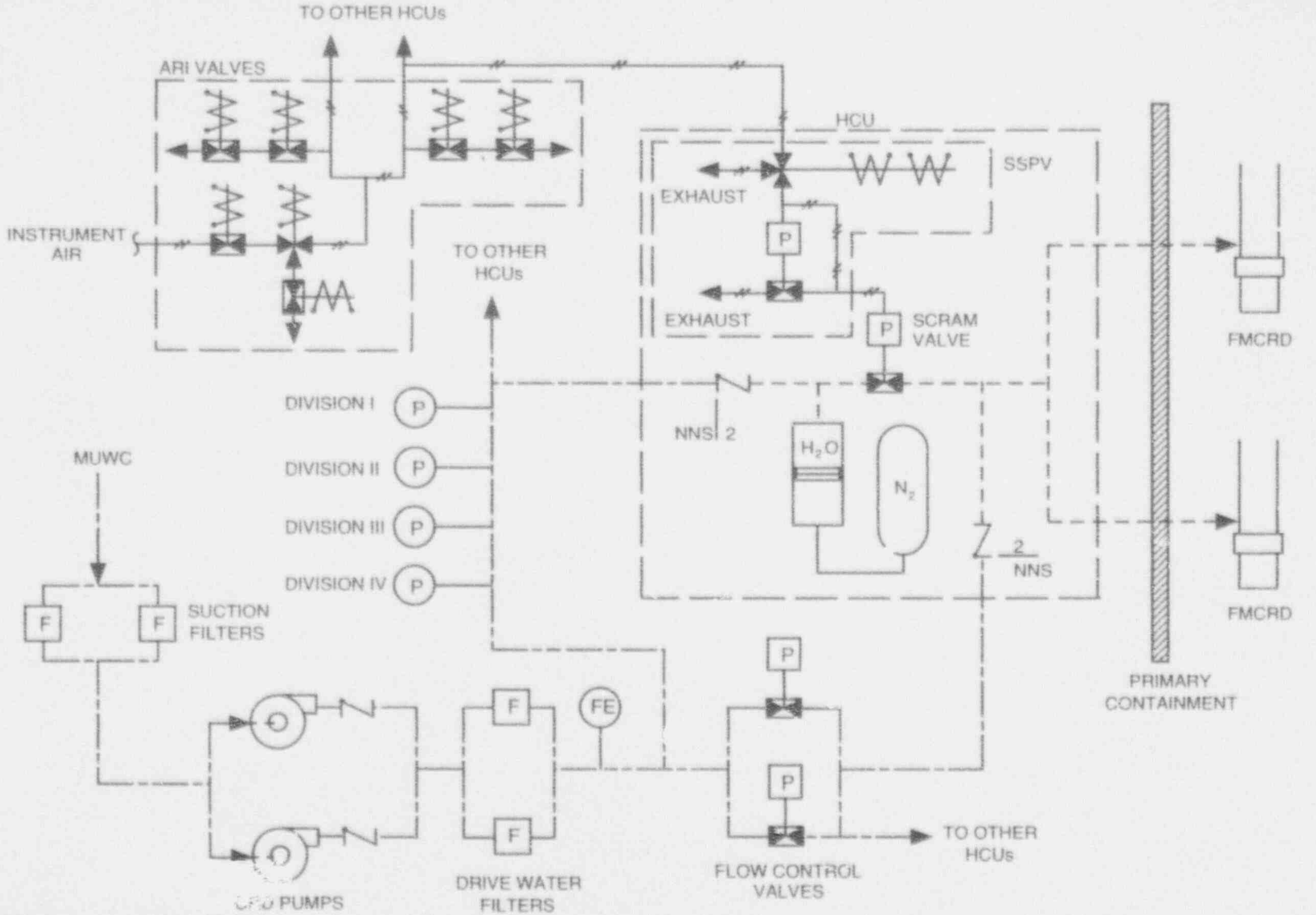
(4) Status indication for the scram valve position.

The following CRD System safety-related electrical equipment are located in either the Reactor Building or primary containment and are qualified for a harsh environment: the HCU charging header pressure instrumentation, the scram solenoid pilot valves, and FMCRD separation switches.

The piping and components on the suction side of the CRD pumps are designed for 28.8 kg/cm^2g for intersystem loss-of-coolant accident (ISLOCA) conditions.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.2 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the CRD System.



NOTES:
1. THERE ARE A TOTAL OF 205 FMCRDS AND 103 HCUs.
2. THE SSPV FUNCTION IS REPRESENTED BY A SEPARATE SOLENOID VALVE AND A PNEUMATIC VALVE; IN ACTUAL APPLICATION, THEY MAY BE COMBINED INTO A SINGLE VALVE ASSEMBLY THAT IS FUNCTIONALLY EQUIVALENT.

Figure 2.2.2 Control Rod Drive System

Table 2.2.2 Control Rod Drive System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the CRD System is as shown on Figure 2.2.2.	1. Inspections of the as-built system will be conducted.	1. The as-built CRD System conforms with the basic configuration shown on Figure 2.2.2.
2. The ASME Code components of the CRD System retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A hydrostatic test will be conducted on those code components of the CRD System required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME Code components of the CRD System conform with the requirements in the ASME Code, Section III.
3. The FMCRD can move the control rod up or down over its entire range by a ball nut and ball screw driven at a speed of 30 mm/sec ±10% by the electric stepper motor.	3. Tests will be conducted on each installed FMCRD.	3. Each control rod moves up and down over its entire range at a speed of 30 mm/sec ±10%.
4. The average scram times of all FMCRDs with the reactor pressure as measured at the vessel bottom below 76.3 kg/cm^{3}g are: Percent Insertion — Time (sec) 10 — ≤ 0.42 40 — ≤ 1.00 60 — ≤ 1.44 100 — ≤ 2.80 These times are measured starting from loss of signal to the scram solenoid pilot valves in the HCU.	4. Tests will be conducted on each installed HCU and its associated FMCRD.	4. The average scram times of all FMCRDs with the reactor pressure as measured at the vessel bottom below 76.3 kg/cm^{3}g are: Percent Insertion — Time (sec) 10 — ≤ 0.42 40 — ≤ 1.00 60 — ≤ 1.44 100 — ≤ 2.80 These times are measured starting from loss of signal to the scram solenoid pilot valves in the HCU
5. The FMCRD has an electro-mechanical brake with a minimum holding torque of 5 kg-m on the motor drive shaft.	5. Tests of each FMCRD brake will be conducted in a test facility.	5. The FMCRD electro-mechanical brake has a minimum holding torque of 5 kg-m on the motor drive shaft.
6. The FMCRD has a ball check valve at the point of connection with the scram inlet line.	6. Tests will be conducted on each of the as-built FMCRD ball check valves by inducing a reverse flow.	6. The ball check valve actuates to close the scram inlet port under conditions of reverse flow.
7. Two redundant and separate switches in the FMCRD detect separation of the hollow piston from the ball nut.	7. Tests of each as-built FMCRD will be conducted.	7. Both switches in each FMCRD detect separation of the hollow piston from the ball nut.

Table 2.2.2 Control Rod Drive System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. Following receipt of an ARI signal, solenoid valves on the scram air header open to reduce pressure in the header, allowing the HCU scram valves to open.	8. Tests will be conducted on the as-built ARI valves using a simulated actuation signal.	8. Following receipt of a simulated ARI signal, solenoid valves on the scram air header open to reduce pressure in the header, allowing the HCU scram valves to open.
9. For the four HCU charging water header pressure sensors, independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	9a. Tests will be conducted on the as-built charging water header sensors by providing a test signal in only one Class 1E division at a time.	9a. The test signal exists only in the Class 1E Division under test.
	9b. Inspections of the as-installed charging water header sensor Class 1E divisions will be conducted.	9b. Physical separation exists between Class 1E divisions. Physical separation exists between these Class 1E divisions and non-Class 1E equipment.
10. For their preferred source of power, the FMCRDs are collectively powered from one Class 1E division; for their alternate source of power, they are collectively powered from one non-Class 1E PIP bus.	10. Inspections of the as-built CRD System will be conducted.	10. For their preferred source of standby power, the FMCRD motors are collectively powered from one Class 1E division; for their alternate source of standby power, they are collectively powered from one non-Class 1E PIP bus.
11. Main control room alarms, displays and controls provided for the CRD System are defined in Section 2.2.2.	11. Inspections will be performed on the main control room alarms, displays and controls for the CRD System.	11. Alarms, displays and controls exist or can be retrieved in the main control room as defined in Section 2.2.2.

2.2.5 Neutron Monitoring System

Design Description

The Neutron Monitoring System (NMS) is a neutron monitoring and protection system. The functions of the system are to:

(1) Monitor the thermal neutron flux in the reactor core,

(2) Provide trip signals to the Reactor Protection System (RPS), and

(3) Provide power information to the operator and plant control or process systems.

The startup range neutron monitor (SRNM), the local power range monitor (LPRM), and the average power range monitor (APRM) are classified as Class 1E safety-related. The automated incore instrument calibration system and the multi-channel rod block monitor (MRBM) are classified as non-safety-related.

The SRNM monitors neutron flux from the source range to 15% of the rated power. The SRNM has ten SRNM channels which are distributed throughout the reactor core and assigned to four divisions. The SRNM detector is a fixed incore sensor. Detector cables are separated according to different divisional assignment, connected to their designated pre-amplifiers located in the Reactor Building, and then transmitted to signal processing electronic units in the Control Building.

The LPRM monitors local neutron flux in the power range up to 125% of the rated power, and overlaps with part of the SRNM range. LPRM detector assemblies are provided and are distributed in the core, with four sensors per each LPRM assembly. The LPRM assembly also contains space for automated incore calibration detector. The LPRM detector outputs are connected to the APRM signal conditioning units in the control building, where the signals are processed and amplified. LPRM detector signals are divided and assigned to four APRM channels corresponding to four divisions. LPRM signals in each APRM channel are summed and averaged to form an APRM signal which represents the core average power.

The automated incore instrument calibration system provides local power information at various core locations that correspond to LPRM locations. The automated incore instrument calibration system uses its own set of incore detectors for local power measurement and provides local power information for three dimension core power determination and for the calibration of the LPRMs. The measured data are sent to the Process Computer System for such calculation and LPRM calibration.

The MRBM uses LPRM signals to detect local power change during the rod withdrawal. If the averaged LPRM signal exceeds a preset rod block setpoint, a control rod block demand is issued.

Figure 2.2.5 shows the configuration of each NMS division.

In the NMS outside the primary containment, independence is provided between Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment. The SRNM and APRM trip signal outputs are in four divisions. The SRNM trip and the APRM trip logic are independent from each other. The SRNM generates a high neutron flux trip or a short period trip signal. Any single SRNM channel trip causes a trip in its division. The APRM can generate a high neutron flux trip, a simulated thermal power (STP) trip signal, a rapid core flow decrease trip signal, or a core power oscillation trip signal. The NMS provides these trip signals to the RPS.

The SRNM and APRM are fail-safe in the event of loss of electrical power to any division of their logic.

The NMS bypass function is performed within the NMS. Within the NMS, the bypass functions of the SRNM and the APRM are separate and independent from each other. The SRNM channels are grouped into three bypass groups. Individual SRNM channels can be bypassed. At any one time, up to three SRNM channels can be bypassed. At any one time, only one APRM channel can be bypassed. A bypassed SRNM channel or a bypassed APRM channel does not cause a trip output sent to the RPS.

Each of the four divisions of the SRNM, LPRM and APRM instruments are powered by their respective divisional Class 1E power supplies.

The NMS has the following displays and controls in the main control room:

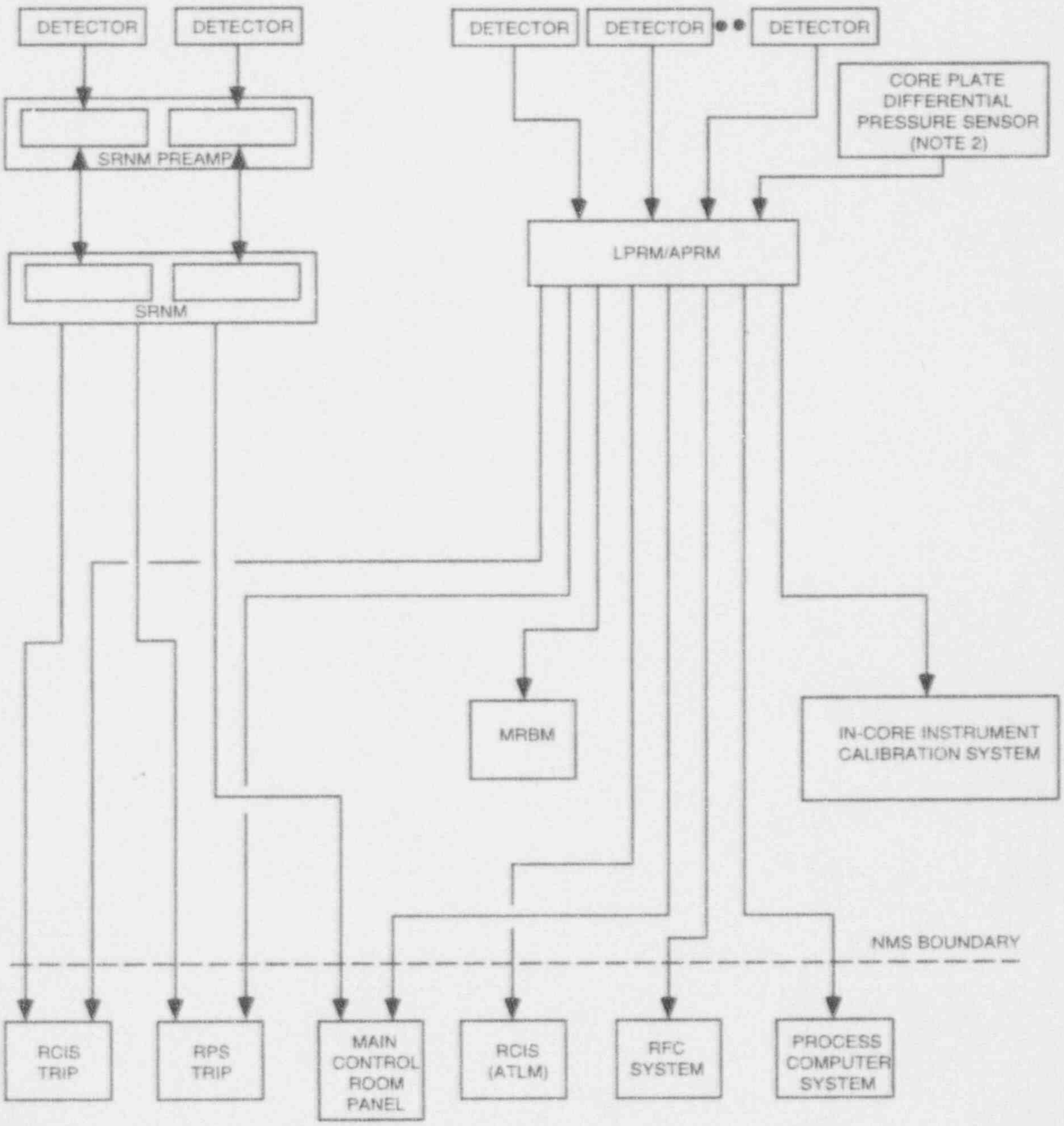
(1) SRNM, LPRM, and APRM neutron flux displays.

(2) Trip and bypass status displays.

(3) Bypass control devices.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.5 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the NMS.



NOTE:
1. DIAGRAM REPRESENTS ONE OF FOUR NMS DIVISIONS.
2. USED FOR RAPID CORE FLOW DECREASE TRIP.

Figure 2.2.5 Neutron Monitoring System

Table 2.2.5 Neutron Monitoring System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The equipment comprising the NMS is defined in Section 2.2.5.	1. Inspection of the as-built system will be conducted.	1. The as-built NMS conforms with the description in Section 2.2.5.
2. The MRBM uses LPRM signals to detect local power change during the rod withdrawal. If the averaged LPRM signal exceeds a preset rod block setpoint, a control rod block demand is issued.	2. Tests will be conducted on MRBM using simulated LPRM input signals.	2. A control rod block demand signal is issued when the simulated averaged LPRM signal exceeds the preset rod block setpoint.
3. The SRNM generates a high neutron flux trip or a short period trip signal. Any single SRNM channel trip causes a trip in the division.	3. Tests will be conducted on the SRNM using simulated neutron flux and period signals.	3. Trip signals are generated when the simulated input signals exceed trip setpoints. Any single SRNM channel trip causes a trip in its division.
4. The APRM can generate high neutron flux trip, a STP trip signal, a rapid core flow decrease trip signal, or a core power oscillation trip signal.	4. Tests will be conducted on the APRM using simulated neutron flux, and core plate differential pressure signals.	4. Trip signals are generated when the trip setpoints for high neutron flux, a high STP, a rapid core flow decrease, and a core power oscillation are exceeded.
5. The SRNM and APRM are fail-safe in the event of loss of electrical power to any division of their logic.	5. Tests will be conducted on the SRNM and APRM by disconnecting electrical power to one division of logic at a time.	5. Upon loss of electrical power to one division of either the SRNM or APRM a trip signal is generated in that division.
6. Within the NMS, the bypass functions of the SRNM and the APRM are separate and independent from each other. The SRNM channels are grouped into three bypass groups. Individual SRNM channels can be bypassed. At any one time, up to three SRNM channels can be bypassed. At any one time, only one APRM channel can be bypassed.	6. Inspections and tests will conducted on the SRNM and APRM bypass functions.	6. Within the NMS, the bypass functions of the SRNM and the APRM are separate and independent from each other. The SRNM channels are grouped into three bypass groups. Individual SRNM channels can be bypassed. At any one time, up to three SRNM channels can be bypassed. At any one time, only one APRM channel can be bypassed.
7. A bypassed SRNM channel or a bypassed APRM channel does not cause a trip output sent to the RPS.	7. Tests will be conducted on the SRNM and APRM bypassed channels using simulated input signals.	7. No trip output signals is sent to the RPS.

Table 2.2.5 Neutron Monitoring System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. In the NMS independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	8a. Tests will be performed on the NMS by providing a test signal to only one Class 1E division at a time.	8a. The test signal exists only in the Class 1E division under test in the NMS.
	8b. Inspection of the as-installed Class 1E divisions in the NMS will be performed.	8b. In the NMS, physical separation exists between Class 1E divisions. Physical separation exists between these Class 1E divisions and non-Class 1E equipment.
9. Main control room displays and controls provided for the NMS are as defined in Section 2.2.5.	9. Inspections will be performed on the main control room displays and controls for the NMS.	9. Displays and controls exist or can be retrieved in the main control room as defined in Section 2.2.5.

2.3.2 Area Radiation Monitoring System

Design Description

The Area Radiation Monitoring (ARM) System measures the gamma radiation levels at assigned locations within the plant, displays the measurements in the main control room, and activates alarms when the detected radiation levels exceed preset limits.

The ARM System is a multiple channel instrumentation system consisting of radiation detectors, radiation process monitors, and local audible alarms. Each ARM channel monitors the radiation level in its assigned area, and initiates its respective main control room (MCR) alarm and a local alarm (if provided) when the radiation level exceeds a preset limit.

The ARM System is classified as non-safety-related.

The ARM System radiation sensors and the audible warning alarms are installed locally in the plant, while the radiation process monitors are located in the Control Building.

The ARM System has the following alarms and displays in the MCR:

(1) Displays of radiation levels.

(2) Channel trip status.

(3) Alarms.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.3.2 provides definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Area Radiation Monitoring System.

Table 2.3.2 Area Radiation Monitoring System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The equipment comprising the ARM System is defined in Section 2.3.2.	1. Inspection of the as-built system will be conducted.	1. The as-built ARM System conforms with the description in Section 2.3.2.
2. Each ARM channel monitors radiation level in its assigned area, and initiates its respective MCR alarm and a local audible alarm (if provided) when the radiation level exceeds a preset limit.	2. Tests will be conducted using simulated input signals for each channel.	2. The MCR alarm and local audible alarm (if provided is initiated when the simulated radiation level exceeds a preset limit.
3. MCR alarms and displays provided for the ARM System are as defined in Section 2.3.2.	3. Inspections will be performed on the MCR alarms and displays for the ARM System.	3. Alarms and displays exist or can be retrieved in the MCR as defined in Section 2.3.2.

2.3.3 Containment Atmospheric Monitoring System

Design Description

The Containment Atmospheric Monitoring System (CAMS) is used for post-accident monitoring of the primary containment. The system monitors the atmospheric conditions in the drywell and in the suppression chamber for radiation levels and for hydrogen and oxygen gas concentration levels, displays the measurements in the main control room (MCR), and activates alarms in the MCR on high levels of radiation and/or gas concentrations.

CAMS consists of two independent divisions and each division is composed of two radiation channels and an oxygen/hydrogen gas monitoring equipment.

The CAMS is classified as a Class 1E safety-related system.

Operation of each CAMS division can be activated manually by the operator or automatically during a post-accident condition by a signal indicating a high drywell pressure or a low reactor water level.

One radiation channel of each CAMS division monitors the radiation level in the drywell and the other channel monitors the radiation level in the suppression chamber.

The oxygen/hydrogen monitoring equipment of each CAMS division analyzes the hydrogen and oxygen gas concentration levels in the drywell and in the suppression chamber and provides separate gas concentration displays in the MCR.

Each CAMS division is powered from its respective divisional Class 1E power source. In the CAMS, independence is provided between the Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment.

Both CAMS divisions are located in the Reactor Building, except for the radiation and the gas process monitors which are located in the Control Building.

The CAMS has the following alarms, displays, and controls in the MCR:

(1) Displays of radiation, hydrogen and oxygen levels.

(2) Alarms for radiation levels, and for hydrogen and gas concentration levels.

(3) Manual system level initiation for each CAMS division.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.3.3 provides definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Containment Atmospheric Monitoring System.

Table 2.3.3 Containment Atmospheric Monitoring System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The equipment comprising the CAMS is defined in Section 2.3.3.	1. Inspection of the as-built system will be conducted.	1. The as-built CAMS conforms with the description in Section 2.3.3.
2. Operation of each CAMS division can be activated manually by the operator or automatically.	2. Tests of each division of the as-built CAMS will be conducted using manual controls and simulated automatic initiation signals.	2. Each CAMS division is activated.
3. In the CAMS, independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	3a. Tests will be performed on the CAMS by providing a test signal to only one Class 1E division at a time.	3a. The test signal exist only in the Class 1E division under test in the CAMS.
	3b. Inspection of the as-built Class 1E divisions in the CAMs will be performed.	3b. In the CAMS, physical separation exists between Class 1E divisions. Physical separation exists between these Class 1E divisions and non-Class 1E equipment.
4. Main control room alarms, displays and controls provided for the CAMS are as defined in Section 2.3.3.	4. Inspections will be performed on the main control room alarms, displays and controls for the CAMS.	4. Alarms, displays and controls exist or can be retrieved in the main control room as defined in Section 2.3.3.

2.4.4 Reactor Core Isolation Cooling System

Design Description

The Reactor Core Isolation Cooling (RCIC) System consists of a turbine, pump, piping, valves, controls and instrumentation. The RCIC turbine is driven by the steam from the reactor pressure vessel (RPV) which then drives the RCIC pump. The function of the RCIC System is to provide makeup water to the RPV.

The RCIC steam supply to the turbine branches off one of the main steamlines inside containment and exhausts to the suppression pool. The primary source of RCIC pump suction is the Condensate Storage Tank (CST). The suppression pool is the secondary source of RCIC pump suction. Figure 2.4.4a shows the basic system configuration and scope. Figure 2.4.4b shows RCIC System control interfaces.

The RCIC System shown on Figure 2.4.4a is classified as safety-related.

The RCIC System operates in the following modes:

(1) RPV water makeup.

(2) Full flow test.

(3) Minimum flow bypass.

RPV Water Makeup Mode

As shown on Figure 2.4.4b, the RCIC System channel measurements are provided to the Safety System Logic and Control (SSLC) System for signal processing, setpoint comparisons, and generating trip signals. The RCIC System is automatically initiated when either a high drywell pressure or low reactor water level condition exists. The SSLC processors use a two-out-of-four voting logic for system initiation and shutdown. Manual RCIC System initiation can be performed.

The RCIC System automatically shuts down when a high reactor water level condition exists. Following RCIC shutdown on high reactor water level signal, the RCIC System automatically restarts to provide RPV water makeup, if the low reactor water level initiation signal recurs.

During this mode, the primary source pump suction is the CST. Automatic transfer of pump suction from CST to suppression pool occurs when a low CST water level or a high suppression pool water level signal exists. The SSLC processor use a two-out-of four logic to initiate suction transfer.

In the RPV water makeup mode, the RCIC pump delivers a flow rate of at least 182 m^3/hr against a maximum differential pressure (between the RPV and the

suction source) of 82.8 kg/cm^2. This flow rate is achieved within 30 seconds of receipt of the system initiation signal. The RCIC pump has sufficient net positive suction head (NPSH) available at the pump.

Full Flow Test Mode

The RCIC System has a full flow test mode to permit pump flow testing during plant operation. During the test, water is pumped from the suppression pool and returned to the suppression pool via the test return line. The vessel injection valve is kept closed to prevent RPV injection during the test.

If a system initiation signal occurs during the full flow test mode, the RCIC System automatically aligns to the RPV water makeup mode.

Minimum Flow Bypass Mode:

The RCIC System has a minimum flow bypass mode that assures there is always flow in the RCIC pump when it is operating. This is accomplished by monitoring pump discharge flow, and opening a minimum flow valve to the suppression pool when flow falls below minimum value. The minimum flow valve closes when the pump flow exceeds the minimum value. Minimum flow bypass operation is automatic based on a flow signal opening the minimum flow valve when the flow is low, with a concurrent high pump discharge pressure signal.

The remaining discussion in this section is not mode specific and applies (unless stated otherwise) to the entire RCIC System.

The RCIC System shown on Figure 2.4.4a is classified as Seismic Category I. Figure 2.4.4a shows the ASME Code Class for the RCIC System. The RCIC System is located inside primary containment and in the Reactor Building.

As shown on Figure 2.4.4a, the RCIC System components are powered from Class 1E Division I except for the steam supply outboard containment isolation valve which is powered from Class 1E Division II. In the RCIC System, independence is provided between Class 1E divisions, and also between Class 1E divisions and non-Class 1E equipment.

Outside the primary containment the RCIC System shown on Figure 2.4.4a, is physically separated from the two divisions of the High Pressure Core Flooder (HPCF) System.

The RCIC System has the following displays and controls in the main control room (MCR):

(1) Parameter displays for the instruments shown on Figure 2.4.4a.

(2) Controls and status indication for the active safety-related components shown on Figure 2.4.4a.

(3) Manual system level initiation capability for RPV water makeup mode.

The safety-related electrical components (including instrumentation and control) shown on Figure 2.4.4a located inside primary containment and in the Reactor Building are qualified for a harsh environment.

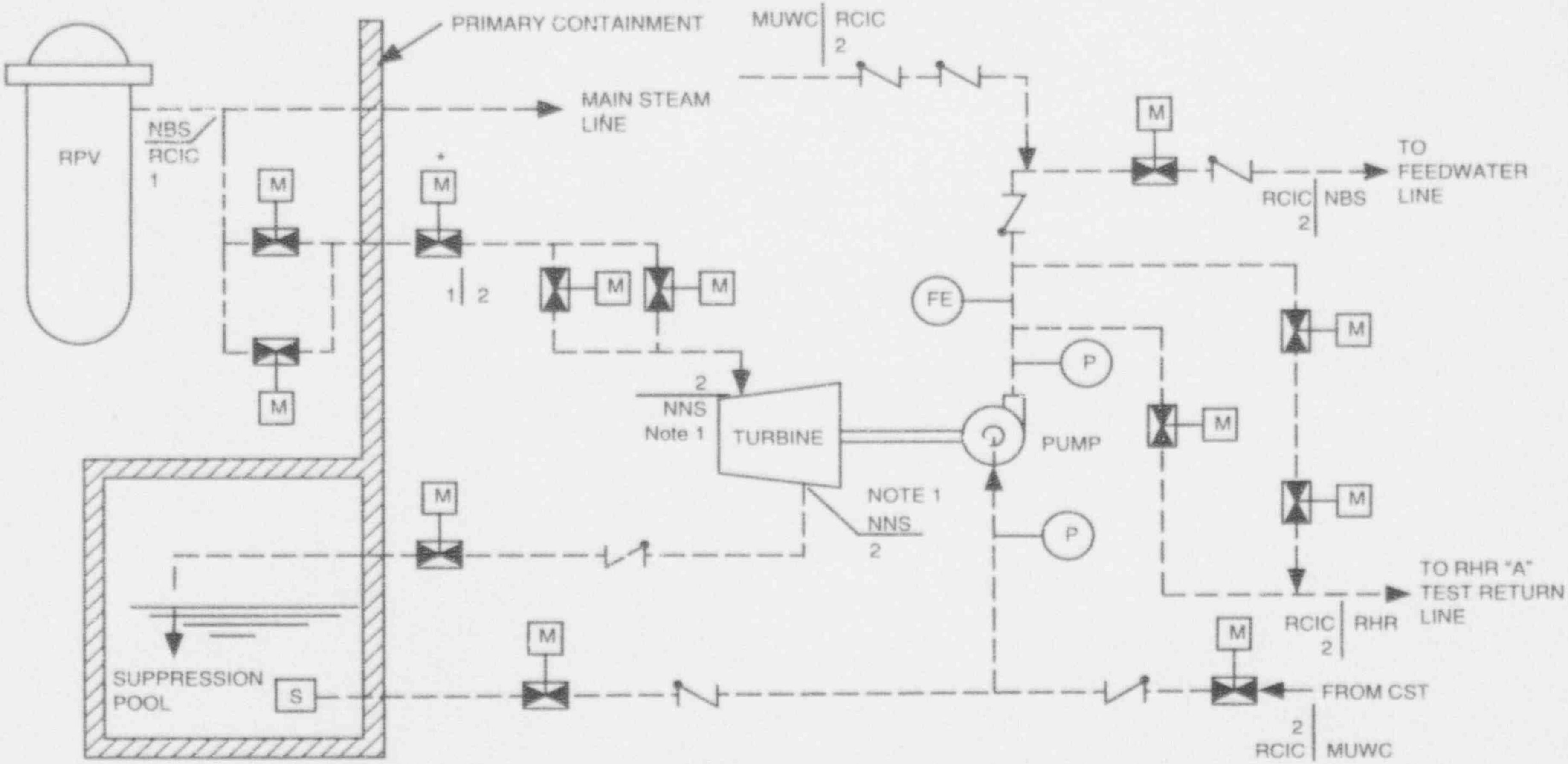
The motor-operated valves (MOVs) shown on Figure 2.4.4a have active safety-related functions and perform these functions under differential pressure, fluid flow, and temperature conditions.

The RCIC turbine is tripped if a low suction pressure condition is present.

The RCIC System pump suction piping and components have a design pressure of 28.8 kg/cm^2g for intersystem loss-of-coolant (ISLOCA) conditions.

Inspections, Tests, Analyses and Acceptance Criterial

Table 2.4.4 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for the RCIC System.



NOTES:

1. RCIC TURBINE IS NOT COVERED BY ASME CODE SECTION III BUT IS DESIGNED, FABRICATED, AND INSTALLED TO SAFETY-RELATED STANDARDS AND IS CONSISTENT WITH THE ASME CODE SECTION III REQUIREMENTS. THE TURBINE INCLUDES THE TURBINE TRIP AND THROTTLE VALVE.

2. ALL RCIC SYSTEM COMPONENTS SHOWN ON THIS FIGURE ARE POWERED FROM CLASS 1E DIVISION I EXCEPT FOR THE OUTBOARD CONTAINMENT ISOLATION VALVE (*) WHICH IS CLASS 1E DIVISION II.

Figure 2.4.4a Reactor Core Isolation Cooling System

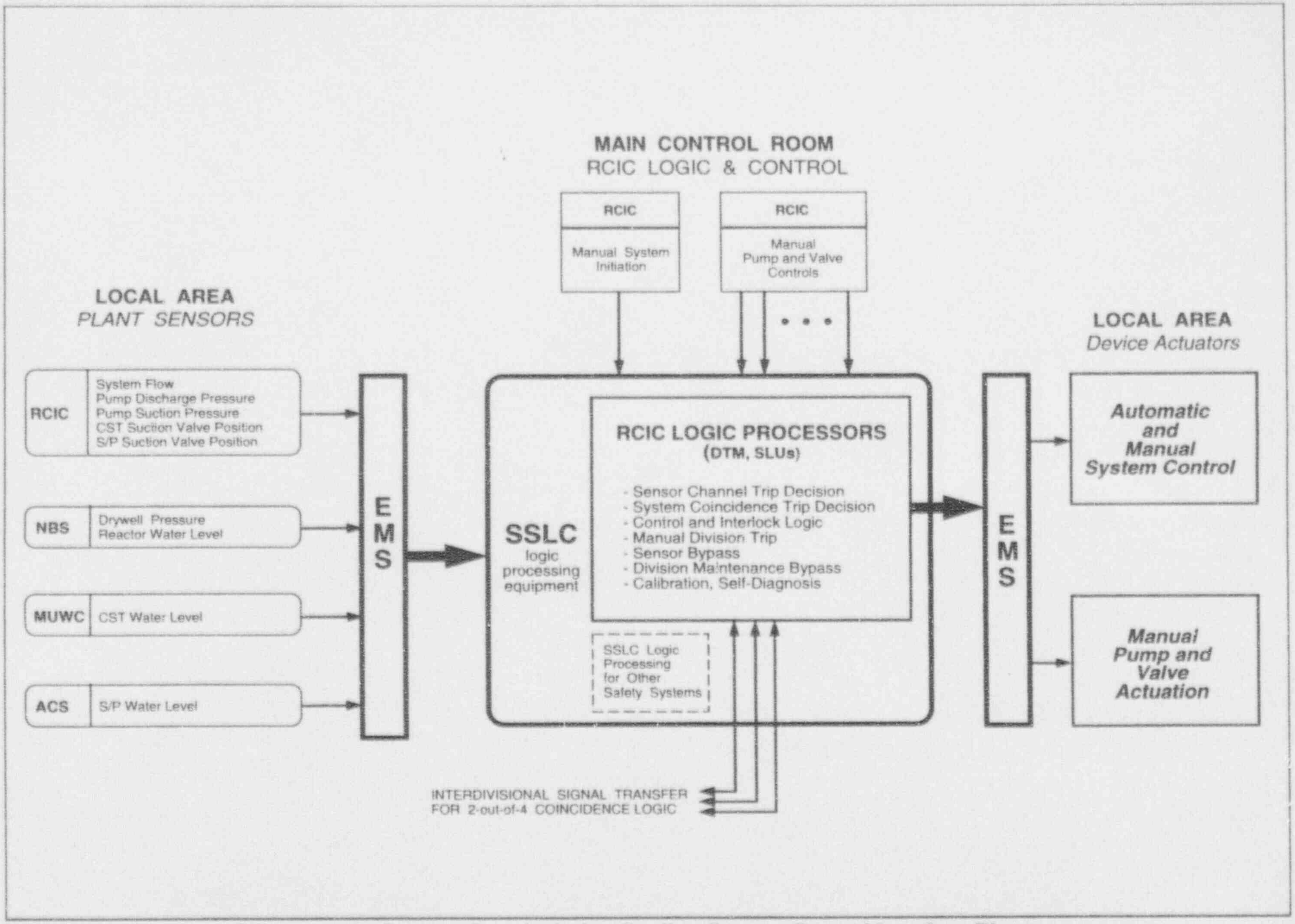


Figure 2.4.4b Reactor Core Isolation Cooling System Control Interface Diagram

Table 2.4.4 Reactor Core Isolation Cooling System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the RCIC System is as shown on Figure 2.4.4a and 2.4.4b.	1. Inspections of the as-built system will be conducted.	1. The as-built RCIC System conforms with the basic configuration shown on Figure 2.4.4a.
2. The ASME Code components of the RCIC System retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A hydrostatic test will be conducted on those Code components of the RCIC System required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME Code components of the RCIC System conform with the requirements in the ASME Code Section III.
3a. The RCIC System is automatically initiated in the RPV water makeup mode when either a high drywell pressure or a low reactor water level condition exists.	3a. Tests will be conducted using simulated input signals for each process variable to cause trip conditions in two, three, and four instrument channels of the same process variable.	3a. RCIC System receives an initiation signal.
3b. Manual RCIC System initiation can be performed.	3b. Test will be conducted by manually initiating RCIC System.	3b. RCIC System receives an initiation signal.
3c. Following receipt of an initiation signal, the RCIC System automatically initiates and operates in the RPV water makeup mode.	3c. Tests will be conducted on the RCIC System using simulated initiation signal.	3c. Upon receipt of a simulated initiation signal, the following occurs: a) Steam supply bypass valve receives open signal. b) Test return valves receive close signal. c) CST suction valve receives open signal. d) Injection valve receives open signal after a 10 second time delay. e) Steam admission valve receives open signal after a 10 second time delay.
3d. The RCIC System automatically shuts down when a high reactor water level condition exists.	3d. Tests will be conducted using simulated high reactor water level signals to cause trip conditions in two, three, and four instrument channels of water level variable.	3d. RCIC System receives shutdown signal.

Table 2.4.4 Reactor Core Isolation Cooling System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3e. Following receipt of shutdown signal, RCIC System automatically terminates RPV water makeup mode.	3e. Tests will be conducted on RCIC System using simulated shutdown signal.	3e. Upon receipt of simulated shutdown signal the following occurs: a) Steam supply bypass valve receives close signal b) RCIC initiation logic resets c) Injection valve receives close signal d) Steam admission valve receives close signal
3f. Following RCIC shutdown on high reactor water level signal, the RCIC System automatically restarts to provide RPV water makeup if low reactor water level signal recurs.	3f. Test will be conducted using simulated low reactor water level signal.	3f. Upon receipt of simulated low reactor water level signal the following occurs: a) Steam supply bypass valve receives open signal. b) Test return valves receive close signal. c) CST suction valve receives open signal. d) Injection valve receives open signal after a 10 second time delay. e) Steam admission valve receives open signal after a 10 second time delay.
3g. RCIC System automatically initiates suction transfer from CST to suppression pool when either a low CST water level or a high suppression pool water level exists.	3g. Tests will be conducted using simulated input signals for each process variable to cause trip conditions in two, three, and four instrument channels of the same process variable.	3g. RCIC System receives suction transfer initiation signal.
3h. Following receipt of suction transfer initiation signal, RCIC System automatic ally switches pump suction.	3h. Test will be conducted using simulated suction transfer initiation signal.	3h. Upon receipt of simulated suction transfer initiation signal, the following occurs: a) CST suction valve receives close signal. b) Suppression pool suction valve receives open signal.

Table 2.4.4 Reactor Core Isolation Cooling System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3i. The RCIC System pump has sufficient NPSH.	3i. Inspections, tests, and analyses will be performed based upon the as-built system. The analyses will consider the effects of: - Pressure losses for pump inlet piping and components - Suction from suppression pool with water level at the minimum value - 50% blockage of pump suction strainers - Design basis fluid temperature (77°C) - Containment at atmospheric pressure.	3i. The available NPSH exceeds the NPSH required.
4. If a system initiation signal occurs during the full flow test mode, the RCIC System automatically aligns to the RPV water makeup mode.	4. Test will be conducted using simulated initiation signal	4. RCIC System automatically aligns to RPV water makeup mode from test mode upon receipt of an initiation signal.
5. The RCIC System has a minimum flow bypass mode that assures there is always flow in the RCIC pump when it is operating.	5. Tests will be conducted on the pump minimum flow valve interlock logic using simulated pressure and flow signals.	5. The pump minimum flow rate receives a signal to open when signals indicative of the following conditions exist concurrently: a) Pump discharge pressure is high when the pump starts, b) Pump flow is low. The pump minimum flow valve receives a signal to close when a signal indicative of the following condition exists.
6. In the RCIC System, independence is provided between Class 1E Divisions and between Class 1E Divisions and non-Class 1E equipment.	6a. Tests will be performed in the RCIC System by providing a test signal in only one Class 1E Division at a time. 6b. Inspections of the as-built Class 1E Divisions in the RCIC System will be performed.	6a. The test signal exists only in the Class 1E Division under test in the RCIC System. 6b. In the RCIC System physical separation exists between Class 1E Divisions in the RCIC System. Physical separation exists between Class 1E Divisions and non-Class 1E equipment.

Table 2.4.4 Reactor Core Isolation Cooling System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. Outside the primary containment, the RCIC System shown on Figure 2.4.4a, is physically separated from the two divisions of the HPCF System.	7. Inspections of the as-installed RCIC System will be performed.	7. Outside the primary containment, the RCIC System shown on Figure 2.4.4a, is physically separated from the two division of the HPCF System by structural and/or fire boundaries.
8. Main control room displays and controls provided for RCIC System are as defined in Section 2.4.4.	8. Inspections will be performed on the main control room displays and controls for the RCIC System.	8. Displays and controls exist or can be retrieved in the main control room as defined in Section 2.4.4.
9. MOV's designated in Section 2.4.4 as having active safety-related function open and/or close under differential pressure, fluid flow, and temperature conditions.	9. Opening and/or closing tests of installed MOV's will be conducted under pre-operational differential pressure, fluid flow, and temperature conditions.	9. Each MOV's opens and/or closes. The following MOV's open and/or close in the following time limits upon receipt of the actuation signal. <u>Valve</u> — <u>Time</u> Steam Supply Containment Isolation Valves — $\leq$ 30 secs Close Injection Valve — $\leq$ 15 secs Open
10. The RCIC turbine is tripped if low suction pressure condition is present.	10. Test will be conducted using a simulated low suction pressure signal.	10. The turbine trip and throttle valve receives a trip signal.

2.5.5 Refueling Equipment

Design Description

The Reactor Building is supplied with a refueling machine for fuel movement and servicing plus an auxiliary platform for servicing operations from the vessel flange level.

The refueling machine is a gantry crane, which spans the reactor vessel and the storage pools on bedded tracks in the refueling floor. A telescoping mast and grapple suspended from a trolley system is used to lift and orient fuel bundles for placement in the core and/or storage racks. Two auxiliary hoists, one main and one auxiliary monorail trolley-mounted, are provided for in-core servicing. Control of the machine is from an operator station on the refueling floor.

The refueling machine is classified as non-safety-related.

A position indicating system and travel limit computer is provided to locate the grapple over the vessel core and prevent collision with pool obstacles. The mast grapple has a redundant load path so that no single component failure results in a fuel bundle drop. Interlocks on the platform: (1) prevent hoisting a fuel bundle over the vessel unless an all control-rod in permissive is present; (2) limit vertical travel of the fuel grapple to provide shielding over the grappled fuel during transit; (3) prevent lifting of fuel without grapple hook engagement and load engagement.

The refueling machine is classified as Seismic Category I.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.5.5 provides definition of the inspection, test, and/or analyses, together with associated acceptance criteria, which will be undertaken for the refueling machine. No entries are proposed for the auxiliary platform.

Table 2.5.5 Refueling Equipment

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the refueling machine is described in Section 2.5.5.	1. Inspections of the as-built refueling machine will be conducted.	1. The as-built refueling machine conforms with the basic configuration described in Section 2.5.5.
2. Interlock on the platform	2. Tests will be conducted on the as-built refueling machine using simulated signals and loads.	2. Interlock on the platform
a. Prevent hoisting a fuel bundle over the vessel unless an all-control-rod-in permissive is present.		a. Prevent hoisting a fuel bundle over the vessel unless an all-control-rod-in permissive is present.
b. Limit vertical travel of the fuel grapple to provide shielding over the grappled fuel during transit.		b. Limit vertical travel of the fuel grapple to provide shielding over the grappled fuel during transit.
c. Prevent lifting of fuel without grapple hook engagement and load engagement.		c. Prevent lifting of fuel without grapple hook engagement and load engagement.

2.5.6 Fuel Storage Facility

Design Description

The Fuel Storage Facility provides storage racks for the temporary and long term storage of new and spent fuel and associated equipment. The new and spent fuel storage racks use the same configuration and prevent inadvertent criticality.

The racks are classified as safety-related.

Racks provide storage for spent fuel in the spent fuel storage pool in the Reactor Building. New fuel is stored in the new fuel storage vault in the Reactor Building. The racks are top loading, with fuel bail extended above the rack. The spent fuel racks have a minimum storage capacity of 270% of the reactor core which is equivalent to a minimum of 2354 fuel storage positions. The new and spent fuel racks maintain a subcriticality of at least 5% Δk under dry or flooded conditions. The rack arrangement prevents accidental insertion of fuel assemblies between adjacent racks and allows flow to prevent the water from exceeding 100°C.

The racks are classified as Seismic Category I.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.5.6 provides a definition of the inspections, tests, and/or analyses and associated acceptance criteria, which will be undertaken for the new and spent fuel storage racks.

Table 2.5.6 Fuel Storage Facility

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the new and spent fuel racks is described in Section 2.5.6.	1. Inspections of the as-built system will be conducted	1. The as-built new and spent fuel storage racks conform with the basic configuration described in Section 2.5.6.
2. The new and spent fuel racks maintain a subcriticality of at least 5% Δk under dry or flooded conditions.	2. Analysis will be performed to determine the k_{eff} of the as-built new and spent fuel racks.	2. An analysis report exists which concludes that the new and spent fuel racks have a subcriticality of at least 5%Δk under dry or flooded conditions.
3. The rack arrangement prevents accidental insertion of fuel assemblies between adjacent racks.	3. Inspections of the as-built new and spent fuel racks will be performed.	3. The rack arrangement prevents accidental insertion of fuel assemblies between adjacent racks.
4. The rack arrangement allows flow to prevent the water from exceeding 100°C.	4. An analysis of the as-built spent fuel rack will be performed to determine the maximum water temperature	4. An analysis report exists which concludes that the rack arrangement allows flow to prevent the water from exceeding 100°C.

2.8.4 Loose Parts Monitoring System

Design Description

The Loose Parts Monitoring System (LPMS) monitors the reactor pressure vessel (RPV) and appurtenances for indications of loose metallic parts within the reactor pressure vessel. The LPMS detects structure borne sound that can indicate the presence of loose parts impacting against the reactor pressure vessel and internals. The system alarms when the signal amplitude exceeds preset limits.

The LPMS consists of sensors, signal analysis and data acquisition equipment. LPMS has multiple channels which process signals from sensors mounted on the external surfaces of the reactor coolant pressure boundary. The LPMS is classified as non-safety-related.

The LPMS has provisions for both automatic and manual start-up of data acquistion equipment with automatic activation in the event the preset alert level is reached or exceeded. The system also initiates an alarm in the main control room when an alert condition is reached.

Inspections, Tests, Analyses and Acceptance Criteria

Tables 2.8.4 provides a definition of the inspections, tests and/or analyses, together with associated acceptance criteria, which will be undertaken for Loose Parts Monitoring System.

Table 2.8.4 Loose Parts Monitoring System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Equipment comprising the LPMS is defined in Subsection 2.8.4.	1. Inspection of the as-built system will be conducted.	1. The as-built LPMS conforms with the description in Subsection 2.8.4.
2. LPMS monitors the RPV and appurtenances for indication of loose metallic parts.	2. Inspections will be conducted on the as-built LPMS.	2. The LPMS sensitivity, without the background noise associated with plant operation, is such that it can detect a metallic loose part that weighs from 0.11 kg to 13.6 kg and impacts with a maximum kinetic energy of 0.68 joules on inside surface of RPV within 0.91 meter of a sensor.

2.10.2 Condensate Feedwater and Condensate Air Extraction System

The Condensate Feedwater and Condensate Air Extraction System consists of two subsystems, the Condensate and Feedwater System (CFS) and the Main Condenser Evacuation System (MCES).

Design Description

Condensate and Feedwater System

The function of the CFS is to receive condensate from the condenser hotwells, supply condensate to the Condensate Purification System, and deliver feedwater to the reactor. Condensate is pumped from the main condenser hotwell by the condensate pumps, passes through the low pressure feedwater heaters to the feedwater pumps, and then is pumped through the high pressure heaters to the reactor. Figure 2.10.2a shows the basic system configuration. The CFS boundaries extend from the main condenser outlet to (but not including) the seismic interface restraint outside the containment.

The CFS is classified as non-safety-related.

The CFS is controlled by signals from the Feedwater Control System.

The CFS is located in the steam tunnel and Turbine Building.

The CFS has parameter displays for the instruments shown on Figure 2.10.2a in the main control room.

Main Condenser Evacuation System

The MCES removes the hydrogen and oxygen produced by the radiolysis of water in the reactor, and other power cycle noncondensable gases. The system exhausts the gases to the Off-Gas System during plant operation, and to the Turbine Building compartment exhaust system at the beginning of each startup. The MCES consists of redundant steam jet air ejectors (SJAE) units for power plant operation, and a mechanical vacuum pump for use during startup. Figure 2.10.2b shows the basic system configuration.

The MCES is classified as non-safety-related.

The MCES is located in the Turbine Building.

Steam supply to the SJAE provides dilution of the hydrogen and prevents the offgas from reaching the flammable limit of hydrogen. When the steam flow drops below the setpoint for stream dilution, the Off-Gas System is isolated.

The vacuum pump is tripped and its discharge valve is closed upon receiving a main steamline high radiation signal.

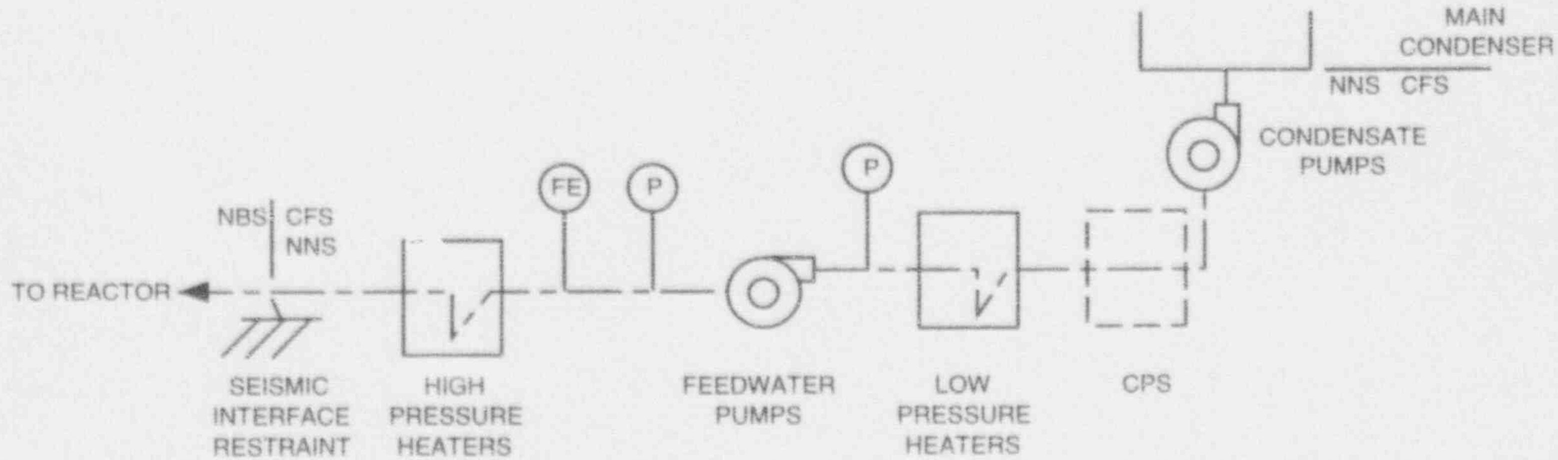
The MCES has the following displays in the main control room:

(1) Parameter displays for the instruments shown on Figure 2.10.2b.

(2) Status indication for the vacuum pump and SJAE discharge valves.

Inspections, Tests, Analyses and Acceptance Criteria

Tables 2.10.2a and 2.10.2b provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the CF and MCE System respectively.



NOTES:
1. RELIEF VALVE DISCHARGE AND VENTS ARE CHANNELED THROUGH CLOSED SYSTEMS.
2. FEEDWATER PUMP REDUNDANCY IS PROVIDED.

Figure 2.10.2a Condensate and Feedwater System

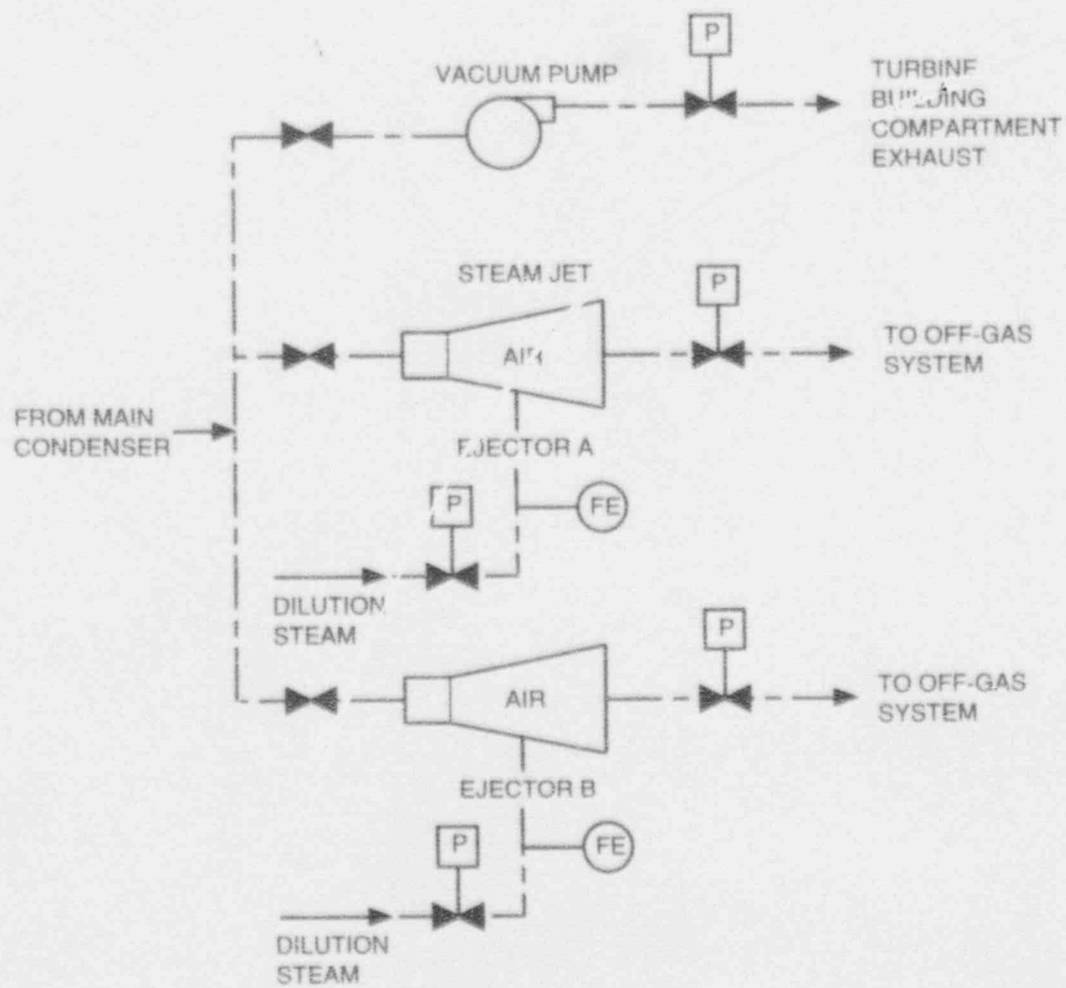


Figure 2.10.2b Main Condenser Evacuation System

Table 2.10.2a Condensate and Feedwater System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the CFS is as shown on Figure 2.10.2a.	1. Inspections of the as-built CFS will be conducted.	1. The as-built CFS conforms with the basic configuration shown in Figure 2.10.2a.
2. The CFS is controlled by signals from the Feedwater Control System	2. Tests of the as-built CFS will be conducted using simulated input signals.	2. The CFS starts and responds to the simulated signals.
3. Main control room displays provided for the CF System are as defined in Section 2.10.2.	3. Inspections will be performed on the main control room displays for the CFS.	3. Displays exist or can be retrieved in the main control room as defined in Section 2.10.2.

Table 2.10.2b Main Condenser Evacuation System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the MCES is as shown on Figure 2.10.2b.	1. Inspections of the as-built MCES will be conducted.	1. The as-built MCES conforms with the basic configuration shown in Figure 2.10.2b.
2. When the steam flow drops below the setpoint for steam dilution, the Off-Gas System is isolated.	2. Tests will be conducted on the as-built MCES using simulated signals for steam flow.	2. The SJAE discharge valves, close on receipt of a simulated low flow signal.
3. The vacuum pump is tripped and its discharge valve is closed upon receiving a main steamline high radiation signal.	3. Tests will be conducted on the as-built MCES using simulated signals for radiation in the main steamlines.	3. The vacuum pump trips and the discharge valve closes upon receipt of a simulated high radiation signal.
4. Main control room displays provided for the MCES are as defined in Section 2.10.2b.	4. Inspections will be performed on the main control room displays for the MCES.	4. Displays exist or can be retrieved in the main control room as defined in Section 2.10.2b.

2.10.4 Condensate Purification System

Design Description

The Condensate Purification System (CPS) purifies and treats the condensate, using filtration to remove insoluble solids, and ion exchange demineralizer to remove soluble solids. The CPS consists of full flow high efficiency particulate filters followed by full flow deep bed demineralizers. Figure 2.10.4 shows the basic system configuration.

The CPS is classified as non-safety-related.

The CPS is located in the Turbine Building.

The CPS has alarms and display for efficient conductivity in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.10.4 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the CP System.

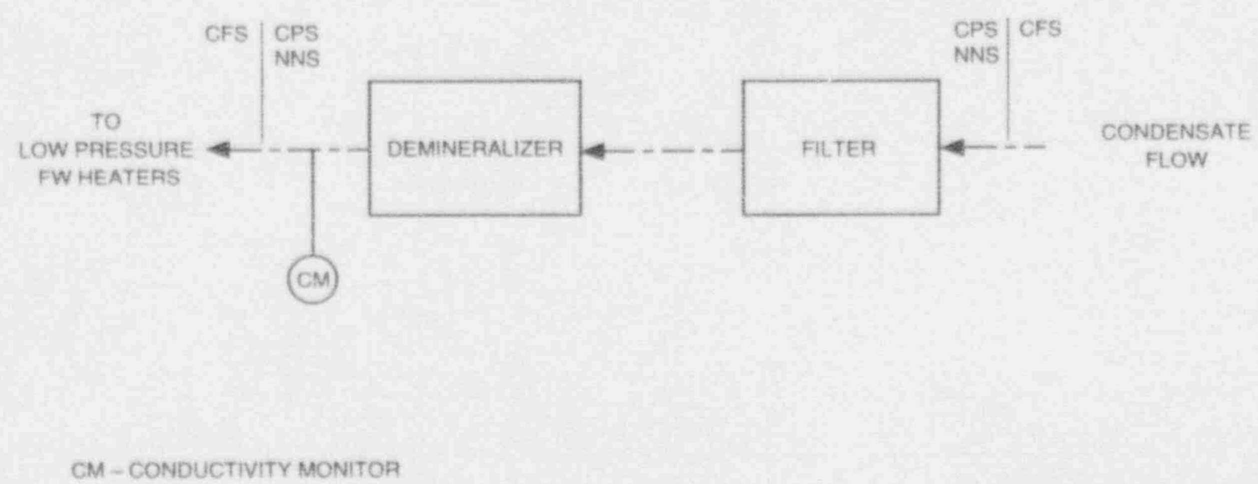


Figure 2.10.4 Condensate Purification System

Table 2.10.4 Condensate Purification System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the CPS is as shown on Figure 2.10.4.	1. Inspections of the as-built System will be conducted.	1. The as-built CPS conforms with the basic configuration shown in Figure 2.10.4.
2. Main control room alarm and display provided for the CPS are as defined in Section 2.10.4.	2. Inspections will be performed on the main control room alarm and display for the CPS.	2. Alarm and display exist or can be retrieved in the main control room as defined in Section 2.10.4.

2.10.7 Main Turbine

Design Description

The Main Turbine (MT) System uses the energy in steam from the reactor to drive the plant generator.

The MT System is classified as non-safety-related.

The MT System has the following features that prevent overspeed:

(1) Main stop valves (MSV)/Control valves (CV) [MSVs trip/CVs trip and modulate].

(2) Combined intercept valves (CIVs) [CIVs trip].

(3) Extraction line non-return valves (trip).

(4) Redundant valve closure mechanisms, i.e., fast acting solenoid valves and emergency trip fluid system.

(5) Redundant normal speed control.

Three levels of signals to MT System valves, i.e., normal speed control/overspeed trip/backup overspeed trip.

Overspeed trip occurs as follows:

Overspeed Condition	**Protective Action**
(1) Exceeds normal speed control setpoint	Normal speed control signals the CVs and CIVs to close.
(2) Exceeds overspeed trip setpoint	Overspeed trip signals MSVs, CVs, CIVs, and extraction line non-return valves to close.
(3) Exceeds backup overspeed trip setpoint	Backup overspeed trip signals MSVs, CVs, CIVs, and extraction line non-return valves to close.

The turbine MSV closes in 0.1 seconds or greater. The turbine CV trip closure is 0.15 seconds or greater. In the modulating mode, the full stroke servo-closure of the turbine CV is 2.5 seconds or greater.

The MT System has the following alarms and displays in the main control room:

(1) Overspeed alarm.

(2) Parameter displays for turbine speed and inlet steam pressure.

The MT System is located within the Turbine Building. The axis of the turbine and generator is orientated within the Turbine Building to be inline with the Reactor and Control Buildings.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.10.7 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria which will be undertaken for the TG System.

Table 2.10.7 Main Turbine Generator System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The equipment comprising the MT System is defined in Section 2.10.7.	1. Inspection of the as-built system will be conducted.	1. The as-built MT System conforms with the description in Section 2.10.7.
2. MT System overspeed protective actions are as defined in Section 2.10.7.	2. Tests will be conducted on the as-built MT System using simulated overspeed signals.	2. The following protective actions occur: **Overspeed Condition** — **Protective Action** (a) Exceeds normal speed control setpoint. — Normal speed control signals the CVs and CIVs to close. (b) Exceeds overspeed trip setpoint. — Overspeed trip signals MSVs, CVs, CIVs, and extraction line non-return valves to close. (c) Exceeds backup overspeed trip setpoint. — Backup overspeed trip signals MSVs, CVs, CIVs, and extraction line non-return valves to close.
3. The turbine MSV closes in 0.10 seconds or greater.	3. Tests will be conducted on the as-built turbine MSV.	3. The turbine MSV closes in 0.10 seconds or greater
4. The turbine CV trip closure is 0.15 seconds or greater.	4. Test s will be conducted on the as-built turbine CV.	4. The turbine CV trip closure is 0.15 seconds or greater.
5. In the modulating mode, the full stroke servo closure of the turbine CV is 2.5 seconds or greater.	5. Tests will be conduced on the as-built turbine CV.	5. In the modulating mode, the full stroke servo closure of the turbine CV is 2.5 seconds or greater.
6. Main control room alarms and displays provided for the MT System are as defined in Section 2.10.7.	6. Inspections will be performed on the main control room alarms and displays for the MT System.	6. Alarms and displays exist or can be retrieved in the main control room as defined in Section 2.10.7.

Table 2.10.7 Main Turbine Generator System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. The axis of the turbine and generator is oriented within the Turbine Building to be inline with the Reactor and Control Buildings.	7. Inspections will be conducted of the as-built turbine and generator.	7. The axis of the turbine and generator is oriented within the Turbine Building to be inline with the Reactor and Control Buildings.

2.10.9 Turbine Gland Seal System

Design Description

The Turbine Gland Seal (TGS) System prevents the escape of steam from the turbine shaft casing penetrations and valve stems and prevents air inleakage through subatmospheric turbine glands. Figure 2.10.9 shows the basic system configuration.

The TGS System consists of a sealing steam pressure regulator, steam seal header and a gland seal condenser (GSC) with two exhaust blowers.

The TGS System is bounded by the Main Turbine System and the Turbine Bypass System. The TGS System receives steam from either the Turbine Main Steam Systems, the feedwater heater drain tank or auxiliary steam sources. The exhaust blowers discharge to the Turbine Building compartment exhaust system.

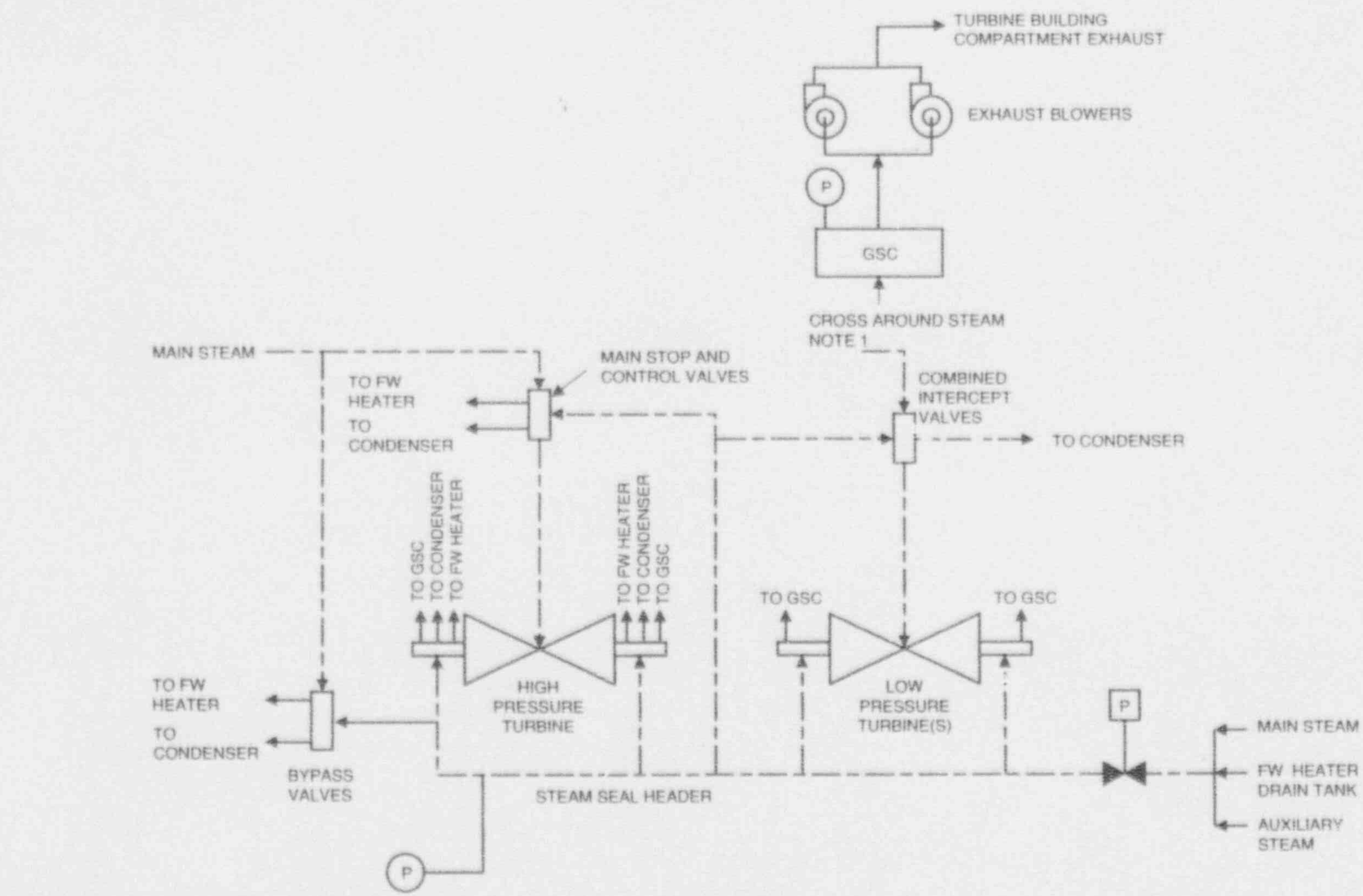
The TGS System is classified as non-safety-related.

The TGS System is located in the Turbine Building.

The TGS System has displays for gland seal condenser and steam seal header pressure in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.10.9 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the TGS System.



NOTES:
1. STEAM PATH BETWEEN HIGH AND LOW PRESSURE TURBINES.

Figure 2.10.9 Turbine Gland Seal

Table 2.10.9 Turbine Gland Seal System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the TGS System is as shown on Figure 2.10.9.	1. Inspections of the as-built system will be conducted.	1. The as-built TGS System conforms with the basic configuration shown on Figure 2.10.9.
2. Main control room displays provided for the TGS System are as defined in Section 2.10.9	2. Inspections will be performed on the main control room displays for the TGS System.	2. Displays exist or can be retrieved in the main control room as defined in Section 2.10.9.

2.10.13 Turbine Bypass System

Design Description

The Turbine Bypass System (TBS) discharges main steam directly to the condenser. The TBS is bounded by the Turbine Main Steam System and the Main Condenser.

The TBS is classified as non-safety-related.

The turbine bypass valves are opened by a signal from the Steam Bypass and Pressure Control System.

The TBS is analyzed to demonstrate the structural integrity under safe shutdown earthquake (SSE) loading conditions.

The TBS is located in the turbine building.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.10.13 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the TBS. The TBS SSE structure integrity evaluation is addressed in Table 2.10.1.

Table 2.10.13 Turbine Bypass System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration for the TBS is described in Section 2.10.13 and Turbine Main Steam System, Figure 2.10.1.	1. Inspections of the as-built TBS will be conducted.	1. The as-built TBS conforms with the basic configuration of Section 2.10.13 and Turbine Main Steam System, Figure 2.10.1.
2. The turbine bypass valves are opened by a signal from the Steam Bypass and Pressure Control System.	2. Tests will be conducted using a simulated signal.	2. Turbine bypass valves open upon receipt of simulated signal from the Steam Bypass and Pressure Control System.

2.10.23 Circulating Water System

Design Description

The Circulating Water (CW) System provides a supply of cooling water to the Main Condenser to remove the heat rejected by the turbine cycle and auxiliary systems. Figure 2.10.23 shows the system basic configuration.

The CW System is classified as non-safety-related.

The CW System is located inside and outside the Turbine Building.

For the CW System, condenser area high level alarm is provided in the main control room (MCR).

To prevent flooding of the Turbine Building, the CW System automatically isolates in the event of 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 level switches.

Inspections, Tests, 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 for the CW System.

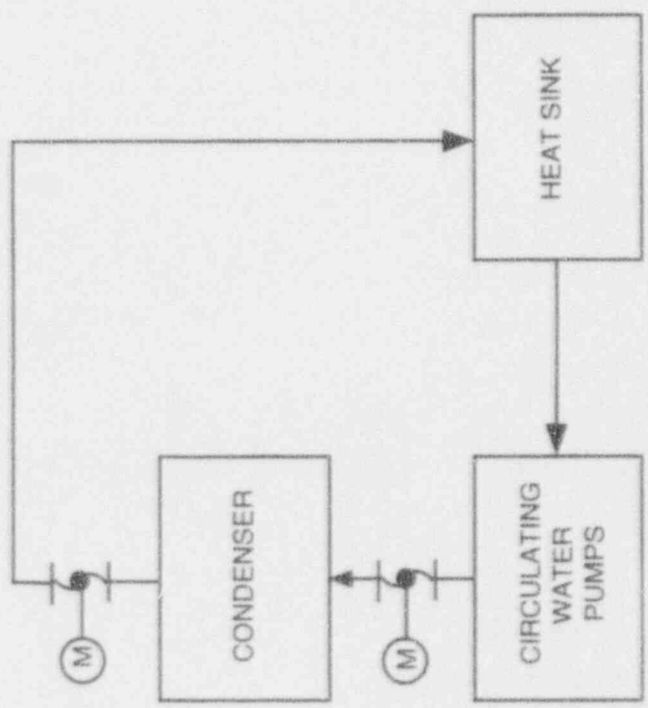


Figure 2.10.23 Circulating Water System

Table 2.10.23 Circulating Water System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. A basic configuration for the CW System is as shown on Figure 2.10.23.	1. Inspections of the as-built system will be conducted.	1. The as-built CW System conforms with the basic configuration shown on Figure 2.10.23.
2. 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 level switches.	2. Testing of the as-built CW System will be performed using simulated signals.	2. 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 level switches.
3. MCR alarms provided for the CW System are as defined in Section 2.10.23.	3. Inspections will be performed on the MCR alarms for the CW System.	3. Alarms exist or can be retrieved in the MCR as defined in Section 2.10.23.

2.11.8 Ultimate Heat Sink

The Ultimate Heat Sink (UHS) is not part of the certified design. Interface requirements for the UHS are provided in Section 4.1.

2.11.11 Station Service Air System

Design Description

The Station Service Air (SA) System consists of two air compressing trains, an air receiver tank, two trains of filters, piping, valves, controls and instrumentation. Figure 2.11.11 shows basic SA System configuration and scope.

The SA System provides compressed air for general plant use. The SA System also provides backup to the Instrument Air (IA) System in the event that IA System pressure is lost.

Except for the containment penetration and isolation valves, the SA System is classified a non-safety-related.

The containment penetration and isolation valves are classified as Seismic Category I. Figure 2.11.11 shows the ASME Code class for the SA System components.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.11.11 provides definition of inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the SA System.

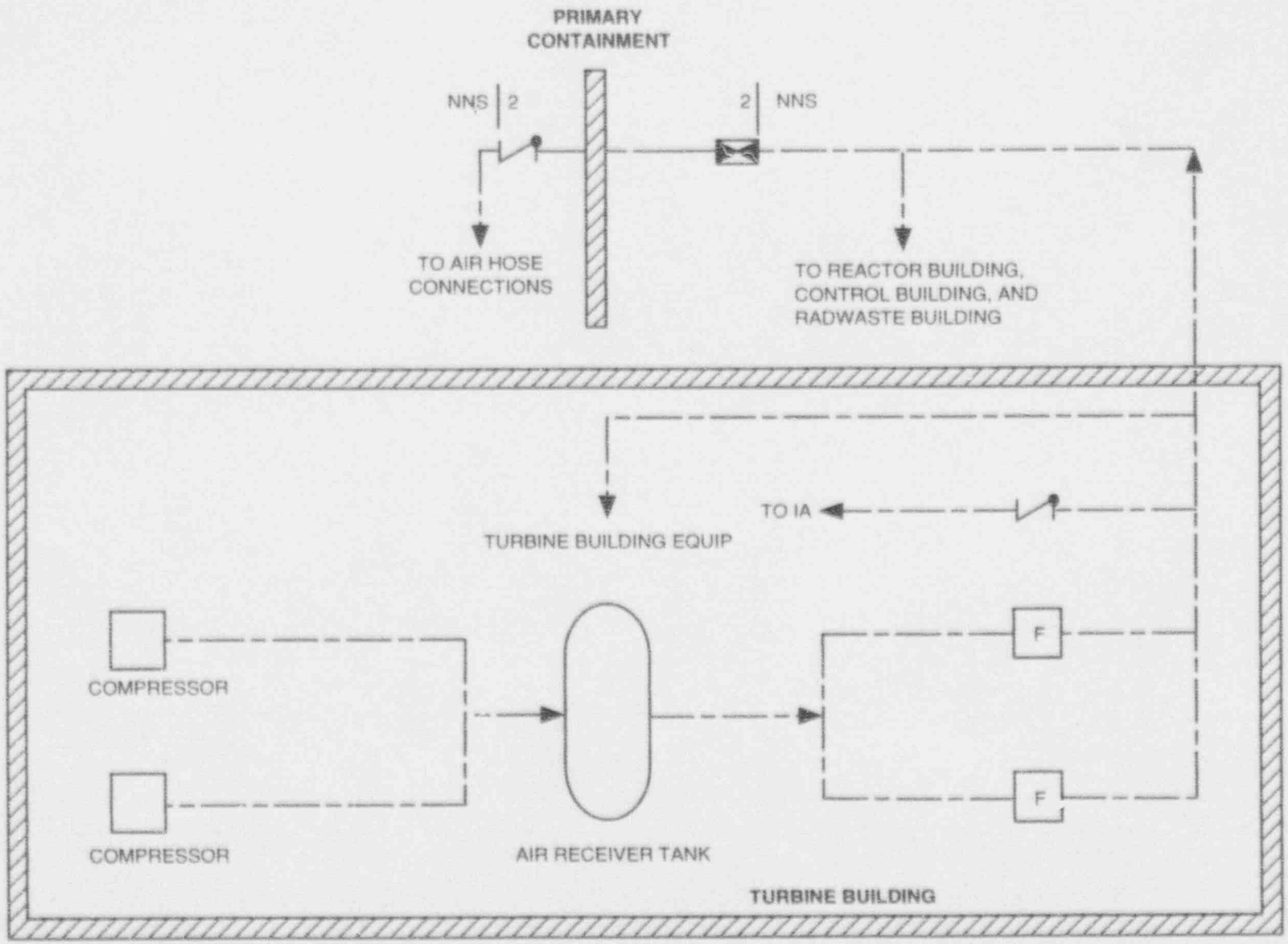


Figure 2.11.11 Station Service Air System

Table 2.11.11 Station Service Air System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the SA System is as shown on Figure 2.11.11.	1. Inspections of the as-built system will be conducted.	1. The as-built SA System conforms with the basic configuration shown on Figure 2.11.11.
2. The ASME Code components of the SA System retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A pressure test will be conducted on those Code components of the SA System required to be pressure tested by the ASME Code.	2. The results of the pressure test of the ASME Code components of the SA System conform with the requirements in ASME Code Section III.

2.11.12 Instrument Air System

Design Description

The Instrument Air (IA) System consists of two air compressing trains, an air receiver tank, two drying trains, piping, valves, controls and instrumentation. Figure 2.11.12 shows the basic IA System configuration and scope.

The IA System provides compressed air for pneumatic equipment, valves, controls and instrumentation outside the primary containment.

The IA System distribution piping penetrates the primary containment. During plant operation, this line is supplied with nitrogen by the High Pressure Nitrogen Gas Supply (HPIN) System. In the event that HPIN System pressure is lost, the IA System provides air backup by remote manual alignment of IA System.

Except for the containment penetration and isolation valves, the IA System is classified as non-safety-related.

The IA containment penetration and isolation valves are classified as Seismic Category I. Figure 2.11.12 shows the ASME Code class for the IA System piping and components.

The IA System containment isolation valve is powered from Class 1E Division I. In the IA System, independence is provided between Class 1E divisions, and also between the Class 1E division and non-Class 1E equipment.

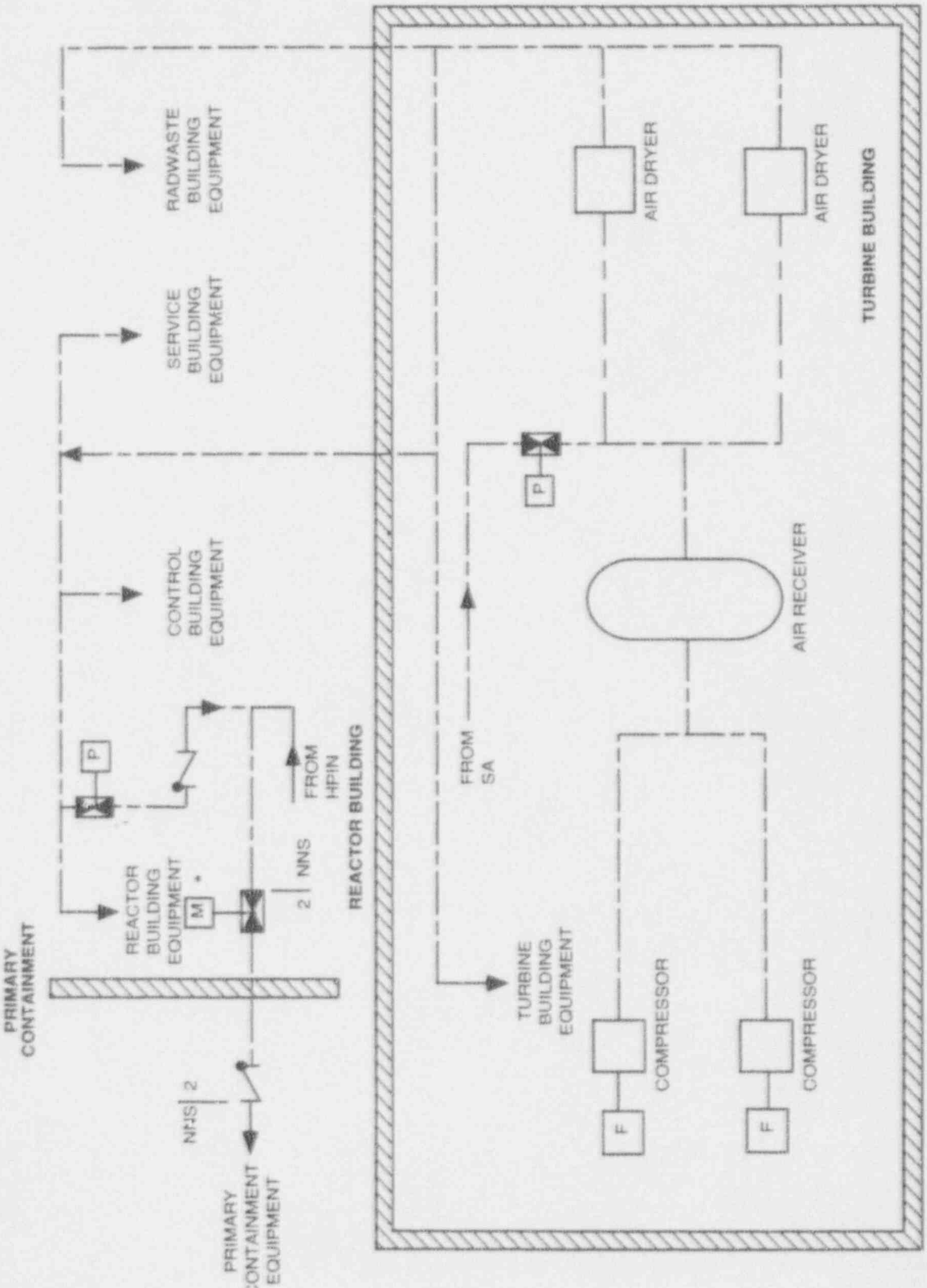
The main control room has controls and open/close status indication for the containment isolation valve.

The safety-related electrical equipment that provides containment isolation and is located outside primary containment in the Reactor Building and is qualified for a harsh environment.

The motor-operated valve (MOV) shown on Figure 2.11.12 has an active safety-related function and closes under differential pressure, fluid flow, and temperature conditions.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.11.12 provides definition of inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the IA System.



* CONTAINMENT ISOLATION VALVE IS POWERED FROM CLASS 1E DIVISION I.

Figure 2.11.12 Instrument Air System

Table 2.11.12 Instrument Air System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the IA System is shown on Figure 2.11.12.	1. Inspections of the as-built IA System will be conducted.	1. The as-built IA System conforms with the basic configuration shown on Figure 2.11.12.
2. The ASME Code components of the IA System retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A pressure test will be conducted on those Code components of the IA System required to be pressure tested by the ASME Code.	2. The results of the pressure test of the ASME Code components of the IA System conform with the requirements in ASME Code Section III.
3. In the IA System, independence is provided between Class 1E divisions and also between the Class 1E division and non-Class 1E equipment.	3a. Tests will be performed on the IA System by providing a test signal in only one Class 1E division at a time.	3a. The test signal exists in the IA System only when the signal is applied to the division associated with the IA System.
	3b. Inspection of the as-installed Class 1E division in the IA System will be performed	3b. In the IA System, physical separation exists between the Class 1E division and non-Class 1E equipment.
4. Main control room displays and controls provided for the IA System are as defined in Section 2.11.12.	4. Inspections will be performed on the main control room displays and controls for the IA System.	4. Displays and controls exist or can be retrieved in the main control room as defined in Section 2.11.12.
5. The MOV designated in Section 2.11.12 as having an active safety-related function closes under differential pressure, fluid flow, and temperature conditions.	5. Closing tests of the installed valve will be conducted under preoperational differential pressure, fluid flow, and temperature conditions.	5. The MOV closes.

2.11.13 High Pressure Nitrogen Gas Supply System

Design Description

The High Pressure Nitrogen Gas Supply (HPIN) System provides nitrogen to pneumatic equipment inside the primary containment. Figure 2.11.13 shows the basic HPIN System configuration and scope.

The HPIN System consists of:

(1) Two divisional systems (Divisions A and B) which are supplied from bottled nitrogen supplies. These systems can supply nitrogen to the automatic depressurization system (ADS) accumulators on the safety/relief valves (SRVs).

(2) A non-divisional system that is supplied from the Atmospheric Control (AC) System. This system can supply nitrogen to the non-ADS and ADS accumulators on the SRVs.

The two divisional systems and the containment penetrations and isolation valves on the non-divisional system are classified as safety-related.

During operation, all SRV accumulators are supplied from the non-divisional system. If the pressure sensor in either of the safety-related systems indicates low pressure, the valve between that system and the non-divisional system closes and the supply valve to the bottled nitrogen supply in that division opens. If the pressure sensor in the non-divisional system indicates a low pressure, the valves between the non-divisional and the divisional systems close.

The two divisional systems and the containment penetration and isolation valves in the non-divisional system are classified as Seismic Category I. Figure 2.11.13 shows the ASME Code class for the HPIN System piping and components.

Except for the isolation valves and distribution piping inside the primary containment, the HPIN System is located in the Reactor Building.

Each of the two HPIN divisions is powered from the respective Class 1E division as shown on Figure 2.11.13. In the HPIN System, and independence is provided between the Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment.

Outside the primary containment and except for the interconnection through the non-divisional system, each mechanical division (Divisions A and B) is physically separated from the other division.

The HPIN System has the following displays and controls in the main control room:

(1) Parameter displays for the sensors shown on Figure 2.11.13.

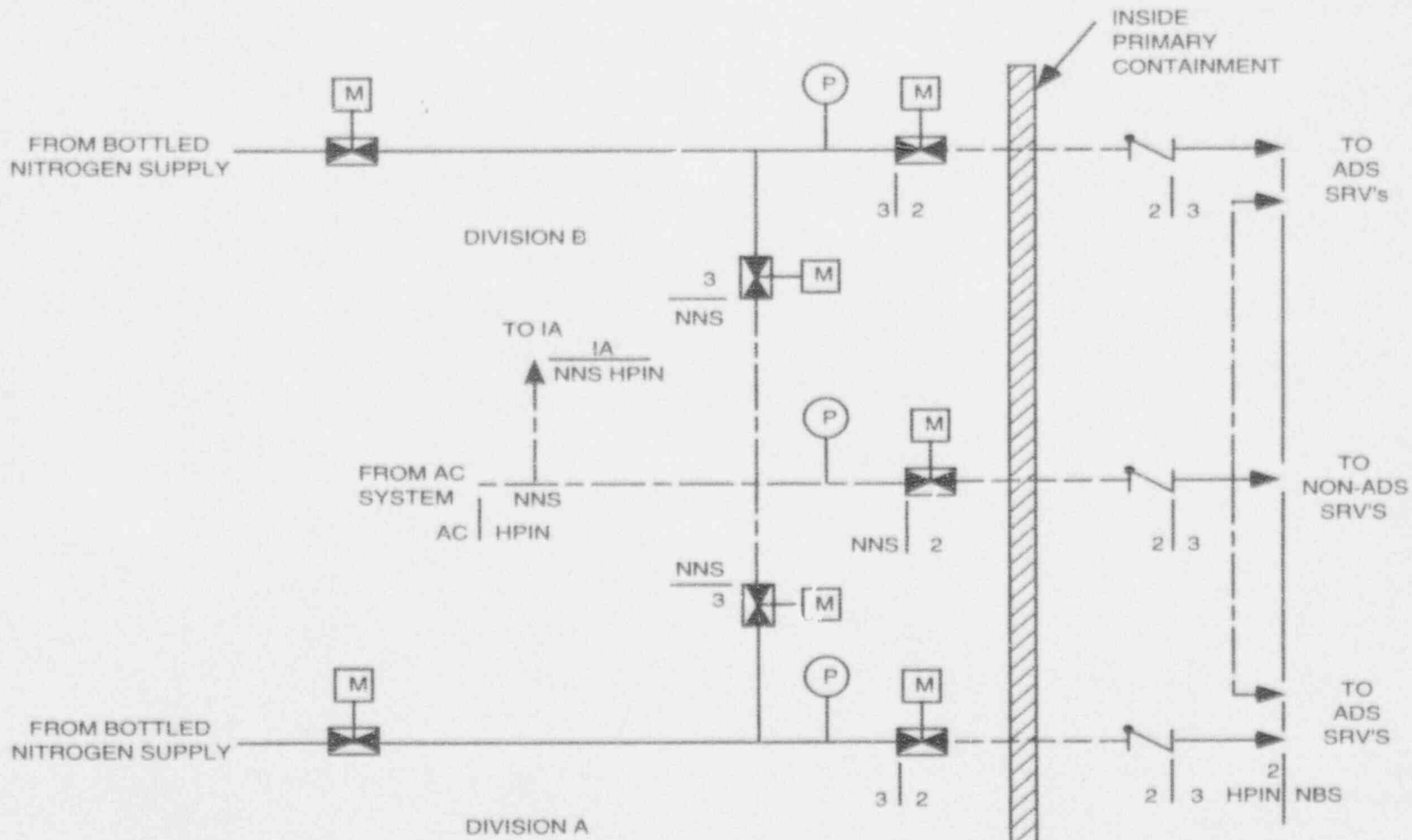
(2) Control and status indication for the active safety-related components shown on Figure 2.11.13.

The safety-related electrical equipment shown on Figure 2.11.13 located in the Reactor Building is qualified for a harsh environment.

The motor operated valves (MOVs) shown on Figure 2.11.13 have active safety-related functions and perform these functions under differential pressure, fluid flow, and temperature conditions.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.11.13 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the HPIN System.



NOTES:
1. HPIN SYSTEM DIVISION-A IS POWERED FROM CLASS 1E DIVISION I, AND HPIN SYSTEM DIVISION-B IS POWERED FROM CLASS 1E DIVISION II. THE CONTAINMENT ISOLATION VALVE OF NON-DIVISIONAL PORTION OF THE HPIN SYSTEM IS POWERED FROM CLASS 1E DIVISION I.

Figure 2.11.13 High Pressure Nitrogen Gas Supply System Schematic

Table 2.11.13 High Pressure Nitrogen Gas Supply System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the HPIN System is as shown on Figure 2.11.13.	1. Inspections of the as-built system will be conducted.	1. The as-built HPIN System conforms with the basic configuration shown on Figure 2.11.13.
2. The ASME Code components of the HPIN System retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A pressure test will be conducted on those code components of the HPIN System required to be pressure tested by the ASME Code.	2. The results of the pressure test of the ASME Code components of the HPIN System conform with the requirements in ASME Code Section III.
3. If the pressure sensor in either of the safety-related systems indicates low pressure, the valve between that system and the non-divisional system closes and the supply valve to the bottled nitrogen supply in that division opens.	3. Tests will be conducted on each division of the as-built HPIN System using simulated pressure signals.	3. If the pressure sensor in either of the safety-related systems indicates low pressure, the valve between that system and the non-divisional system closes and the supply valve to the bottled nitrogen supply in that division opens.
4. If the pressure sensor in the non-divisional system indicates a low pressure, the valves between the non-divisional and the divisional systems close.	4. Tests will be conducted on the as-built HPIN system using simulated pressure signals.	4. If the pressure sensor in the non-divisional system indicates a low pressure, the valves between the non-divisional and the divisional systems close.
5. In the HPIN System, independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	5a. Tests will be performed in the HPIN System by providing a test signal in only Class 1E division at a time.	5a. The test signal exists only in the Class 1E division under test in the HPIN System.
	5b. Inspections of the as-installed Class 1E divisions in the HPIN System will be performed.	5b. In the HPIN System, physical separation exists between Class 1E divisions. Physical separation exists between these Class 1E divisions and non-Class 1E equipment.
6. Outside the primary containment and except for the interconnection through the non-divisional system, each mechanical division (Divisions A and B) of the HPIN System is physically separated from the other division.	6. Inspections of the as-built HPIN System will be conducted.	6. Outside the primary containment and except for the interconnection through the non-divisional system, each mechanical division (Divisions A and B) of the HPIN System is physically separated from the other division by structural and/or fire barriers.

Table 2.11.13 High Pressure Nitrogen Gas Supply System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. Main control room displays and controls provided for the HPIN System are as defined in Section 2.11.13.	7. Inspections will be performed on the main control room displays and controls for the HPIN System.	7. Displays and controls exist or can be retrieved in the main control room as defined in Section 2.11.13.
8. MOVs designated in Section 2.11.13 as having an active safety-related function open or close under differential pressure, fluid flow, and temperature conditions.	8. Opening and closing tests of installed valves will be conducted under preoperational differential pressure, fluid flow, and temperature conditions.	8. Each MOV opens or closes.

2.12.13 Emergency Diesel Generator System (Standby AC Power Supply)

Design Description

The Emergency Diesel Generator (DG) System consists of three diesel engines and their respective combustion air intake system, starting air system, fuel oil system (from the day tank to the engine), lubricating oil system, engine jacket cooling water system, engine exhaust system and silencer, governor system, and generator with its excitation and voltage regulation systems.

The three DGs are classified as Class 1E, safety-related and supply standby AC power to their respective Class 1E Electrical Power Distribution (EPD) System divisions (Divisions I, II, and III). The DG interfaces to the EPD System are shown on Figure 2.12.1.

The DG units and their auxiliary systems are classified Seismic Category I and Class 1E, and are located in their respective divisional areas in the Reactor Building. The DG combustion air intakes are located above the maximum flood level. The DG combustion air intakes are separated from DG exhaust ducts. Each divisional DG (Divisions I, II, and III) with its auxiliary systems is physically separated from the other divisions. Electrical independence is provided between Class 1E divisions and between Class 1E divisions and non-Class 1E equipment.

The DGs are sized to supply their load demand following a Lose-of-Coolant Accident (LOCA). The DG air start receiver tanks are sized to provide five DG starts without recharging the tanks.

A Loss of Preferred Power (LOPP) signal (bus under-voltage) from an EPD System medium voltage divisional bus automatically starts its respective DG, and initiates automatic connection of the DG to its divisional bus. A DG automatically connects to its respective bus when DG rated voltage and frequency conditions are established. After a DG connects to its respective bus, the non-accident loads are automatically sequenced onto the bus.

LOCA signals from the Residual Heat Removal (RHR) Systems automatically start their respective divisional DG. After starting, the DGs remain in a standby mode (i.e. running at rated voltage and frequency, but not connected to their busses), unless a LOPP signal exists. When LOCA and LOPP signals exist, the DG automatically connects to its respective divisional bus. After a DG connects to its respective bus, the LOCA loads are automatically sequenced onto the bus. DGs can start and run their largest motor load at the end of the automatic loading sequence.

A manual start signal from the main control room (MCR) starts a DG. After starting, the DG remains in a standby mode, unless a LOPP signal exists.

DGs start, attain rated voltage and frequency, and are ready to load in 20 seconds after receiving an automatic or manual start signal.

When a DG is operating in parallel (test mode) with off-site power, a LOPP or a LOCA signal overrides the test mode by disconnecting the DG from its respective divisional bus.

Except for the DG engine overspeed trip and the generator differential relay trip, the DG unit protective trip circuits are bypassed by a LOCA signal or are provided with independent measurements and coincident trip logic.

The DG System has the following displays and controls in the MCR. (1) displays for the DG output voltage, amperes, watts, vars, frequency, and engine speed. (2) controls for manually starting and stopping the DG units.

The DG System has displays at the Remote Shutdown System (RSS) for DC run and stop indication.

Inspections, Tests, 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 (DG) System.

Table 2.12.13 Emergency Diesel Generator System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the DG System is described in Section 2.12.13.	1. Inspection of the as-built system will be conducted.	1. The as-built DG System conforms with the basic configuration described in Section 2.12.13.
2. Each divisional DG (Divisions I, II, and III) with its auxiliary systems is physically separated from the other divisions.	2. Inspection of the as-built DG Systems will be conducted.	2. Each DG with its auxiliary systems is physically separated from the other divisions by structural and/or fire barriers.
3. Electrical independence is provided between Class 1E divisions and between Class 1E divisions and non-Class 1E equipment.	3. Tests will be conducted in the as-built DG Systems by providing a test signal in only one Class 1E division at a time.	3. The test signal exists in only the Class 1E division under test in the DG System.
4. The DGs are sized to supply their load demand following a LOCA.	4. Analyses based on the as-built DG load profile will be performed.	4. Analyses exist and conclude that the as-built DG System capacities exceed, as determined by their nameplate ratings, their load demand following a LOCA.
5. DG air start receiver tanks have capacity for five DG starts without recharging the tanks.	5. Tests will be conducted in the as-built DG Systems by starting the DGs five times.	5. As-built DGs will start five times without recharging the air start receiver tanks.
6. A LOPP signal (bus under-voltage) from an EPD System medium voltage divisional bus automatically starts its respective DG, and initiates automatic connection of the DG to its divisional bus. A DG automatically connects to its respective bus when DG rated voltage and frequency conditions are established. After a DG connects to its respective bus, the non-accident loads are automatically sequenced onto the bus.	6. Tests will be conducted in the as-built DG Systems by providing a simulated LOPP signal.	6. As-built DGs automatically start on receiving a LOPP signal, attain rated voltage (±10%), and rated frequency (±2%) in ≤ 20 seconds, automatically connect to their respective divisional bus, and sequence their non-accident loads onto the bus.

Table 2.12.13 Emergency Diesel Generator System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. LOCA signals from the RHR Systems automatically start their respective divisional DG. After starting, the DGs remain in a standby mode (i.e. running at rated voltage and frequency, but not connected to their busses), unless a LOPP signal exists.	7. Tests will be conducted in the as-built DG Systems by providing a simulated LOCA signal, without a LOPP signal.	7. As-built DGs automatically start on receiving a LOCA signal, attain rated voltage (±10%), and rated frequency (±2%) in ≤ 20 seconds, and remain in the standby mode.
8. When LOCA and LOPP signals exist, the DG automatically connects to its respective divisional bus. After a DG connects to its respective bus, the LOCA loads are automatically sequenced onto the bus.	8. Tests will be conducted in the as-built DG System by providing simulated LOCA and LOPP signals.	8. When LOCA and LOPP signals exist, the DG automatically connects to its respective divisional bus. After a DG connects to its respective bus, the LOCA loads are automatically sequenced onto the bus.
9. DGs can start and run their largest motor load at the end of the automatic loading sequence.	9. Tests will be conducted in the as-built DG System by tripping the largest motor load, after load sequencing is completed, and then restarting it.	9. As-built DGs can start and run their largest motor load at the end of the automatic loading sequence, and bus voltage and frequency recover to rated voltage (±10%) and rated frequency (±2%) within 10 seconds.
10. A manual start signal from the MCR starts a DG. After starting, the DG remains in a standby mode, unless a LOPP signal exists.	10. Tests will be conducted in the as-built DG System by providing a manual start signal from the MCR, without a LOPP signal.	10. As-built DGs automatically start on receiving a manual start signal from the MCR, attain rated voltage (±10%), and rated frequency (±2%) in ≤ 20 seconds and remain in the standby mode.
11. When a DG is operating in parallel (test mode) with off-site power, a LOPP or a LOCA signal overrides the test mode by disconnecting the DG from its respective divisional bus.	11. Tests will be conducted in the as-built DG Systems by providing LOPP and LOCA signals while operating the DGs in the test mode.	11. When operating in the test mode with off-site power and a LOPP or a LOCA signal is received, as-built DGs automatically disconnect from their respective divisional buses.
12. Except for the DG engine overspeed trip and the generator differential relay trip, the DG unit protective trip circuits are bypassed by a LOCA signal or are provided with independent measurements and coincident trip logic.	12. Tests will be conducted in the as-built DG Systems by providing simulated test signals in only one Class 1E protective circuit at a time.	12. Test signals, except for the DG engine overspeed trip and the generator differential relay trips, do not trip the DG.

Table 2.12.13 Emergency Diesel Generator System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
13. MCR displays and controls provided for the DG System are as defined in Section 2.12.13	13. Inspections will be conducted on the MCR displays and controls for the as-built DG Systems.	13. Displays and controls exist or can be retrieved in the MCR as defined in Section 2.12.13.
14. RSS displays provided for the DG System are as defined in Section 2.12.13	14. Inspections will be conducted on the RSS displays for the as-built DG Systems.	14. Displays exist or can be retrieved on the RSS as defined in Section 2.12.13.

2.14.6 Atmospheric Control System

Design Description

The Atmospheric Control (AC) System consists of a nitrogen supply, injection lines, exhaust lines, bleed line, valves, controls, and instrumentation. The AC System also has the containment overpressure protection system. Figure 2.14.6 shows the basic system configuration and scope.

The AC System provides an inert atmosphere within the primary containment during plant operation.

Except for the primary containment penetrations isolation valves, and suppression pool level sensors, the AC System is classified as non-safety-related.

The AC primary containment penetrations, isolation valves, and suppression pool level sensors are classified as Seismic Category I. Figure 2.14.6 shows the ASME Code class for the AC System piping and components.

AC System components are located in the Reactor Building, except for the nitrogen supply.

Figure 2.14.6 shows the Class 1E divisional power assignments for the AC System components. In the AC System, the independence is provided between the Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment.

The main control room has control and open/close status indication for the containment isolation valves.

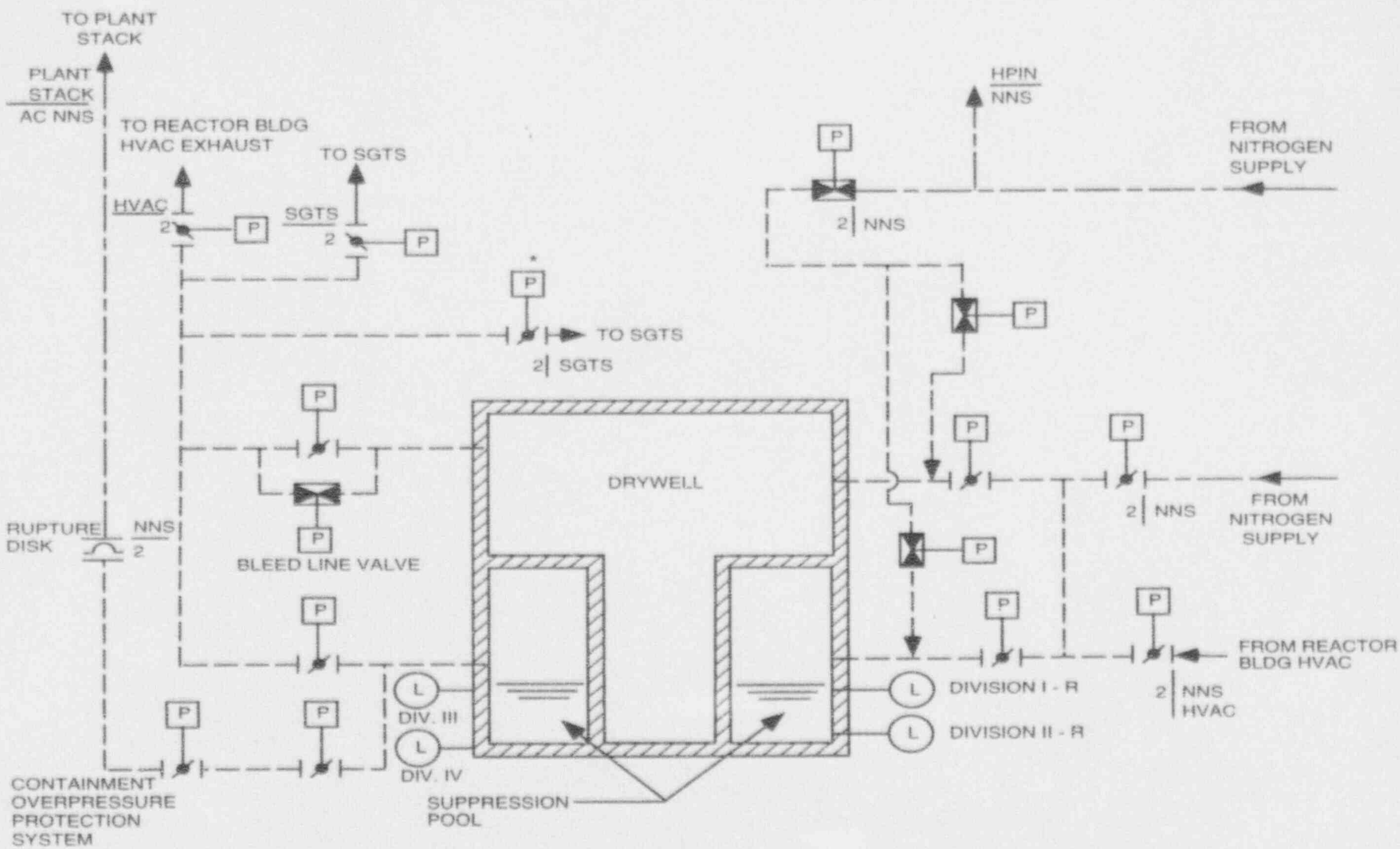
AC System components with display interfaces with the Remote Shutdown System (RSS) are shown on the Figure 2.14.6.

The safety-related electrical equipment located in the Reactor Building is qualified for a harsh environment.

The two valves in the containment overpressure protection system fail open on loss of pneumatic pressure or loss of electrical power to the valve actuating solenoid. The other pneumatic valves shown on Figure 2.14.6 fail close on loss of pneumatic pressure or loss of electrical power to the valve actuating solenoids.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.14.6 provides a definition of the inspections, tests and/or analyses, together with associated criteria, which will be undertaken for the AC System.



NOTES:

1. INBOARD CONTAINMENT ISOLATION VALVES ARE POWERED FROM CLASS 1E DIVISION II OUTBOARD CONTAINMENT ISOLATION VALVES ARE POWERED FROM CLASS 1E DIVISION I EXCEPT AS NOTED WITH "*" WHICH IS POWERED FROM CLASS 1E DIVISION III

Figure 2.14.6 Atmospheric Control System

Table 2.14.6 Atmospheric Control System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the AC System is as shown on Figure 2.14.6.	1. Inspections of the as-built AC System will be conducted.	1. The as-built AC System conforms with the basic configuration shown on Figure 2.14.6.
2. The ASME Code components of the AC System retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A pressure test will be conducted on those code components of the AC System required to be pressure tested by the ASME Code.	2. The results of the pressure test of the ASME Code components of the AC System conform with the requirements in ASME Code Section III.
3. In the AC System, independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	3a. Tests will be performed in the AC System by providing a test signal in only Class 1E division at a time.	3a. The test signal exists only in the Class 1E division under test in the AC System.
	3b. Inspection of the as-installed Class 1E divisions in the AC System will be performed.	3b. In the AC System physical separation exists between Class 1E divisions. Physical separation exists between these Class 1E divisions and non-Class 1E equipment.
4. Main control room displays and controls provided for the AC System are as defined in Section 2.14.6.	4. Inspections will be performed on the main control room displays and controls for the AC System.	4. Displays and controls exist or can be retrieved in the main control room as defined in Section 2.14.6.
5. RSS displays provided for the AC System are as defined in Section 2.14.6.	5. Inspections will be performed on the RSS displays for the AC System.	5. Displays exist on the RSS as defined in Section 2.14.6.
6. The two valves in the containment overpressure protection system fail open on loss of pneumatic pressure or loss of electrical power to the valve actuating solenoid. The other pneumatic valves shown on Figure 2.14.6 fail close on loss of pneumatic pressure or loss of electrical power to the valve actuating solenoids.	6. Tests will be conducted on the as-built AC System pneumatic valves.	6. The two valves in the containment overpressure protection system fail open on loss of pneumatic pressure or loss of electrical power to the valve actuating solenoid. The other pneumatic valves shown on Figure 2.14.6 fail close on loss of pneumatic pressure or loss of electrical power to the valve actuating solenoids.

2.14.7 Drywell Cooling System

Design Description

The Drywell Cooling (DWC) System circulates the drywell atmosphere through coolers, thus maintaining its temperature during plant operation. Figure 2.14.7 shows the basic system configuration and scope.

The DWC system consists of three fan coil units and two chilled water units. Each fan coil unit consists of a cooling coil and a fan. These units are cooled by the Reactor Building Cooling Water (RCW) System. Each chilled water unit consists of a cooling coil only. These units are cooled by the Heat Ventilating and Air Conditioning Normal Cooling (HNCW) System.

The DWC System is classified as a non-safety-related.

The DWC System is located inside the drywell.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.14.7 provides a definition of the inspections, tests and/or analyses, together with associated acceptance criteria, which will be undertaken for the DWC System.

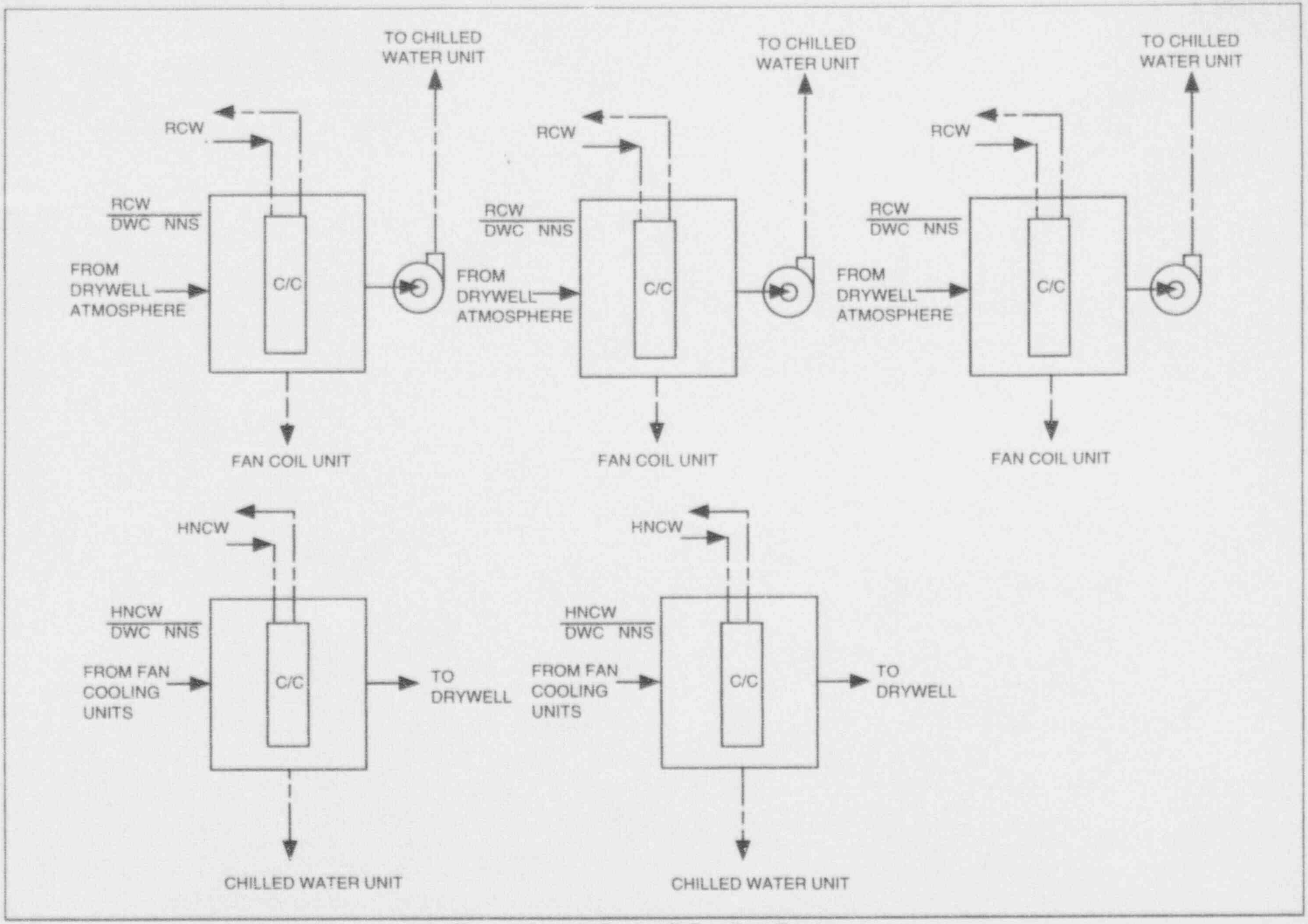


Figure 2.14.7 Drywell Cooling System

Table 2.14.7 Drywell Cooling System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the DWC System is as shown on Figure 2.14.7.	1. Inspections of the as-built system configuration will be conducted.	1. The as-built DWC System conforms with the basic configuration shown in Figure 2.14.7.

2.14.8 Flammability Control System

Design Description

The Flammability Control System (FCS) is provided to control the potential buildup of hydrogen and oxygen in the containment from radiolysis of water after a design basis loss-of-coolant accident (LOCA). The system consists of two independent and redundant hydrogen and oxygen recombiners. Figure 2.14.8 shows the basic system configuration and scope.

The FCS is classified as safety-related.

After a LOCA, the system can be manually actuated from the main control room if high oxygen concentrations exist in the primary containment. Each recombiner removes gas from the drywell, recombines the oxygen with hydrogen, and returns the gas mixture, along with the condensate to the wetwell.

The system is classified as Seismic Category I. Figure 2.14.8 shows ASME Code class for the FCS piping and components.

Each of the two FCS divisions is powered from the respective Class 1E division as shown on Figure 2.14.8. In the FCS, independence is provided between Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment.

Each mechanical division of the FCS (Divisions B and C) is physically separated from the other division.

The FCS has the following displays and controls in the main control room:

(1) Controls and status indication for the valves shown on Figure 2.14.8.

(2) Controls and status indication for the recombiner unit.

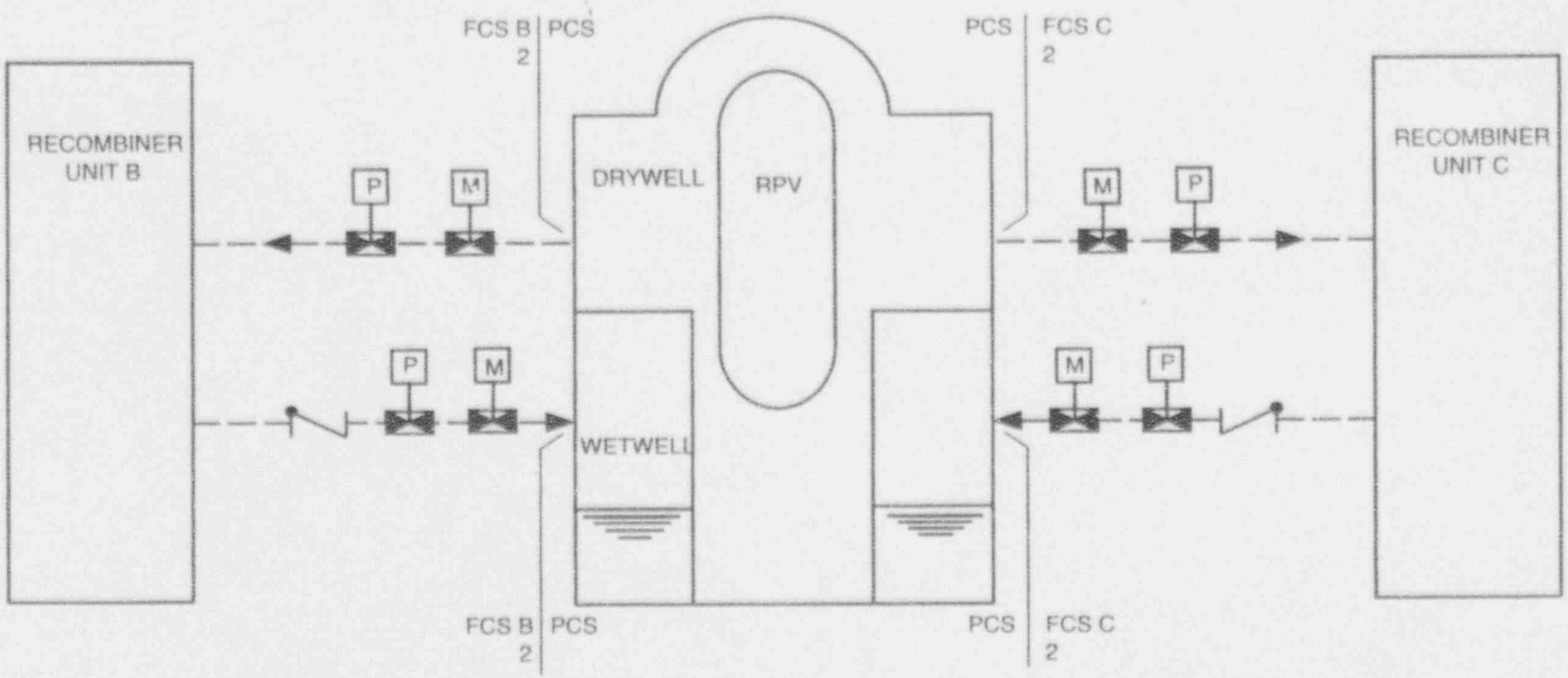
The safety-related electrical equipment shown on Figure 2.14.8, and included in the recombiner units, is qualified for a harsh environment.

The motor operated valves (MOVs) shown on Figure 2.14.8 and active safety-related MOVs in the recombiners, if any, have active safety-related functions and perform these functions under differential pressure, fluid flow, and temperature conditions.

The pneumatic valves shown on Figure 2.14.8 fail to close in the event of loss of pneumatic pressure or loss of electrical power to the valve actuating solenoids.

Inspections, Tests, Analyses and Acceptance Criterial

Table 2.14.8 provides a definition of the inspections, tests, and/or ana'yses, together with associated acceptance criteria, which will be undertaken for the FCS.



NOTE:
1. CLASS 1E ELECTRICAL POWER FOR FCS UNIT B IS SUPPLIED FROM DIVISION II EXCEPT FOR THE PNEUMATIC ISOLATION VALVE DUAL SOLENOIDS, WHICH IS DIVISIONS I AND III. UNIT C IS SUPPLIED FROM DIVISION III EXCEPT FOR THE OUTBOARD PNEUMATIC ISOLATION VALVE DUAL SOLENOIDS, WHICH IS DIVISION I AND II.

Figure 2.14.8 Flammability Control System

Table 2.14.8 Flammability Control System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration for the FCS is as shown on Figure 2.14.8.	1. Inspections of the as-built system will be conducted.	1. The as-built FCS conforms with the basic configuration shown on Figure 2.14.8.
2. The ASME Code components of the FCS retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A pressure test will be conducted on those Code components of the FCS required to be pressure tested by the ASME code.	2. The results of the pressure test of the ASME code components of the FCS conform with the requirements in the ASME Code, Section III.
3. In the FCS, independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	3a. Tests will be performed in the FCS by providing a test signal in only one Class 1E division at a time.	3a The test signal exists only in the Class 1E division under test in the FCS.
	3b. Inspection of the as-installed Class 1E divisions in the FCS will be performed.	3b. Physical separation exists between Class 1E divisions in the FCS. Physical separation exists between Class 1E divisions and non-Class 1E equipment in the FCS.
4. Each mechanical division of the FCS (Divisions B, C) is physically separated from the other divisions.	4. Inspections of the as-built FCS will be conducted.	4. Each mechanical division of the FCS is physically separated from the other mechanical divisions of FCS by structural and/or fire barriers.
5. Main control room displays and controls provided for the FCS are as defined in Section 2.14.8.	5. Inspections will be performed on the main control room displays and controls for the FCS.	5. Displays and controls exist or can be retrieved in the main control room as defined in Section 2.14.8.
6. MOVs designated in Section 2.14.8 as having an active safety-related function open and/or close under differential pressure and fluid flow and temperature conditions.	6. Opening and/or closing tests of installed valves will be conducted under preoperational differential pressure, fluid flow, and temperature conditions.	6. Each MOV opens and/or closes.
7. The pneumatic valves shown on Figure 2.14.8 fail close in the event of loss of pneumatic pressure or loss of electrical power to the valve actuating solenoid.	7. Tests will be conducted on the as-built FCS pneumatic valves.	7. The pneumatic valves shown on Figure 2.14.8 fail close in the event of loss of pneumatic pressure or loss of electrical power to the valve actuating solenoid.

2.15.3 Cranes and Hoists

Design Description

Cranes and Hoists are used for maintenance and refueling tasks.

During refueling/servicing, the Reactor Building (R/B) crane handles the shield plugs, drywell and reactor vessel heads, and the steam dryer/separators. The minimum crane coverage includes the R/B refueling floor laydown area, and the R/B equipment storage pit. During plant operation the crane handles new fuel shipping containers and the spent fuel shipping casks. For these activities, the minimum crane coverage includes the new fuel vault, the R/B equipment hatches, and the spent fuel cask loading and washdown pits.

The upper drywell hoists are used during outages to service valves and equipment inside the upper drywell.

The lower drywell hoists service valves and equipment inside the Lower Drywell during outages.

The cranes and hoists are classified as non-safety-related.

The R/B crane is interlocked to prevent movement of heavy loads over the spent fuel storage portion of the spent fuel storage pool. The hoisting and braking system of the R/B crane are redundant.

The R/B crane has a lifting capacity greater than or equal to the heaviest expected load.

The upper drywell hoists and lower drywell hoists are classified as Seismic Category I.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.3 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Cranes and Hoists.

Table 2.15.3 Cranes and Hoists

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Equipment comprising the Cranes and Hoists System is described in Section 2.15.3.	1. Inspection of the as-built system will be conducted.	1. The as-built Cranes and Hoists System conforms with the description in Section 2.15.3
2. The R/B crane is interlocked to prevent movement of heavy loads over the spent fuel storage portion of the spent fuel storage pool.	2. Tests will be conducted of the as-built R/B crane movement using a heavy load.	2. The R/B crane interlock prevents the carrying of a load greater than one fuel assembly and its associated handling devices over the spent fuel storage portion of the spent fuel storage pool.
3. The R/B crane has a lifting capacity greater than or equal to the heaviest expected load.	3. Tests of the as-built R/B crane will be conducted.	3. The R/B crane meets: a. A static load test at 125% of rated load. b. A full operational test at 100% of rated load.

2.15.6 Fire Protection System

Design Description

The Fire Protection System (FPS) detects, alarms and extinguishes fires. Fire detection and alarm systems are provided in fire areas. The FPS consists of sprinkler systems, standpipes and hose reels, and portable extinguishers. The foam systems are also used for special applications. The basic configuration of the FPS water supply system is shown on Figure 2.15.6.

Areas covered by sprinklers or foam systems are also covered by the manual hose system. Areas covered only by manual hoses can be reached from at least two hose stations. A hose reel and fire extinguisher are located no greater than 30.5 m from any location within the buildings.

The FPS is classified as a non-safety-related. The sprinkler systems and the standpipe systems in the Reactor and Control Buildings and portions of the FPS water supply system identified in Figure 2.15.6 remain functional following a safe shutdown earthquake (SSE). These portions of the water supply are separated from the remainder of the system by valves as shown in Figure 2.15.6.

Fresh water is used for the water supply system. Two sources with a minimum capacity of 1140 m^3 for each source are provided. A minimum of 456 m^3 is reserved for use by the portion of the suppression system used for the Reactor and Control Buildings. The FPS for the Reactor and Control Buildings supplies a minimum flow of 1890 liters/min at a pressure greater than 4.57 kg/cm^2g at the most hydraulically remote hose connection.

A fire water supply connection to the Residual Heat Removal System piping is provided from the portion of the FPS used for Reactor and Control Buildings.

Automatic foam water extinguishing systems are provided for the diesel generator rooms and day tank rooms.

Fire detection and alarm systems are supplied with power from a non-Class 1E uninterruptible power supply.

The FPS has the following displays and alarms in the Main Control Room (MCR):

(1) Detection system fire alarms.

(2) Status of FPS pumps.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.6 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Fire Protection System.

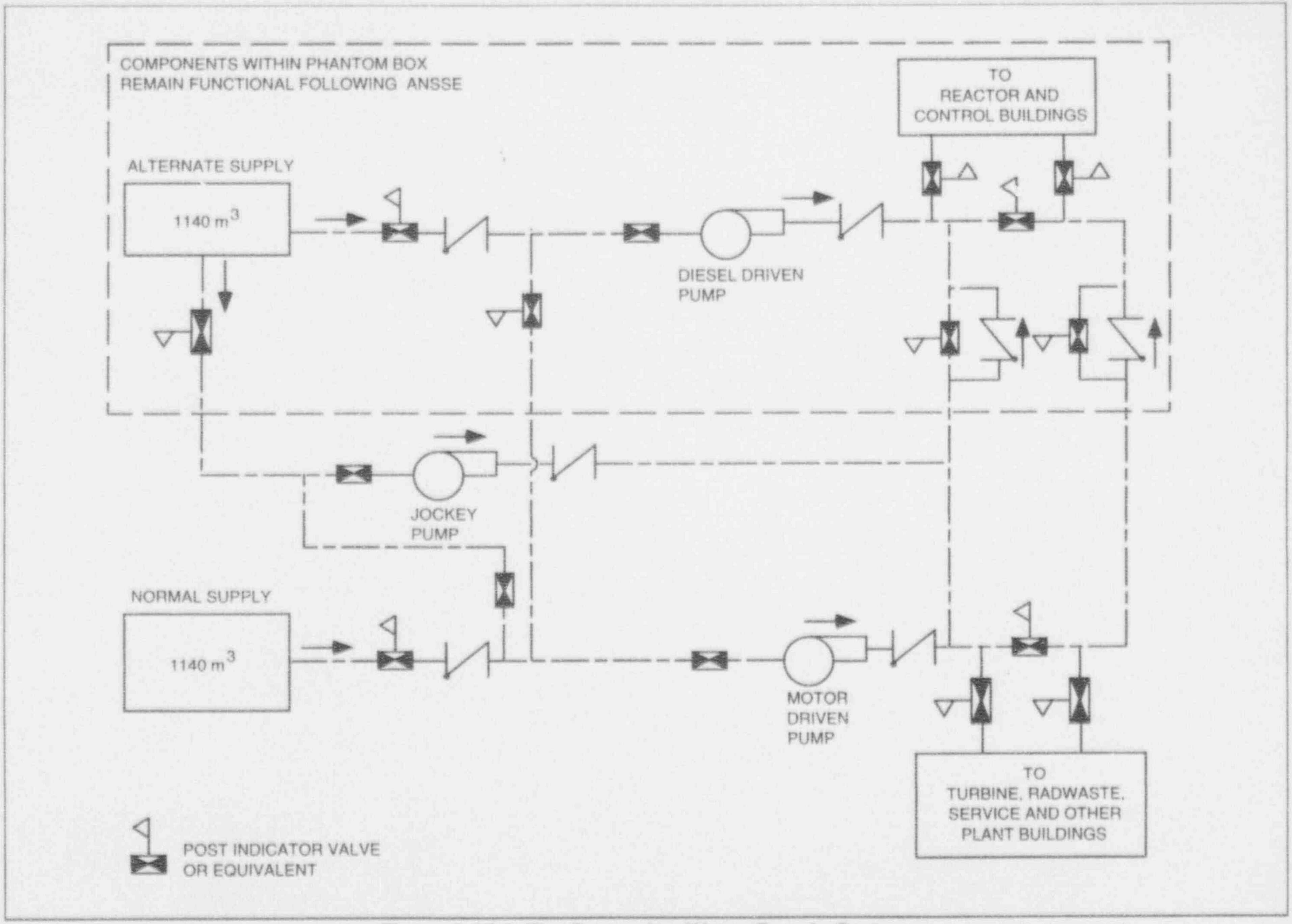


Figure 2.15.6 Fire Protection Water Supply System

Table 2.15.6 Fire Protection System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration for the FPS is defined in Section 2.15.6	1. Inspections of the as-built FPS will be conducted.	1. 1. The as-built configuration of the FPS is in accordance with Section 2.15.6.
2. The FPS for the Reactor and Control Buildings supplies a minimum flow of 1890 liters/min. at a pressure greater than 4.57 kg/cm^2 at the most hydraulically remote hose connection.	2. Tests will be conducted of the as-built FPS.	2. The FPS for the Reactor and Control Buildings supplies a minimum flow of 1890 liters/min. at a pressure greater than 4.57 kg/cm^2 at the most hydraulically remote hose connection.
3. Automatic foam-water extinguishing systems are provided for the diesel generator and day tank rooms.	3. Inspections of the as-built foam-water extinguishing systems will be conducted. The automatic logic will be tested using simulated fire conditions.	3. The automatic foam-water suppression system are present and initiation logic is actuated under simulated fire conditions.
4. The sprinkler systems and the standpipe systems in the Reactor and Control Buildings and the portions of the FPS water supply system identified in Figure 2.15.6 remain functional following an SSE.	4. Seismic analysis of the as-built FPS will be performed.	4. An analysis report exists which concludes that as-built sprinkler systems and the standpipe systems in the Reactor and Control Buildings and the portions of the FPS water supply system identified in Figure 2.15.6 remain functional following an SSE.
5. The fire detection and alarm systems are supplied with power from a non-Class 1E uninterruptible power supply.	5. Inspections of the as-built FPS will be conducted.	5. The FPS is supplied with power from a non-Class 1E uninterruptible power supply.
6. MCR alarms and displays provided for the FPS are as defined in Section 2.15.6.	6. Inspections will be performed on the MCR alarms, and displays for the FPS.	6. Alarms and displays exist or can be retrieved in the MCR as defined in Section 2.15.6.

2.15.12 Control Building

Design Description

The Control Building (C/B) is a structure which houses and provides protection and support for plant control and electrical equipment, batteries, portions of the Reactor Building Cooling Water (RCW) System, and C/B heating, ventilating and air conditioning equipment. The C/B is located between the Reactor and Turbine Buildings. Figures 2.15.12a through 2.12.12g show the basic configuration and scope of the C/B.*

The C/B is constructed of reinforced concrete and structural steel. The C/B is a shear-wall structure which accommodates seismic loads with its perimeter walls and steam-tunnel walls, together with their supporting floors. Columns carry vertical loads to the basemat. The top of the C/B basemat is located 20.5m±1m below the finished grade elevation.

The C/B, except for the main control area envelope, is divided into three separate divisional areas for mechanical and electrical equipment and four divisional areas for instrumentation and control equipment (including batteries). Interdivisional boundaries have the following features:

(1) Interdivisional walls, floors, doors and penetrations which have three-hour fire rating.

(2) Watertight doors to prevent flooding in one division from propagating to other divisions.

(3) Divisional walls in the basement are 0.6m thick or greater.

The main control area envelope is separated from the rest of the C/B by walls, floors, doors and penetrations which have three-hour fire rating.

Watertight doors between flood divisions have open/close sensors with status indication in the main control room.

The C/B flooding that results from component failures in any of the C/B divisions does not prevent safe shutdown of the reactor. The basement floor is the collection point for floods. Except for the basement and main control area envelope, safety-related electrical equipment and instrumentation and control equipment is located at least 20 centimeters above the floor surface. Level sensors are located in the basement area of each of the three mechanical divisions. These sensors send signals to the corresponding divisions of the

* The overall building dimensions provided in Figures (later) through (later) are provided for information only and are not intended to be part of the certified ABWR information.

Reactor Service Water (RSW) System indicating flooding in that division of the C/B.

The basement area level sensors are powered from their respective divisional Class 1E power supply. Independence is provided between the Class 1E divisions for these sensors and also between the Class 1E divisions and non-Class 1E equipment.

To protect the C/B against an external flood. The following design features are provided:

(1) External walls below flood level are equal to or greater than 0.6m thick.

(2) Penetrations in the external walls below flood level are provided with flood protection features.

Within the C/B, the steam tunnel has no penetrations from the steam tunnel into other areas of the C/B. The concrete thickness of the steam tunnel walls, floor and ceiling within the C/B are equal to or greater than 1.6m.

The C/B is classified as Seismic Category I. It is designed to accommodate the dynamic and static loading conditions associated with the various loads and load combinations which form the structural design basis. The loads are those associated with:

(1) Natural phenomena, including wind, floods, tornadoes, earthquakes, rain and snow.

(2) Internal events, including fires, floods, pipe breaks and missiles.

(3) Normal plant operation, including live loads, dead loads and temperature effects.

The steam tunnel is protected against pressurization effects that occur in the steam tunnel as a result of postulated rupture of pipes containing high energy fluid.

Inspections, Tests, 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

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the C/B is shown on Figures 2.15.12a through 2.15.12g.	1. Inspections of the as-built structure will be conducted.	1. The as-built C/B conforms with the basic configuration shown on Figures 2.15.12a through 2.15.12g.
2. The top of the C/B basemat is located 20.5m±1m below the finished grade elevation.	2. Inspections of the as-built structure will be conducted.	2. The top of the C/B basemat is located 20.5m±1m below the finished grade elevation.
3. Inter-divisional walls, floors, doors and penetrations in the C/B have a three-hour fire rating.	3. Inspections of the as-installed inter-divisional boundaries will be conducted.	3. The as-installed walls, floors, doors and penetrations that form the inter-divisional boundaries have a three-hour fire rating.
4. The C/B has divisional areas with walls and watertight doors as shown on Figures 2.15.12a through 2.15.12g.	4. Inspections of the as-built walls, and doors will be conducted.	4. The as-built C/B has walls and watertight doors as shown on Figures 2.15.12a through 2.15.12g
5. The main control area envelope is separated from the rest of the C/B by walls, floors, doors and penetrations which have a three-hour fire rating.	5. Inspections of the as-built structure will be conducted.	5. The as-built C/B has a main control area envelope separated from the rest of the C/B by walls, floors, doors and penetrations which have a three-hour fire rating.
6. Main control room displays provided for the C/B are as defined in Section 2.15.12.	6. Inspections will be performed on the main control room displays for the C/B.	6. Displays exist or can be retrieved in the main control room as defined in Section 2.15.12.
7. Except for the basemat and main control area envelope, safety-related electrical equipment and instrumentation, and control equipment is located at least 20 cm. above the floor surface.	7. Inspections will be conducted of the as-built equipment.	7. Except for the basemat and main control area envelope, safety-related electrical equipment and instrumentation, and control equipment is located at least 20 cm above the floor surface.
8. Level sensors are located in the basement area of each of the three mechanical divisions.	8. Inspections of the as-built equipment will be conducted.	8. Level sensors are located in the basement area of each of the three mechanical divisions.

Table 2.15.12 Control Building (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. For the basement area level sensors, independence is provided between the Class 1E divisions for these sensors and also between the Class 1E divisions and non-Class 1E equipment.	9a. Tests will be conducted on the as-built sensors by providing a test signal in only one Class 1E division at a time. 9b. Inspections of the as-installed Class 1E divisions will be conducted.	9a. The test signal exists only in the Class 1E division under test. 9b. Physical separation exists between Class 1E divisions. Physical separation exists between these Class 1E divisions and non-Class 1E equipment.
10. The C/B is protected against external floods by having: a) External walls below flood level that are equal to or greater than 0.6m. b) Providing penetrations in the external walls below flood level with flood protection features.	10. Inspections of the as-built structure will be conducted.	10. The C/B is protected against external floods by having a) External walls below flood level that are equal to or greater than 0.6m. b) Providing penetrations in the external walls below flood level with flood protection features.
11. Within the C/B, the steam tunnel has no penetrations from the steam tunnel into other areas of the C/B.	11. Inspections of the as-built structure will be conducted.	11. Within the C/B, the steam tunnel has no penetrations from the steam tunnel into other areas of the C/B.
12. The concrete thickness of the steam tunnel walls, floor and ceiling within the C/B are equal to or greater than 1.6m.	12. Inspections of the as-built structure will be conducted.	12. The concrete thickness of the steam tunnel walls, floor and ceiling within the C/B are equal to or greater than 1.6m.
13. The C/B is able to withstand the structural design basis loads as defined in Section 2.15.12.	13. A structural analysis will be performed which reconciles the as-built data with structural design basis as defined in Section 2.15.12.	13. A structural analysis report exists which concludes that the as-built C/B is able to withstand the structural design basis loads as defined in Section 2.15.12.

2.15.13 Radwaste Building

Design Description

The Radwaste Building (RW/B) is a structure which houses the solid and liquid radwaste treatment systems. The RW/B is classified as non-safety-related.

Flood conditions in the RW/B are prevented from propagating into the Reactor Building and Turbine Building by providing the penetrations in external walls below flood level with flood protection features.

The external walls of the RW/B below grade are classified as Seismic Category I. The exterior walls above grade, the floor slabs, the interior columns, and the roof are classified as non-seismic.

The external walls of the RW/B below grade accommodate the dynamic and static loading conditions associated with the various loads and load combinations which form the structural design basis. The loads are those associated with:

(1) Natural phenomena including wind, floods, tornados, earthquakes, rain, and snow.

(2) Internal events including fires, and floods.

(3) Normal plant operations including live loads, dead loads, and temperature effects.

The exterior walls above grade, the floor slabs, the interior columns, and the roof does not collapse under seismic loads corresponding to the safe shutdown earthquake (SSE) ground acceleration.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.13 provides a definition of the inspections, tests, and/or analyses together with associated acceptance criteria which will be undertaken for the Radwaste Building.

Table 2.15.13 Radwaste Building

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the RW/B is described in Section 2.15.13.	1. Inspections of the as-built structure will be conducted.	1. The as-built RW/B conforms with the basic configuration in Section 2.15.13.
2. The external walls of the RW/B below grade are able to withstand the design basis loadings as defined in Section 2.15.13.	2. A structural analysis will be performed which reconciles the as-built data with the structural design basis as defined in Section 2.15.13.	2. A structural analysis report exists which concludes that the as-built RW/B is able to withstand the structural design basis loads as defined in Section 2.15.13.
3. The exterior walls above grade, the floor slabs, the interior columns, and the roof are designed to not collapse under seismic loads corresponding to the SSE ground accelerations.	3. A seismic analysis will be performed.	3. A structural analysis report exists which concludes building collapse does not occur under seismic loads corresponding to the SSE ground accelerators.

2.15.14 Service Building

Design Description

The Service Building (S/B) is a structure which houses the Technical Support Center, Emergency Operations Center, and the counting room. The S/B is classified as non-safety-related. It is located adjacent to the Control Building.

The S/B is not classified as a Seismic Category I structure.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.14 provides a definition of the inspections, tests, and/or analyses together with associated acceptance criteria, which will be undertaken for the Turbine Building.

Table 2.15.14 Service Building

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the S/B is described in Section 2.15.14.	1. Inspections of the as-built structure will be conducted.	1. The as-built S/B conforms with the basic configuration described in Section 2.15.14.

3.4 Instrumentation and Control

Introduction

Subsection A provides a description of the configuration of safety-related, digital instrumentation and control (I&C) equipment encompassed by Safety System Logic and Control (SSLC). Subsection B contains a description of the hardware and software development process used in the design, testing, and installation of all safety-related equipment. This includes descriptions of the processes used to establish programs that assess and mitigate the effects of electromagnetic interference, establish setpoints for instrument channels, and ensure the qualification of the installed equipment.

The devices addressed in this section are electronic components of the ABWR's safety-related systems. These components are configured as real-time microcontrollers that use microprocessors and other programmable logic devices to perform data acquisition, data communications, system logic processing. These components also contain automatic, on-line self-diagnostic features to monitor these tasks. The operating programs for these controllers are integrated into the hardware as firmware [software permanently stored in programmable read-only memory (PROM)].

A. Safety System Logic and Control

Design Description

Safety-related monitoring and trip logic for the plant protection systems resides in Safety System Logic and Control (SSLC) equipment. SSLC integrates the automatic and manual decision-making and trip logic functions associated with the safety actions of the safety-related systems. SSLC generates the protective function signals that activate reactor trip and provide safety-related mitigation of reactor accidents. The relationship between SSLC and systems for plant protection is shown in Figure 3.4a.

SSLC equipment comprises microprocessor-based, software-controlled, signal processors that perform signal conditioning, setpoint comparison, trip logic, self-test, calibration, and bypass functions. The signal processors associated with a particular safety-related system are an integral part of that system. Functions in common, such as self-test, calibration, bypass control, power supplies and certain switches and indicators, belong to SSLC. However, SSLC is not by itself a system; SSLC is the aggregate of signal processors for several safety-related systems. SSLC hardware and software are classified as Class 1E, safety-related.

Sensors used by the safety-related systems can be either analog, such as process control transmitters, or discrete, such as limit switches and other contact closures. While some sensor signals are hardwired directly to the SSLC processors, most sensor signals are transmitted from the instrument racks in the

Reactor Building to the SSLC equipment in the Control Building via the Essential Multiplexing System (EMS). Both analog and discrete sensors are connected to remote multiplexing units (RMUs) in local areas, which perform signal conditioning, analog-to-digital conversion for continuous process inputs, change-of-state detection for discrete inputs, and message formatting prior to signal transmission. The RMUs are limited to acquisition of sensor data and the output of control signals. Trip decisions and other control logic functions are performed in SSLC processors in the main control room area.

The basic hardware configuration for one division of SSLC is shown in Figure 3.4b. Each division runs independently (i.e., asynchronously) with respect to the other divisions. The following steps describe the processing sequence for incoming sensor signals and outgoing control signals. These steps are performed simultaneously and independently in each of the four divisions:

(1) The digitized sensor inputs received in the control room are decoded by a microprocessor-based function, the Digital Trip Module (DTM). For sensor signals hardwired to the control room, the DTM also performs digitizing and signal conditioning tasks. For each system function, the DTM then compares these inputs to preprogrammed threshold levels (setpoints) for possible trip action.

(2) For Reactor Protection System (RPS) trip and main steam isolation valve (MSIV) closure functions, trip outputs from the DTM are then compared, using a 2–out-of-4 coincidence logic format, with trip outputs from the DTMs of the other three divisions. The trip outputs are compared in the trip logic unit (TLU), another microprocessor-based device. The logic format for the DTM and TLU is fail-safe (i.e., de-energize-to-operate). Thus, a reactor trip or MSIV closure signal occurs on loss of input signal or power to the DTM, but, because of the 2–out-of-4 logic format in the TLU, a tripped state does not appear at the output of the TLU. Loss of signal or power to a division's TLU also causes a tripped state, but the 2–out-of-4 configuration of actuator load drivers prevents de-energization of the pilot valve solenoids.

(3) Trip outputs are sent from the TLU to the RPS and MSIV output logic units (OLUs). The OLUs use non-microprocessor circuitry to provide a diverse interface for the following manual functions:

(a) Manual reactor trip (per division: 2-out-of-4 for completion).

(b) MSIV closure (per division: 2-out-of-4 for completion).

(c) MSIV closure (eight individual control switches).

(d) RPS and MSIV trip reset.

(e) TLU output bypass

The OLUs distribute the automatic and manual trip outputs to the MSIV pilot valve and scram pilot valve actuating devices and provide control of trip seal-in, reset, and TLU output bypass (division-out-of-service bypass). Bypass inhibits automatic trip but has no effect on manual trip. The OLUs also provide a manual test input for de-energizing a division's parallel load drivers (part of the 2-out-of-4 output logic arrangement) so that scram or MSIV closure capability can be confirmed without solenoid de-energization. The OLUs are located external to the TLU so that manual MSIV closure or manual reactor trip (per division) can be performed either when a division's microprocessor logic is bypassed or when failure of sensors or microprocessor logic equipment causes trip to be inhibited.

(4) Trips are transmitted across divisions for 2–out-of-4 voting via fiber optic data links to preserve signal isolation among divisions. The TLU also receives inputs directly from the trip outputs of the Neutron Monitoring System, manual control switches, and contact closures from limit switches and position switches used for equipment interlocks. In addition, plant sensor signals and contact closures that do not require transmittal to other divisions for 2–out-of-4 trip comparison are provided as inputs directly to the TLU. In this case, the TLU also performs the trip setpoint comparison (DTM) function.

(5) For Leak Detection and Isolation System (LDS) functions (except MSIV), emergency core cooling system (ECCS) functions, other safety-related supporting functions, and Electrical Power Distribution System functions such as diesel generator start, logic processing is performed as above, but in separate DTMs and Safety System Logic Units (SLUs). The SLUs are similar to TLUs, but are dual redundant in each processing channel for protection against inadvertent initiation. Dual SLUs both receive the same inputs from the DTM, manual control switch inputs, and contact closures. Both SLU outputs must agree before the final trip actuators are energized. The logic format for the DTM and SLUs is fail-as-is (i.e., energize-to-operate) for ECCS and other safety-related supporting functions. Thus, loss of power or equipment failure will not cause a trip or initiation action. However, containment isolation signals are in fail-safe format and will cause an isolation signal output on loss of power or signal. Besides performing 2-out-of-4 voting logic, the SLUs also provide interlock logic functions conforming to the logic diagram requirements of each supported safety system.

(6) For reactor trip or MSIV closure, if a 2–out-of-4 trip condition of sensors is satisfied, all four divisions' trip outputs will produce a simultaneous coincident trip signal (for example, reactor trip) and transmit the signal through hardwired connections (and isolators where necessary) to load drivers that control the protective action of the actuators. The load drivers are themselves arranged in a 2–out-of-4 configuration, so that at least two divisions must produce trip outputs for protective action to occur.

(7) For engineering safety feature (ESF) functions, the trip signals in three divisions are transmitted by the Essential Multiplexing System to the RMUs, where a final 2–out-of-2 logic comparison is made prior to distribution of the control signals to the final actuators. ESF outputs do not exist in Division IV.

The DTM, TLU, and OLUs for RPS and MSIV in each of the four instrumentation divisions are powered from their respective divisional Class 1E AC sources. The DTMs and SLUs for ESF 1 and ESF 2 in Div. I, II, and III are powered from their respective divisional Class 1E DC sources. In SSLC, independence is provided between Class 1E divisions, and also between Class 1E divisions, and also non-Class 1E equipment.

Bypassing of any single division of sensors (i.e., those sensors whose trip status is confirmed by 2–out-of-4 logic) is accomplished from each divisional SSLC cabinet by means of the manually-operated bypass unit. When such bypass is made, all four divisions of 2–out-of-4 input logic become 2–out-of-3 while the bypass state is maintained. During bypass, if any two of the remaining three divisions reach trip level for any sensed input parameter, then the output logic of all four divisions trips (for RPS and MSIV functions) or the three ECCS divisions initiate the appropriate safety system equipment.

Bypassing of any single division of output trip logic (i.e., taking a logic channel out of service) is also accomplished by means of the bypass unit. This type of bypass is limited to the fail-safe (de-energize-to-operate) reactor trip and MSIV closure functions, since removal of power from energize-to-operate signal processors is sufficient to remove that channel from service.

When a trip logic output bypass is made, the TLU trip output in a division is inhibited from affecting the output load drivers by maintaining that division's load drivers in an energized state. Thus, the 2–out-of-4 logic arrangement of output load drivers for the RPS and MSIV functions effectively becomes 2-out-of-3 while the bypass is maintained.

Bypass status is indicated in the main control room until the bypass condition is removed. An electrical interlock rejects attempts to remove more than one SSLC division from service at a time.

ESF output logic is processed in two redundant channels within each divisional train of ESF equipment. In order to prevent spurious actuation of ESF equipment, final output signals are voted 2-out-of-2 at the remote multiplexing units by means of series-connected load drivers at the RMU outputs. However, in the event of a failure within either processing channel, a bypass is applied automatically (with manual backup) such that the failed channel is removed from service. The remaining channel provides 1-out-of-1 operation to maintain availability during the repair period.

Equipment failure is detected by self-test and other diagnostic software that is part of each SSLC processor and the essential multiplexing system. The bypass unit contains the failure detector that evaluates trouble signals from the results of self-test. If a failure is confirmed by the detector, the bypass unit causes the output load driver of the failed channel to be energized. Thus, the remaining load driver of the operating channel will control the output to the device actuators.

Channel failures are alarmed in the main control room. If a failed channel is not automatically bypassed, the operator is able to manually bypass the channel by a hardwired connection from the main control room.

A portion of the anticipated transient without scram (ATWS) mitigation features is provided by SSLC circuitry, with initiating conditions as follows:

(1) Initiation of automatic depressurization system (ADS) Inhibit: High reactor vessel dome pressure and average power range monitor (APRM) not downscale for 2 minutes or greater, or low reactor water level and APRM not downscale (with no delay). Reset permitted only after APRMs are down scale and water level is rising.

(2) Initiation of automatic Standby Liquid Control System (SLCS) injection: High dome pressure and APRM not downscale for 3 minutes or greater, or low reactor water level and APRM not downscale for 3 minutes or greater.

(3) Initiation of feedwater runback: High dome pressure and startup range neutron monitoring (SRNM) not downscale for 2 minutes or greater. Reset permitted only when both signals drop below the setpoints.

These ATWS features are implemented in four divisions of SSLC control circuitry that are functionally independent and diverse from the circuitry used for the Reactor Protection System (see Figure 3.4c).

SSLC has the following alarms, displays, and controls in the main control room:

(1) SSLC signal processor inoperative (INOP).

(2) SSLC manual controls for bypass as described above.

(3) Displays for bypass status.

(4) Divisional flat display panels that provide display and control capability for manual ESF functions.

(5) Technician interface for display and control of calibration and off-line self-test functions

Inspections, Tests, Analyses and Acceptance Criteria

Table 3.4, Items 1 through 5, provides a definition of the inspections, tests and analyses, together with associated acceptance criteria, which will be undertaken for SSLC.

B. I & C Development and Qualification Processes

Hardware and Software Development Process

The ABWR design uses programmable digital equipment to implement operating functions of instrumentation and control systems. The equipment is in the form of embedded controllers; i.e., a control program developed in software is permanently stored in programmable read-only memory (PROM), and thus becomes part of the controller's hardware.

A controlled process for software development, hardware integration, and final product and system testing is employed. The development process for safety-related hardware and software includes a formal verification and validation (V&V) program. Non-safety-related hardware and software will be developed using a planned design process similar to the safety-related development program, but with emphasis on periodic design reviews rather than formal V&V.

System functional performance testing for each system using the software-based controllers discussed herein is addressed in Section 2 system entries.

An overall software development plan establishes the requirements and methodology for software design and development. The plan also defines methods for auditing and testing software during the design, implementation, and integration phases. These phases are part of the software life cycle, a planned development method to ensure the quality of software throughout its period of usage. The relationship between components of the plan and I&C design activities is shown in Figure 3.4d.

As part of the design of software for safety-related applications, the software development plan, at each defined phase of the software life cycle, addresses software requirements that have been defined as safety-critical. Safety-critical is defined as those computer software components (processes, functions, values or computer program states) in which errors (inadvertent or unauthorized occurrence, failure to occur when required, occurrence out of sequence, occurrence in combination with other functions, or erroneous values) can result in a potential hazard or loss of predictability or control of a system. Potential hazards are failure of a safety-related function to occur on demand and spurious occurrence of a safety-related function in an unsafe direction.

The overall software development plan comprises the following plans:

(1) A Software Management Plan (SMP) that establishes standards, conventions and design processes for I&C software.

A SMP shall be instituted which establishes that software for embedded control hardware shall be developed, designed, evaluated, and documented per a design development process that addresses, for safety-related software, software safety issues at each defined life-cycle phase of the software development.

The SMP defines the following software life-cycle phases:

(a) Planning

(b) Design definition

(c) Software design

(d) Software coding

(e) Integration

(f) Validation

(g) Change control

The SMP shall state that the output of each defined life-cycle phase shall be documents that define the current state of that design phase and the design input for the next design phase

(2) A Configuration Management Plan (CMP) that establishes the standards and procedures controlling software design and documentation.

A CMP shall be instituted that establishes the methods for maintaining, throughout the software design process, the design documentation, procedures, evaluated software, and the resultant as-installed software.

The CMP addresses:

(a) Identification of CMP software documentation

(b) Management of software change control

(c) Control and traceability of software changes

(d) Verification of software to design requirements

(3) A Verification and Validation (V&V) Plan that establishes verification reviews and validation testing procedures.

A V&V Plan shall be developed which establishes that developed software shall be subjected to structured and documented verification reviews and validation testing, including testing of the software integrated into the target hardware.

The V&V plan addresses:

(a) Independent design verification

(b) Baseline software reviews

(c) Testing

(d) Procedure for software revisions.

Electromagnetic Compatibility

Electromagnetic compatibility (EMC) is the ability of equipment to function properly when subjected to an electromagnetic environment, and, in addition, to add minimal electromagnetic energy to that environment. An EMC compliance plan to confirm the level of immunity to electrical noise is part of the design, installation, and pre-operational testing of I&C equipment.

Electrical and electronic components in the systems listed below are qualified according to the established plan for the anticipated levels of electrical interference at the installed locations of the components:

(1) Safety System Logic and Control

(2) Essential Multiplexing System

(3) Non-essential Multiplexing System

(4) Other microprocessor-based, software controlled systems or equipment as referenced in Table 3.0.

The plan is structured on the basis that EMC of I&C equipment is verified by factory testing and site testing of both individual components and interconnected systems to meet electromagnetic compatibility requirements for protection against the effects of:

(1) Electromagnetic Interference (EMI)

(2) Radio Frequency Interference (RFI)

(3) Electrostatic Discharge (ESD)

(4) Electrical surge [Surge Withstand Capability (SWC)]

To be able to predict the degree of electromagnetic compatibility of a given equipment design, the following information is developed:

(1) Characteristics of the sources of electrical noise

(2) Means of transmission of electrical noise

(3) Characteristics of the susceptibility of the system

(4) Techniques to attenuate electrical noise

After these characteristics of the equipment are identified, noise susceptibility is tested for four different paths of electrical noise entry:

(1) Power feed lines

(2) Input signal lines

(3) Output signal lines

(4) Radiation

Instrument Setpoint Methodology

Setpoints for initiation of safety-related functions are determined, documented, installed and maintained using a process that establishes a general program for:

(1) Specifying requirements for documenting the bases for selection of trip setpoints

(2) Accounting for instrument inaccuracies, uncertainties, and drift

(3) Testing of instrumentation setpoint dynamic response

(4) Replacement of setpoint-related instrumentation.

The determination of nominal trip setpoints includes consideration of the following factors:

Design Basis Analytical Limit

In the case of setpoints that are directly associated with an abnormal plant transient or accident analyzed in the safety analysis, a design basis analytical limit is established as part of the safety analysis. The design basis analytical limit is the value of the sensed process variable prior to or at the point which a desired action is to be initiated. This limit is set so that associated licensing safety limits are not exceeded, as confirmed by plant design basis performance analysis.

Allowable Value

An allowable value is determined from the analytical limit by providing allowances for the specified or expected calibration capability, the accuracy of the instrumentation, and the measurement errors. The allowable value is the limiting value of the sensed process variable at which the trip setpoint may be found during instrument surveillance.

Nominal Trip Setpoint

The nominal trip setpoint value is calculated from the analytical limit by taking into account instrument drift in addition to the instrument accuracy, calibration capability, and the measurement errors. The nominal trip setpoint value is the limiting value of the sensed process variable at which a trip action will be set to operate at the time of calibration.

Signal processing devices in the instrument channel

Within an instrument channel, there may exist other components or devices that are used to further process the electrical signal provided by the sensor; e.g., analog-to-digital converters, signal conditioners, temperature compensation circuits, and multiplexing and demultiplexing components. The worst-case

instrument accuracy, calibration accuracy, and instrument drift contributions of each of these additional signal conversion components are separately or jointly accounted for when determining the characteristics of the entire instrument loop.

Not all parameters have an associated design basis analytical limit (e.g., main steam line radiation monitoring). An allowable value may be defined directly based on plant licensing requirements, previous operating experience or other appropriate criteria. The nominal trip setpoint is then calculated from this allowable value, allowing for instrument drift. Where appropriate, a nominal trip setpoint may be determined directly based on operating experience.

Procedures will be used that provide a method for establishing instrument nominal trip setpoint and allowable value. Because of the general characteristics of the instrumentation and processes involved, two different methods are applied:

(1) Computational

(2) Historical data

The computational method is used when sufficient information is available regarding a dynamic process and the associated instrumentation. The procedure takes into account channel instrument accuracy, calibration accuracy, process measurement accuracy, primary element accuracy, and instrument drift. If the resulting nominal trip setpoint and allowable value are not acceptable when checked to ensure that they will not result in an unacceptable level of trips caused by normal operational transients, then more rigorous statistical evaluation or the use of actual operational data may be considered.

Some setpoint values have been historically established as acceptable, both for regulatory and operational requirements. These setpoints have non critical functions or are intended to provide trip actions related to gross changes in the process variable. The continued recommendation of these historically accepted setpoint values is another method for establishing nominal trip setpoint and allowable values. This approach is only valid where the governing conditions remain essentially unaltered from those imposed previously and where the historical values have been adequate for their intended functions.

The setpoint methodology plan requires that activities related to instrument setpoints be documented and stored in retrievable, auditable files.

Equipment Qualification (EQ)

Qualification of safety-related instrumentation and control equipment is implemented by a program that assures this equipment is able to complete its safety-related function under the environmental conditions that exist up to and including the time the equipment has finished performing that function. Qualification specifications consider conditions that exist during normal, abnormal, and design basis accident events in terms of their cumulative effect on equipment performance for the time period up to the end of equipment life.

The material discussed herein identifies an EQ program that addresses the spectrum of environmental conditions that may occur in plant areas where I&C equipment is installed. Not all safety-related I&C equipment will experience all of these conditions; the intent is that qualification be performed by selecting the conditions applicable to each particular piece of equipment and performing the necessary qualification.

As-built instrumentation and control components are environmentally qualified if they can withstand the environmental conditions associated with design basis events without loss of their safety functions for the time needed to be functional. These environmental conditions are as follows, as applicable to the bounding design basis events: Expected time-dependent temperature and pressure profiles, humidity, chemical effects, radiation, aging, seismic events, submergence, and synergistic effects which have a significant effect on equipment performance.

Electrical equipment environmental qualification is demonstrated through analysis of the environmental conditions that would exist in the location of the equipment during and following a design basis accident and through a determination that the equipment is qualified to withstand those conditions for the time needed to be functional. This determination may be demonstrated by:

(1) Testing of an identical item of equipment under identical or similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

(2) Testing of 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.

The installed condition of safety-related I&C equipment is assured by a program whose objective is to verify that the installed configuration is bounded by the test configuration and test conditions.

Inspections, Tests, Analyses and Acceptance Criteria

Table 3.4, Items 6 through 13, provides a definition of the inspections, tests and analyses, together with associated acceptance criteria, which will be used to demonstrate compliance with the above commitments for hardware and software development, electromagnetic compatibility, instrument setpoint methodology, and equipment qualification.

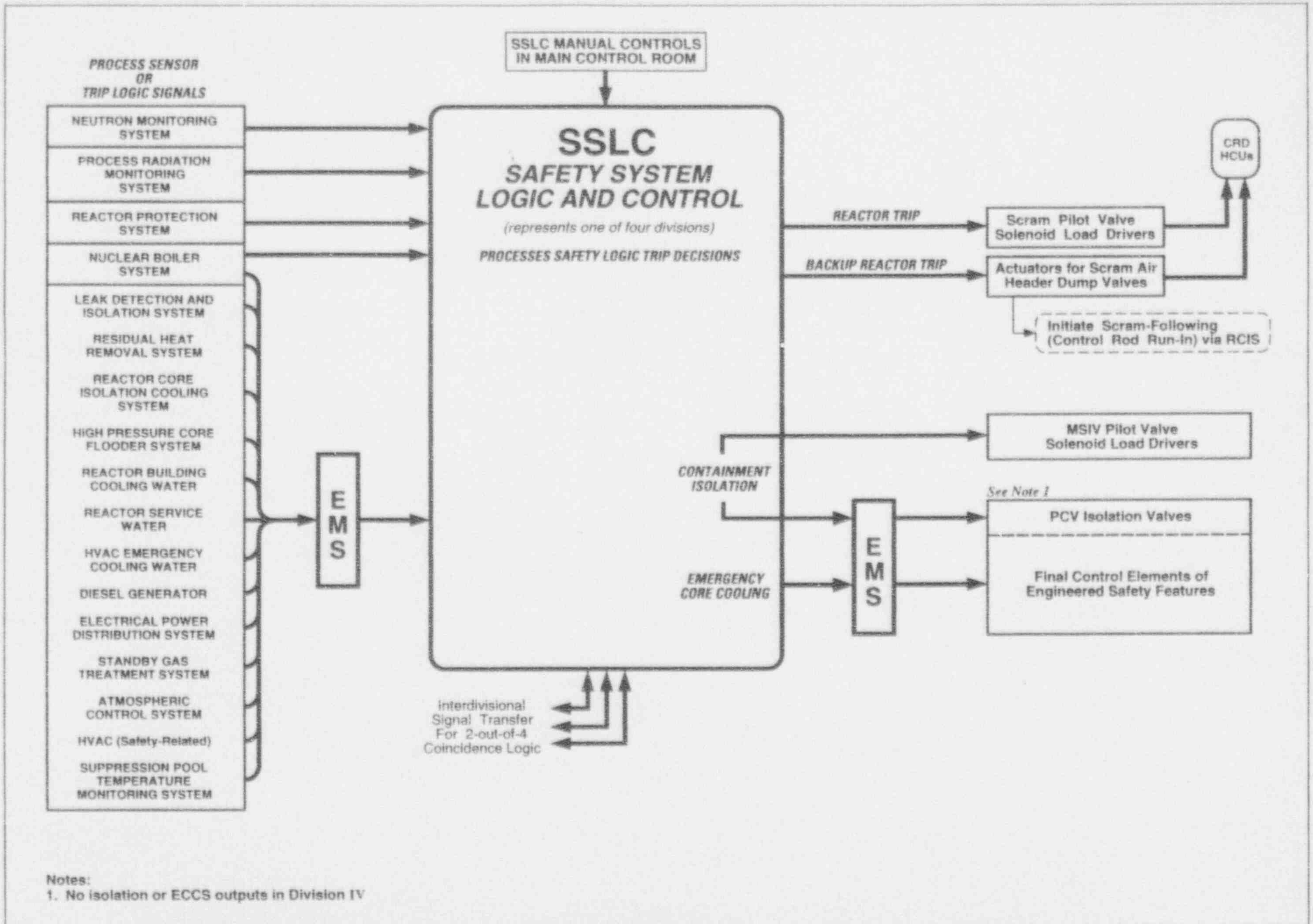
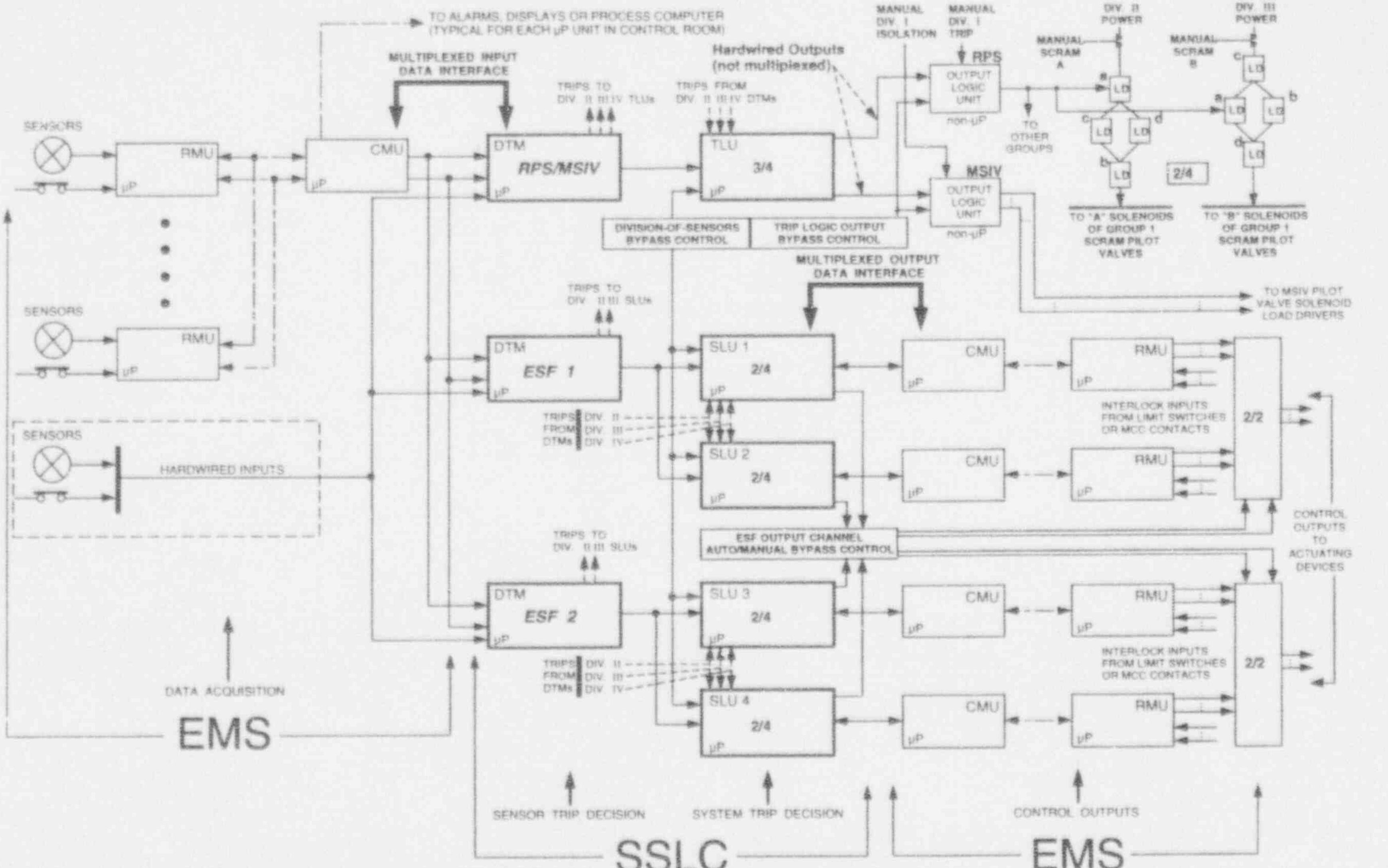


Figure 3.4a Safety System Logic and control (SSLC) Control Interface Diagram

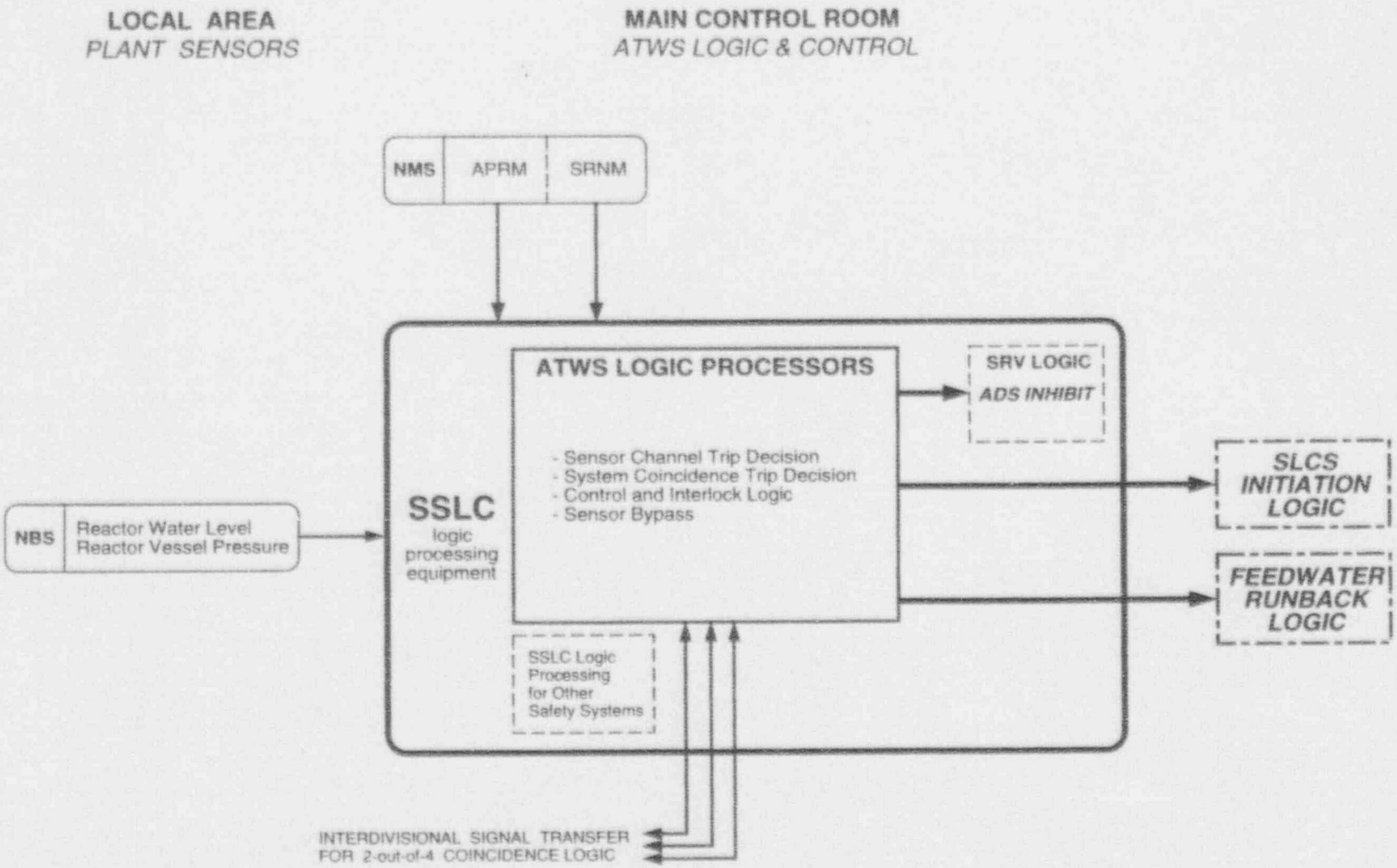


NOTES

1. EMS ARRANGEMENT SHOWN IS A SIMPLIFIED EXAMPLE FOR **ONE DIVISION**. ACTUAL QUANTITY AND INTERCONNECTIONS OF RMUs AND CMUs WILL BE DETERMINED WITHIN SCOPE OF EMS DESIGN. THERE ARE NO SLUs, CMUs, OR RMUs in DIV. IV.
2. A. OUTPUTS TO PROCESS COMPUTER, ALARMS OR DISPLAYS ON DEDICATED FIBER OPTIC DATA LINKS.
   B. CONTROL SWITCH INPUTS TO SSLC NOT SHOWN.
   C. INPUTS FROM NMS AND PRRM NOT SHOWN.
   D. INTERDIVISIONAL COMMUNICATIONS USE FIBER OPTIC DATA LINKS.
3. – – – – → FIBER OPTIC CABLE
   ———→ METALLIC CABLE
4. ABBREVIATIONS

CMU	-	CONTROL ROOM MULTIPLEXING UNIT
DTM	-	DIGITAL TRIP MODULE
EMS	-	ESSENTIAL MULTIPLEXING SYSTEM
ESF	-	ENGINEERED SAFETY FEATURES
LD	-	LOAD DRIVER
MSIV	-	MAIN STEAM ISOLATION VALVE
NMS	-	NEUTRON MONITORING SYSTEM
PRRM	-	PROCESS RADIATION MONITORING SYSTEM
RMU	-	REMOTE MULTIPLEXING UNIT
RPS	-	REACTOR PROTECTION SYSTEM
SLU	-	SAFETY SYSTEM LOGIC UNIT
TLU	-	TRIP LOGIC UNIT
μP	-	MICROPROCESSOR

Figure 3.4b Safety System Logic & Control Block Diagram



Notes:
1. Diagram Represents One Of Four ATWS Divisions
2. Remaining ATWS functions are processed as part of Recirculation Flow Control System logic

Figure 3.4c Anticipated Transient Without Scram (ATWS) Control Interface Diagram

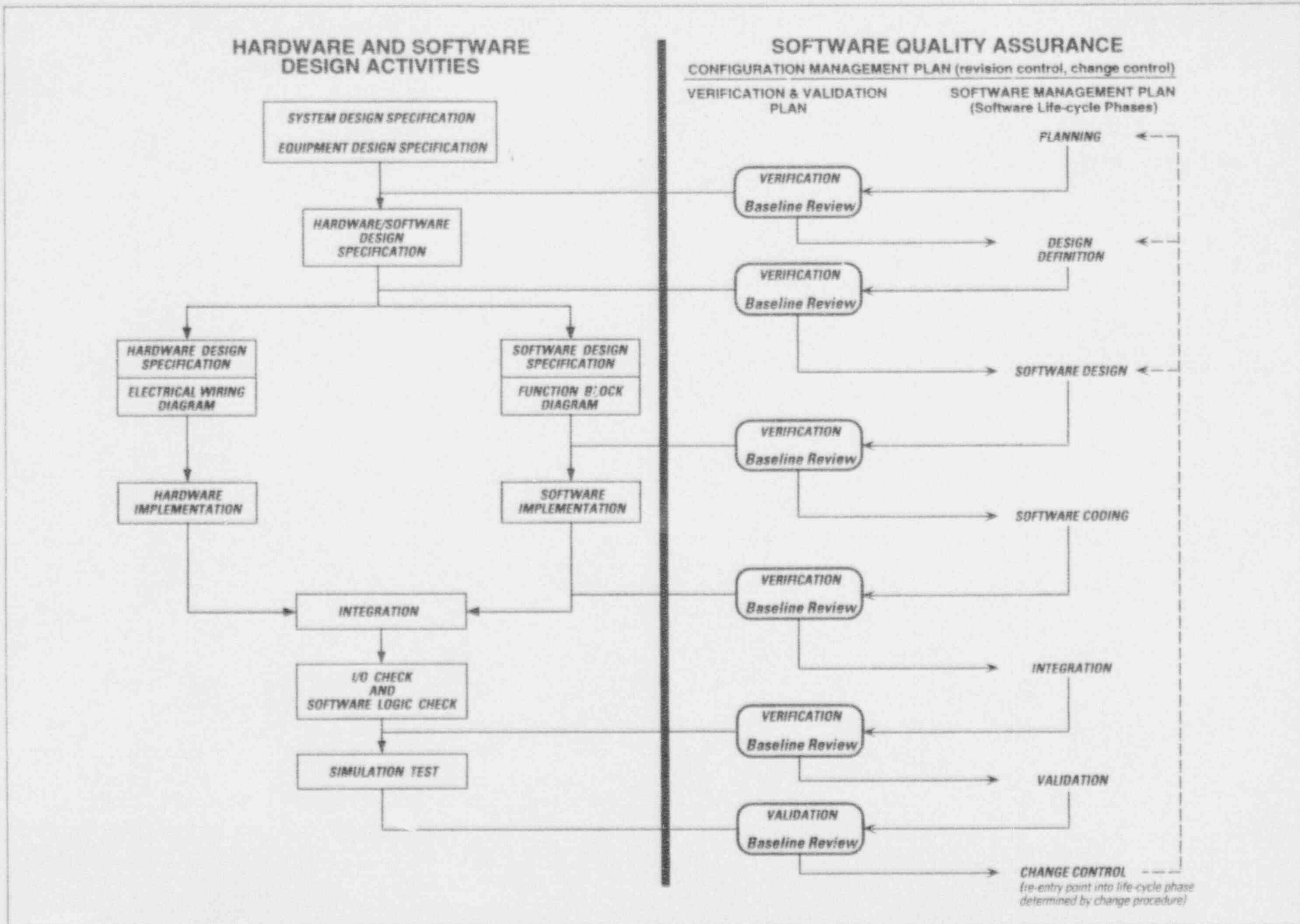


Figure 3.4d Integrated Hardware/Software Development Process

Table 3.4: Instrumentation and Control

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Safety System Logic and Control		
1. The equipment comprising SSLC is defined in Section 3.4(A)	1. Inspections of the as-built SSLC configuration will be conducted.	1. The as-built SSLC configuration conforms with the description in Section 3.4(A)
2. In SSLC, independence is provided between Class 1E divisions and between Class 1E divisions and non-Class 1E equipment.	2a. Tests will be performed on SSLC by providing a test signal to only one Class 1E division at a time.	2a. The test signal exists only in the Class 1E division under test in SSLC.
	2b. Inspection of the as-installed Class 1E divisions in the RPS will be performed.	2b. In SSLC, physical separation exists between Class 1E divisions. Physical separation exists between these Class 1E divisions and non-Class 1E equipment.
3. SSLC provides the following bypass functions:	3. Tests will be performed on the as-built SSLC as follows:	3. Results of bypass tests are as follows:
a. Division-of-sensors bypass b. Trip logic output bypass c. ESF output channel bypass	a(1)Place one division of sensors in bypass. Apply a trip test signal in place of each sensed parameter that is bypassed. At the same time, apply a redundant trip signal for each parameter in each other division, one division at a time. Monitor the voted trip output at each TLU and SLU. Repeat for each division.	a(1)No trip change occurs at the voted trip output of each TLU and SLU Bypass status is indicated in main control room.
	a(2)For each division in bypass, attempt to place each other division in division-of-sensors bypass, one at a time.	a(2)Each division not bypassed cannot be placed in bypass; bypass status in main control room indicates only one division of sensors is bypassed.

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Safety System Logic and Control		
3. (continued)	3. (continued)	3. (continued)
	b(1) Place one division in trip-logic-output bypass. Operate manual auto-trip test switch. Monitor the trip output at the RPS OLU. Operate manual auto-isolation test switch. Monitor the trip output at the MSIV OLU. Repeat for each division.	b(1) No trip change occurs at the trip output of the RPS OLU or MSIV OLU, respectively. Bypass status is indicated in main control room.
	b(2) For each division in bypass, attempt to place the other divisions in trip-logic-output bypass, one at a time.	b(2) Each division not bypassed cannot be placed in bypass; bypass status in main control room indicates only one trip logic output is bypassed.
	c(1) Apply common test signal to any one pair of dual-SLU signal inputs. Monitor test signal at voted 2-out-of-2 output in RMU area. Remove power from one SLU, restore power, then remove power from other SLU. Repeat test for all pairs of dual SLUs in each division.	c(1) Monitored test output signal does not change state when power is removed from eitl:er SLU. Bypass status and loss of power to SLU are indicated in main control room.
	c(2) Disable auto-bypass circuit in bypass unit. Repeat test c(1), but operate manual ESF loop bypass switch for each affected loop.	c(2) Monitored test output signal is lost when power is removed from either SLU, but is restored when manual bypass switch is operated. Bypass status, auto-bypass inoperable, and loss of power to SLU are indicated in main control room.

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Safety System Logic and Control		
4. A portion of the anticipated transient without scram (ATWS) mitigation features is provided by SSLC circuitry, with initiating conditions as follows:	4. Tests will be conducted using simulated input signals for the process variables used by the ATWS logic. For ADS inhibit logic, reset attempts will be made before APRMs are downscale and water level is rising. For feedwater runback logic, reset attempts will be made before initiating test signals drop below setpoints.	4. Four redundant output signals occur for each of the following ATWS mitigating functions (one set in each of the four divisions of ATWS outputs) that lead to initiation of these functions:
a. Initiation of ADS inhibit on high reactor vessel dome pressure and APRM not downscale for 2 minutes or greater, or low reactor water level and APRM not downscale (with no delay). Reset is permitted only after APRMs are down scale and water level is rising.		a. Initiation of ADS inhibit on high reactor vessel dome pressure and APRM not downscale for 2 minutes or greater, or low reactor water level and APRM not downscale (with no delay). Reset is permitted only after APRMs are down scale and water level is rising.
b. Initiation of automatic SLCS injection on high dome pressure and APRM not downscale for 3 minutes or greater, or low reactor water level and APRM not downscale for 3 minutes or greater.		b. Initiation of automatic SLCS injection on high dome pressure and APRM not downscale for 3 minutes or greater, or low reactor water level and APRM not downscale for 3 minutes or greater.
c. Initiation of feedwater runback on high dome pressure and SRNM not downscale for 2 minutes or greater. Reset is permitted only when both signals drop below the setpoints.		c. Initiation of feedwater runback on high dome pressure and SRNM not downscale for 2 minutes or greater. Reset is permitted only when both signals drop below the setpoints.
5. Main control room alarms, displays and controls provided for SSLC are as defined in Section 3.4.	5. Inspections will be performed on the main control room alarms, displays and controls for SSLC	5. Alarms, displays and controls exist or can be retrieved in the main control room as defined in Section 3.4.

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Hardware/Software Development		
6. A Software Management Plan (SMP) shall be instituted which establishes that software for embedded control hardware shall be developed, designed, evaluated, and documented per a design development process that addresses, for safety-related software, software safety issues at each defined life-cycle phase of the software development. The SMP shall state that the output of each defined life-cycle phase shall be documents that define the current state of that design phase and the design input for the next design phase.	6. The Software Management Plan shall be reviewed.	6. The Software Management Plan shall define: a. The organization and responsibilities for development of the software design; the procedures to be used in the software development; the interrelationships between software design activities; and the methods for conducting software safety analyses. b. That the software safety analyses to be conducted for safety-related software applications shall: (i) Identify software requirements having safety-related implications (ii) Document the identified safety-critical software requirements in the software requirements specification for the design (iii) Incorporate in to the software design the safety-critical software functions specified in the software requirements specification (iv) Identify in the coding and test of the developed software, those software modules which are safety-critical (v) Evaluate the performance of the developed safety-critical software modules when operated within the constraints imposed by the established system requirements, software design, and computer hardware requirements

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Hardware/Software Development		
6. (continued)	6. (continued)	6b. (continued) (vi) Evaluate software interfaces of safety-critical software modules (vii) Perform equipment integration and validation testing that demonstrate that safety-related functions identified in the design input requirements are operational. c. The software engineering process, which is composed of the following life-cycle phases: (i) Planning (ii) Design Definition (iii) Software Design (iv) Software Coding (v) Integration (vi) Validation (vii) Change control d. The Planning phase design activities, which shall address the following system design requirements and software development plans:

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Hardware/Software Development		
6. (continued)	6. (continued)	6d. (continued) (i) Software Management Plan (ii) Software Configuration Management Plan (iii) Verification and Validation Plan (iv) Equipment design requirements (v) Safety analysis of design requirements (vi) disposition of design and/or documentation nonconformances identified during this phase e. The Design Definition phase design activities, which shall address the development of the following implementing equipment design and configuration requirements: (i) Equipment schematic (ii) Equipment hardware and software performance specification (iii) Equipment user's manual (iv) Data communications protocol (v) Safety analysis of the developed design definition

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Hardware/Software Development		
6. (continued)	6. (continued)	6e. (continued) (vi) Disposition of design and/or documentation nonconformances identified during this phase f. The Software Design phase, which shall address the design of the software architecture and program structure elements, and the definition of software module functions: (i) Software Design Specification (ii) Safety analysis of the software design (iii) Disposition of design and/or documentation nonconformances identified during this phase g. The Software Coding phase, which shall address the following software coding and testing activities of individual software modules: (i) Software source code (ii) Software module test reports (iii) Safety analysis of the software coding

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Hardware/Software Development		
6. (continued)	6. (continued)	6g. (continued) (iv) Disposition of nonconformances identified in this phase's design documentation and test results h. The Integration phase, which shall address the following equipment testing activities that evaluates the performance of the software when installed in hardware prototypical of that defined in the Design Definition phase: (i) Integration test reports (ii) Safety analysis of the integration test results (iii) Disposition of nonconformances identified in this phase's design documentation and test results i. The Validation phase, which comprises the development and implementation of the following documented test plans and procedures:

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Hardware/Software Development		
6. (continued)	6. (continued)	6i. (continued) (i) Validation test plans and procedures (ii) Validation test reports (iii) Description of as-tested software (iv) Safety analysis of the validation test results (v) Disposition of nonconformances identified in this phase's design documentation and test results (vi) Software change control procedures, and j. The Change Control phase, which begins with the completion of validation testing, and addresses changes to previously validated software and the implementation of the established software change control procedures.

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Hardware/Software Development		
7. A Configuration Management Plan (CMP) shall be instituted that establishes the methods for maintaining, throughout the software design process, the design documentation, procedures, evaluated software, and the resultant as-installed software.	7. The Configuration Management Plan shall be reviewed.	7. The Configuration Management Plan shall define: a. The specific product or system scope to which it is applicable. b. The organizational responsibilities for software configuration management. c. Methods to be applied to: (i) Identify design interfaces (ii) Produce software design documentation (iii) Process changes to design interface documentation and software design documentation (iv) Process corrective actions to resolve deviations identified in software design and design documentation, including notification to end user of errors discovered in software development tools or other software (v) Maintain status of design interface documentation and developed software design documentation

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Hardware/Software Development		
7. (continued)	7. (continued)	7c. (continued) (vi) Designate and control software revision status. Such methods shall require that software code listings present direct indication of the software code revision status. d. Methods for, and the sequencing of, reviews to evaluate the compliance of software design activities with the requirements of the CMP. e. The configuration management of tools (such as compilers) and software development procedures. f. Methods for the dedication of commercial software for safety-related usage. g. Methods for tracking error rates during software development, such as the use of software metrics h. The methods for design record collection and retention.

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Hardware/Software Development		
8. A Verification and Validation Plan (V&VP) shall be developed which establishes that developed software shall be subjected to structured and documented verification reviews and validation testing, including testing of the software integrated into the target hardware.	8. The Verification and Validation Plan shall be reviewed.	8. The Verification and Validation Plan shall define: a. That baseline reviews of the software development process are to be conducted during each phase of the software development life cycle. b. The scope and methods to be used in the baseline reviews to evaluate the implemented design, design documentation, and compliance with the requirements of the Software Management Plan and Configuration Management Plan. c. The requirements for use of commercial software and commercial development tools for safety-related applications and that such use is a controlled and documented procedure d. That verification shall be performed as a controlled and documented evaluation of the conformity of the developed design to the documented design requirements at each phase of baseline review.

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Hardware/Software Development		
8. (continued)	8. (continued)	8. (continued) e. That validation shall be performed through controlled and documented testing of the developed software as installed in the target hardware that demonstrates compliance of the software with the software requirements specifications and compliance of the device(s) under test with the system design specifications. f. That for safety-related software, verification reviews and validation testing are to be conducted by personnel who are knowledgeable in the technologies and methods used in the design, but who did not develop the software design to be reviewed and tested. g. That for safety-related software, design verification reviews shall be conducted as part of the baseline reviews of the design material developed during the Planning through Integration phases of the software development life-cycle (as defined in Criterion 1b, above), and that validation testing shall be conducted as part of the baseline review of the Validation phase of the software development life-cycle.

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Hardware/Software Development		
8. (continued)	8. (continued)	8. (continued) h. That validation testing shall be conducted per a documented test plan and procedure. i. That for non-safety-related software development, verification and validation shall be performed through design reviews conducted as part of the baseline reviews completed at the end of the phases in the software development life cycle. These design reviews shall be performed by personnel knowledgeable in the technologies and methods used in the design development. j. The products which shall result from the baseline reviews conducted at each phase of the software development life-cycle; and that the defined products of the baseline reviews and the V&V Plan shall be documented and maintained under configuration management.

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Hardware/Software Development		
8. (continued)	8. (continued)	8. (continued) k. The methods for identification, closure, and documentation of design and/or design documentation nonconformances. l. That the software development is not complete until the specified verification and validation activities are complete and design documentation is consistent with the developed software.
9. Software development shall be performed in accordance with the software management plan, configuration management plan, and verification and validation plan.	9. Review software development results.	9. Software development has been completed as defined in the SMP, CMP, and V&VP.

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Electromagnetic Compatibility		
10. Electrical and electronic components in the systems listed below are qualified for the anticipated levels of electrical interference at the installed locations of the components according to an established plan: a. Safety System Logic and Control b. Essential Multiplexing System c. Non-essential Multiplexing System d. Other microprocessor-based, software controlled systems or equipment as referenced in Table 3.0. The plan is structured on the basis that electromagnetic compatibility (EMC) of I&C equipment is verified by factory testing and site testing of both individual components and interconnected systems to meet EMC requirements for protection against the effects of: a. Electromagnetic Interference (EMI) b. Radio Frequency Interference (RFI) c. Electrostatic Discharge (ESD) d. Electrical surge [Surge Withstand Capability (SWC)]	10. The EMC compliance plan will be reviewed.	10. An EMC compliance plan is in place. The plan requires, for each system qualified, system documentation that includes confirmation of component and system testing for the effects of high electrical field conditions and current surges. As a minimum, the following information is documented in a qualification file and subject to audit: a. Expected performance under test conditions for which normal system operation is to be ensured. b. Normal electrical field conditions at the locations where the equipment must perform as above. c. Testing methods used to qualify the equipment, including: (1) Types of test equipment. (2) Range of normal test conditions. (3) Range of abnormal test conditions for expected transient environment. (4) Location of testing and exact configuration of tested components and systems, including interconnecting cables, connections to electrical power distribution system, and connections to interfacing devices used during normal plant operation.

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Electromagnetic Compatibility 10. (continued)	10. (continued)	10. (continued) d. Test results that show the component or system is qualified for its application and remains qualified after being subjected to the range of normal and abnormal test conditions specified above. The plan establishes separate test regimes for each element of EMC, using the following approaches: a. EMI and RFI Protection. An EMC compliance plan for each component or system identified in the design commitment includes tests to ensure that equipment performs its functions in the presence of the specified EMI/RFI electrical noise environment, including the low range of the EMI spectrum, without equipment damage, spurious actuation, or inhibition of functions. As part of the pre-operational test program, the EMC compliance plan calls for each system to be subjected to EMI/RFI testing. Tests cover potential EMI and RFI susceptibility over four different paths: (1) Power feed lines (2) Input signal lines (3) Output signal lines (4) Radiation

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Electromagnetic Compatibility		
10. (continued)	10. (continued)	10a.(continued) The test program includes sensitivity of components identified in the design commitment to radiation from plant communication transmitters and receivers b. ESD Protection. An EMC compliance plan for each component or system identified in the design commitment includes tests to ensure that equipment performs its functions in the presence of the specified ESD environment without equipment damage, spurious actuation, or inhibition of functions. The plan is structured on the basis that ESD protection is confirmed by factory tests that determine the susceptibility of instrumentation and control equipment to electrostatic discharges. The EMC compliance plan includes standards, conventions, design considerations, and test procedures to ensure ESD protection of the plant instrumentation and control equipment. The plan requires test documentation confirming that, for each component tested, the following conditions have been met:

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Electromagnetic Compatibility		
10. (continued)	10. (continued)	10b.(continued) (1) No change in output signal status was observed during the test. (2) The equipment performed its normal functions after the test. c. SWC Protection. An EMC compliance plan for each component or system identified in the design commitment includes tests to ensure that equipment performs its functions for the specified SWC environment without equipment damage, spurious actuation, or inhibition of functions. The EMC compliance plan includes standards, conventions, design considerations, and test procedures to ensure SWC protection of the plant instrumentation and control equipment. The plan is structured on the basis that SWC protection is confirmed by factory tests that determine the surge withstand capability of the plant instrumentation and control equipment. The plan documents the level of compliance of each system with the grounding and shielding practices of the standards specified under this certified design commitment

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Setpoint Methodology		
11. Setpoints for initiation of safety-related functions are determined, documented, installed and maintained using a process that establishes a plan for: a. Specifying requirements for documenting the bases for selection of trip setpoints b. Accounting for instrument inaccuracies, uncertainties, and drift c. Testing of instrumentation setpoint dynamic response d. Replacement of setpoint-related instrumentation. The setpoint methodology plan requires that activities related to instrument setpoints be documented and stored in retrievable, auditable files.	11. Inspections will be performed of the setpoint methodology plan used to determine, document, install, and maintain instrument setpoints.	11. The setpoint methodology plan is in place. The plan generates requirements for: a. Documentation of data, assumptions, and methods used in the bases for selection of trip setpoints. b. Consideration of instrument channel inaccuracies (including those due to analog-to-digital converters, signal conditioners, temperature compensation circuits, and multiplexing and demultiplexing components), instrument calibration uncertainties, instrument drift, and uncertainties due to environmental conditions (temperature, humidity, pressure, radiation, EMI, power supply variation), measurement errors, and the effect of design basis event transients are included in determining the margin between the trip setpoint and the safety limit. c. The methods used for combining uncertainties. d. Use of written procedures for preoperational testing and tests performed to satisfy the Technical Specifications. e. Documented evaluation for equivalent or better performance of replacement instrumentation which is not identical to the original equipment.

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Equipment Qualification		
12. Qualification of safety-related instrumentation and control equipment is implemented by a program that assures this equipment is able to complete its safety-related function under the environmental conditions that exist up to and including the time the equipment has finished performing that function. Qualification specifications consider conditions that exist during normal, abnormal, and design basis accident events in terms of their cumulative effect on equipment performance for the time period up to the end of equipment life.	12. A review will be conducted of the equipment qualification program	12. An equipment qualification program is in place. Documentation for the EQ program is recorded in a product qualification file that includes a list of safety-related equipment accompanied by the following equipment information: a. Performance specifications under conditions existing during and after design basis accidents. These include voltage, frequency, load, and other electrical characteristics that assure specified equipment performance. b. Environmental conditions at the location where the equipment is installed. These conditions include: (1) number and /or duration of equipment functional and test cycles/events (2) process fluid conditions (where applicable) (3) voltage, frequency, load, and other electrical characteristics of the equipment (4) dynamic loads associated with seismic events (5) containment response to hydrodynamic conditions

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Equipment Qualification		
12. (continued)	12. (continued)	12b.(continued) (6) system transients and other vibration inducing events (7) pressure, temperature, humidity (8) chemical and radiation environments (9) electromagnetic compatibility (10) aging (11) submergence (if any) (12) environmental conditions defined in 10 CFR 50.49 for electrical items (13) consideration of synergistic effects and margins for unquantified uncertainty. c. One (or a combination) of the following testing methods used to qualify the equipment: (1) Testing of an identical item of equipment under identical or similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable. (2) Testing of a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable.

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Equipment Qualification		
12. (continued)	12. (continued)	12c.(continued) (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. d. Documented results of the qualification that show the equipment: (1) Is qualified for its application, and (2) Meets its specified performance requirements when subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life.

Table 3.4: Instrumentation and Control (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Equipment Qualification		
13. The installed condition of safety-related I&C equipment is assured by a program whose objective is to verify that the installed configuration is bounded by the test configuration and test conditions.	13. A review will be conducted of the program established for as-built verification of safety-related I&C equipment.	13. A program for as-built verification is in place and contains documented evaluation of the following elements: a. The installed configuration is bounded by the test configuration and conditions. b. No physical interferences exist with adjacent plant features which have not been addressed by the qualification process. c. Inspection of installed safety-related I&C equipment has been performed in order to assess compatibility with the methods and assumptions used to qualify the equipment.

3.5 Reliability Assurance Program

Applicants utilizing the design defined in this certification will perform a Reliability Assurance Program (RAP). The RAP will have the following two elements:

(1) A Design Reliability Assurance Program (D-RAP) and

(2) An Operational Reliability Assurance Program (O-RAP)

The O-RAP is related to plant operating issues and is not discussed further.

The following is a summary of the ABWR D-RAP.

Introduction

The ABWR Design Reliability Assurance Program (D-RAP) is a program that will be performed during the detailed design and equipment selection phases of a project to assure that the important ABWR reliability assumptions of the PRA will be considered throughout the plant design process. The PRA evaluates plant response to initiating events to assure that plant damage has a very low probability and that risk to the public is very low. Input to the PRA includes details of the plant design and assumptions about the reliability of plant risk-significant structures, systems and components (SSCs) throughout plant life.

The D-RAP will include the design evaluation of ABWR and will identify relevant aspects of plant operation, maintenance, and performance monitoring of important plant SSCs for owner/operator consideration in assuring safety of the equipment and limited risk to the public. The policy and implementation procedures will be specified by the COL applicant.

Scope

The ABWR D-RAP will include the future design evaluation of the ABWR, and it will identify relevant aspects of plant operation, maintenance, and performance monitoring of plant risk-significant SSCs. The PRA for the ABWR and other industry sources will be used to identify and prioritize those SSCs that are important to prevent or mitigate plant transients or other events that could present a risk to the public.

Purpose

The purpose of the D-RAP is to assure that plant safety as estimated by the PRA is maintained as the detailed design evolves through the implementation and procurement phases and pertinent information is provided in the design documentation to assure that equipment reliability, as it affects plant safety, can be maintained through operation and maintenance during the entire plant life.

Objective

The objective of the D-RAP is to identify those plant SSCs that are significant contributors to risk, as identified by the PRA or other sources, and to assure that, during the implementation phase, plant design continues to utilize risk-significant SSCs whose reliability is commensurate with the PRA assumptions. The D-RAP will also identify key assumptions regarding operation, maintenance and monitoring activities that the plant designer should consider in developing the O-RAP.

Summary

This section represents a commitment that combined operating license applicants referencing the certified design will implement a D-RAP program that meets the objectives presented above. There are no inspections, tests, analyses and acceptance criteria (ITAAC) specifically aimed at verifying implementation of the D-RAP commitment.

3.6 Initial Test Program

Design Description

The ABWR Initial Test Program (ITP)is a program that will be conducted following completion of construction and construction-related inspections and tests and extends to commercial operation. The test program will be composed of preoperational and startup test phases. The general objective of the ITP is to confirm that performance of the as-built facility is in compliance with the design characteristics used for SSAR safety evaluations.

The preoperational test phase of the ITP will consist of those test activities conducted prior to fuel loading. Preoperational testing will be conducted to demonstrate proper performance of structures, systems, components, and design features in the assembled plant. Tests will include, as appropriate, logic and interlocks test, control and instrumentation functional tests, equipment functional tests, system operational test, and system vibration and expansion measurements.

The startup test phase of the ITP will begin with fuel loading and extends to commercial operation. The primary objective of the startup phase testing will be to confirm integrated plant performance with the nuclear fuel in the reactor pressure vessel and the plant at various power levels. Startup phase testing will be conducted at five test conditions during power ascension: open vessel, heatup, low power, mid-power, and high power. The normal sequence of tests at a test condition will be:

(1) Core performance analysis,

(2) Steady-state testing,

(3) Control system tuning and demonstration, and

(4) Minor and major transients.

Testing during all phases of the ITP will be conducted using detailed, step by step written procedures to control the conduct of each test. Such detailed test procedures will delineate established test methods and applicable acceptance criteria. The test procedures will be developed from preoperational and startup test specifications. Approved test procedures will be made available to the NRC approximately 60 days prior to their intended use for preoperational tests and 60 days prior to scheduled fuel loading for startup phase tests. The preoperational and startup test specifications will also be made available to the NRC. Administratively, the ITP will be controlled in accordance with a startup administrative manual. This manual will contain the administrative

requirements that govern the conduct of test program, review, evaluation and approval of test results, and test records retention.

Inspections, Tests, Analyses and Acceptance Criteria

This section represents a commitment that combined operating license applicants referencing the certified design will implement an ITP that meets the objectives presented above. Inspections, tests, analyses and acceptance criteria (ITAAC) aimed at verification of ITP implementation are neither necessary nor required.

ABWR DESIGN CERTIFICATION
MEMORANDUM

ROAD MAPS

THE RELATIONSHIP BETWEEN PLANT SAFETY ANALYSES AND OTHER SAFETY-RELATED ISSUES AND TIER 1 ENTRIES

May 21, 1993
GE Nuclear Energy

ABSTRACT

This memorandum provides eleven road maps defining the linkage between ABWR Safety Analysis Report (SAR) sections and the plant inspections, tests, analyses and acceptance criteria (ITAAC). The road maps are intended to show which ITAAC entries will be used to confirm the as-built facility is consistent with SAR safety analysis assumptions and resolutions of other safety-related issues. GE believes the road maps are informal review aids for the NRC staff and does not propose to include this material in the formal ABWR Design Certification documentation.

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Introduction

As part of their review of the ABWR Design Certification application, NRC staff have requested that GE provide so-called road maps that summarize the relationship between the following:

a) The plant safety analyses and other safety-related issues described in the Safety Analysis Report (SAR) and

b) The inspections, tests, analyses and acceptance criteria (ITAAC) included in the Tier 1 material being prepared by GE for the ABWR.

The road-map concept is intended to be a review aid which will assist the NRC staff in concluding that plant characteristics of particular importance to safety will be adequately confirmed by the Tier 1 ITAAC process; i.e., the as-built facility has the characteristics which are consistent with the plant safety analyses assumptions and the resolutions of the plant safety-related issues described in the SAR.

This memorandum provides road maps for eleven safety-related areas described in the SAR. GE believes the material in this memorandum is aimed at assisting the staff review process; consequently, GE does not propose to include road maps in either Tier 2 or Tier 1 of the ABWR Design Certification documentation.

Road Map Selection Criteria

There are no guidelines for clearly identifying which subjects merit road map treatment. However, the request for road map material derives from the structural differences between the ABWR SAR and the proposed Tier 1 material. Typically, the various SAR safety-related issues are treated in one place in the SAR. The Tier 1 material for ABWR is being prepared on a system-by-system basis. Since the resolution of plant safety-related issues typically involve contributions from multiple systems, there is not a direct correlation between the resolution of the plant safety-related issues and the corresponding Tier 1 treatment of the systems involved. In order to provide some correlation, GE has prepared road maps for SAR subjects with the following characteristics:

> Important plant safety-related issues for which acceptable plant performance is dependent upon contributions from multiple ABWR systems.

The eleven subjects identified in Table 1 are in this category.

Table 1

Summary of Major Safety-related Issues Presented in SAR

Issue	SAR Section
Core Cooling Analysis	(CH.6)
Containment Pressure/Temperature Response	(CH.6)
Transient Analysis	(CH.15)
Radiological Analysis	(CH.15,19)
Overpressure Protection	(CH. 5)
Flooding Protection (Includes Internal and External Events)	(CH. 3)
Fire Protection	(CH.9)
ATWS Analysis	(CH. 15)
PRA Analysis (Includes Severe Accidents, Station Blackout and Shutdown Risk)	(CH.19)
Generic Safety Issues	(CH. 19B)
TMI Issues	(CH. 19B)

Parameter Selection Criteria

The selection of the road map parameters is based on the significant design parameters identified in the SAR to support a given safety-related issue or analysis. The selection of plant design parameters to be addressed in ITAAC are defined by the principles and guidelines underlying the tiered approach that has been adopted for the 10 CFR Part 52 design certification process. As a result, not all design parameters discussed in the SAR have supporting ITAAC entries. Table 2 summarizes the basic ITAAC selection criteria.

Table 2

Safety-Related Parameters Confirmed by ITAAC Entries

Characteristic	Bases
1. The parameter has a primary influence on the safety analysis results such that small variations could possibly result in changes to analysis results of some safety significance.	Consistent with the tiered approach to design certification.
2. The parameter is a major plant design basis assumption.	Consistent with the ITAAC process which addresses important plant design basis assumptions.
3. The parameter is a measurable or observable characteristic of the plant which can be verified prior to fuel loading.	Consistent with Part 52 regulations that the ITAAC process must be completed before fuel loading.
4. The parameter is a plant design characteristic and <u>not</u> (1) an operating condition, (2) dependent on as-built, as-procured equipment characteristics, or (3) a procedural commitment.	Consistent with the ITAAC process which does not encompass plant operating conditions.

Road Maps

Tables 3 through 13 present the proposed road maps for the eleven safety-related subjects identified in Table 1.

Table 3
Core Cooling Analysis

SSAR Entry	Parameter	Value	Verifying ITAAC
6.3.3.5	Following a LOCA the RHR System is Automatically Directed to the LPFL Mode	----	2.4.1
6.3.3.7.4	The Safety Related Systems Will Operate as Designed with the Loss of All Offsite AC Power	----	2.12.13
Table 6.3-1	Plant Parameters at 102% of Rated Core Thermal Power		
	Core Thermal Power (MW_t)	4005	none
	Vessel Steam Output (kg/hr x 10^6)	7.85	none
	Vessel Steam Dome Pressure (kg/cm^2a)	74.2	none
	Low Pressure Flooder System		
	Vessel Pressure at which Flow May Commence (kg/cm^2d -- vessel to drywell)	15.8	2.4.1
	Min. Rated Flow (m^3/hr per pump)	954	2.4.1
	at Vessel Pressure (kg/cm^2d -- vessel to drywell)	2.8	2.4.1
	Initiating Signals		
	Low Water Level	----	2.4.1
	Analytical Setpoint (cm above TAF)	18.3	none
	or		
	High Drywell Pressure	----	2.4.1
	Setpoint (kg/cm^2g)	0.14	none

Table 3
Core Cooling Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
Table 6.3-1	Low Pressure Flooder System (Cont.)		
	Maximum Allowable Time Delay from Initiating Signal to Pumps at Rated Speed (sec)	29.0	2.4.1
	Maximum Allowable Time Delay from Low Pressure Permissive Signal to Injection Valve Fully Open (sec)	36.0	2.4.1
	Reactor Core Isolation Cooling System		
	Vessel Pressure at which Flow May Commence (kg/cm^2d -- vessel to drywell)	82.75	2.4.4
	Min. Rated Flow (m^3/hr)	182	2.4.4
	at Vessel Pressures (kg/cm^2d -- vessel to pump suction) to	82.75 to 10.55	2.4.4
	Initiating Signals		
	Low Water Level	----	2.4.4
	Setpoint (cm above TAF)	246.9	none
	or		
	High Drywell Pressure	----	2.4.4
	Setpoint (kg/cm^2g)	0.14	none
	Maximum Allowable Time Delay from Initiating Signal to Rated Flow Available and Injection Valve Fully Open (sec)	29.0	none

Table 3
Core Cooling Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
Table 6.3-1	High Pressure Core Flooder System		
	Vessel Pressure at which Flow May Commence (kg/cm^2d -- vessel to drywell)	82.75	2.4.2
	Minimum Rated Flows (m^3/hr per subsystem)	182 to 727	2.4.2
	at Vessel Pressures (kg/cm^2d -- vessel to pump suction)	82.75 to 7.0	2.4.2
	Initiating Signals		
	Low Water Level	----	2.4.2
	Setpoint (cm above TAF)	103.6	none
	or		
	High Drywell Pressure	----	2.4.2
	Setpoint (kg/cm^2g)	0.14	none
	Maximum Allowable Time Delay from Initiating Signal to Rated Flow Available and Injection Valve Fully Open (sec)	36.0	2.4.2
	Automatic Depressurization System		
	Total Number of Relief Valves with ADS Function	8	2.1.2
	Min. Flow Capacity ($kg/hr \times 10^6$)	2.903	2.1.2
	at Vessel Pressure (kg/cm^2g)	79.1	2.1.2

Table 3
Core Cooling Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
Table 6.3-1	Automatic Depressurization System (Cont.)		
	Initiating Signals		
	Low Water Level	----	2.1.2
	Setpoint (cm above TAF)	18.3	none
	and		
	High Drywell Pressure	----	2.1.2
	Setpoint (kg/cm^2g)	0.14	none
	or		
	High Drywell Pressure Bypass Timer Timed Out	----	2.1.2
	Setpoint (sec)	480	2.1.2
	Delay Time from All Initiating Signals Completed to the Time Valves are Open (sec)	29.0	2.1.2
	Fuel Parameters		
	Fuel Type	Initial Core	none
	Fuel Bundle Geometry	8x8	none
	Lattice	C	none
	Number of Rods per Bundle	62	none
	Peak Technical Specification Linear Heat Generation Rate (kw/m)	44.0	none
	Initial MCPR	1.13	none
	Design Axial Peaking Factor	1.40	none

Table 3
Core Cooling Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
Table 6.3-3	The RHR Subsystems are Divisionally Separated	----	2.4.1
	The HPCF Subsystems are Divisionally Separated	----	2.4.2
	RCIC Operation Does not Required AC Power	----	2.4.4
	A Single Failure Will not Prevent the Operation of More Than One ADS Valve	----	2.1.2
Table 6.3-4	LOCA Break Sizes		
	Steamline (cm^2)	984.8	2.1.1
	Feedwater Line (cm^2)	838.9	2.1.1
	RHR Shutdown Cooling Suction Line (cm^2)	791.5	2.1.1
	RHR Injection Line (cm^2)	205.3	2.1.1
	High Pressure Core Flooder (cm^2)	92.0	2.1.1
	Bottom head Drain Line (cm^2)	20.25	2.1.1
Table 15.6-4	MSIV Closure Initiated by High Steam Flow	----	2.4.3
	Scram Initiated by MSIV Closure	----	2.2.7
Table 15.6-15	Scram Initiated by Low Water Level 3	----	2.2.7

Table 4
Containment Pressure/Temperature Response

SSAR Entry	Parameter	Value	Verifying ITAAC
6.2.1.1.4.1	Suppression Pool Surface Area (m^2)	506.6	none
	Total Surface of Drywell Connecting Vents (m^3)	11.3	2.14.1
	Vacuum Breakers		
	Diameter (mm)	500	2.14.1
	Quantity	8	2.14.1
Table 6.2-2	Drywell		
	Net Free Volume (m^3)	7350	none
	Leak Rate (%/Day)	0.5	2.14.1
	Wetwell		
	Air Space Volume (m^3)	5960	none
	Leak Rate (%/Day)	0.5	2.14.1
	Min. Suppression Pool Water Volume (m^3)	3580	2.14.1
	Vent System		
	Number of Vents	30	2.14.1
	Nominal Vent Diameter (m)	0.7	2.14.1
	Total Horizontal Vent Area (m^2)	11.55	2.14.1

Table 4
Containment Pressure/Temperature Response (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
Table 6.2-2	Vent System (Cont.)		
	Vent Centerline Submergence (at Low Level)		
	Top Row (m)	3.5	2.14.1
	Middle Row (m)	4.87	2.14.1
	Bottom Row (m)	6.24	2.14.1
Table 6.2.2-a	Containment Spray		
	Number of RHR Subsystems (Pump Plus Heat Exchanger)	2	2.4.1
	Drywell Spray Flow Rate per RHR Subsystem (kg/hr x 10^5)	8.21	none
	Wetwell Spray Flow Rate per RHR Subsystem (kg/hr x 10^5)	1.12	2.4.1
	Containment Cooling System		
	Number of RHR Subsystems (Pump Plus Heat Exchanger)	3	2.4.1
	Pump Capacity (m^3/hr per pump)	954	2.4.1
	Overall Heat Transfer Coefficient (kcal/sec-^oC)	88.5	2.4.1

Table 4
Containment Pressure/Temperature Response (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
Table 6.3-4	LOCA Break Sizes		
	Steamline (cm^2)	984.8	2.1.1
	Feedwater Line (cm^2)	838.9	2.1.1
	RHR Shutdown Cooling Suction Line (cm^2)	791.5	2.1.1
	RHR Injection Line (cm^2)	205.3	2.1.1
	High Pressure Core Flooder (cm^2)	92.0	2.1.1
	Bottom head Drain Line (cm^2)	20.25	2.1.1

Table 5
Transient Analysis

SSAR Entry	Parameter	Value	Verifying ITAAC
Table 15.0-1	Plant Parameters at 102% of Rated Core Thermal Power		
	Core Thermal Power (MW_t)	4005	none
	Vessel Steam Output (kg/hr x 10^6)	7.84	none
	Vessel Steam Dome Pressure (kg/cm^2g)	73.1	none
	Vessel Core Flow		
	Rated (kg/hr x 10^6)	52.2	none
	Maximum (kg/hr x 10^6)	58.0	none
	Feedwater Temperature (^oC)	217	none
	Turbine Bypass Capacity (% NBR)	33	none
	Turbine Inlet Pressure (kg/cm^2a)	69.9	none
	Fuel Lattice Type	N	none
	Core Leakage Flow (%)	11.67	none
	Reactor Internal Recirculation Pumps		
	Number of Pumps	10	2.1.3
	RPT Trip Delay (sec)	0.16	none
	Pump Inertia Time Constant (sec)	0.62	2.1.3

Table 5
Transient Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
Table 15.0-1	Relief Valve (Relief Function)		
	Capacity (% NBR Steam Flow at 80.5 kg/cm^2g)	91.3	2.1.2
	Number of Valves	18	2.1.2
	Time Delay (sec)	0.4	none
	Opening Time (sec)	0.15	2.1.2
	Setpoint Range	80.5 to 84.0	none
	Number of Valve Groupings	6	none
	High Flux Trip Scram	----	2.2.5
	Analytical Setpoint (%NBR)	127.5	none
	Vessel Level Analytical Trip Setpoints (m above bottom of separator skirt bottom)		
	Level 8	1.73	none
	Level 4	1.08	none
	Level 3	0.57	none
	Level 2	-0.75	none
	APRM Simulated Thermal Power Trip Scram	----	2.2.5
	Time Constant (sec)	7	none
	Analytical Setpoint (% NBR)	117.3	none
	Total Steamline Volume (m^3)	113.2	2.1.2

Table 5
Transient Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
Table 15.0-6	FMCRD Scram Times		
	10% Rod Insertion (sec)	0.46	2.2.2
	40% Rod Insertion (sec)	1.208	2.2.2
	60% Rod Insertion (sec)	1.727	2.2.2
	100% Rod Insertion (sec)	3.719	2.2.2
15.1.1.1.1	Maximum Feedwater Temperature Loss Due to a Single Operator Error or Equipment Failure (^{o}C)	55.6	none
Table 15.1-5	Max Feedwater Runout Capacity (% of rated flow at the design pressure of 74.9 kg/cm^2g)	130	none
	High Water Level 8 Initiates		
	Turbine Trip	----	none
	Feedwater Pump Trip	----	2.2.3
	Turbine Stop Valve Position Switches Initiate		
	Reactor Scram	----	2.2.7
	Trip of 4 RIPs	----	2.2.8
Table 15.1-6	Low Water Level 2 Initiates		
	Trip of 6 RIPs	----	2.2.8
	RCIC System	----	2.4.4
	Maximum Startup Time (sec)	30	none
	MSIV Closure on Low Turbine Inlet Pressure	----	2.4.3
15.1.3.3.1	Maximum MSIV Closure Time (sec -- assumes 0.5 sec for instrument delay)	5.0	2.1.2

Table 5
Transient Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
15.1.4.3	RHR Suppression Pool Cooling Initiated on High Pool Temperature	----	none
	Setpoint (^{o}C)	43.3	none
	Reactor Scram Initiated on High Pool Temperature	----	none
	Setpoint (^{o}C)	48.9	none
15.2.1.3.1	Main Steam Flow with 1 TCV Closed (% of rated flow)	85	none
	TCV Full Stroke Servo Closure (sec)	2.5	2.10.7
Table 15.2-1a	Low Water Level 3 Initiates Trip of 4 RIPs	----	2.2.8
Table 15.2-2	High Dome Pressure Initiates Trip of 4 RIPs	----	2.2.8
Table 15.2-3	T/G Load Rejection Initiates		
	Turbine Control Valve Fast Closure	----	2.10.7
	Turbine Bypass System Operation on High Pressure	----	2.10.13
	Fast Control Valve Closure Initiates		
	Scram	----	2.2.7
	Trip of 4 RIPs	----	2.2.8
15.2.2.3.1	TCV Full Stroke Fast Closure (sec)	0.15	2.10.7
Table 15.2-6	Turbine Trip Initiates		
	Turbine Control Valve Fast Closure	----	2.10.7
	Turbine Bypass System Operation on High Pressure	-----	2.10.13

Table 5
Transient Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
Table 15.2-6 (Cont.)	Turbine Stop Valve Position Switch Setpoint (% of open position)	85	none
15.2.3.3.1	Turbine Stop Valve Full Stroke Closure (sec)	0.10	2.10.7
Table 15.2-9	MSIV Position Switches Initiate		
	Scram	----	2.2.7
	Setpoint (% of open position)	85	none
15.2.4.3.1	Minimum MSIV Closure Time (sec)	3.0	2.1.2
15.2.5.3.1	Condenser Vacuum Decay Rate (cm Hg/sec)	5.1	none
Table 15.2-14	Low Condenser Vacuum Initiates		
	Turbine Trip	----	none
	MSIV Closure	----	2.4.3
	Main Bypass Valve Closure	----	none
15.2.6.1.1.2	RIP M/G Set		
	Number of RIPs	6	2.2.8
	Length of Time Hold Original Speed (sec)	1.0	none
	RIP Coastdown		
	Rate (% per sec)	10	none
	Length of Time (sec)	2.0	none
	Time of RIP Trip (sec)	3.0	none

Table 5
Transient Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
Table 15.2-17	Low Water Level 4 Initiates Recirculation Flow Runback	----	none
	Low Water Level 3 Initiates Reactor Scram	----	2.2.7
15.2.7.2.2	Maximum Delay Time (sec)	1.0	none
	Meets Single-failure Criterion	----	2.2.7
15.2.9	RHR System has 3 Independent Divisions	----	2.4.1
15.3.1.1.1	No More Than 3 RIPs on One Elecrical Power Bus	----	2.2.8
15.3.1.2.2.2	Rapid Core Flow Coastdown Initiates Reactor Scram	----	2.2.5
15.3.2.3.1.1	Maximum RIP Speed Decreasing Rate		
	One RIP (%/sec)	40	none
	All RIPs Simultaneously (%/sec)	5	none
15.4.1.1.2.2	Mode Switch in the **Refuel** Position		
	Refueling Platform Cannot Be Moved Over the Core If a Control Rod is Withdrawn and Fuel is on the Hoist	----	2.5.5
	Control Rods Cannot Be Moved if the Refueling Platform is Over the Core and the Fuel is on the Hoist	----	2.5.5
	Only One or Two Control Rods Associated with the Same HCU Can Be Withdrawn	----	2.2.1

Table 5
Transient Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
15.4.1.2.1	On Short Flux Period SRNMs Generate		
	Rod Block	----	none
	Setpoint (sec)	20	none
	Reactor Scram	----	2.2.5
	Setpoint (sec)	10	none
15.4.1.2.3.2	FMCRD Withdrawal Speed (mm/sec)	30	2.2.2
15.4.2.1	At Power the ATLM of the RCIS Prevents Rod Withdrawal Based on MCPR and APLHGR Limits	----	2.2.7
15.4.4.1.1	Overcurrent Protection Logic on the Electrical Bus Which Supplies the Power to the RIPs	----	2.12.1
15.4.5.1.1	Maximum RIP Speed Increasing Rate		
	One RIP (%/sec)	40	none
	All RIPs Simultaneously (%/sec)	5	none
15.4.5.3.1	Maximum Core Flow (% of rated)	120	none
15.4.8.1	FMCRD Designed to Prevent Rod Ejection	----	2.2.2
15.4.9.1	FMCRD Designed to Prevent Separation of Control Blade and Drive	----	2.2.2
15.5.1.3.1	Min.HPCF System Water Temperature (^{o}C)	4.4	none
15.5.1.3.2	Max. HPCF Flow (% of flow at rated pressure)	138	none

Table 6
Radiological Analysis

SSAR Entry	Parameter	Value	Verifying ITAAC
Table 15.6-5	Maximum MSIV Closure Time (sec) (Assumes 0.5 sec for instrument delay.)	5.0	2.1.2
Table 15.6-8	Core Thermal Power at 102% of Rated (MW_t)	4005	none
	Primary Containment Leakage Rate (% per day)	0.5	2.14.1
	MSIV Total Leakage Rate for All Lines (SCFH)	140	2.1.2
	Reactor Building Secondary Containment Leakage Rate (% per day)	50	none
	SGTS		
	Filter Efficiency (%)	99	2.14.4
	Drawdown Time (min)	20	none
	Parameters for MSIV Leakage Calculation		
	Main Steamline		
	Minimum Length (m)	47.9	none
	Nominal Inside Radius (cm)	31.98	none
	Nominal Outside Radius (cm)	35.55	none
	Nom. Insulation Thickness (cm)	12.0	none
	Main Steam Drain Line Outboard of Outboard MSIV		
	Minimum Length (m)	71.6	none
	Nominal Inside Radius (cm)	3.33	none
	Nominal Outside Radius (cm)	4.45	none
	Nom. Insulation Thickness (cm)	6.5	none

Table 6
Radiological Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
Table 15.6-8	Min. Condenser Free Air Volume (m^3)	1246	none
	Control Room		
	Free Air Volumes		
	Total (m^3)	5509	none
	Largest Single Room (m^3)	1400	none
	Recirculation Rates		
	Max. Filtered Intake (m^3/sec)	0.1	none
	Unfiltered Intake	none	none
	Min. Filtered Recirculation (m^3/sec)	0.65	none
	Min. Charcoal Efficiency (%)	95	none
15.7.3.1	All Compartments Containing High Liquid Radwastes are Steeled Lined Up to a Height Capable of Containing the Release of All the Liquid Radwastes into the Compartment.	----	none
Table 15.7-8	Min. Height of Water Above Fuel in Spent Fuel Pool (m)	7.0	none

Table 7
Overpressure Protection

SSAR Entry	Parameter	Value	Verifying ITAAC
5.2.2.1.4	Direct Scram Signal Generated By:		
	Position Switches on		
	MSIVs	----	2.2.7
	Turbine Stop Valves	----	2.2.7
	Pressure Swiches on		
	TCV Hydraulic Actuation System Dump Valve	----	2.2.7
5.2.2.2.2.1	Plant Parameters at 102% of Rated Core Thermal Power		
	Core Thermal Power (MW_t)	4005	none
	Vessel Steam Output (kg/hr x 10^6)	7.844	none
	Vessel Steam Dome Pressure (kg/cm^2g)	73.1	none
Table 5.2-2	Scram Signal on		
	High Flux	----	2.2.5
	Recirculation Pump Trip on		
	High Vessel Pressure	----	2.2.8
	Setpoint (kg/cm^2g)	79.1	none

Table 7
Overpressure Protection (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
Table 5.2-3	Safety/Relief Valve		
	Analytical Spring Set Pressure		
	2 SRVs (kg/cm^2g)	80.8	none
	Capacity per valve (kg/hr) (103% Spring Set Pressure)	395000	2.1.2
	4 SRVs (kg/cm^2g)	81.5	none
	Capacity per valve (kg/hr) (103% Spring Set Pressure)	399000	2.1.2
	4 SRVs (kg/cm^2g)	82.2	none
	Capacity per valve (kg/hr) (103% Spring Set Pressure)	402000	2.1.2
	4 SRVs (kg/cm^2g)	82.9	none
	Capacity per valve (kg/hr) (103% Spring Set Pressure)	406000	2.1.2
	4 SRVs (kg/cm^2g)	83.6	none
	Capacity per valve (kg/hr) (103% Spring Set Pressure)	409000	2.1.2
	No. of Valves	18	2.1.2
Figure 5.2-1	SRV Safety Function Opening Time (sec)	0.3	2.1.2

Table 8
Flooding Protection

SSAR Entry	Parameter	Value	Verifying ITAAC
	Reactor and Control Building Flood Protection (from External Sources)		
3.4.1.1	Pipe Penetrations Below Design Flood Level Will Be Sealed Against Hydrostatic Head Inside Tunnel or Connecting Building	----	2.15.10 2.15.12
3.4.1.1.1	Min. Wall Thicknesses Below Design Flood Level (cm)	60	2.15.10 2.15.12
	Water Stops Provided in All Expansion and Construction Joints Below Design Flood Level	----	none
	Watertight Doors and Equipment Hatches Installed Below Design Flood Level	----	2.15.10 2.15.12
	Minimum Height of Waterproof Coating on External Surfaces (cm above ground Level)	8	none
	Roofs Designed to Facilitate Drainage and Prevent Pooling of Water	----	2.15.10 2.15.12
	Plant Entry Elevation (cm above grade)	30	none
	Procedures Assure Watertight Doors and Hatch Covers Are Locked in the Event of a Flood Warning	----	none

Table 8
Flooding Protection (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
	Reactor Building Flood Protection (from Internal Sources)		
3.4.1.1.2	All Piping, Vessels and Head Exchangers with Flooding Potential are Seismically Analyzed		
	Standby Liquid Control System	----	2.2.4
	Residual Heat Removal System	----	2.4.1
	High Pressure Core Flooder System	----	2.4.2
	Reactor Core Isolation Cooling System	----	2.4.4
	Reactor Water Cleanup System	----	none
	Fuel Pool Cooling and Cleanup System	----	none
	Suppression Pool Cleanup System	----	none
	Makeup Water (Purified) System	----	none
	Makeup Water (Condensate)	----	none
	Reactor Building Cooling Water System	----	2.11.3
	HVAC Normal Cooling Water System	----	none
	HVAC Emergency Cooling Water Sys.	----	2.11.6
	Reactor Service Water System	----	2.11.9
	Hot Water Heating System	----	none
	Fire Protection System	----	none
	Oil Storage and Transfer System	----	2.16.2
	Main Steamlines (Inside Reactor Bldg)	----	2.1.2
	Feedwater Lines (Inside Reactor Bldig)	----	2.1.2

Table 8
Flooding Protection (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
	Reactor Building Flood Protection (from Internal Sources) -- Cont.		
	MSIVs Automatically Close on High Radiation or Temperature in Main SteamlineTunnel	----	2.4.3
	All Rooms Are Supplied With Floor Drains	----	2.9.2
	Manual Firefighting Flowrate (m^3/min)	.57	none
	Safety-Related Equipment Raised Off the Floor (mm)	200	2.15.10
3.4.1.1.2.1.1	Evaluation of Floor 100 (B3F)		
	Maximum RHR Suction Line Size (mm)	450	none
	Watertight Doors on Compartments Containing ECCS Equipment	----	2.15.10
	Sump Pump Alarms in RHR HX Rooms	----	none
	Maximum Suppression Pool Cleanup System Suction Line Size (mm)	200	none
3.4.1.1.2.1.2	Evaluation of Floor 200 (B2F)		
	Maximum RHR Pressure Line Size (mm)	250	none
	RHR Pressure Lines Inside Pipe Chases	----	2.15.10
	Maximum RCW Pressure Line Size (mm)	400	none
	Maximum Line Size in Fourth Quadrant (mm)	200	none
	Minimum Floor Spread Area (m^2)	300	2.15.10

Table 8
Flooding Protection (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
	Reactor Building Flood Protection (from Internal Sources) -- Cont.		
3.4.1.1.2.1.3	Evaluation of Floor 300 (B1F)		
	Maximum RHR Pressure Line Size (mm)	250	none
	Maximum Capacity of Line Inventory Plus Surge Tank Capacity of HVAC Chilled Water Supply or Emergency Cooling Water System That Will Be Released Following A Break (m^3)	8	none
3.4.1.1.2.1.4	Evaluation of Floor 400 (1F)		
	Maximum RHR Pressure Line Size (mm)	250	none
	RHR, HPCF and RCIC Lines in Pipe Chases	----	2.15.10
	Maximum RCW Line Size in Emergency Diesel Generator Rooms (mm)	200	none
	Foam Sprinkler System in Diesel Generator Areas	----	2.15.16

Table 8
Flooding Protection (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
	Reactor Building Flood Protection (from Internal Sources) -- Cont.		
3.4.1.1.2.1.5	Evaluation of Floor 500 (2F)		
	Maximum FPC Line Size (mm)	200	none
	Divisional DG Equipment Areas are Separated and Mechanically Isolated from Each Other	----	2.15.10
	Maximum RCW Line Size (mm)	300	none
	FPC Pools have Stainless Steel Liners to Prevent Leakage into the Pool Structure	----	none
	No Water Line Connections in SGTS Monitor and Stack Monitor Rooms	----	none
	Steamline Tunnel Area Isolated by Sealed Doors and Firewalls	----	2.15.10
3.4.1.1.2.1.6	Evaluation of Floor 600 (3F)		
	Emergency Diesel Generator Fuel Day Tank Maximum Capacity (m^3 per Tank)	11.4	none
	Raised Sills on Entry Ways to		
	Areas Containing Emergency Diesel Fuel (cm)	20	none
	Foam Sprinkler System in Fuel Storage Tank Areas	----	2.15.6
	Low Water Level Alarms on Standby Liquid Control Tanks	----	2.2.4

Table 8
Flooding Protection (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
	Reactor Building Flood Protection (from Internal Sources) -- Cont.		
3.4.1.1.2.1.7	Evaluation oı Floor 700 (M4F)		
	(No Additional Requirements)		
3.4.1.1.2.1.8	Evaluation of Floor 800 (4F)		
	Each RCW Surge Tank A,B & C and Its Associated Piping Is in a Separate Compartment	----	2.15.10
3.4.1.1.2.2	Control Building Flood Protection (from Internal Sources)		
	No Openings into the Control Building from the Steam Tunnel	----	2.15.12
	The Steam Tunnel		
	Sealed At the Reactor Building End	----	2.15.10
	Open At the Turbine Building End	----	none
	Thickness of Steam Tunnel Walls (cm)	160	none
	All Rooms Are Supplied With Floor Drains	----	2.9.2
	Maximum Service Water Line Size (mm)	700	none
	High Water Level in RCW/RSW Heat Exchanger Room Will Automatically Close RSW Isolation Valves and Stop Pumps	----	2.15.12
	Setpoint (mm above basemat)	1500	none
	Maximum Length of Service Water Piping Out to Ultimate Heat Sink (km)	4	none

Table 8
Flooding Protection (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
	Control Building Flood Protection (from Internal Sources) -- Cont.		
	Water Tight Doors on RCW/RSW Heat Exchanger Rooms	----	2.15.12
	Maximum Total Released Inventory of Chilled Water Line and Surge Tank Following a Break (m^3)	6	none
	Elevation Differences Separate Control Area from Water Sources	----	2.15.12
	Redundant Mechanical Functions are Physically Separated	----	2.15.12
	Safety-Related Equipment Raised Off the Floor (mm)	200	2.15.12
	Radwaste Building Flood Protection (from Internal Sources)		
	Radwaste Tanks in Sealed Compartments to Contain Any Spillage or Leakage	----	none
	Piping from Other Buildings Sealed in Water Tight Tunnel	----	none
	Service Building Flood Protection (from Internal Sources)		
	All Rooms Are Supplied With Floor Drains	----	2.9.2

Table 8
Flooding Protection (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
	Turbine Building Flood Protection (from Internal Sources)		
	Normally Closed Alarmed Door in Passage From Service Building	----	2.15.11
	Radwaste Tunnel Sealed at All Ends	----	none
	Non-Water Tight Truck Door at Grade	----	none
	High Water Level in Condenser Pit Automatically Shuts Down Circulating Water System	----	2.10.23

Table 9
Fire Protection

(Reactor and Control Building)

SSAR Entry	Parameter	Value	Verifying ITAAC
9A.2.4	Suspended Ceilings, Including Lighting Fixtures, are of Noncombustible Construction	----	none
	Electrical Cable Fire-stops Have Fire Rating Equal to Rating of Barrier They Penetrate	----	2.15.10 2.15.11
	Control, Power or Instrument Cables of Systems Having Similar Safety Related or Shutdown Functions are Located in Separate Fire-resistive Enclosures.	----	2.12.1
	A Minimum of Two Fire Suppresssion Means is Available to Each Fire Area	----	2.15.6
	HVAC Penetrations at Fire Wall Have Fire Dampers Which Have Fire Rating Equal to Rating of Barrier They Penetrate	----	none
9A.3.2	No Openings in the Steam Tunnel Walls Within the Control Building	----	2.15.11
	Water Curtain Spray System at Steam Tunnel Exit from the Control Building to the Turbine Building	----	none
9A.4.1.1.1	Drywell Inerted During Plant Operation	----	2.14.6
	Drywell Has Purge and Vent System	----	2.15.5
9A.4.1.1.2	Wetwell Inerted During Plant Operation	----	2.14.6
	Wetwell Has Spray System	----	2.4.1
Appendix 9A	Systems Having Similar Safety Related or Shutdown Functions are Located in Separate Fire-resistive Enclosures.	----	2.15.11

Table 9
Fire Protection (Cont.)

(Reactor and Control Building)

SSAR Entry	Parameter	Value	Verifying ITAAC
Appendix 9A	A Means of Fire Detection, Alarming and Suppression is Provided and Accessible.	----	2.15.6
	Fire Stops Are Provided for Cable Tray and Piping Penetrations Through Rated Fire Barriers	----	2.15.10 2.15.11
	Non-Safety Related Equipment is Located in Rooms Separate from Rooms Which Contain Safety Related Equipment	----	2.15.10 2.15.11
	Stair Towers are Located in Separate Fire-resistive Enclosures	----	none
	Alternate Means of Access and Egress are Provided by a Separate Stair Tower, Elevator or Corridor	----	2.15.10 2.15.11
	Elevator Shafts are Located in Separate Fire-resistive Enclosures	----	none

Table 10
ATWS Analysis

SSAR Entry	Parameter	Value	Verifying ITAAC
	Nominal Initial Operating Conditions		
Table 15E.3-1	Plant Parameters at 100% of Rated Core Thermal Power		
	Core Thermal Power (MW_t)	3926	none
	Vessel Steam Output (kg/hr x 10^6)	7.64	none
	Vessel Steam Dome Pressure (kg/cm^2g)	72.1	none
	Feedwater Temperature (^oC)	215.6	none
	Minimum Suppression Pool Volume (m^3)	3580	2.14.1
	Suppression Pool Temperature (^oC)	37.7	none
	Condensate Storage Temperature (^oC)	48.9	none
	Equipment Performance Characteristics		
15.8.2	Minimum SLCS Capacity (m^3/hr)	22.7	2.2.4
Table 15E.3-2	Minimum Closure Time of MSIV (sec)	3.0	2.1.2
	Relief Valve		
	Capacity (%NBR Steam Flow at 80.5 kg/cm^2g)	91.3	2.1.2
	Number of Valves	18	2.1.2
	Analytical Setpoint Range (kg/cm^2g)	80.5 to 84.0	none
	Opening Time (sec)	0.15	2.1.2

Table 10
ATWS Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
	Equipment Performance Characteristics		
Table 15E.3-2	Reactor Core Isolation Cooling System		
	Min. Rated Flow (kg/hr)	50.4	2.4.4
	at Vessel Pressures (kg/cm^{2}d -- vessel to pump suction)	82.75 to 10.55	2.4.4
	Initiates on Low Water Level	----	2.4.4
	Maximum Allowable Time Delay from Initiating Signal to Rated Flow Available and Injection Valve Fully Open (sec)	29.0	2.4.4
	High Pressure Core Flooder System		
	Number of Subsystems	2	2.4.2
	Minimum Rated Flows (kg/sec per subsystem)	50.4 to 201.6	2.4.2
	at Vessel Pressures (kg/cm^{2}d -- vessel to pump suction)	82.75 to 7.0	2.4.2
	Initiates on Low Water Level	----	2.4.2
	Injection Terminated on High Water Level	----	2.4.2
	Maximum Allowable Time Delay from Initiating Signal to Rated Flow Available and Injection Valve Fully Open (Does not include diesel start time and Loading sequence --sec)	20.0	2.4.2

Table 10
ATWS Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
	Equipment Performance Characteristics		
Table 15E.3-2	ATWS Dome Pressure Sensor Time Constant (sec)	0.3	none
	ATWS Logic Time Delay (sec)	0.5	none
	Nominal Recirculation Pump System Inertia (kg-m^2)	21.5	2.1.3
	Delay Before Start of Electro-Hydraulic Rod Insertion		
	With Off-site Power Available (sec)	1.0	none
	Without Off-site Power Available (Assumes 1.0 sec for instrument delay --sec)	39.0	none
	Maximum Electro-Hydraulic Control Rod Insertion Time (sec)	135	2.2.2
	Maximum ARI Rod Insertion Time (sec)	25	none
	Minimum RHR Pool Cooling Capacity (Kcal/sec-^oC)	265	2.4.1
	MSIV Closure Initiated on Low Water Level	----	2.4.3
	MSIV Closure Initiated on Low Steamline Pressure	----	2.4.3
	Analytical Setpoint (kg/cm^2g)	52.7	none

Table 10
ATWS Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
	ATWS Logic and Setpoints		
15E.4	ARI and FMCRD Run-in Initiated on		
	High Dome Pressure	----	2.2.8
	Analytical Setpoint (kg/cm^2g)	79.1	none
	or		
	Low Water Level 2	----	2.2.8
	Analytical Setpoint (cm above TAF)	246.9	none
	SLCS Initiated on an ATWS Trip Signal	----	2.2.4
	ATWS Trip Signals for SLCS Initiation		
	High Dome Pressure	----	2.2.8
	Analytical Setpoint (kg/cm^2g)	79.1	none
	and		
	SRNM Not Downscale	----	2.2.8
	Analytical Time Delay (minutes)	3	2.2.8
	or		
	Low Water Level 2	----	2.2.8
	Analytical Setpoint (cm above TAF)	246.9	none
	and		
	SRNM Not Downscale	----	2.2.8
	Analytical Time Delay (minutes)	3	2.2.8
	or		
	Manual ARI/FMCRD Run-in Signals	----	2.2.8
	and		
	SRNM Not Downscale	----	2.2.8
	Analytical Time Delay (minutes)	3	2.2.8
	RPT (RIPs not Connected to M/G Set) Initiated on		
	High Dome Pressure	----	2.2.8
	Analytical Setpoint (kg/cm^2g)	79.1	none

Table 10
ATWS Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
	ATWS Logic and Setpoints		
15E.4	RPT (RIPs Connected to M/G Set) Initiated on		
	Low Water Level 2	----	2.2.8
	Analytical Setpoint (cm above TAF)	246.9	none
	Recirculation Runback Initated on		
	Any Scram Signal	----	2.2.8
	or		
	Any ARI/FMCRD Run-in Signal	----	2.2.8
	Feedwater Runback Initiated on an ATWS Trip Signal	----	2.2.3
	ATWS Trip Signals for Feedwater Runback		
	High Dome Pressure	----	3.4
	Analytical Setpoint (kg/cm^{2}g)	79.1	none
	and		
	SRNM Not Downscale	----	3.4
	Analytical Time Delay (minutes)	2	3.4

Table 10
ATWS Analysis (Cont.)

SSAR Entry	Parameter	Value	Verifying ITAAC
	ATWS Logic and Setpoints		
	ADS Inhibit Initiated on an ATWS Trip Signal	----	2.1.2
	ATWS Trip Signals for ADS Inhibit		
	High Dome Pressure	----	3.4
	Analytical Setpoint (kg/cm^2g)	79.1	none
	and		
	SRNM Not Downscale	----	3.4
	Analytical Time Delay (minutes)	2	3.4
	or		
	Low Water Level 2	----	3.4
	Analytical Setpoint (cm above TAF)	246.9	none
	and		
	SRNM Not Downscale	----	3.4
	Analytical Time Delay (seconds)	25	3.4

Table 11
PRA Analysis

[Later]

Table 12
Generic Safety Issues

[Later]

Table 13
TMI Issues

[Later]