GE Nuclear Energy

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Docket No. STN 52-001

Chet Poslusny, Senior Project Manager
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of the Nuclear Reactor Regulation

Subject: Submittal Supporting Accelerated ABWR Review Schedule - **Tier 1/ITAAC: Submittal of Revised Material from the January 11 through January 21, 1993 and March 8 through March 12, 1993 GE/NRC Meetings**

Dear Chet:

Enclosed are revised versions of the ABWR Tier 1 ITAAC material that was reviewed during the January 11 through January 21 and March 8 through March 12, 1993, GE/NRC meetings. The attached Tier 1 entries reflect a) changes to the design descriptions and ITAAC tables as a result of the punch list items identified during the meetings; b) restructuring of the system entries so as to give (to the extent practical) consistent system-to-system treatment of technical issues; c) changes that maximize use of standardized entries for issues that appear repetitively in multiple systems. Also enclosed are written dispositions of the punch list items from the January and March meetings.

The enclosed material also includes preliminary versions of the Instrument Setpoint Methodology and Equipment Qualification entries in Section 3.4 I&C. The complete revised version of Section 3.4 will be included with the next Tier 1 submittal which is currently scheduled for 5-21-93. Tom Boyce has details of the proposed schedules for phased submittal of the ABWR Tier 1 material.

GE believes the enclosed material reflects the agreements that were developed during the January and March meetings. The system entries have undergone fairly extensive restructuring because of the reasons noted above, but GE does not believe that there are any substantive technical changes.

The staff is requested to review the enclosed material. If necessary, GE personnel will be happy to meet with staff members to explain the changes that have been incorporated -- and to answer any questions the staff may have.

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

cc: Tom Boyce (NRC)
Norman Fletcher (DOE)
Tony James (GE)
Roy Louison (GE)
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Joe Quirk (GE)

1.1 Definitions

The following definitions apply to terms used in the Design Descriptions and associated ITAAC:

Acceptance Criteria means the performance, physical condition, or analysis result for a structure, system, or component that demonstrates the Design Commitment is met.

Analysis means a calculation, mathematical computation, or engineering or technical evaluation. Engineering or technical evaluations could include, but are not limited to, comparisons with operating experience or design of similar structures, systems, or components.

As-built means the physical properties of a structure, system, or component following the completion of its installation or construction activities at its final location at the plant site.

Basic Configuration (for a Building) means the building arrangement of structural features (e.g., floors, ceilings, walls, columns, and doorways) and of structures, systems, or components which are specified in the building Design Description.

Basic Configuration (for a System) means the functional arrangement of structures, systems, and components specified in the Design Description, and verifications for that system as specified in Section 1.2.

Containment means the Primary Containment System, unless explicitly stated otherwise.

Design Commitment means that portion of the Design Description that is verified by ITAAC.

Design Description means that portion of the design that is certified.

Division (for electrical systems/equipment) is the designation applied to a given safety-related system or set of components which are physically, electrically, and functionally independent from other redundant sets of components.

Division (for mechanical systems/equipment) is the designation applied to a specific set of safety-related components within a system.

Inspect or Inspection means visual observations, physical examinations, or review of records based on visual observation or physical examination that compare the structure, system, or component condition to one or more Design Commitments. Examples include walkdowns, configuration checks, measurements of dimensions, and non-destructive examinations.

Test means the actuation or operation, or establishment of specified conditions, to evaluate the performance or integrity of as-built structures, systems, or components, unless explicitly stated otherwise.

Type Test means a test on one or more sample components of the same type and manufacturer to qualify other components of that same type and manufacturer. A type test is not a test of the as-built structures, systems, or components.

1.2 General Provisions

The following general provisions are applicable to the Design Descriptions and associated ITAAC:

Verifications for Basic Configuration for Systems

Verifications for Basic Configuration of systems include and are limited to inspection of the system functional arrangement and the following inspections, tests, and analyses:

(1) Inspections, including non-destructive examination (NDE), of the as-built, pressure boundary welds for ASME Code Class 1, 2, or 3 components identified in the Design Description to demonstrate that the requirements of ASME Code Section III for the quality of pressure boundary welds are met.

(2) Tests, or tests and analyses, of the Seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) identified in the Design Description, including associated anchorage, to demonstrate that the equipment is qualified to withstand design basis dynamic loads without loss of its safety function.

(3) Tests, or tests and analyses, of the Class 1E electrical equipment identified in the Design Description (or on accompanying figures) to demonstrate that it is qualified to withstand the environmental conditions that would exist during and following a design basis accident without loss of its safety function for the time needed to be functional. These environmental conditions, as applicable to the bounding design basis accident(s), are as follows: expected time-dependent temperature and pressure profiles, humidity, chemical effects, radiation, aging, submergence, and their synergistic effects which have a significant effect on equipment performance. As used in this paragraph, the term "Class 1E electrical equipment" constitutes the equipment itself, connected instrumentation and controls, connected electrical components (such as cabling, wiring, and terminations), and the lubricants necessary to support performance of the safety functions of the Class 1E electrical components identified in the Design Description, to the extent such equipment is not located in a mild environment during or following a design basis accident.

Electrical equipment environmental qualification shall be 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:

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

(b) Testing of a similar item of equipment under identical or similar conditions with a supporting analysis to show that the equipment is qualified; or

(c) Experience with identical or similar equipment under identical or similar conditions with supporting analysis to show that the equipment is qualified; or

(d) Analysis in combination with partial type test data that supports the analytical assumptions and conclusions to show that the equipment is qualified.

(4) Tests or type tests of active safety-related Motor-Operated Valves (MOVs) identified in the Design Description to demonstrate that the MOVs are qualified to perform their safety functions under design basis differential pressure, system pressure, fluid temperature, ambient temperature, minimum voltage, and minimum and/or maximum stroke times.

Treatment of Individual Items

The absence of any discussion or depiction of an item in the Design Description or accompanying figures shall not be construed as prohibiting a licensee from utilizing such an item, unless it would prevent an item from performing its safety functions as discussed or depicted in the Design Description or accompanying figures.

When the term "operate," "operates," or "operation" is used with respect to an item discussed in the Acceptance Criteria, it refers to the actuation and running of the item. When the term "exist," "exists," or "existence" is used with respect to an item discussed in the Acceptance Criteria, it means that the item is present and meets the Design Description.

Implementation of ITAAC

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) are provided in tables with the following three-column format:

Design Commitment	Inspections, Tests,Analyses	Acceptance Criteria

Each Design Commitment in the left-hand column of the ITAAC tables has an associated Inspections, Tests, or Analyses (ITA) requirement specified in the middle column of the tables. The identification of a separate ITA entry for each Design Commitment shall not be construed to require that separate inspections, tests, or analyses must be performed for each Design Commitment. Instead, the activities associated with more than one ITA entry may be combined, and a single inspection, test, or analysis may be sufficient to implement more than one ITA entry.

An ITA may be performed by the licensee of the plant, or by its authorized vendors, contractors, or consultants. Furthermore, an ITA may be performed by more than a single individual or group, may be implemented through discrete activities separated by time, and may be performed at any time prior to fuel load (including before issuance of the Combined Operating License for those ITAAC that do not necessarily pertain to as-installed equipment). Additionally, an ITA may be performed as part of the activities that are required to be performed under 10CFR Part 50 (including, for example, the Quality Assurance (QA) program required under Appendix B to Part 50); therefore, an ITA need not be performed as a separate or discrete activity.

Discussion of Matters Related to Operations

In some cases, the Design Descriptions in this document refer to matters that relate to operation, such as normal valve or breaker alignment during normal operation modes. Such discussions are provided solely to place the Design Description provisions in context (e.g., to explain automatic features for opening or closing valves or breakers upon off-normal conditions). Such discussions shall not be construed as requiring operators during operation to take any particular action (e.g., to maintain valves or breakers in a particular position during normal operation).

Interpretation of Figures

In many but not all cases, the Design Descriptions in Section 2 include one or more Figures. The Figures may represent a functional diagram, general structural representation, or other general illustration. For I&C systems, the Figures also represent aspects of the relevant logic of the system or part of the system. Unless specified explicitly, these Figures are not indicative of the scale, location, dimensions, shape, or spatial relationships of as-built structures, systems, and components. In particular, the as-built attributes of structures, systems, and components may vary from the attributes depicted on these Figures, provided that those safety functions discussed in the Design Description pertaining to the Figure are not adversely affected.

2.1.2 Nuclear Boiler System

Design Description

General System Description

The primary functions of the Nuclear Boiler System (NBS) are:

(1) To deliver steam from the Reactor Pressure Vessel (RPV) to the Main Steam (MS)System,

(2) To provide containment isolation of the Main Steam Lines (MSLs),

(3) To deliver feedwater from the Condensate, Feedwater, and Air Extraction (CFDWA) System to the RPV,

(4) To provide overpressure protection of the Reactor Coolant Pressure Boundary (RCPB),

(5) To provide automatic depressurization of the RPV in the event of a loss-of-coolant accident (LOCA) where the RPV does not depressurize rapidly and the high pressure makeup systems fail to adequately maintain the water level in the RPV, and

(6) To provide instrumentation to monitor the drywell pressure and RPV pressure, metal temperature, and water level.

Figures 2.1.2a, 2.1.2b, 2.1.2c, 2.1.2d, and 2.1.2e show the basic system configuration ands scope. Figure 2.1.2f shows the NBS control interfaces.

The NBS equipment shown on Figures 2.1.2a, 2.1.2b, 2.1.2c, 2.1.2d, and 2.1.2e is classified as safety-related except for the non-nuclear-safety (NNS) part of the MSL drains, equipment associated with the power actuated relief mode of the SRVs, the SRV discharge pipe temperature sensors, and the non-safety-related instruments shown on Figure 2.1.2e.

Main Steam Lines

The MSLs direct steam from the RPV to the MS System. The NBS contains only the portion of the MSLs from their connection to the RPV to the boundary with the MS System, which occurs at the seismic interface located downstream of the outboard main steam isolation valves (MSIVs). Figures 2.1.2a and 2.1.2b show the general configuration of the MSLs and the MSL drain lines. The MSL drain lines provide a flow path for the MSIV leakage during an accident.

The combined volume of the steam lines, from the RPV to the main steam turbine stop valves and turbine bypass valves, is greater than or equal to 113.2 m^3.

Each MSL has a flow limiter. The MSL flow limiter consists of a flow restricting venturi which is located in each RPV MSL outlet nozzle. The restrictor limits the coolant blowdown rate from the RPV in the event a MSL break occurs outside the containment to a flow rate equal to or less than 200% of rated steam flow at 72.1 kg/cm^2 g upstream pressure. The throat diameter of the MSL flow limiters is not greater than 355 mm.

The pneumatic-operated valve in the MSL drain line shown in Figure 2.1.2b opens, if either electric power to the valves actuating solenoid is lost, or pneumatic pressure to the valve is lost.

The MSLs and the MSL drain lines are located in the drywell and the steam tunnel.

Main Steam Isolation Valves

Two isolation valves are located in a horizontal run of each of the four main steam lines; one valve is inside of the drywell, and the other is near the outside of the primary containment pressure boundary.

The MSIV closing time is equal to or greater than 3 seconds and less than or equal to 4.5 seconds when N_2 or air is admitted to the MSIV actuator. When all the MSIVs are closed, the combined leakage through the MSIVs for all four MSLs is less than or equal to 66.1 liter per minute at 20°C and one atmosphere absolute pressure.

The MSIV's primary actuation mechanism for opening and closing is pneumatic. Springs close the MSIV if pneumatic pressure to the MSIV actuator is lost.

Feedwater Lines

The Feedwater (FW) lines direct Feedwater from the CFDWA System to the RPV. The NBS contains only the portion of the FW lines from the seismic interface located upstream of the Motor-Operated Valves (MOVs) to their connections to the RPV. Figure 2.1.2c shows the portion of the FW lines within the NBS.

Isolation of each feedwater line is accomplished by two containment isolation valves consisting of one check valve inside the drywell and one positive closing check valve outside the containment. The FW line isolation check valves are qualified to withstand a FW line break outside containment. The feedwater line upstream of the outboard isolation valve contains a MOV, and a seismic interface restraint.

Safety/Relief Valves

The Safety/Relief Valves (SRVs) are located on the MSLs between the RPV and the inboard MSIV. These valves protect against overpressurization of the RCPB. Figures 2.1.2a, 2.1.2b and 2.1.2d show the general configuration of the SRVs, and the SRV discharge lines.

The rated capacity of the pressure-relieving devices is sufficient to prevent a rise in pressure within the RPV of more than 110% of the design pressure (96.7 kg/ cm^2 gauge) for design basis events.

The SRV discharge lines are sized so that critical flow conditions occurs through the valve. Each SRV has its own discharge line. The SRV discharge lines terminate at quenchers located below the surface of suppression pool.

The SRVs provide three main protection functions:

(1) Overpressure safety operation: The valves function as spring loaded safety valves and open to prevent RCPB overpressurization. The valves are self-actuated by inlet steam pressure.

The following table identifies the SRV spring set pressures and flow capacities. The opening time for the SRVs, from the time the pressure exceeds the valve set pressure to the time the valve is fully open, is less than or equal to 0.3 seconds.

Set Pressures and Capacities

SRVs	Number* of Valves	Nameplate Spring Set Pressure (kg/cm^2g)†	ASME Rated Capacity at 103% Spring Set Pressure (kg/hr each)‡	Used For ADS
J, P	2	80.8	395,000	
B, G, M, S	4	81.5	399,000	
D, E, K, U	4	82.2	402,000	
C, H, N, T	4	82.9	406,000	X
A, F, L, R	4	83.6	409,000	X

* Eight of the SRVs serve in the automatic depressurization system function.
† Spring set pressure tolerances as permitted by the ASME Boiler and Pressure Vessel Code, Section III.
‡ Minimum capacity per the ASME Boiler and Pressure Vessel, Section III.

(2) Overpressure relief operation: The valves are opened using a pneumatic actuator upon receipt of an automatic or manually initiated signal.

For overpressure relief valve operation (power-actuated mode), reactor vessel pressure sensors generate a high pressure trip signal which is used to initiate opening the SRVs. Valve opening is initiated when an

electrical signal is received at the solenoid valve associated with power actuated relief (Figure 2.1.2d).

The SRV pneumatic operator is so arranged that, if it malfunctions, it does not prevent the SRV from opening when steam inlet pressure reaches the spring lift setpoint.

(3) Automatic Depressurization System (ADS) operation: The ADS valves open automatically or manually in the power actuated mode during a loss-of-coolant accident (LOCA). Eight of the eighteen SRVs are designated as ADS valves and are capable of operating from either ADS LOCA logic or overpressure relief logic signals. The above table identifies the ADS SRVs.

The ADS accumulator capacity can open the SRV with the drywell pressure at design pressure following failure of the pneumatic supply to the accumulator.

The SRVs can be operated individually in the power-actuated mode by remote manual switches located in the main control room. They are provided with position sensors which provide positive indication of SRV disk/stem position.

Automatic Depressurization System

As shown in Figure 2.1.2f, the NBS system channel measurements are provided for the SSLC for signal processing, setpoint comparisons, and generating trip signals. Except for the pump running permissive, the SSLC uses a two-out-of-four voting logic for ADS initiation. The ADS logic is automatically initiated when a low reactor water level signal is present. If the RPV low water level signal is present concurrently with high drywell pressure signal, both the main ADS timer (less than or equal to 29 seconds) and the high drywell pressure bypass timer (less than or equal to 8 minutes) are initiated. Absent a concurrent high drywell pressure signal, only the ADS high drywell pressure bypass timer is initiated. Upon the time out of the ADS high drywell pressure bypass timer, concurrent with RPV low water level signal, the main ADS timer is initiated, if not already initiated. The main timer continues to completion and times out only in the continued presence of an RPV low water level signal. Upon time out of the main ADS timer, concurrent with positive indication by pump discharge pressure of at least one RHR or one HPCF pump running, the ADS function is initiated.

Signals from all four divisions for low reactor water level and high drywell pressure and Division I control logic signal actuate one set of pilots, and sensors from all four divisions for low reactor water and high drywell pressure and Division II control logic signal actuate the second set of pilots, either of which initiates the opening of the ADS SRVs.

ADS initiation is accomplished by redundant trip channels arranged in two divisionally separated logics that control two separate solenoid-operated pneumatic pilots on each ADS SRV. Either pilot can operate the ADS valve. These pilots control the pneumatic pressure applied by the accumulators and the High Pressure Nitrogen Gas Supply (HPIN) System. The DC power for the logic is obtained from the Safety System Logic and Control (SSLC) Division I and II.

For anticipated transient without scram (ATWS) mitigation, the ADS has an automatic and manual inhibit of the automatic ADS initiation. Automatic initiation of ADS is inhibited unless there is a coincident low reactor water level signal and an average power range monitors (APRMs) downscale signal. There are main control room switches for the manual inhibit of automatic initiation of ADS.

The ADS can also be initiated manually. On a manual initiation signal, concurrent with positive indication of at least one RHR or one HPCF pump is running, the ADS function is initiated.

NBS Instrumentation

The NBS contains the instrument lines and instrumentation for monitoring the reactor pressure and water level. For drywell pressure, turbine inlet pressure, main condenser vacuum, and RPV metal temperature, the NBS contains the sensors. Figure 2.1.2e shows the drywell pressure and RPV instrumentation in the NBS.

The mechanical portion of each division of the safety-related NBS instrumentation located in the Reactor Building is physically separated from the other divisions.

The reactor vessel outside surface (metal) temperatures are measured at the head flange and the bottom head locations.

Figure 2.1.2e shows the water level instrumentation. The instruments that sense the water level are differential pressure devices calibrated for specific RPV pressure and temperature conditions. Instrument zero for the RPV water level ranges is the top of the active fuel.

With the exception of turbine inlet pressure sensor and main condenser vacuum sensor located in the Turbine Building, the NBS instrumentation is located in the steam tunnel and the Reactor Building.

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

The NBS equipment identified as safety-related is classified as Seismic Category I except for the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 3 equipment shown on Figure 2.1.2c. The MSL drain lines from the MSLs to the Main Condenser are seismically analyzed to withstand the Safe Shutdown Earthquake (SSE).

The MSL drain lines from the MSLs to the main condenser are seismically analyzed to withstand the Safe Shutdown Earthquake (SSE).

Figures 2.1.2a, 2.1.2b, and 2.1.2c show the ASME Boiler and Pressure Vessel Code Classes.

The divisional equipment in the NBS is powered from its respective Class 1E divisions as shown in Figures 2.1.2b, 2.1.2d, and 2.1.2e. In the NBS, independence is provided between Class 1E Divisions, and also between Class 1E Divisions and Non-Class 1E Equipment.

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

(1) Parameter displays for the instruments shown on Figures 2.1.2b and 2.1.2e. This includes the reactor vessel pressure, reactor vessel water level, drywell pressure, main condenser vacuum, and turbine inlet pressure.

(2) Controls and status indication for the active safety-related components shown on Figures 2.1.2b, 2.1.2c (excluding the inboard FW line check valves, and the ASME Boiler and Pressure Vessel Code Class 2 check valves), and 2.1.2d.

(3) Manual system level initiation capability for the ADS.

(4) Manual capability to inhibit automatic initiation of ADS.

NBS components with displays and control interfaces with the Remote Shutdown System (RSS) are shown on Figures 2.1.2a and 2.1.2e.

The safety-related electrical equipment (including instrumentation and controls) shown on Figures 2.1.2b, 2.1.2c, 2.1.2d, and 2.1.2e, located in the containment, steam tunnel and Reactor Building, is qualified for a harsh environment.

The MOVs shown on Figure 2.1.2b (except for the ASME Boiler and Pressure Vessel Code Class 2 MOV) have safety-related functions and close under differential pressure, fluid flow, and temperature conditions.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.1.2 provides a definition of the inspections, tests and/or analyses together with associated acceptance criteria which will be undertaken for the NBS.

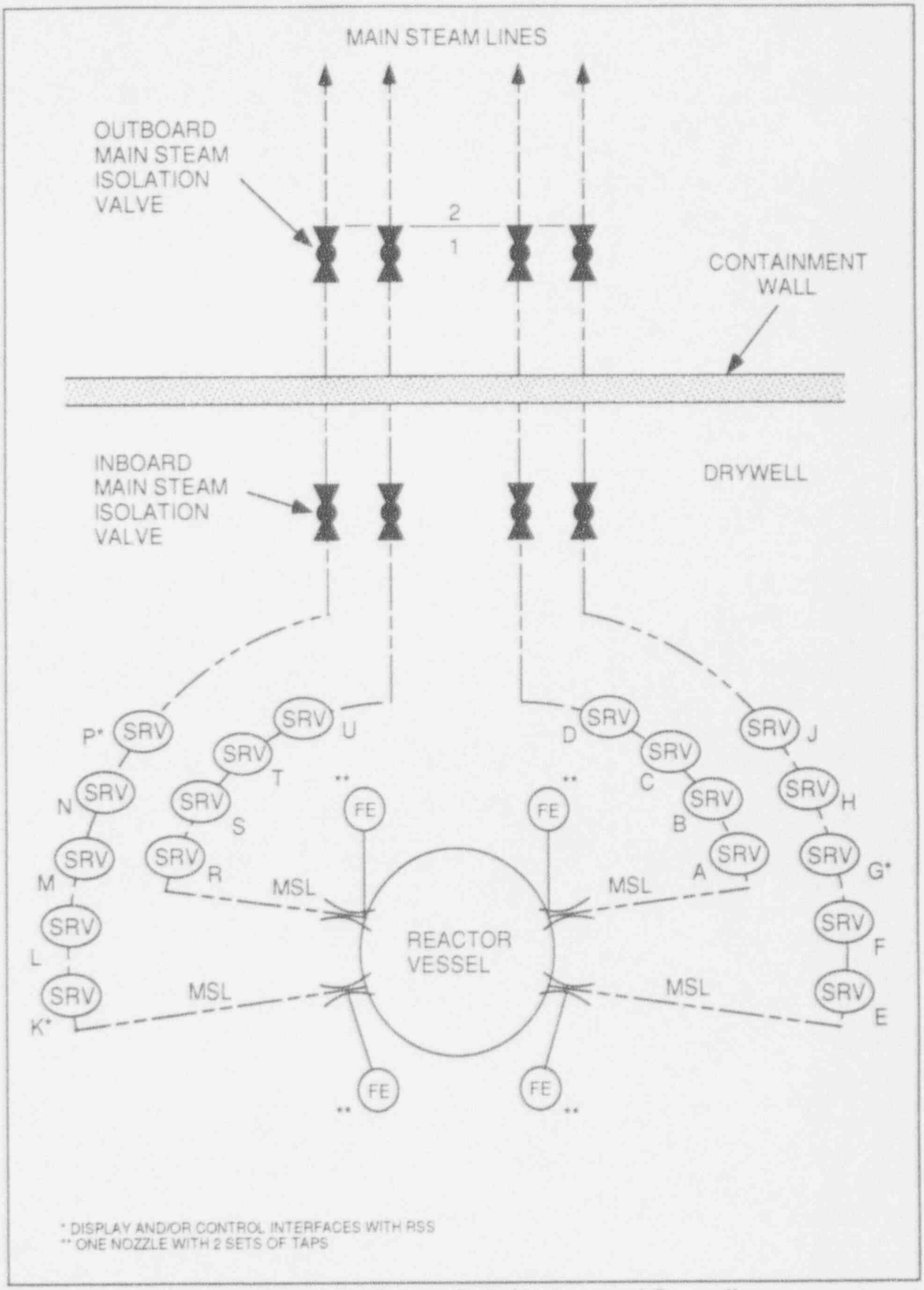
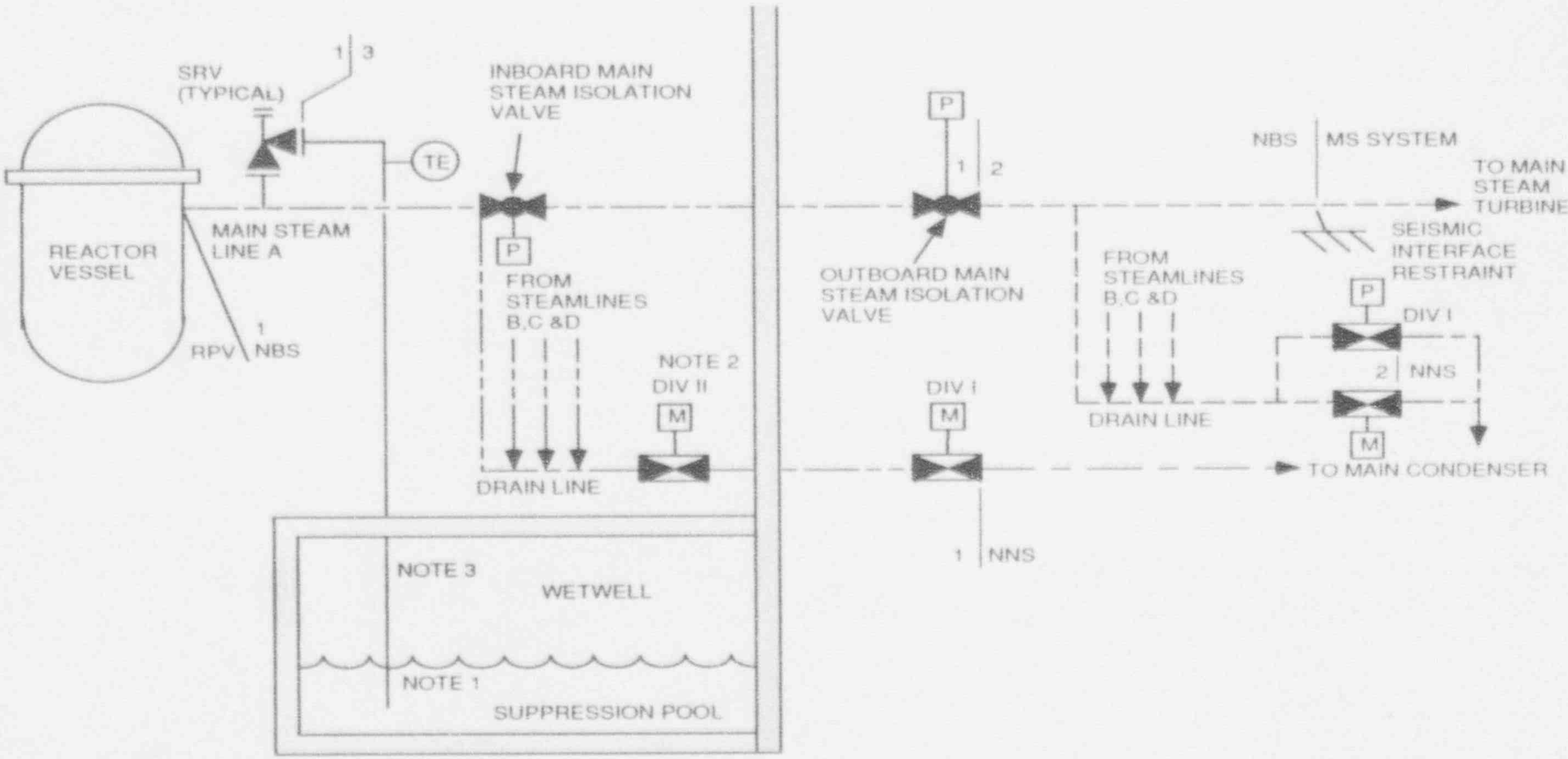


Figure 2.1.2a NBS Safety/Relief Valves and Steamline



NOTES:
1. AT MINIMUM LEVEL
2. MAY BE PNEUMATIC
3. THE PIPING PRESSURE WELDS IN THE WETWEELL AIRSPACE SHALL BE EXAMINED USING ASME CODE CLASS 2 REQUIREMENTS.
4. EACH MSIV HAS CLASS 1E POSITION SWITCHES WHICH RECEIVE POWER FROM ITS RESPECTIVE CLASS 1E DIVISION.
5. EACH SRV HAS POSITION SENSORS.

Figure 2.1.2b NBS Steamline

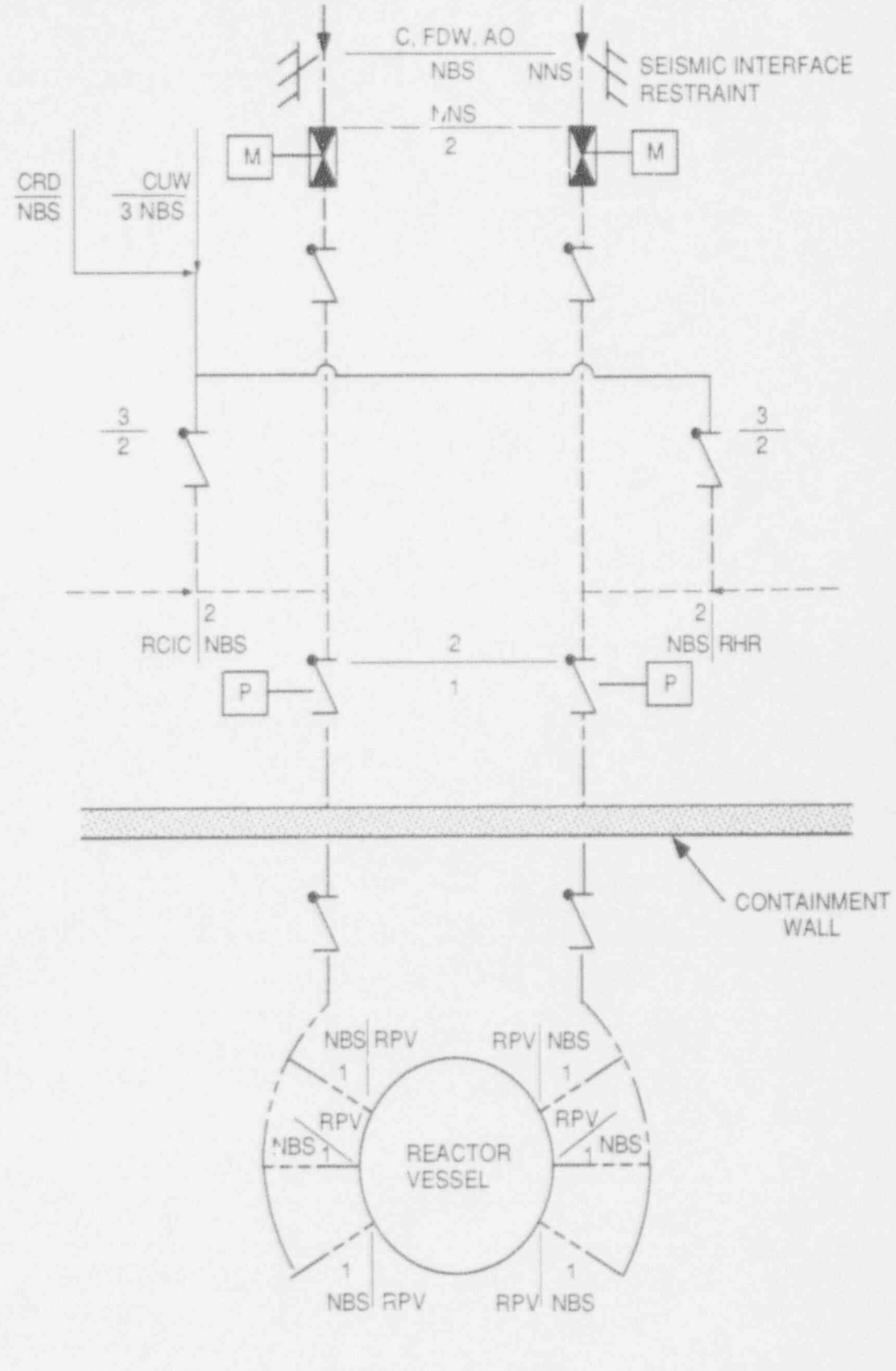


Figure 2.1.2c NBS Feedwater Line

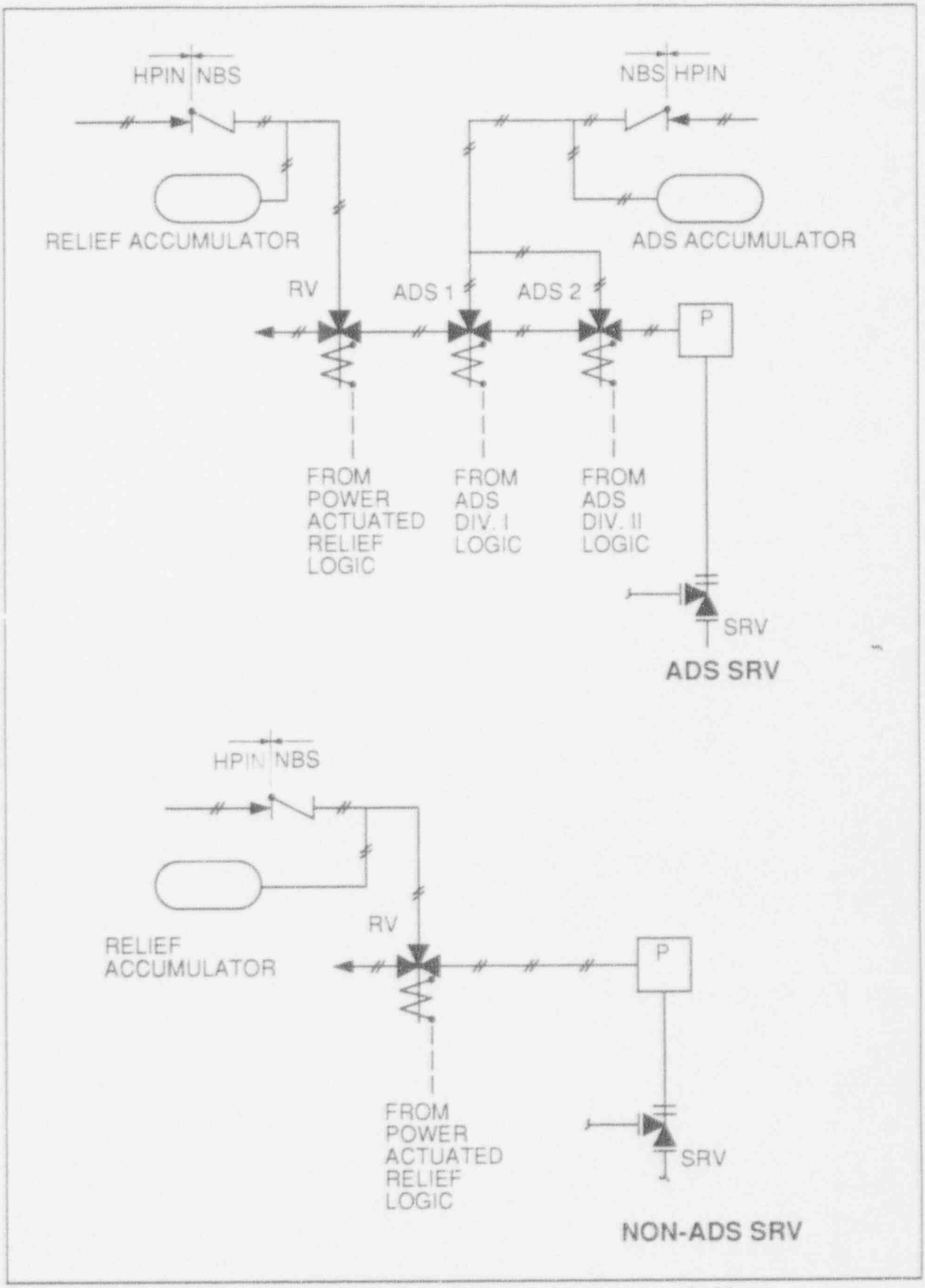
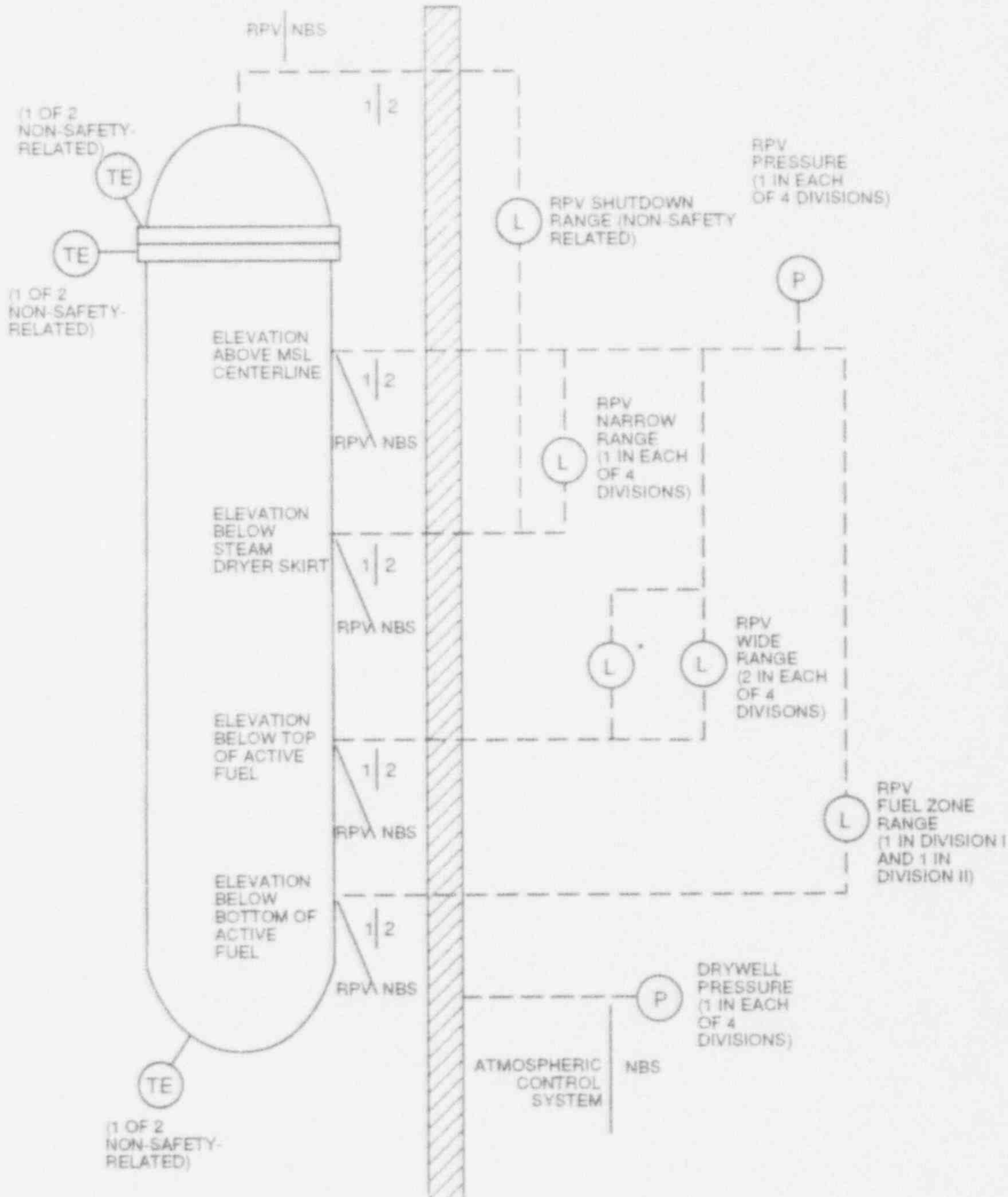


Figure 2.1.2d NBS Safety/Relief Valve Pneumatic Lines



* DISPLAY AND/OR CONTROL INTERFACES WITH RSS

NOTES:
1. EACH OF FOUR DIVISIONS OF NBS INSTRUMENTATION HAS ONE TURBINE INLET PRESSURE SENSOR AND ONE MAIN CONDENSER VACUUM SENSOR.
2. EACH DIVISION IS POWERED BY ITS RESPECTIVE CLASS 1E DIVISION.
3. THE MAIN CONDENSER VACUUM SENSORS ARE LOCATED ABOVE THE TURBINE BUILDING DESIGN BASIS FLOOD LEVEL.

Figure 2.1.2e NBS Drywell Pressure and Reactor Vessel Instrumentation

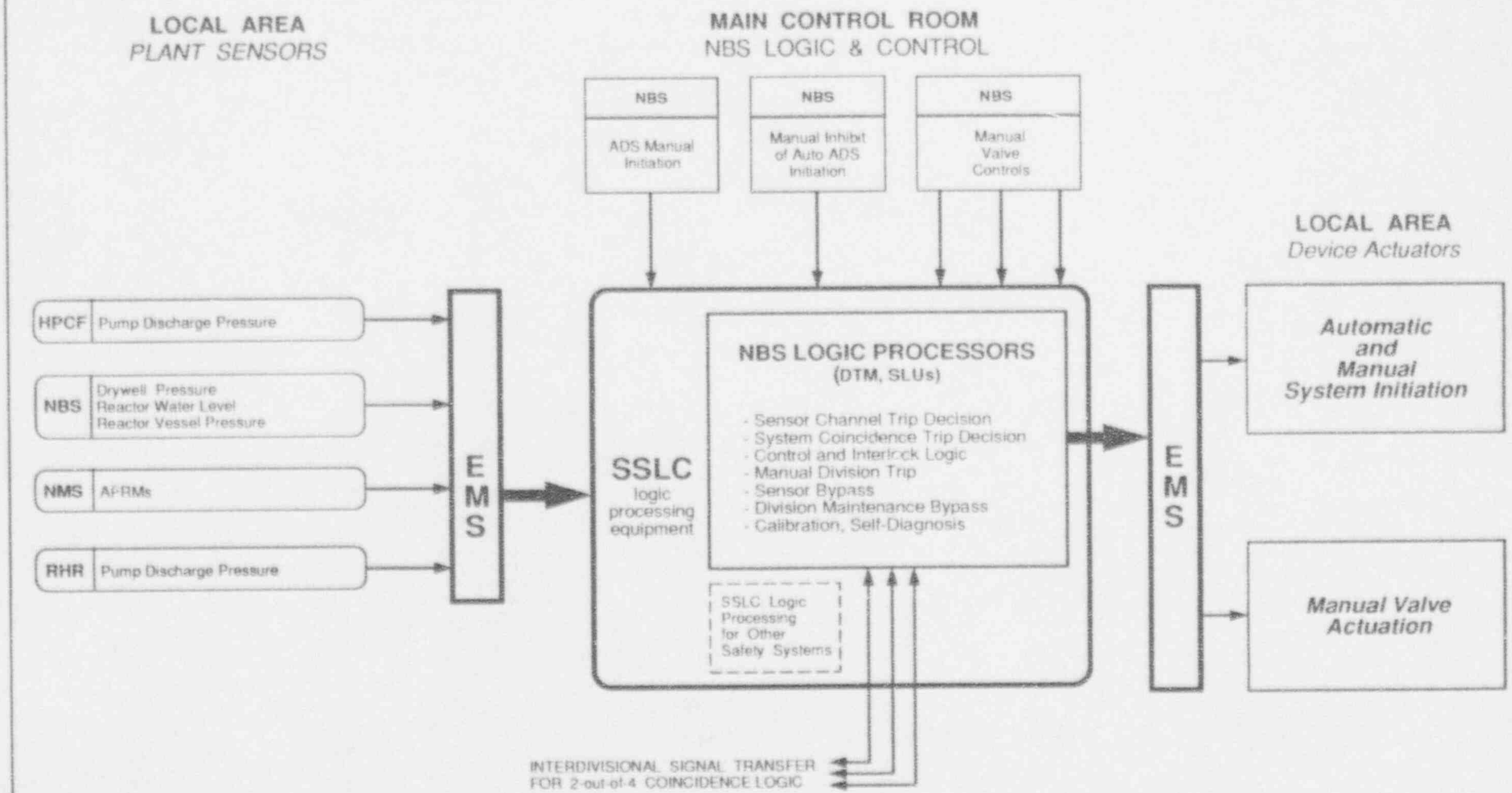


Figure 2.1.2f Nuclear Boiler System

Table 2.1.2 Nuclear Boiler System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the NBS is shown in Figures 2.1.2a, b, c, d, e, and f.	1. Inspections will be conducted for the NBS System.	1. The as-built NBS conforms with the basic configuration shown in Figures 2.1.2a, b, c, d, e, and f.
2. The ASME Code components of the NBS System retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A hydrostatic test of the ASME Code components of the NBS System will be conducted.	2. The results of the hydrostatic test of the ASME Code components of the NBS System conform with the requirements in the ASME Code, Section III.
3. The combined volume of the four Main Steam Lines (MSLs) and branch lines from the RPV to the main steam turbine stop valves and steam bypass valves is greater than or equal to 113.2 m^3.	3. Analyses will be performed using as-built dimensions of the steamlines to determine the combined steamline volume.	3. The combined steamline volume is greater than or equal to 113.2 m^3.
4. The throat diameter for the MSL flow limiter is less than or equal to 355 mm.	4. Inspections of the as-built MSL flow limiters will be conducted.	4. The throat diameter of the MSL flow limiters is less than or equal to 355 mm.
5. The pneumatic operated valve in the MSL drain line shown in Figure 2.1.2b opens if either electric power to the valve actuating solenoid is lost, or pneumatic pressure to the valve is lost.	5. Tests will be conducted on the as-built MSL drain valve.	5. The MSL pneumatic drain line valve shown in Figure 2.1.2b opens when either electric power to the valve actuating solenoid is lost, or pneumatic pressure to the valve is lost.
6. MSIV closing time is equal to or greater than 3 seconds and less than or equal to 4.5 seconds when N_2 or air is admitted into the MSIV actuator.	6. Tests of the as-built MSIV will be conducted to determine the closure time.	6. The MSIV closing time is equal to or greater than 3 and less than or equal to 4.5 seconds.
7. When all MSIVs are closed, the combined leakage through the MSIVs for all four MSLs is less than or equal to 66.1 liters per minute at 20° C and 1 atmosphere absolute pressure.	7. Tests will be conducted on the as-built MSIVs to determine the leakage.	7. MSIV leakage for all four MSLs is less than or equal to 66.1 liters per minute at 20° C and 1 atmosphere absolute pressure.
8. Springs close the MSIV if pneumatic pressure to the MSIV actuator is lost.	8. Tests will be conducted on the as-built MSIV.	8. The MSIV closes when pneumatic pressure is removed from the MSIV actuator.

Table 2.1.2 Nuclear Boiler System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. The SRV spring set pressure and flow capacities are given in Section 2.1.2. The opening time for the SRVs from the time flow the pressure exceeds the valve set pressure to the time the valve is fully open, is less than or equal to 0.3 seconds.	9. Analysis and tests (at a test facility) will be conducted in accordance with the ASME Boiler and Pressure Vessel Code.	9. The SRVs have the capacities and set pressures shown on Section 2.1.2. The opening time for the SRVs from the time the pressure exceeds the valve set pressure to the time the valve is fully open is less than or equal to 0.3 seconds.
10. The ADS accumulator can open the SRV with the drywell pressure at design pressure following failure of the pneumatic supply to the accumulator.	10. An analysis and/or type test will be performed to demonstrate the capacity of the SRV ADS accumulators.	10. Either: a. The SRV ADS accumulators have the capacity to lift the stem of the SRVs to the full open position one time with the drywell pressure at, or above the drywell design pressure, or b. the SRV ADS accumulators have the capacity to lift the stem of the SRVs to the full open position five time with the drywell at atmospheric pressure, and an analysis that shows that 5 SRV lifts at atmospheric pressure demonstrates the capability to open one time with the drywell at the drywell design pressure.
11. For overpressure relief valve operation, reactor vessel pressure sensors generate a high pressure trip signal which is used to initiate opening of the SRVs.	11. Tests will be conducted on the power actuated relief logic using simulated input signal to cause trip conditions.	11. The valve solenoid receives an initiation signal.

Table 2.1.2 Nuclear Boiler System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. The ADS logic is automaticall6y initiated when a low reactor water level signal is present.	12. Tests will be conducted using simulated input signals for each NBS process variable to cause trip conditions in two, three, and four instrument channels of the same process variable associated with each of the two ADS logic divisions.	12a. Upon receipt of a low water level signal, concurrent with a high drywell pressure signal, at the input to the ADS initiation logic, the following occurs: 1) The main ADS timer initiates and continues to time out in the continued presence of the RPV low water level signal. The time delay for the main ADS timer is less than or equal to 29 seconds. 2) Upon time out of the main ADS timer, a concurrent signal that represents positive indication of at least one RHR or HPCF pump running, an ADS actuation signal is generated to the associated ADS valve solenoids.

Table 2.1.2 Nuclear Boiler System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. (Continued)	12. (Continued)	12b. Upon receipt of a low water level signal, in the absence of a high drywell pressure signal, at the input to the ADS initiation logic, the following occurs. 1) The ADS high drywell pressure bypass timer initiates. The time delay for the ADS high drywell pressure bypass timer is less than or equal to 8 minutes. 2) Upon time out of the ADS high drywell pressure bypass timer, concurrent with an RPV low water level signal, the main ADS timer initiates and continues to time out in the continued presence of the RPV low water level signal. 3) Upon time out of the main ADS timer, concurrent with a pump discharge pressure signal that represents positive indication of at least one RHR or HPCF pump running, an ADS actuation signal is generated to the associated ADS valve solenoids.
13. For ATWS mitigation, the ADS has an automatic and manual inhibit of the automatic ADS initiation.	13a. The tests defined in item 12a. will be conducted with a simulated APRM not-downscale signal.	13a. ADS actuation does not occur.
	13b. The test defined in 12a will be conducted with the ADS manual inhibit device set to inhibit.	13b. ADS actuation does not occur
14. The ADS can be initiated manually.	14. Tests will be conducted by initiating each ADS division manually, concurrent with a simulated RHR or HPCF pump running signal.	14. Upon receipt of a manual initiation signal, an ADS actuation signal is generated to the associated ADS valve solenoids.

Table 2.1.2 Nuclear Boiler System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
15. The mechanical portion of each division of the safety-related NBS instrumentation located in the Reactor Building is physically separated from the other divisions.	15. Inspections of the as-built NBS instrumentation will be conducted	15. The mechanical portion of each NBS instrumentation division is physically separated from the other divisions by structural and/or fire barriers
16. The MSL drain lines from the MSLs to the main condenser are seismically analyzed to withstand the SSE.	16. An inspection of the stress report containing the dynamic analysis of the piping will be conducted.	16. A stress report exists. This report documents that a dynamic seismic analysis has been performed.
17. In the NBS, independence is provided between Class 1E Divisions, and between Class 1E Divisions and non-Class 1E equipment.	17a. Tests will be performed in the NBS by providing a test signal in only one Class 1E Division at a time.	17a. The test signal exists only in the Class 1E Division under test in the NBS.
	17b. Inspection of the as-installed Class 1E Divisions in the NBS will be performed.	17b. Physical separation exists between Class 1E Divisions in the NBS. Physical separation exists between Class 1E Divisions and non-Class 1E equipment.
18. Main control room displays and controls provided for NBS are as defined in Section 2.1.2.	18. Inspections will be performed on the main control room displays and controls for the NBS.	18. Displays and controls exist or can be retrieved in the main control room as defined in Section 2.1.2.
19. RSS displays and controls provided for the NBS are defined in Section 2.1.2.	19. Inspections will be performed on the RSS displays and controls for the NBS.	19. Displays and controls exist on the RSS as defined in Section 2.1.2.
20. MOVs designated in Section 2.1.2 as having an active safety function will close under differential pressure, fluid flow, and temperature conditions.	20. Closing tests of installed valves will be conducted under preoperational differential pressure, fluid flow, and temperature conditions.	20. Each MOV closes on receipt of an actuation signal.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.1.2 Nuclear Boiler System

PUNCH LIST ITEM: A. Need to reference Figure 2.1.2D and include it.

GE DISPOSITION: This figure is now included in the revised Section 2.1.2 and is referenced from the text.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.1.2 Nuclear Boiler System

PUNCH LIST ITEM: B. Insert boilerplate and/or ITAAC to address BWR pressure vessel level problem.

GE DISPOSITION: Potential design changes to BWR pressure vessel level instrumentation is an active, ongoing program within the industry at this time. The Tier 1 material currently does not include any design changes aimed at the recently identified issues. Furthermore, the SSAR description and the associated Tier 1 entries may not be compatible with the eventual design changes selected by GE and/or the BWR Owner's Group working on this problem.

GE believes this issue will be resolved in the next one to two months and proposes to make SAR modifications and Tier 1 changes to the final ABWR certification material to be submitted July 30, 1993.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.1.2 Nuclear Boiler System

PUNCH LIST ITEM: C. Add standard boilerplate as shown by John Chambers (capture specific times for timers).

GE DISPOSITION: A new section has been added to the design description (with corresponding ITAAC entries) to specifically define the ADS logic and necessary ITAAC. GE believes this revised material will resolve this punch list item.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.1.2 Nuclear Boiler System

PUNCH LIST ITEM: D. Analysis algorithm to be provided in Tier 2.

GE DISPOSITION: This material will be added to the SSAR.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.1.2 Nuclear Boiler System

PUNCH LIST ITEM: E. Add boilerplate

GE DISPOSITION: Complete. The standardized entry for electrical independence developed during the January 1993 GE/NRC meetings has been added to Section 2.1.2.

SYSTEM: 2.1.2 Nuclear Boiler System

PUNCH LIST ITEM: F. Standard electrical/I&C separation ITAAC

GE DISPOSITION: Covered by response to punch list Item E.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
<u>FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS</u>

SYSTEM: 2.1.2 Nuclear Boiler System

PUNCH LIST ITEM: G. Check main condenser design description for inlet pressure and condenser vacuum

GE DISPOSITION: GE has concluded that the NBS (Section 2.1.2) should cover the instruments provided for main condenser vacuum and turbine inlet pressure. Consequently, they have been added to this section and are not addressed in the main condenser section.

2.2.3 Feedwater Control System

Design Description

The Feedwater Control (FDWC) System controls the flow of feedwater into the reactor pressure vessel (RPV) to maintain the water level in the vessel during plant operation. The FDWC System consists of redundant, microprocessor based controllers, and flow sensors for main steamlines and feedwater lines, as shown in the control interface diagram in Figure 2.2.3.

The FDWC digital controllers determine narrow range level signal using three reactor level measurement inputs from the NBS. Sensor signals are transmitted to the FDWC digital controllers by the Non-Essential Multiplexing System (NEMS).

The steam flow in each of four main steamlines is sensed at the RPV nozzle venturis. Sensor signals are transmitted to the FDWC digital controllers by the NEMS. These measurements are processed in the digital controllers to give the total steam flow out of the vessel.

Feedwater flow is sensed at a flow element in each of the two feedwater lines. Sensor signals are transmitted to the FDWC digital controllers by the NEMS. These measurements are processed in the digital controllers to give the total feedwater flow into the vessel.

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

The FDWC System operates in either manual, automatic single-element or automatic three-element control modes. At low feedwater flow, the FDWC System utilizes only water level measurement in automatic single-element control mode. At higher flow rates, the FDWC System in three-element control mode uses water level, steam flow, and feedwater flow measurements for water level control.

The FDWC System monitors reactor water level signals and, if a high RPV water level setpoint is reached, sends trip signals to the Turbine Control System and to the Condensate, Feedwater and Condensate Air Extraction (CF&CAE) System. If a low RPV water level setpoint is reached, the FDWC System ends trip signals to the Recirculation Flow Control (RFC) System.

If the FDWC System receives an Anticipated Transient Without Scram (ATWS) trip signal from the Safety System and Logic Control (SSLC), FDWC issues signals to runback feedwater flow.

The FDWC System is powered by separate non-Class 1E uninterruptible power supplies.

The total feedwater flow is displayed on the main control panel. The FDWC System operating mode is selectable from the main control room. The FDWC System microprocessors are located in the Control Building.

Digital controllers used for the FDWC System are redundant, with diagnostic capabilities that identify and isolate failure of level input signals.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.3 provides definition of the inspections, tests, and/or analysis, together with associated acceptance criteria which will be undertaken for the Feedwater Control System.

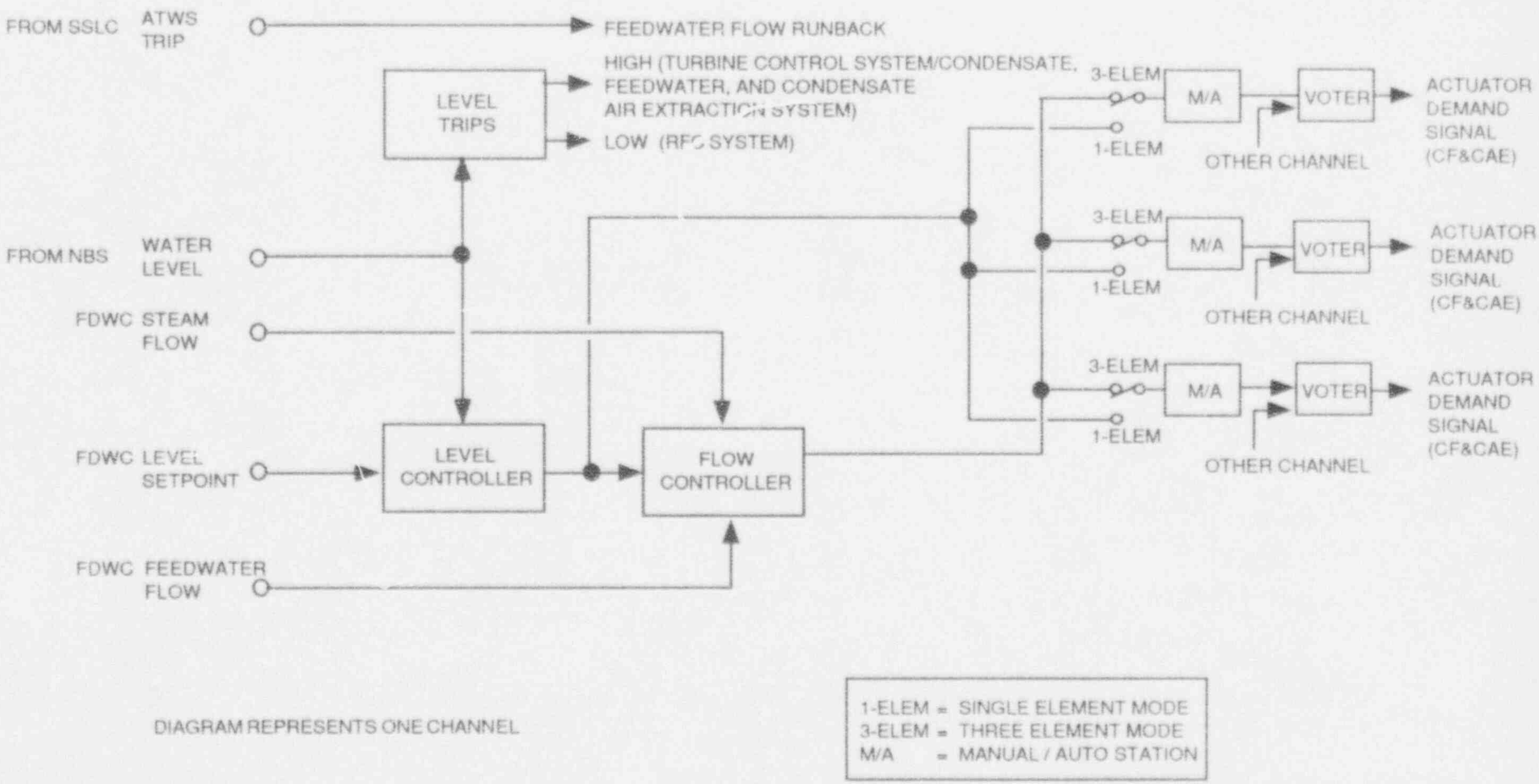


Figure 2.2.3 FDWC Control Interface Diagram

Table 2.2.3 Feedwater Control System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The equipment comprising the FDWC System is defined in Section 2.2.3.	1. Inspections of the as-built system will be conducted.	1. The as-built FDWC System conforms with the description in Section 2.2.3.
2. The FDWC System controls the flow of feedwater into the RPV.	2. A test will be performed by simulating a decreasing reactor level signal.	2. A signal to increase feedwater flow occurs.
3. The FDWC System monitors reactor water level signals and, if a high RPV water level setpoint is reached, sends trip signals to the Turbine Control System and to the CF&CAE System. If a low RPV water level setpoint is reached, the FDWC System sends trip signals to the RFC System.	3. Tests will be performed on the FDWC System, using simulated RPV water level signals.	3. When a high RPV water level setpoint is reached, trip signals are sent to Turbine Control System and CF&CAE System. When a low RPV water level setpoint is reached, a trip signal is sent to the RFC System.
4. If the FDWC System receives an ATWS trip signal from SSLC, FDWC issues signals to runback feedwater flow.	4. Tests will be performed on the FDWC System, using a simulated ATWS trip signal.	4. When an ATWS trip signal is received, FDWC issues feedwater runback signals.
5. The FDWC System digital controllers are powered by separate non-Class 1E uninterruptible power supplies.	5. Tests will be performed by providing a test signal in only one non-Class 1E uninterruptible power supply at a time.	5. The test signal exists in only one digital control channel at a time.
6. Main control room controls and displays provided for the FDWC System are as defined in Section 2.2.3.	6. Inspections will be performed on the main control room controls and displays for the FDWC System.	6. Controls and displays exist or can be retrieved in the main control room as defined in Section 2.2.3.
7. Digital controllers used for the FDWC System are redundant.	7. Tests will be performed by simulating failure of each operating FDWC System digital controller.	7. There is no loss of FDWC System output upon loss of any one digital controller.
8. Digital controllers used for the FDWC System have diagnostic capabilities that identify and isolate failure of level input signals.	8. Tests will be performed by simulating level input signal failures to the FDWC System digital controllers.	8. There is no loss of FDWC System output upon loss of any one level input signal.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.2.3 Feedwater Control System

PUNCH LIST ITEM: 1. Add NNS level measurements to NBS figures/design description.

GE DISPOSITION: The non-safety-related instruments used to provide input to the feedwater control system have been added to the Nuclear Boiler System's instrumentation figure.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.2.3 Feedwater Control System

PUNCH LIST ITEM: 2. Add L3 ATWS input text and logic to Section 7.7.1.4.

GE DISPOSITION: The SAR has been modified to reflect this ATWS interface with the feedwater control system.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.2.3 Feedwater Control System

PUNCH LIST ITEM: 3. Revise SAR to show current FW pump speed controller in system description.

GE DISPOSITION: The SAR has been revised to include this material.

SYSTEM: 2.2.3 Feedwater Control System

PUNCH LIST ITEM: 4. Identify that FDWC power supply is non-safety-related in the SAR.

GE DISPOSITION: The SAR has been modified to reflect this power supply configuration.

2.2.4 Standby Liquid Control System

The Standby Liquid Control (SLC) System injects neutron absorbing poison into the reactor using a boron solution, thus providing backup reactor shutdown capability independent of the normal reactivity control system based on insertion of control rods into the core. The SLC System is designed to bring the reactor from full power to a subcritical condition without control rod movement, at any time in a core cycle, and at design basis conditions with the reactor in the most reactive xenon-free state. The SLC System operates over a range of reactor pressure conditions which bound the elevated pressures associated with an anticipated transient without scram (ATWS). Figure 2.2.4 shows the basic system configuration and scope.

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two motor-operated injection valves which are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged through the high pressure core flooder (HPCF) Division B subsystem sparger.

The SLC System uses a dissolved solution of sodium pentaborate as the neutron-absorbing poison. This solution is held in the storage tank which has a heater to maintain solution temperature above the saturation temperature. The heater has automatic actuation and automatic shutoff.

A test tank and associated piping and valves permit testing of the SLC System during plant operation. The tank is supplied with demineralized water which is pumped in either a closed loop or is injected into the reactor.

Key SLC System equipment performance requirements are:

(1)	Pump flow (minimum)	378 l/min with both pumps operating 189 l/min with one pump operating
(2)	Maximum reactor pressure (for injection)	88.9 kg/cm^{2}a
(3)	Pumpable volume in storage tank (minimum)	23.1 m^3

The SLC System can be manually initiated from the main control room. Each of the two divisions is controlled by a separate switch. When it is manually initiated to inject a liquid neutron absorber into the reactor, the following devices and actions are initiated by each divisional switch:

(1) The specified division injection valve is opened.

(2) The specified division storage tank discharge valve is opened.

(3) The specified division injection pump is started.

(4) The reactor water cleanup isolation valves are closed.

Both divisions of the SLC System are automatically initiated during an ATWS condition. With the storage tank at minimum level and both pumps operating, the system is designed to inject the minimum required boron solution.

The SLC System provides borated water to the reactor core to compensate for the various reactivity effects. These effects are xenon decay, elimination of steam voids, changing water density due to the reduction in water temperature, Doppler effect in uranium, changes in neutron leakage, and changes in control rod worth. To meet this objective, it is necessary to inject a quantity of boron which produces a minimum concentration of 850 parts per million (ppm) by weight of natural boron in the reactor core at 20°C. To allow for potential leakage and imperfect mixing in the reactor system, an additional approximately 25% (220 ppm) is added to the above requirement, resulting in a total requirement of greater than or equal to 1070 ppm. The required concentration is thus achieved in a mass of water equal to the sum of the mass of water in the RPV at normal water level (equal to or less than 455×10^3 kg) plus the mass of water in the RPV shutdown cooling piping (equal to or less than 130×10^3 kg). The quantity of boron solution contained in the storage tank above the pump suction shutoff level provides the required concentration of 1070 ppm when injected into the reactor.

The SLC System pumps have sufficient net positive suction head (NPSH) available at the pump. The SLC System pumps are designed to produce discharge pressure to inject the solution into the reactor when the reactor is at pressure conditions corresponding to the system relief valve (109.7 kg/cm^{2}g), which is above peak ATWS pressure in the RPV.

SLC System components required for RPV injection are classified as Seismic Category I.

Figure 2.2.4 shows the ASME Code class for the SLC System piping and components.

The SLC System is located in the Reactor Building. The storage tank, test water tank, the two positive displacement pumps, and associated valving are located in the secondary containment on the floor elevation below the operating floor.

Each of the SLC System divisions is powered from the respective Class 1E division. The power supplied to one motor-operated injection valve, suction valve, and injection pump is powered from Division I. The power supply to the

other motor-operated injection valve, suction valve, and injection pump is powered from Division II.

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

- Alarms for storage tank temperature and level.
- Parameter displays for the instruments shown on Figure 2.2.4.
- Controls and status indication for the pumps, injection valves, suction valves.
- A manual system initiation switch for each division.

The motor-operated valves (MOVs) shown on Figure 2.2.4have safety-related functions and perform these functions under differential pressure, fluid flow and temperature conditions.

The SLC System is physically separated from and independent of the hydraulic portion of the Control Rod Drive (CRD) System.

Inspections, Tests, Analyses and Acceptance Criteria

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

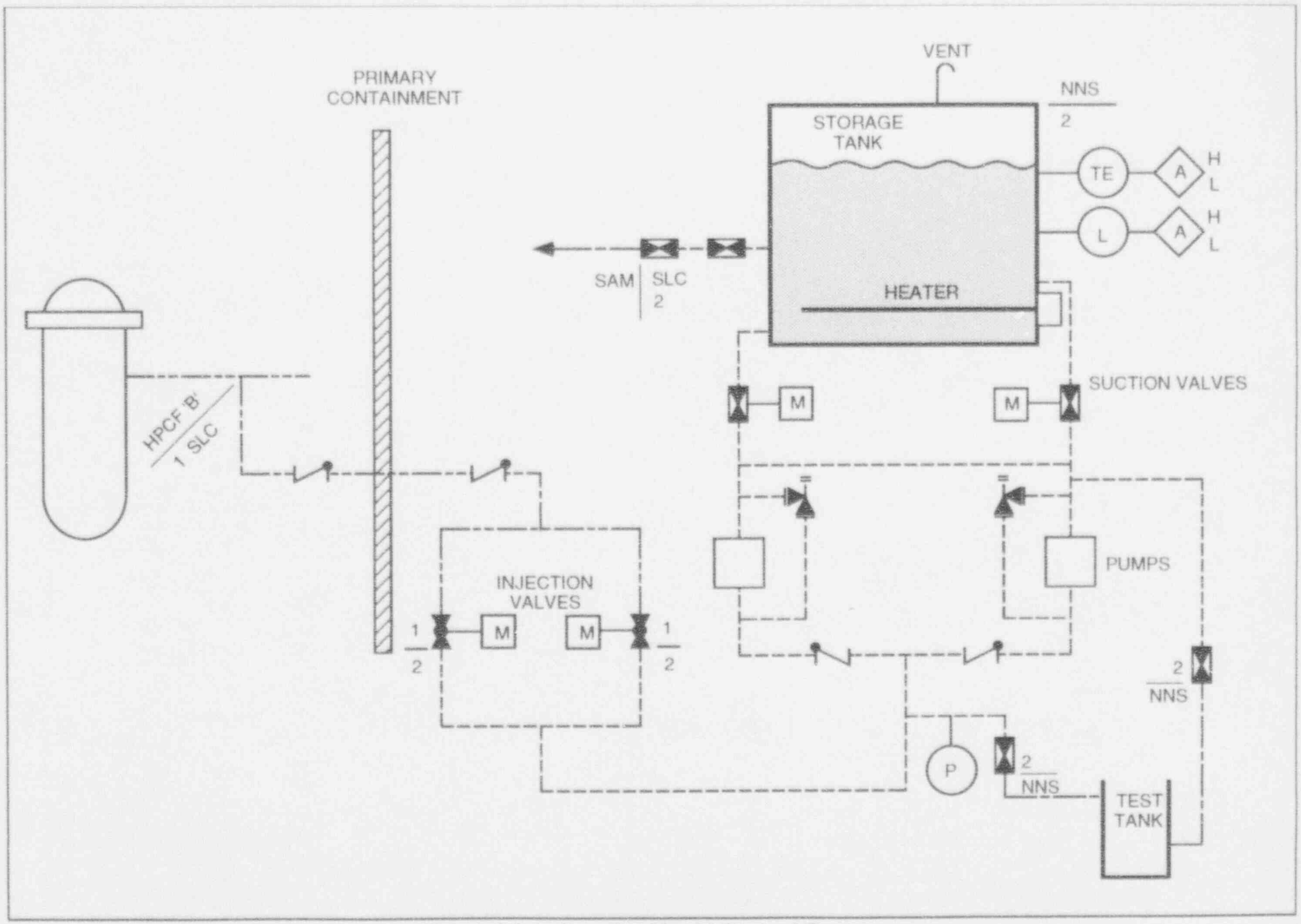


Figure 2.2.4 Standby Liquid Control System

Table 2.2.4 Standby Liquid Control System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the SLC System is shown in Figure 2.2.4.	1. Inspections of the as-built system will be conducted.	1. The as-built SLC System conforms with the basic configuration shown in Figure 2.2.4.
2. The ASME Code components of the SLC 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 portions of the SLC System that are required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME Code components of the SLC System conform with the requirements in the ASME Code, Section III.
3a. A test tank and associated piping and valves permit testing of the SLC System during plant operation. The tank is supplied with demineralized water which is pumped in either a closed loop or is injected into the reactor.	3a. Tests will be conducted on each division of the as-built SLC System using installed controls, power supplies and other auxiliaries The following tests will be conducted:	3a.
	(1) Demineralized water will be pumped against a pressure greater than or equal to 88.9 kg/cm^{2}a in a closed loop on the test tank.	(1) Demineralized water is pumped with a flow rate greater than or equal to 189 l/min in the closed loop.
	(2) Demineralized water will be injected from the test tank into the reactor.	(2) Demineralized water is injected from the test tank into the reactor.
3b. The SLC System delivers at least 378 l/min of solution with both pumps operating when the reactor pressure is less than or equal to 88.9 kg/cm^{2}a.	3b. Tests will be conducted on the as-built SLC System using installed controls, power supplies and other auxiliaries. Demineralized water will be injected from the storage tank into the reactor with both pumps running against a discharge pressure of less than or equal to 88.9 kg/cm^{2}a.	3b. The SLC System injects greater than or equal to 378 liters/minute into the reactor with both pumps running against a discharge pressure of greater than or equal to 88.9 kg/cm^{2}a.

Table 2.2.4 Standby Liquid Control System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3c. The SLC System delivers at least 189 l/min of solution with either pump operating when the reactor pressure is less than or equal to 88.9 kg/cm^{2}a.	3c. Tests will be conducted on the as-built SLC System using installed controls, power supplies and other auxiliaries. Demineralized water will be injected from the storage tank into the reactor with one pump running against a discharge pressure of less than or equal to 88.9 kg/cm^{2}a.	3c. The SLC System injects greater than or equal to 189 liters/minute into the reactor with either pump running against a discharge pressure greater than or equal to 88.9 kg/cm^{2}a.
3d. The SLC System can be manually initiated from the main control room.	3d. Tests will be conducted on the as-built SLC System using the manual initiation switch.	3d. Each division of the SLC System initiates when the manual initiation switch for that division is actuated.
3e. Both divisions of the SLC System are automatically initiated during an ATWS.	3e. Tests will be conducted on the as-built SLC System using simulated ATWS signals.	3e. Upon receipt of a simulated ATWS signal, both divisions of SLC automatically initiate.
3f. The performance of the SLC System is based on the following plant parameters:	3f. The as-built dimensions will be used in a volumetric analysis to calculate the volumes listed below:	3f.
(1) Storage Tank pumpable volume is greater than or equal to 23.1 m^3	(1) Minimum Storage tank pumpable volume	(1) Storage tanks pumpable volume is greater than or equal to 23.1 m^3.
(2) RPV water inventory is less than or equal to 455 x 10^3 kg at normal water level and 20°C.	(2) RPV water inventory at normal water level and 20°C.	(2) RPV water inventory is less than or equal to 455 x 10^3 kg at 20°C.
(3) RHR shutdown cooling system inventory is less than or equal to 130 x 10^3 kg at 20°C.	(3) RHR shutdown cooling system water inventory at 20°C.	(3) RHR shutdown cooling system inventory is less than or equal to 130 x 10^3 kg at 20°C.
3g. The SLC pumps have sufficient NPSH.	3g. Tests will be conducted on the as-built SLC System on the as-built SLC System by injecting demineralized water using both SLC System pumps from the storage tank to the RPV with the storage tank at the low level (pump trip level) and a temperature of greater than or equal to 43°C.	3g. The available NPSH exceeds the NPSH required when the SLC System injects greater than or equal to 378 liters/minute.

Table 2.2.4 Standby Liquid Control System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3h. The SLC System pump relief valves open when the inlet pressure to the valve equals or exceeds the setpoint (109.7 kg/cm^{2}g).	3h. Shop or field tests will be conducted to determine the relief valve setpoint.	3h. The SLC System pump relief valves open when the inlet pressure to the valve equals or exceeds 109.7 kg/cm^{2}g.
4. Class 1E loads of the SLC System are powered from Class 1E Divisions, as described in Section 2.2.4.	4. Tests will be conducted on the SLC System by providing a test signal in only one Class 1E Division at a time.	4. The test signal exists only in the Class 1E Division under test in the SLC System.
5. Main control room alarms, displays, and controls provided for the SLC System are defined in Section 2.2.4.	5. Inspections will be performed on the main control room alarms, displays, and controls for the SLC System.	5. Alarms, displays, and controls exist or can be retrieved in the main control room as defined in Section 2.2.4.
6. The MOVs designated in Section 2.2.4 open under differential pressure, fluid flow and temperature conditions.	6. Opening tests of installed valves will be conducted under pre-op differential pressure, fluid flow, and temperature conditions.	6. Each MOV opens upon receipt of the actuation signal.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
<u>FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS</u>

SYSTEM: 2.2.4 Standby Liquid Control

PUNCH LIST ITEM: A. Verify preoperational test procedure tests ability of each pump to deliver capacity with opposite relief valve open or removed.

GE DISPOSITION: Chapter 14 of the SSAR has been modified to include a commitment that the preoperational tests of the pumps in this mode will be conducted.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.2.4 Standby Liquid Control

PUNCH LIST ITEM: B. Add MOV boilerplate.

GE DISPOSITION: The revised Standby Liquid Control System entry 2.2.4 now includes a standardized MOV entry in the ITAAC table.

2.2.7 Reactor Protection System

Design Description

The Reactor Protection System (RPS) is an instrumentation and control system and its purpose is to initiate reactor scram whenever RPS logic requirements for scram initiation are satisfied.

As shown in Figure 2.2.7.a, the RPS interfaces with Neutron Monitoring System (NMS), Process Radiation Monitoring System (PRRM), Nuclear Boiler System (NBS), Control Rod Drive System (CRD), Rod Control and Information System (RCIS), Recirculation Flow Control System (RFCS), Suppression Pool Temperature Monitoring System (SPTM), Essential Multiplexing System (EMS), and the Process Computer System. Figure 2.2.7.a also depicts the implementation of RPS logic within Safety System Logic and Control (SSLC).

The RPS has four divisions. Figure 2.2.7.b shows the RPS divisional aspects and the signal flow paths from sensors to scram pilot valve solenoids. Equipment within an RPS division consists of sensors (transducers or switches), multiplexers, Digital Trip Modules (DTM), Trip Logic Unit (TLU), Output Logic Unit (OLU), and Load Drivers (LD). The LDs are only in divisions II and III.

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

The RPS consists of logic and circuitry for initiation of both automatic and manual scrams. The automatic scram function is comprised of four independent divisions of sensor instrument channels, hardware/software based logic, and two independent divisions of actuating devices. Automatic scram is initiated whenever a scram condition is detected by two or more automatic divisions of RPS logic. For automatic scram, the sensor input signals to RPS originate either from RPS's own sensors or other systems' sensors. For determination of the existence of an automatic scram condition, within each automatic scram channel of the RPS, the DTM of a given RPS channel compares the monitored process variable with the stored set-point in its memory and issues a trip signal if the monitored process variable exceeds the set-point. The DTM then sends the trip signal to the TLU of its own channel and the TLUs of the other three channels of RPS, where two-out-of-four voting is performed (see Figure 2.2.7.b).

In the case of main steam line high radiation and inboard/outboard MSIV closure signals, the PRRM system and NBS provide their divisional discrete trip signals (trip/no-trip) directly to their corresponding divisional RPS DTM. However, in the case of NMS, the four channels of NMS each provide their trip signals to each RPS divisional TLU. A list of conditions that can cause automatic

reactor scram is provided below. The name of the system that provides the sensor input signal or the trip signal is shown in brackets.

(1) Turbine Stop Valves Closure at above 40% power levels [RPS]

(2) Low Turbine Control Valves Oil Pressure (Fast Closure) at above 40% power levels [RPS]

(3) NMS Trips [Discrete trip signals to RPS TLUs]

(4) High Reactor Pressure [NBS]

(5) Low Reactor Water Level [NBS]

(6) High Drywell Pressure [NBS]

(7) Main Steam Lines Isolation [NBS discrete signals to RPS DTMs]

(8) Low Control Rod Drive Accumulator Charging Header Pressure [CRD]

(9) High Suppression Pool Temperature [SPTM discrete signals to RPS DTMs]

The TLUs provide their trip signals to their divisional OLUs which are used to control the solid-state LDs that control the Class 1E AC power to the scram solenoids, and relays that control DC power to back-up scram valves. For automatic scram initiation, the TLUs trip signals cause the LDs to interrupt Class 1E AC power to the scram solenoids (fail-safe logic), cause the back-up scram relays to supply DC power to back-up scram solenoids, and provide scram follow signals to the RCIS. Each division of RPS controls eight LDs. The LDs are arranged to switch AC power to the scram solenoids in a two-out-of-four format. That is, reactor scram will occur only if two or more divisions of RPS provide trip signals to their associated LDs.

Manual scram function, which is separate and independent from automatic scram, is implemented in divisions II and III of RPS. For manual scram initiation two manual scram push-buttons of the RPS must be simultaneously depressed. When manual scram is initiated, RPS, through manual scram switches, interrupts Class 1E AC power to the scram solenoids, connects divisional Class 1E DC power to scram air header dump valves (backup scram valves) and provides scram follow signals to RCIS. The RPS logic seals in the scram signal, and permits reset of scram logic after a time delay of at least 10 seconds.

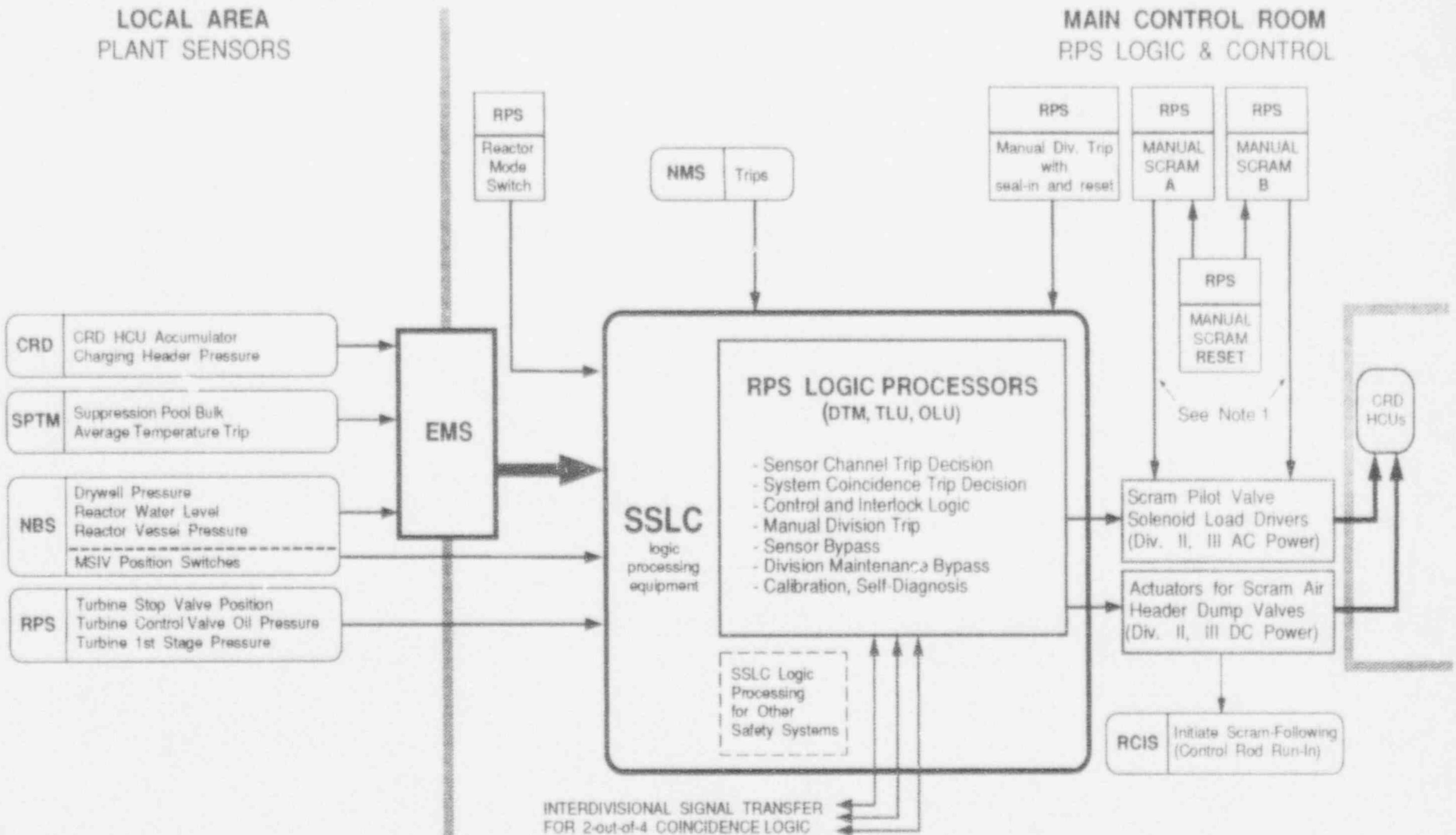
The RPS design is single-failure-proof, and redundant. Also, the RPS design is fail-safe in the event of loss of electrical power to one division of RPS logic.

Each of the four RPS divisional logic and associated sensors are powered from their respective divisional Class 1E power supply. In the RPS, independence is provided between Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment.

As shown on Figure 2.2.7a, the RPS has manual divisional trip switches, reactor mode switch, manual scram switches, and scram reset switches for manual controls. Divisional trip displays, and scram solenoids electrical power status lights are also provided. These RPS controls and displays are provided in the main control room. RPS sensors are turbine control valve oil pressure switches, turbine stop valve position switches, and turbine first stage pressure sensors. These sensors are located in the turbine building.

Inspections, Tests, Analyses and Acceptance Criteria

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



NOTE:
1. Hardwire connection.
2. Diagram Represents One Of Four Divisions.

Figure 2.2.7a Reactor Protection System Control Interface Diagram

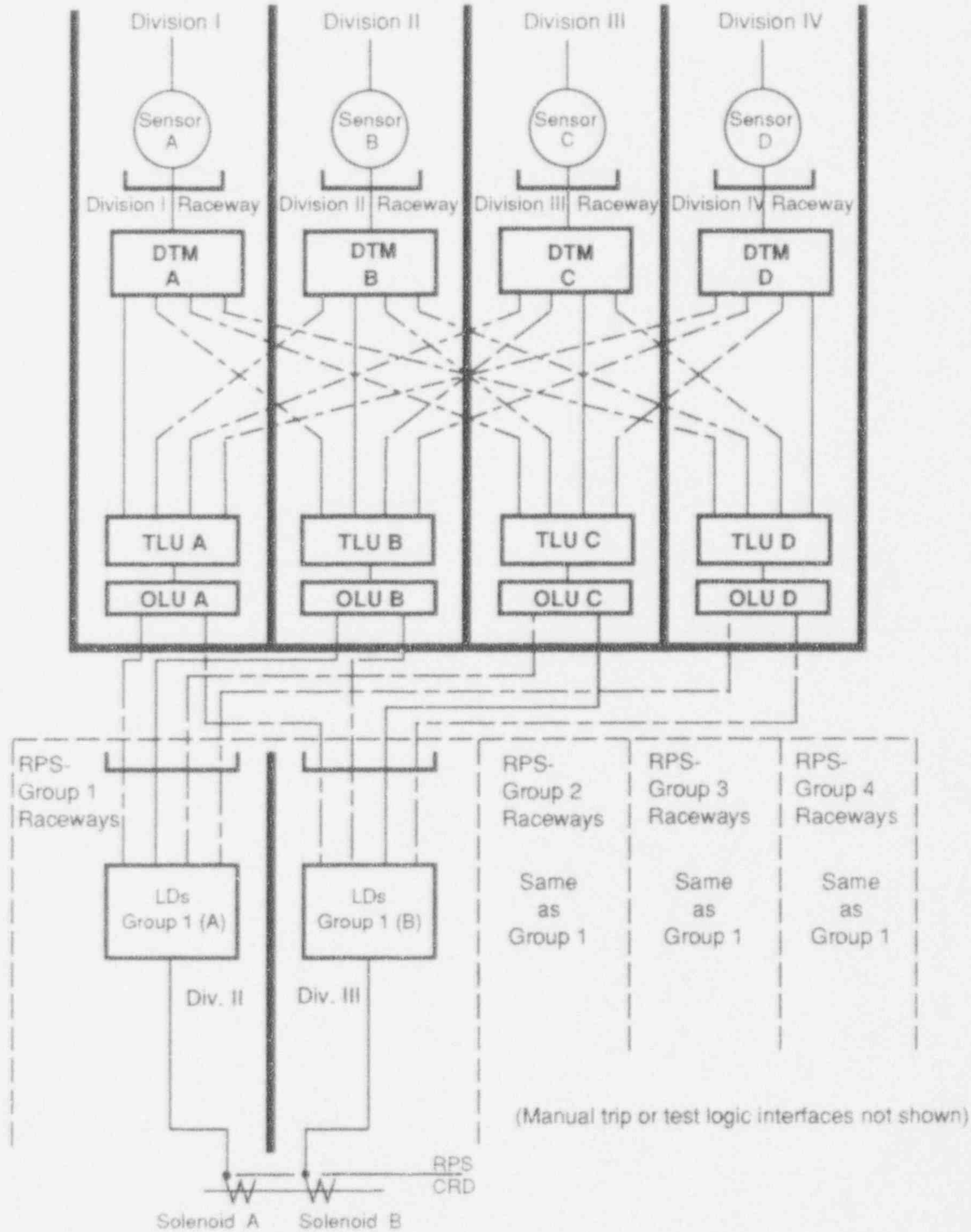


Figure 2.2.7b Reactor Protection System

Table 2.2.7 Reactor Protection System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Equipment comprising the RPS is defined in Section 2.2.7.	1. Inspection of the as-built system will be conducted.	1. The as-built RPS conforms with the description in Section 2.2.7.
2. RPS logic uses four independent sensor instrument channels of each process variable described in Section 2.2.7 for its automatic scram function.	2. 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 of RPS.	2. The RPS LDs change their states to interrupt electrical power to scram solenoids. RPS back-up scram relays close and RCIS relays close to provide signals to RCIS.
3. For manual scram initiation two manual scram push-buttons of the RPS must be simultaneously depressed.	3. Tests will be conducted by depressing the scram push-button A, the B scram push-button, and both.	3. When manual scram push-button A is depressed Div II AC power to A scram solenoids is interrupted. When scram push-button B is depressed Div III AC power to B scram solenoids is interrupted. When both A & B scram push-buttons are depressed reactor scram occurs. RPS back-up scram relays close to energize the solenoids of scram air header dump valves and RCIS relays close to provide signals to RCIS.
4. The RPS logic seals in the scram signal, and permits reset of scram logic after a time delay of at least 10 seconds.	4. Tests will be conducted by attempting to reset RPS scram circuitry during the 10 seconds time period after scram initiation.	4. During the 10 second time period after scram initiation, reset does not occur.
5. RPS design is fail-safe in the event of loss of electrical power to one division of RPS logic.	5. Tests will be conducted by disconnecting electrical power to one division of RPS logic at a time.	5. Upon loss of electrical power to one division of RPS logic, the LDs of that division change their state to interrupt electrical power to scram solenoids.

Table 2.2.7 Reactor Protection System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. In the RPS independence is provided between Class 1E divisions, and between Class 1E divisions and Non-Class 1E equipment.	6. a. Tests will be performed on the RPS by providing a test signal to only one Class 1E division at a time. b. Inspection of the as-installed Class 1E divisions in the RPS will be performed.	6. a. The test signal exists only in the Class 1E division under test in the RPS. b. In the RPS physical separation exists between Class 1E divisions. Physical separation exists between these Class 1E divisions and Non-Class 1E equipment.
7. Main control room displays and controls provided for the RPS are as defined in Section 2.2.7.	7. Inspections will be performed on the main control room displays and controls for the RPS.	7. Displays and controls exist or can be retrieved in the main control room as defined in Section 2.2.7.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.2.7 Reactor Protection System

PUNCH LIST ITEM: 1. Provide SAR amendment that revises SAR:
- to remove high seismic activity scram
- to remove TCV fast-closure limit switch inputs
- to modify 7.2 text tables and logic.

GE DISPOSITION: These changes have been incorporated in the SSAR.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.2.7 Reactor Protection System

PUNCH LIST ITEM: 2. Check Turbine Building ITAAC for instrument list consistency. [Turbine first stage pressure].

GE DISPOSITION: This review has been completed and it was necessary to add the turbine first-stage pressure sensor to the list of instruments discussed in the Turbine Building design description.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.2.7 Reactor Protection System

PUNCH LIST ITEM: 3. For RPS, add configuration check per LDS ITAAC No. 1.

GE DISPOSITION: A configuration check using the same language as the LDS item has been added to the RPS ITAAC as ITAAC No. 1.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.2.7 Reactor Protection System

PUNCH LIST ITEM: 4. Review the need for the work "power" in ITAAC No. 9. (Why did GE choose to deviate from the agreed-to boilerplate?)

GE DISPOSITION: The Reactor Protection System design description and ITAAC table have been revised to utilize the standardized Independence material developed during tne January 1993 GE/NRC/NUMARC meetings on ABWR Tier 1.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.2.7 Reactor Protection System

PUNCH LIST ITEM: 5. Address Divisional issue

GE DISPOSITION: Since the March 1993 GE/NRC discussions of the RPS, GE has reviewed this punch list item and has reached the conclusion that no changes are necessary to Figure 2.2.7b. The issue centers on the use of the word "division" for both the incoming channels of sensor information and the outgoing divisions of signals which initiate reactor scram. GE understands the NRC's concern, but does not believe the figure as defined is ambiguous or misleading. Furthermore, there appears to be no obvious other description that would be any more technically correct than the approach currently being used. GE is certainly open to any NRC suggestions for clarifying changes to this figure.

2.4 Core Cooling

2.4.1 Residual Heat Removal System

Design Description

The Residual Heat Removal (RHR) System has three separate divisions. The major functions of the RHR System are:

(1) Containment heat removal,

(2) Reactor decay heat removal,

(3) Emergency reactor vessel level makeup, and

(4) Augmented fuel pool cooling.

Figures 2.4.1a, 2.4.1b, and 2.4.1c show the basic system configuration and scope. Figure 2.4.1d shows the RHR System control interfaces.

The RHR System operates in the following modes:

(1) Low pressure core flooder (LPFL) (Divisions A, B, and C),

(2) Suppression pool cooling (Divisions A, B, and C),

(3) Wetwell spray (Divisions B, and C),

(4) Drywell spray (Divisions B, and C),

(5) Shutdown cooling (Divisions A, B, and C),

(6) Augmented fuel pool cooling, and fuel pool makeup (Divisions B, and C),

(7) Alternating current power source (AC) independent water addition (Division C),

(8) Full flow test (Divisions A, B, and C), and

(9) Minimum flow bypass (Divisions A, B, and C.).

Except for the non-ASME Code Class components of the AC independent water addition feature (Figure 2.4.1c), the entire RHR System shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c is classified as safety-related.

Low Pressure Core Flooder Mode

As shown on Figure 2.4.1d, the RHR system channel measurements are provided to the Safety System Logic and Control (SSLC) System for signal processing, setpoint comparisons, and generating trip signals. The RHR System is automatically initiated when either a high drywell pressure or low reactor water level condition exists, i.e., a loss-of-coolant accident (LOCA) signal. A RHR initiation signal is provided to the systems as identified on Figure 2.4.1d. The SSLC processors use a two-out-of-four voting logic for System initiation. Each RHR division can also be initiated manually (LPFL mode).

Following receipt of an initiation signal, the RHR System automatically initiates and operates in the LPFL mode to provide emergency makeup to the reactor vessel. The LPFL injection flow for each division begins when the reactor pressure vessel (RPV) dome pressure is no less than 15.8 kg/cm^2 above the drywell pressure. When the RPV dome pressure is no less than 2.8 kg/cm^2 greater than the drywell pressure, the LPFL injection flow for each division is 954 m^3/hr minimum. The LPFL mode is accomplished by all three divisions of the RHR System by transferring water from the suppression pool to the RPV, via the RHR heat exchangers. The system automatically reverts to the LPFL mode of operation from the test mode, the suppression pool cooling, or wetwell spray modes upon receipt of an initiation signal. If a drywell spray valve is open in Division B or C, that RHR division automatically reverts to the LPFL mode in response to the injection valve beginning to open. The RPV injection valve in each division requires a low reactor pressure permissive signal to open and closes automatically on receipt of a high reactor vessel pressure signal.

Suppression Pool Cooling Mode

The suppression pool cooling mode of the RHR System limits the long-term post-LOCA temperature of the suppression pool, and limits the long-term peak temperatures and pressures within the wetwell and drywell regions of the containment. In this mode, the RHR System circulates water through the RHR heat exchangers and returns it directly to the suppression pool. This mode is manually initiated by control of individual system components. In the suppression pool cooling mode, the total heat removal capacity between the RHR and ultimate heat sink of no less than 88.5 kcal/sec°C, for each division. The heat removal path is the RHR heat exchanger, the Reactor Building Cooling Water (RCW) System, and the Reactor Service Water (RSW) System. In the suppression pool cooling mode, the RHR tube side heat exchanger (Hx) flow rate is 954 m^3/hr minimum per division. The RHR pumps have sufficient net positive suction head (NPSH) available at the pump.

Containment Spray Mode

The containment spray mode of the RHR System is available in Divisions B and C, and consists of the wetwell spray and drywell spray operating together. In this mode, the RHR System pumps suppression pool water to a single wetwell spray

header and single drywell spray header through the associated RHR heat exchanger. The containment spray mode of the RHR System is initiated manually by control of individual system components. The drywell spray inlet valves can only be opened if a high drywell pressure condition exists and if the injection valves are fully closed. The wetwell spray flow rate for either Division B or C, is no less than 114 m^3/hr.

Shutdown Cooling Mode

In the shutdown cooling mode of operation, the RHR System removes decay heat from the reactor core, and is used to achieve and maintain a cold shutdown condition by removing decay and sensible heat from the core and reactor vessel. This mode reduces reactor pressure and temperature to cold shutdown conditions. In this mode, each division takes suction from the RPV via its dedicated suction line, pumps the water through its respective heat exchanger tubes, and returns the cooled water to the RPV. Two divisions (B and C) discharge water back to the RPV via dedicated spargers, while the third division (A) utilizes the vessel spargers of one of the two feedwater lines (FW-A). Shutdown cooling is initiated manually once the RPV has been depressurized below the system low pressure permissive. In any division, the shutdown cooling suction valve cannot be opened unless the following valves in that division are closed:

(1) Suppression pool suction valve,

(2) Suppression pool return valve,

(3) Drywell spray valves, and

(4) Wetwell spray valve

Each shutdown cooling suction valve automatically closes on low reactor water level. The low pressure portions of the shutdown cooling piping are protected from high reactor pressure by automatic closure of the shutdown cooling suction valves on a high reactor vessel pressure. The shutdown cooling flow rate for any division is no less than 954 m^3/hr.

Augmented Fuel Pool Cooling and Fuel Pool Makeup

The augmented fuel pool cooling mode of the RHR System (Divisions B and C) can supplement the Fuel Pool Cooling (FPC) System as follows: (1) Directly cooling the fuel pool by circulation fuel pool water through the RHR heat exchanger and returning it to the fuel pool; (2) While providing shutdown cooling during refueling operations, return the cooled RHR shutdown cooling flow to the fuel pool. Also, this mode provides for fuel pool emergency makeup capability by permitting the RHR pumps (Divisions B and C) to transfer suppression pool water to the fuel pool. This mode is accomplished manually by

control of individual system components. In the augmented fuel pool cooling mode, the RHR tube side heat exchanger flow rate for Division B or C is no less than 350 m^3/hr.

AC Independent Water Addition Mode

Division C of the RHR System also functions in an AC independent water addition mode. This mode provides a means of injecting emergency makeup water to the reactor by cross connecting the Reactor Building Fire Protection (FP) System header to the RHR System just outside the containment. This makes it independent of the normal safety-related AC power distribution network. This mode is accomplished by opening two in-series valves on the cross-connection piping just upstream of the tie-in to the normal RHR piping. This is accomplished by local manual action at the valves. Fire Protection System water can be directed to either the RPV or the drywell spray sparger by local manual opening of the Division C RHR injection valve or the two Division C drywell spray valves.

Full Flow Test Mode

Each division of the RHR System has a full flow test mode to permit pump flow testing during plant operation. In this mode, the system is essentially operated in the suppression pool cooling mode, drawing suction from and discharging back to the suppression pool.

Minimum Flow Bypass Mode

Each division of the RHR System has a minimum flow bypass mode that assures there is always flow in the RHR pumps when they are 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 RHR System.

The RHR System is classified as Seismic Category I. Figures 2.4.1a, and 2.4.1b, and 2.4.1c show the ASME Code Class for the RHR System. The RHR System is located in the Reactor Building.

Each of the three divisions is powered from the respective Class 1E division as shown on Figures 2.4.1a, 2.4.1b, 2.4.1c. In the RHR System, independence is provided between Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment.

Outside the containment, each mechanical Division of the RHR System (Divisions A, B, and C) is physically separated from the other divisions.

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

(1) Parameter displays for the instruments shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c.

(2) Controls and status indication for the active safety-related components shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c.

(3) Manual system level initiation capability for LPFL mode.

RHR System components with displays and control interfaces with the Remote Shutdown System (RSS) are shown on Figures 2.4.1a and 2.4.1b.

The safety-related electrical equipment shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c located inside the primary containment and the Reactor Building is qualified for a harsh environment.

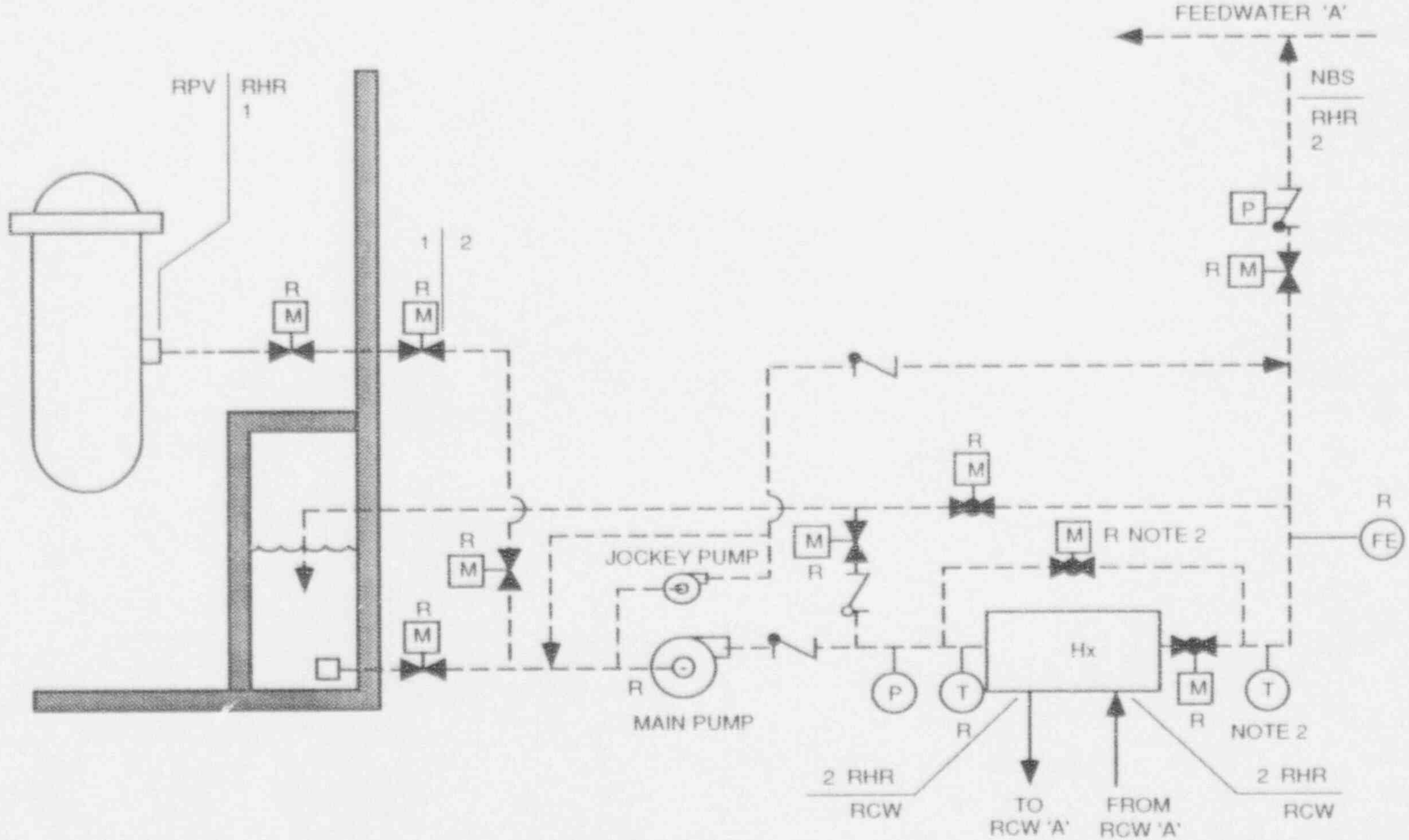
The motor-operated valves shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c (except the heat exchanger bypass and discharge valves) have active safety-related functions and perform these functions under differential pressure, fluid flow and temperature conditions.

The pumps are interlocked to prevent starting with a closed suction path.

The piping and components outside the containment isolation valves and on the suction side of the pump have a design pressure of 28.8 kg/cm^2g for intersystem LOCA (ISLOCA) conditions.

Inspections, Tests, Analyses and Acceptance Criteria

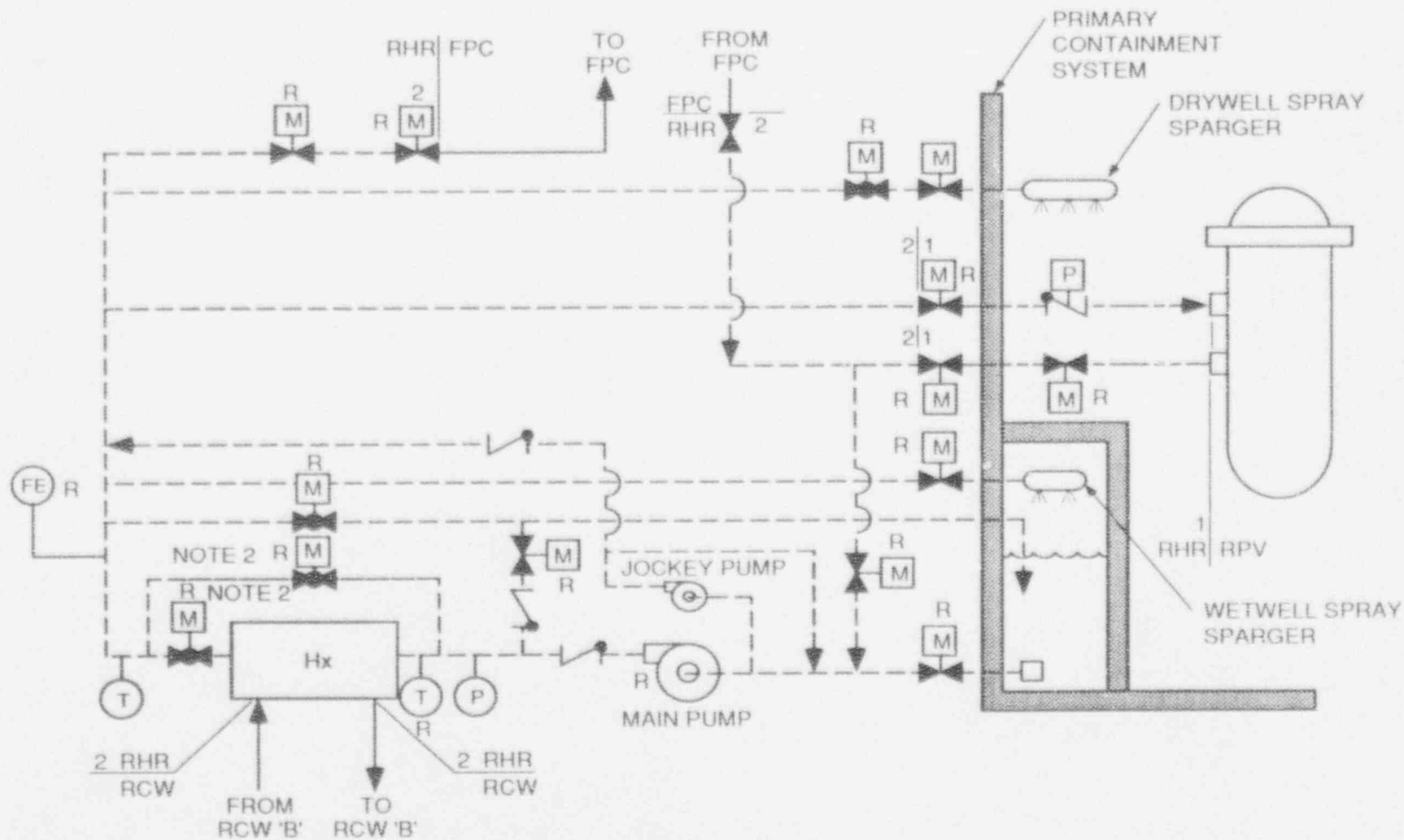
This section provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria which will be undertaken for the RHR System.



NOTES:

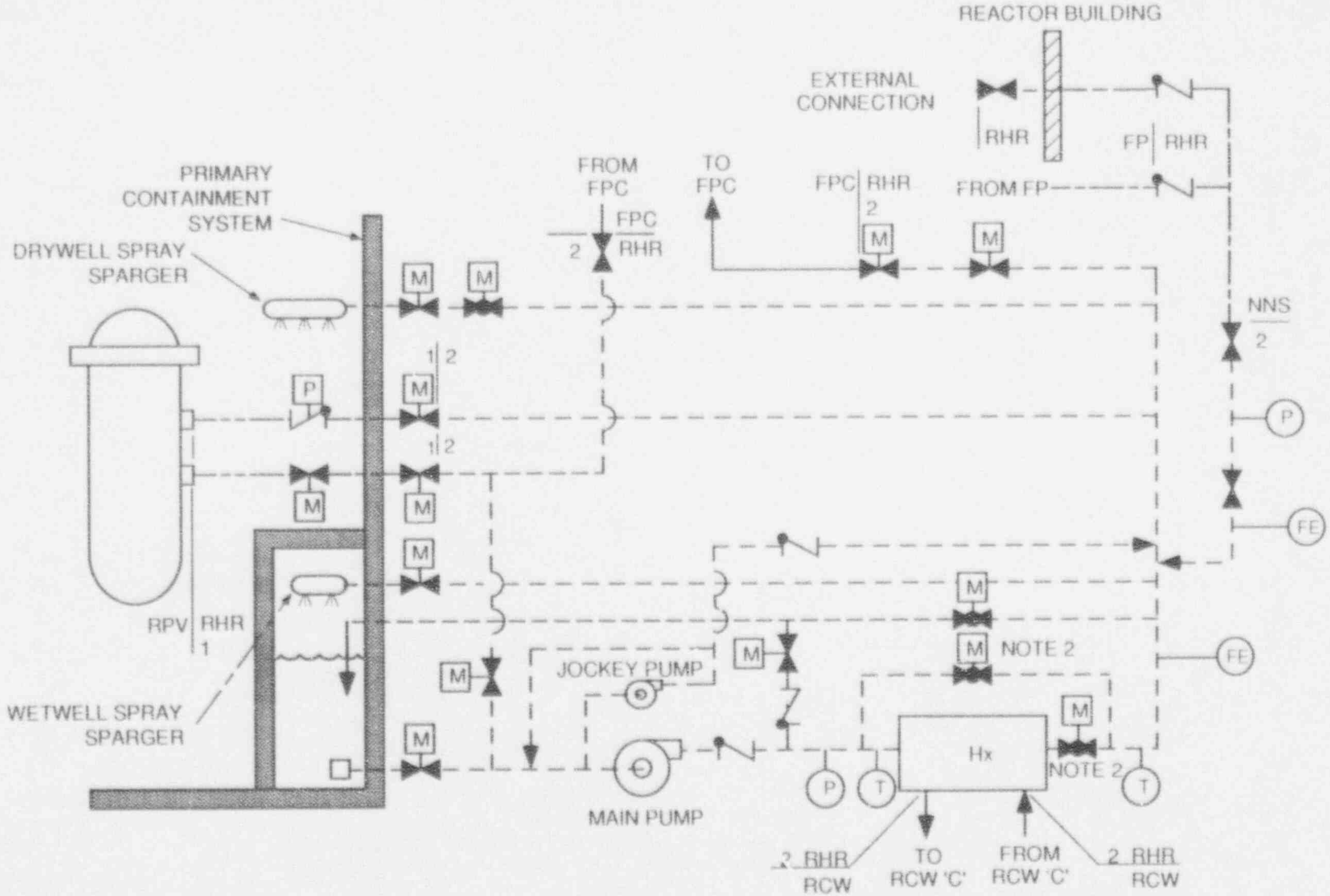
1. ALL ELECTRICAL POWER LOADS FOR THE CLASS 1E COMPONENTS SHOWN ON THIS FIGURE ARE POWERED FROM CLASS 1E DIVISION I EXCEPT FOR THE OUTBOARD CONTAINMENT ISOLATION VALVE, WHICH IS DIVISION II.
2. NOT AN ACTIVE SAFETY-RELATED VALVE.

Figure 2.4.1a Residual Heat Removal (RHR-A) System



NOTES:
1. ALL ELECTRICAL POWER LOADS FOR THE CLASS 1E COMPONENTS SHOWN ON THIS FIGURE ARE POWERED FROM CLASS 1E DIVISION II EXCEPT FRO THE OUTBOARD CONTAINMENT VALVE, WHICH IS DIVISION III.
2. NOTAN ACTIVE SAFETY-RELATED VALVE.
3. DRYWELL AND WETWELL SPRAY SPARGERS ARE COMMON TO DIVISION B AND C.

Figure 2.4.1b Residual Heat Removal (RHR-B) System



NOTES:
1. ALL ELECTRICAL POWER LOADS FOR THE CLASS 1E COMPONENTS SHOWN ON THIS FIGURE ARE POWERED FROM CLASS 1E DIVISION III EXCEPT FOR THE OUTBOARD CONTAINMENT ISOLATION VALVE, WHICH IS DIVISION I.
2. NOT AN ACTIVE SAFETY-RELATED VALVE.
3. DRYWELL AND WETWELL SPRAY SPARGERS ARE COMMON TO DIVISIONS B AND C.

Figure 2.4.1c Residual Heat Removal (RHR-C) System

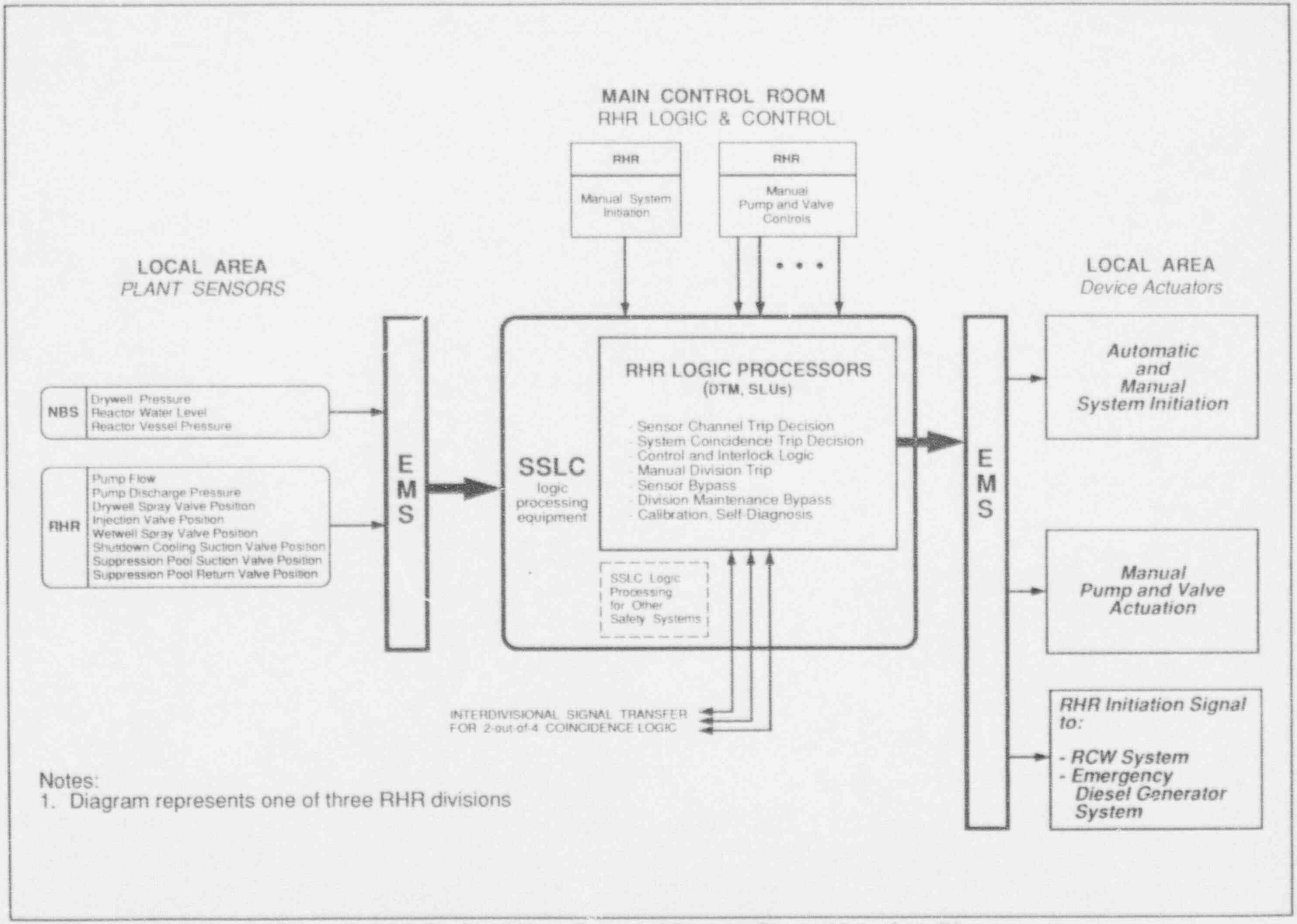


Figure 2.4.1d Residual Heat Removal System Control Interface Diagram

Table 2.4.1 Residual Heat Removal System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the RHR System is shown in Figures 2.4.1a, 2.4.1b, 2.4.1c, and 2.4.1d.	1. Inspections of the as-built system will be conducted.	1. The as-built RHR System conforms with the basic configuration shown in Figures 2.4.1a, 2.4.1b, 2.4.1c, and 2.4.1d.
2. The ASME Code components of the RHR 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 RHR System that are required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME Code components of the RHR System conform with the requirements in the ASME Code, Section III.
3a. The RHR System is automatically initiated in the LPFL 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. Each division of RHR receives an initiation signal.
3b. Each RHR division can be initiated manually (LPFL mode).	3b. Test will be conducted by initiating each division manually.	3b. Each division of RHR receives an initiation signal.
3c. Following receipt of an initiation signal, the RHR System automatically initiates and operates in the LPFL mode to provide emergency makeup to the reactor vessel.	3c. Tests will be conducted on each RHR division using a simulated initiation signal.	3c. Upon receipt of a simulated initiation signal the following occurs: a) The RHR pump receives a signal to start, b) The RPV injection valve receives a signal to open provided a low reactor pressure permissive is present, c) The suppression pool return valve receives a signal to close, d) The wetwell spray valve receives a signal to close (Divisions B and C only).

Table 2.4.1 Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3d. The LPFL injection flow for each division begins when the RPV dome pressure is no less than 15.8 kg/cm^2 above the drywell pressure. When the RPV dome pressure is no less than 2.8 kg/cm^2 greater than the drywell pressure, the LPFL injection flow for each division is 954 m^3/hr minimum.	3d. Tests will be conducted on the as-built RHR System in the RHR LPFL mode. Analyses will be performed to convert the test results to the conditions of the Design Commitment.	3d. The converted RHR flow satisfies the following: The LPFL injection flow for each division begins when the RPV dome pressure is no less than 15.8 kg/cm^2 above the drywell pressure. When the RPV dome pressure is no less than 2.8 kg/cm^2 greater than the drywell pressure, the LPFL injection flow for each division is 954 m^3/hr minimum.
3e. The system automatically reverts to the LPFL mode of operation from the test mode, the suppression pool cooling or wetwell spray modes upon receipt of an initiation signal.	3e. Tests will be conducted on each RHR division using simulated LPFL initiation signals.	3e. Each division automatically reverts to the LPFL mode of operation from the test mode, the suppression pool cooling or wetwell spray modes upon receiving an initiation signal.
3f. If a drywell spray valve is open in Division B or C, that RHR division automatically reverts to the LPFL mode in response to the injection valve beginning to open.	3f. Tests will be conducted on RHR Division B and C drywell spray mode using a simulated injection valve opening signal.	3f. Drywell spray valves in a division close on receipt of injection valve not fully closed signal in that division.
3g. The RPV injection valve in each division requires a low reactor vessel pressure permissive signal to open and closes automatically on receipt of a high reactor vessel pressure signal.	3g. Tests will be conducted on the injection valves in each RHR division using a simulated reactor vessel pressure signal.	3g. The RPV injection valve in each division requires a low reactor vessel pressure permissive signal to open and closes automatically on receipt of a high reactor vessel pressure signal.
4a. The total heat removal capacity requirements between the RHR System and ultimate heat sink is no less than 88.5 kcal/sec°C for each division.	4a. Inspections and analyses will be performed to determine the heat exchanger's effective heat removal capacity, for each division.	4a. The total heat removal capacity requirements between the RHR System and ultimate heat sink is no less than 88.5 kcal/sec°C for each division.
4b. In the suppression pool cooling mode, the RHR tube side heat exchanger, (Hx) flow rate is 954 m^3/hr minimum, per division.	4b. Tests will be performed on each RHR division.	4b. In the suppression pool cooling mode, the RHR tube side heat exchanger, (Hx) flow rate is 954 m^3/hr minimum, per division.

Table 2.4.1 Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4c. The RHR pumps have sufficient NPSH.	4c. 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 the suppression pool with water level at the minimum value – 50% blockage of pump suction strainers, – design basis fluid temperature (100°C), – containment at atmospheric pressure.	4c. The available NPSH exceeds the NPSH required.
5a. The drywell spray inlet valves can only be opened if a high drywell pressure exists and if the injection valves are fully closed.	5a. Tests will be performed of the drywell spray valve interlock logic using simulated drywell pressure and valve position signals.	5a. The two in-series drywell spray valves are blocked from being open simultaneously unless signals indicative of the following conditions exist concurrently: a) Drywell pressure is high, b) The RPV injection valve is fully closed, c) The shutdown cooling suction valve is fully closed. The drywell spray valves will automatically close if signals indicative of the following condition exists: a) The RPV injection valve is not fully closed.
5b. The wetwell spray flow rate for either Division B or C, is no less than 114 m^3/hr.	5b. Tests will be conducted on Divisions B and C in the wetwell spray mode.	5b. RHR division B provides wetwell spray flow greater than or equal to 114 m^3/hr. RHR division C provides wetwell spray flow greater than or equal to 114 m^3/hr.

Table 2.4.1 Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6a. Shutdown cooling is initiated manually once the RPV has been depressurized below the system low pressure permissive.	6a. Tests will be conducted on the RHR shutdown cooling mode for manual initiation, using simulated reactor vessel pressure signals.	6a. The RHR shutdown mode operates when reactor vessel pressure is below system low pressure permissive. The RHR shutdown mode is not manually initiated when reactor vessel pressure is not less than the low pressure permissive.
6b. In any division, the shutdown cooling suction valve cannot be opened unless the following valves in that division are closed: Suppression pool suction valve, Suppression pool return valve, Drywell spray valves, and Wetwell spray valve.	6b. Tests will be conducted on each RHR division, to open the shutdown cooling suction valve.	6b. In any division, the shutdown cooling suction valve cannot be opened unless the following valves in that division are closed: Suppression pool suction valve, Suppression pool return valve, Drywell spray valves, and Wetwell spray valve.
6c. Each shutdown cooling suction valve automatically closes on low reactor water level.	6c. Tests will be conducted on each RHR division using a simulated reactor water level signal.	6c. Each shutdown cooling suction valve automatically closes on low reactor water level.
6d. The low pressure portions of the shutdown cooling piping are protected from high reactor pressure by automatic closure of the shutdown cooling suction valves on a high reactor vessel pressure signal.	6d. Tests will be conducted on the shutdown cooling suction valves in each RHR division using a simulated reactor vessel pressure signal.	6d. The shutdown cooling suction valves close when the RHR System receives a simulated high reactor vessel pressure signal.
6e. In the shutdown cooling mode, the RHR tube side heat exchanger flow rate is greater than or equal to 954 m^3 /hr (heat exchanger heat removal capacity in this mode is bounded by suppression pool cooling requirements).	6e. In the shutdown cooling mode, system functional tests will be performed to determine system flow rate through each heat exchanger. Inspections and analysis shall be performed to verify the shutdown cooling mode is bounded by suppression pool cooling requirements.	6e. The RHR heat exchangers tube side flow rate is greater than or equal to 954 m^3/hr. Heat exchanger removal capacity in this mode is bounded by suppression pool cooling requirements.

Table 2.4.1 Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. In the augmented fuel pool cooling mode, the RHR tube side heat exchanger flow rate for Divisions B or C is no less than 350 m^3/hr (heat exchanger heat removal capacity in this mode is bounded by suppression pool cooling requirements).	7. Tests will be performed to determine system flow rate through each heat exchanger in the augmented fuel pool cooling mode. Inspections and analysis shall be performed to verify the shutdown cooling mode is bounded by suppression pool cooling requirements.	7. The RHR tube side heat exchanger flow rate is greater than or equal to 350 m^3/hr. in the augmented fuel pool cooling mode. Heat exchanger heat removal capacity in this mode is bounded by suppression pool cooling requirements.
8. Each division of the RHR has a minimum flow bypass mode that assures there is always flow in the RHR pumps when they are operating.	8. Tests will be conducted on the pump minimum flow valve interlock logic using simulated pressure and flow signals.	8. The pump minimum flow valve 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: a) Pump flow exceeds minimum value.
9. In the RHR System independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	9a. Tests will be performed on the RHR System by providing a test signal to only one Class 1E division at a time.	9a. The test signal exists only in the Class 1E division under test in the RHR System.
	9b. Inspection of the as-installed Class 1E divisions in the RHR System will be performed.	9b. In the RHR System, physical separation exists between Class 1E divisions. Physical separation exists between these Class 1E divisions and non-Class 1E equipment.
10. Each mechanical division of the RHR System (Divisions A, B, C) is physically separated from the other divisions.	10. Inspections of the as-built RHR System will be performed.	10. Each mechanical division of the RHR System is physically separated from other mechanical divisions of RHR System by structural and/or fire boundaries with the exception of primary containment.

Table 2.4.1 Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. Main control room displays and controls provided for RHR System are as defined in Section 2.4.1.	11. Inspections will be performed on the main control room displays and controls for the RHR System.	11. Displays and controls exist or can be retrieved in the main control room as defined in Section 2.4.1.
12. RSS displays and controls provided for the RHR System are as defined in Section 2.4.1.	12. Inspections will be performed on the RSS displays and controls for the RHR System.	12. Displays and controls exist on the RSS as defined in Section 2.4.1.
13. MOVs designated in Section 2.4.1 as having an active safety function open and/or close under differential pressure, fluid flow, and temperature conditions.	13. Opening and/or closing tests of installed valves will be conducted under pre-operational differential pressure, fluid flow, and temperature conditions.	13. Each MOV opens and/or closes.
14. The RHR pumps are interlocked to prevent starting with a closed suction path.	14. Tests will be conducted on the RHR pump start logic using simulated valve position signals.	14. The RHR pump is prevented from starting unless signals indicative of one of the following conditions exists: a) A suction path from the suppression pool is available. (The suppression pool suction valve is fully open.) b) A suction path from the RPV via the shutdown cooling suction line is available. (The shutdown cooling suction valve and inboard and outboard isolation valves are all fully open.)

SYSTEM: 2.4.1 Residual Heat Removal System

PUNCH LIST ITEM: A. Analysis Method to be in SSAR

GE DISPOSITION: GE will include a summary of analysis methods that will be used to assess the heat exchanger effective heat removal capability. These will be included in the RHR sections of Chapter 5 and submitted to the Staff along with other RHR system changes which currently are being included in the SSAR.

SYSTEM: 2.4.1 Residual Heat Removal System

PUNCH LIST ITEM: B. ITAAC 6B and 7d need to be coordinated to ensure consistency of test function.

GE DISPOSITION: The logic associated with the RHR injection valve in each RHR division has been consolidated in new ITAAC table entry 3g. This resolves any issues of consistency between the two original ITAAC entries 6B and 7d.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.4.1 Residual Heat Removal System

PUNCH LIST ITEM: C. Check consistency of interlock logic and testing in ITAAC 5 through 7.

GE DISPOSITION: GE has reviewed and restructured the RHR ITAAC table entries. As part of this process, The consistency of interlock logic and testing in the original ITAAC Numbers 5 through 7 has been reviewed and clarified.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.4.1 Residual Heat Removal System

PUNCH LIST ITEM: D. Modify for consistency based on definition acceptance.

GE DISPOSITION: GE has reviewed this item and believes the ITAAC as originally defined (Item 12d - Mechanical Separation) is correct and should remain in the RHR ITAAC table. This ITAAC table has been included in the revised RHR Tier 1 material as ITAAC entry No. 10.

ABWR DESIGN CERTIFICATION: L'SPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.4.1 Residual Heat Removal System

PUNCH LIST ITEM: E. Consistent treatment of codes for figures.

GE DISPOSITION: The ASME Code Class identification on RHR figures has been reviewed and made consistent with the guidelines in Appendix A of the GE Tier 1 material. This includes removing Code Class identification for systems which interface with the RHR.

2.4.3 Leak Detection and Isolation System

Design Description

The Leak Detection and Isolation System (LDS) is a control and instrumentation system whose function is to detect and monitor leakage from the reactor coolant pressure boundary and initiate isolation of the leakage source. The system is designed to initiate automatic isolation of the process lines that penetrate the containment by closing the isolation valves. The functions of the LDS include isolation of the main steamlines, the primary and secondary containment, and individual system process lines; activation of the Standby Gas Treatment System (SGTS); monitoring of leakages inside and outside the primary containment; and providing the monitored leakage parameters in the main control room.

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

The LDS logic design uses four instrument channels to monitor each leakage parameter that initiates an isolation function on a two-out-of-four channel trip.

As shown on Figure 2.4.3, the LDS safety-related channel measurements are provided as inputs to the Safety System Logic and Control System (SSLC) for signal processing, setpoint comparisons, and generation of the trip signals that initiate the isolation functions. The LDS isolation logic consists of safety-related sensors, redundant instrument channels and logic processors that initiate the automatic isolation functions. Once isolation is initiated, the logic seals in the isolation signal and operator action is required to reset the logic to its normal state.

The following primary and secondary containment isolation and automatic control functions are provided by the LDS using four instrument channels to monitor leakage:

(1) Closure of the main steamline isolation valves (MSIVs) and main steamline drain valves on a signal indicating low reactor water level, high main steamline flow in any main steamline, high ambient temperature in the steam tunnel area or in the Turbine Building, low main condenser vacuum, or low steam inlet pressure to the main turbine.

(2) Isolation of the Reactor Water Cleanup (CUW) System process lines on a signal indicating low reactor water level, high ambient MSL tunnel area temperature, high mass differential flow, high ambient temperature in the CUW areas, or when the Standby Liquid Control (SLC) System is activated.

(3) Initiation of the SGTS on a signal indicating high drywell pressure, low reactor water level, high radiation in the secondary containment or high radiation in the fuel handling area.

(4) Isolation of Reactor Building Heating, Ventilation and Air Conditioning (HVAC) System on a signal indicating high drywell pressure, low reactor water level, high radiation in the secondary containment or high radiation in the fuel handling area.

(5) Isolation of containment purge and vent lines on a signal indicating high drywell pressure, low reactor water level, high radiation in the secondary containment or high radiation in the fuel handling area.

(6) Isolation of the Reactor Building Cooling Water (RCW) System and of the HVAC Normal Cooling Water (HNCW) System lines on a signal indicating high drywell pressure or low reactor water level.

(7) Isolation of the Reactor Heat Removal (RHR) System shutdown cooling system loops on a signal indicating high reactor pressure or low reactor water level. Also, each RHR shutdown cooling loop is individually isolated on a signal indicating high ambient temperature in the RHR loop equipment area.

(8) Isolation of the Reactor Core Isolation Cooling (RCIC) System steam line to the RCIC turbine on a signal indicating high steam flow in the RCIC line, low steam pressure in the RCIC line, high RCIC turbine exhaust pressure, or high ambient temperature in the RCIC equipment area.

(9) Isolation of the Suppression Pool Cleanup (SPCU) System on a signal indicating high drywell pressure or low reactor water level.

(10) Isolation of the Flammability Control System (FCS) on a signal indicating high drywell pressure or low reactor water level.

(11) Isolation of the drywell sump pump discharge lines on a signal indicating high drywell pressure or low reactor water level. Also, each discharge line is individually isolated on a signal indicating high radioactivity in the discharged liquid waste.

(12) Isolation of the LDS fission products monitor drywell sample and return lines on a signal indicating high drywell pressure or low reactor water level.

(13) The LDS provides to the neutron monitoring system a signal indicating a high drywell pressure or low reactor water level.

Separate manual controls in the control room are provided in LDS design for logic reset, MSIV operational control, MSIV partial closure tests, and for manual isolation of primary and secondary containment.

Each MSIV has three pilot solenoid valves, two are used for operational control and the third is used to test the MSIV for partial closure. Each MSIV pilot solenoid valve is controlled separately by LDS as follows:

(1) Two of the three pilot solenoid valves of the MSIV are each provided with four divisional control signals to open the valve. MSIV closure occurs on loss of any two of the four divisional signals.

(2) The third MSIV pilot solenoid valve is provided with one-out-of-two manual control signals to test the MSIV for partial closure. Division I or III manual signal is used to close the outboard MSIV, while Division II or IV manual signal is used to close the inboard MSIV.

Except for MSIVs, the LDS provides three separate divisional isolation signals (Divisions I, II and III) for automatic closure of the primary and secondary containment isolation valves. Each LDS divisional isolation signal initiates closure of the isolation valves that are assigned in the same division.

The LDS design includes the following manual controls for separate isolation of the RCIC System, and closure of the MSIVs and the primary and secondary containment isolation valves:

(1) Four MSIV isolation switches–one per Divisions I, II, III, and IV.

Closure of all the MSIVs requires the actuation of any two of the four divisional MSIV isolation switches.

(2) Three primary and secondary containment isolation switches–one per Divisions I, II and III.

Each isolation switch closes its respective divisional isolation valves in the primary and secondary containment, except for the MSIVs and RCIC.

(3) Two RCIC isolation switches–one per Divisions I and II.

Either isolation switch isolates the steamline to the RCIC turbine and causes turbine trip. Division I switch closes the inboard while Division II switch closes the outboard isolation valves.

Manual reset controls are provided at the divisional level to initialize the logic and for logic reset after isolation has cleared. Separate reset functions are

provided in the LDS logic design for the MSIVs, the RCIC, and the containment isolation.

The LDS design is single-failure-proof and redundant. Also, the LDS design is fail-safe in the event of loss of electrical power to one division of LDS logic.

Each of the four LDS divisional logic and associated sensors is powered from its respective divisional Class 1E power supply. In the LDS, independence is provided between Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment.

The LDS sensors are located in the Reactor Building and Turbine Building; the logic processors are located in the Control Building.

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

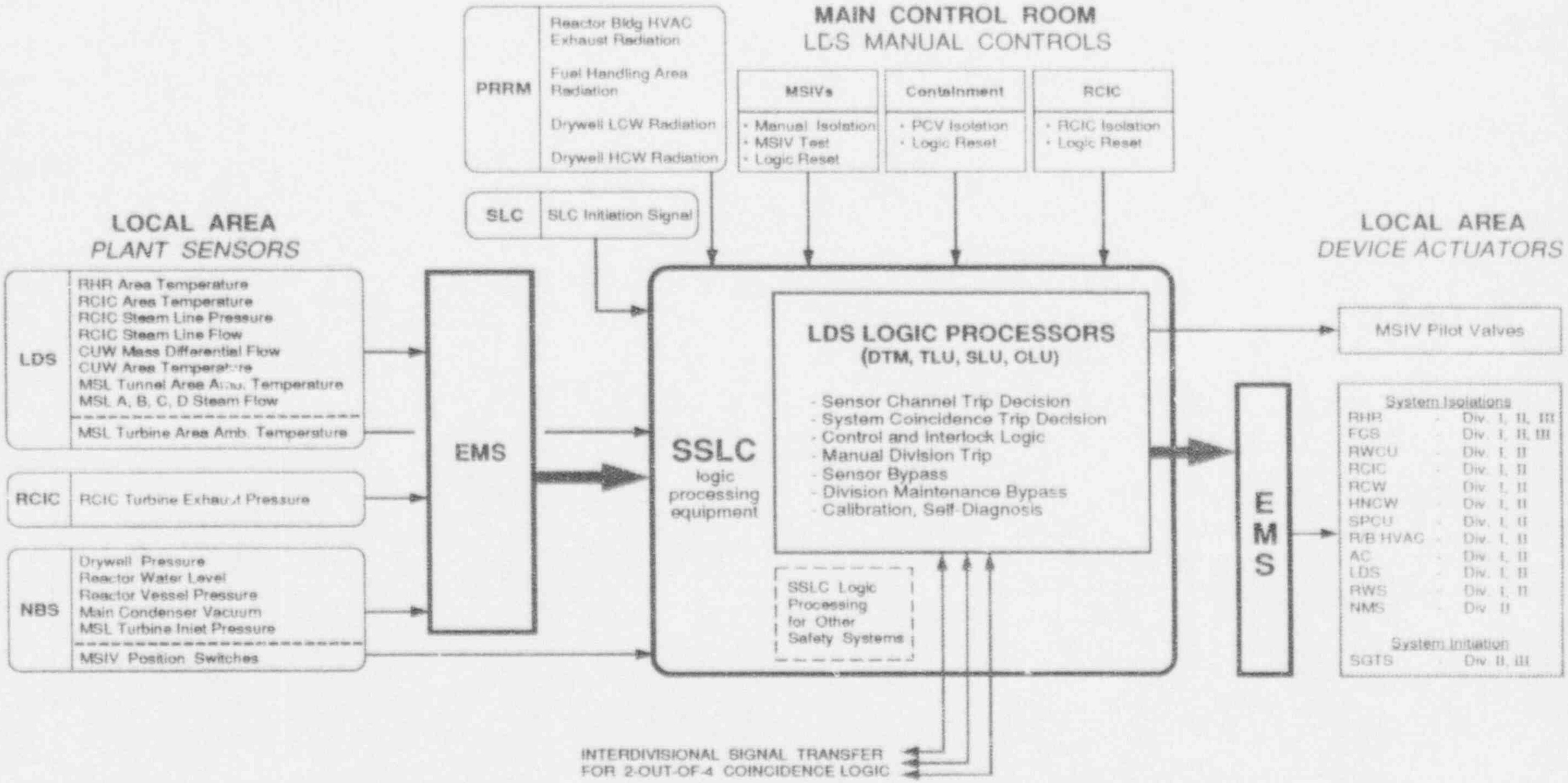
(1) Parameter displays for LDS plant sensors defined on Figure 2.4.3

(2) LDS manual controls as described above

(3) LDS divisional trip status

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.4.3 provides a definition of the inspections, tests, and/or analyses together with associated acceptance criteria, which will be undertaken for the LDS.



Notes:
1. Diagram represents one of four LDS divisions.

Figure 2.4.3 Leak Detection and Isolation System Interface Diagram

Table 2.4.3 Leak Detection & Isolation System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The equipment comprising the LDS is defined in Section 2.4.3.	1. Inspection of the as-built system will be conducted.	1. The as-built LDS conforms with the description in Section 2.4.3.
2. LDS logic uses four independent sensor instrument channels of each process variable described in Section 2.4.3 for its automatic control and isolation functions.	2. 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.	3. Isolation signal is initiated when at least any two out of four channels have tripped.
3. Each MSIV can be subjected to a partial closure test from the main control room.	3. Tests will be conducted by actuating each MSIV test switch.	3. When the test switch is actuated, the MSIV partially closes and then reopens automatically.
4. LDS provides separate manual controls in the main control room for MSIV closure, for isolation of the primary and secondary containment, and for isolation of the RCIC System.	4. Tests will be performed on the as-built system. a. Simultaneously actuate any two of the four MSIV isolation switches to close all MSIVs. b. Actuate each RCIC isolation switch (Div. I and II) c. Actuate each primary and secondary containment isolation switch (Div. I, II and III) to isolate the containment.	4. a. Closure of all the MSIVs occurs only when any two out of four switches are actuated. b. Isolation of the RCIC System occurs when Div. I switch closes the inboard or Div. II switch closes the outboard isolation valves. c. Each divisional primary and secondary containment isolation switch closes only its respective containment isolation valves.
5. Manual reset controls are provided to perform reset functions as described in Section 2.4.3.	5. Tests will be performed using the LDS reset controls.	5. The logic circuitry resets for LDS operation.
6. LDS design is fail-safe in the event of loss of electrical power to one division of LDS logic.	6. Tests will be conducted by disconnecting electrical power to one division of LDS logic at a time.	6. Upon loss of electrical power to one division of LDS logic, the faulted LDS divisional channel trips.

Table 2.4.3 Leak Detection & Isolation System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. In the LDS, independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	7. a. Tests will be performed on the LDS by providing a test signal to only one Class 1E division at a time.	7. a. The test signal exists only in the Class 1E division under test in the LDS.
	b. Inspection of the as-installed Class 1E divisions in the LDS will be performed.	b. In the LDS, physical separation exists between Class 1E divisions. Physical separation exists between these Class 1E divisions and non-Class 1E equipment.
8. Main control room displays and controls provided for the LDS are as defined in Section 2.4.3.	8. Inspections will be performed on the main control room displays and controls for the LDS.	8. Displays and controls exist or can be retrieved in the main control room as defined in Section 2.4.3.

SYSTEM: 2.4.3 Leak Detection System

PUNCH LIST ITEM: A. Add boilerplate for Control Room configuration (minus alarms)

GE DISPOSITION: The revised Section 2.4.3 now includes a standard I&C configuration entry.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.4.3 Leak Detection System

PUNCH LIST ITEM: B. Develop logic test to demonstrate fail-safe provision on loss of divisional power

GE DISPOSITION: Complete. A fail-safe test of the LDS is now included as Item 6 in the revised Section 2.4.3. The tests involve disconnecting electrical power to one division of LDS logic at a time.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.4.3 Leak Detection System

PUNCH LIST ITEM: C. Add divisional power supply boilerplate

GE DISPOSITION: This material has been added as Item 7 in the revised Section 2.4.3.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.4.3 Leak Detection System

PUNCH LIST ITEM: D. Add physical separation boilerplate. (Check electrical write-up side discussions)

GE DISPOSITION: GE believes physical separation is covered in the new ITAAC table Item 7 noted in the response to punch list Item C. No other action is planned.

2.6 Reactor Auxiliary

2.6.1 Reactor Water Cleanup System

Design Description

The Reactor Water Cleanup (CUW) System as shown in Figure 2.6.1 removes particulate and dissolved impurities from the reactor coolant by circulating a portion of the reactor coolant through a filter-demineralizer.

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

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

The CUW System is classified as a non-safety-related system with the exception of the primary containment isolation function. The major portion of the system is located outside of the primary containment in the reactor building.

CUW System piping and components from the RPV out to and including the outboard isolation valves are part of the reactor coolant pressure boundary and are classified as Seismic Category I. The remainder of the piping system is classified as non-Seismic Category I. Figure 2.6.1 shows the ASME Code class for the CUW system components

The inboard containment isolation valve is powered from Class 1E Division II, and the outboard containment isolation valves are powered from Class 1E Division I.

The main control room has displays for the instrumentation shown in Figure 2.6.1 and control and open/close status for the containment isolation valves.

The safety-related electrical equipment that provides containment isolation and is located in the containment and reactor building is qualified for a harsh environment.

The motor-operated valves (MOVs) shown in Figure 2.6.1 have active safety-related functions and perform these functions under differential pressure, fluid flow and temperature conditions.

The CUW suction line is provided with a flow restrictor which provides flow restricting and flow monitoring functions. Maximum throat diameter is 135 mm.

The reactor vessel bottom head drain line is connected to the main CUW suction piping by a tee. The center line of the tee connection is at an elevation of at least 460 mm above the center line of the variable leg nozzle of the RPV wide range water level instrument.

Inspections, Tests, Analyses and Acceptance Criteria

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

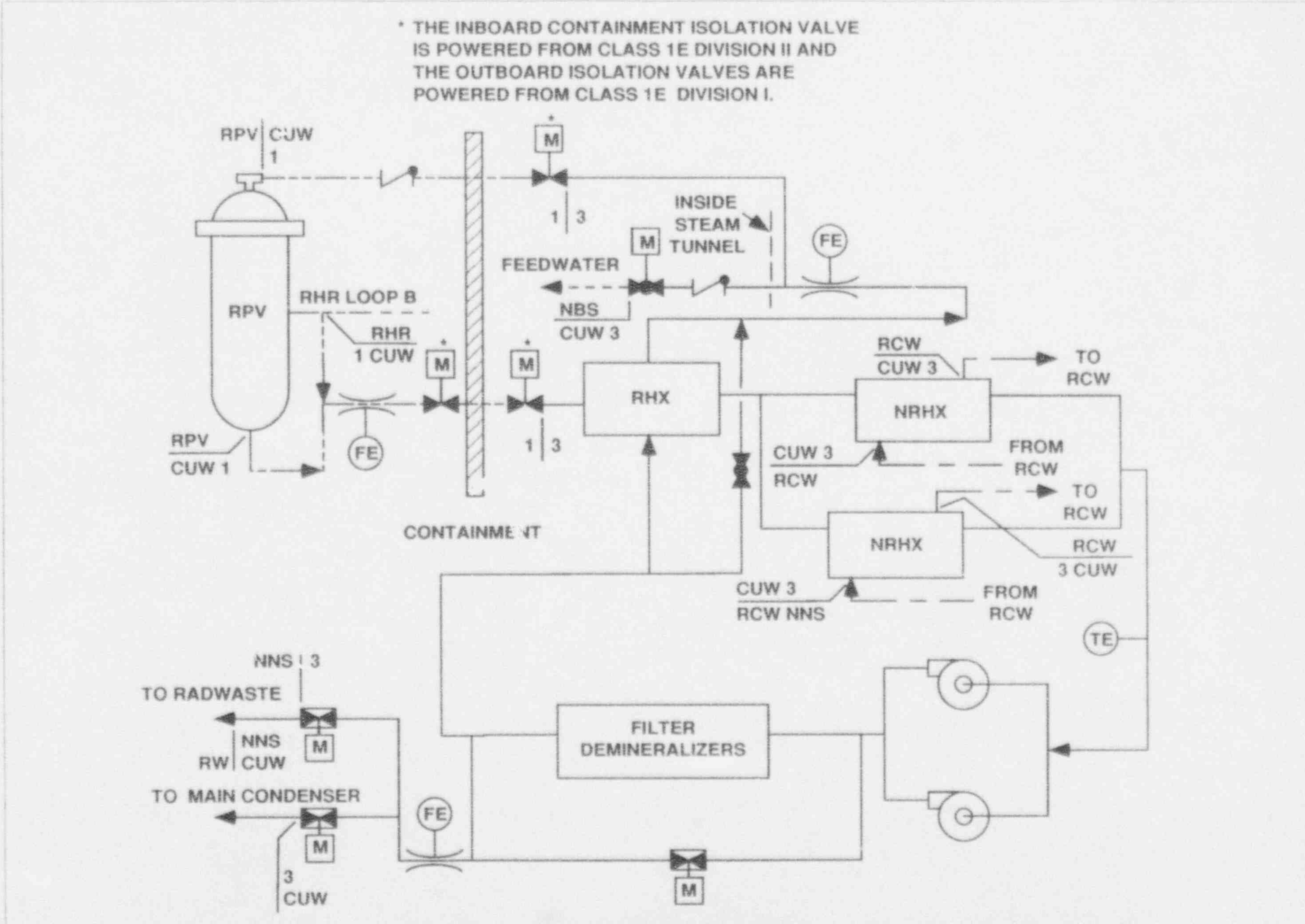


Figure 2.6.1 Reactor Water Cleanup System Schematic

Table 2.6.1 Reactor Water Cleanup System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitments	Inspections, Tests, Analyses	Acceptance Criteria
1. A basic configuration for the CUW System is as shown in Figure 2.6.1.	1. Inspection of the as-built system will be conducted.	1. The as-built CUW System conforms with the basic configuration shown in Figure 2.6.1.
2. The ASME Code components of the CUW 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 CUW System required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME Code components of the CUW System conform with the requirements in the ASME Code, Section III.
3. In the CUW System, independence is provided between Class 1E Divisions and non-1E equipment	3a. Tests will be performed on the CUW System 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 CUW System.
	3b. Inspection of the as-installed Class 1E Divisions in the CUW System will be performed.	3b. Physical separation exists between Class 1E Divisions in the CUW System. Physical separation exists between Class 1E Divisions and non-Class 1E equipment in the CUW System.
4. Main control room displays and controls provided for CUW System are as defined in Section 2.6.1.	4. Inspections will be performed on the main control room displays and controls for the CUW System.	4. Displays and controls exist or can be retrieved in main control room as defined in Section 2.6.1.
5. MOVs designated in Section 2.6.1 as having an active safety- related function close under differential pressure and fluid flow and temperature conditions.	5. Closing tests of installed valves will be conducted under pre-op differential pressure, fluid flow, and temperature conditions.	5. Each MOV closes. The following valves close in the following time limits upon receipt of the actuation signal: Valve — Time (sec) Suction line inboard CIV — ≤ 30 Suction line outboard CIV — ≤ 30
6. Maximum throat diameter of the CUW suction line flow restrictor is 135 mm.	6. Inspection will be conducted on the CUW suction line flow restrictor throat diameter.	6. Maximum throat diameter of the CUW suction line flow restrictor is 135 mm.

Table 2.6.1 Reactor Water Cleanup System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitments	Inspections, Tests, Analyses	Acceptance Criteria
7. The bottom head drain line is connected to the main CUW suction piping by a tee. The center line of the tee connection is at an elevation of at least 460 mm above the center line of the variable leg nozzle of the RPV wide-range water level instrument.	7. Inspections of the as-built CUW and RPV will be performed.	7. The center line of the vessel bottom head drain line tee connection is at least 460 mm above the center line of the variable leg nozzle of the RPV wide-range water level instrument.

SYSTEM: 2.6.1 Reactor Water Cleanup System

PUNCH LIST ITEM: A. GE to confirm 460 MM is at centerline.

GE DISPOSITION: Confirmed and clarified in the CUW entry.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.6.1 Reactor Water Cleanup System

PUNCH LIST ITEM: B. GE to modify words (text also) to reflect flow restrictor function.

GE DISPOSITION: Complete. See revised version of 2.6.1.

SYSTEM: 2.6.1 Reactor Water Cleanup System

PUNCH LIST ITEM: C. GE to check standardization words for classification of piping.

GE DISPOSITION: GE has attempted to standardize the words for ASME Code Class of piping using the following as an example for other systems.

Figure 2.6.1 shows the ASME Code Class for the CUW system piping and components.

2.10 Power Cycle

2.10.1 Turbine Main Steam System

Design Description

The Turbine Main Steam (MS) System, as shown in Figure 2.10.1, supplies steam generated in the reactor to the turbine, steam auxiliaries and steam turbine bypass valves. The MS System ranges between, but does not include, the seismic interface restraint to the turbine stop valves, turbine bypass valves and branch lines up to and including the steam auxiliary valve(s).

The MS System:

(1) Accommodates operational stresses such as internal pressure and dynamic loads without failures.

(2) Provides a seismically analyzed fission product leakage path to the main condenser.

(3) Has suitable access to permit in-service testing and inspections.

(4) Closes the steam auxiliary (SA) valve(s) on a main steam isloation valve (MSIV) isolation signal. These valves fail closed on loss of electrical power to the valve actuating solenoid or on loss of pneumatic pressure.

The MS System main steam piping consists of four lines from the seismic interface restraint to the main turbine stop valves. The header arrangement upstream of the turbine stop valves allows the valves to be tested on-line and also supplies steam to the power cycle auxiliaries.

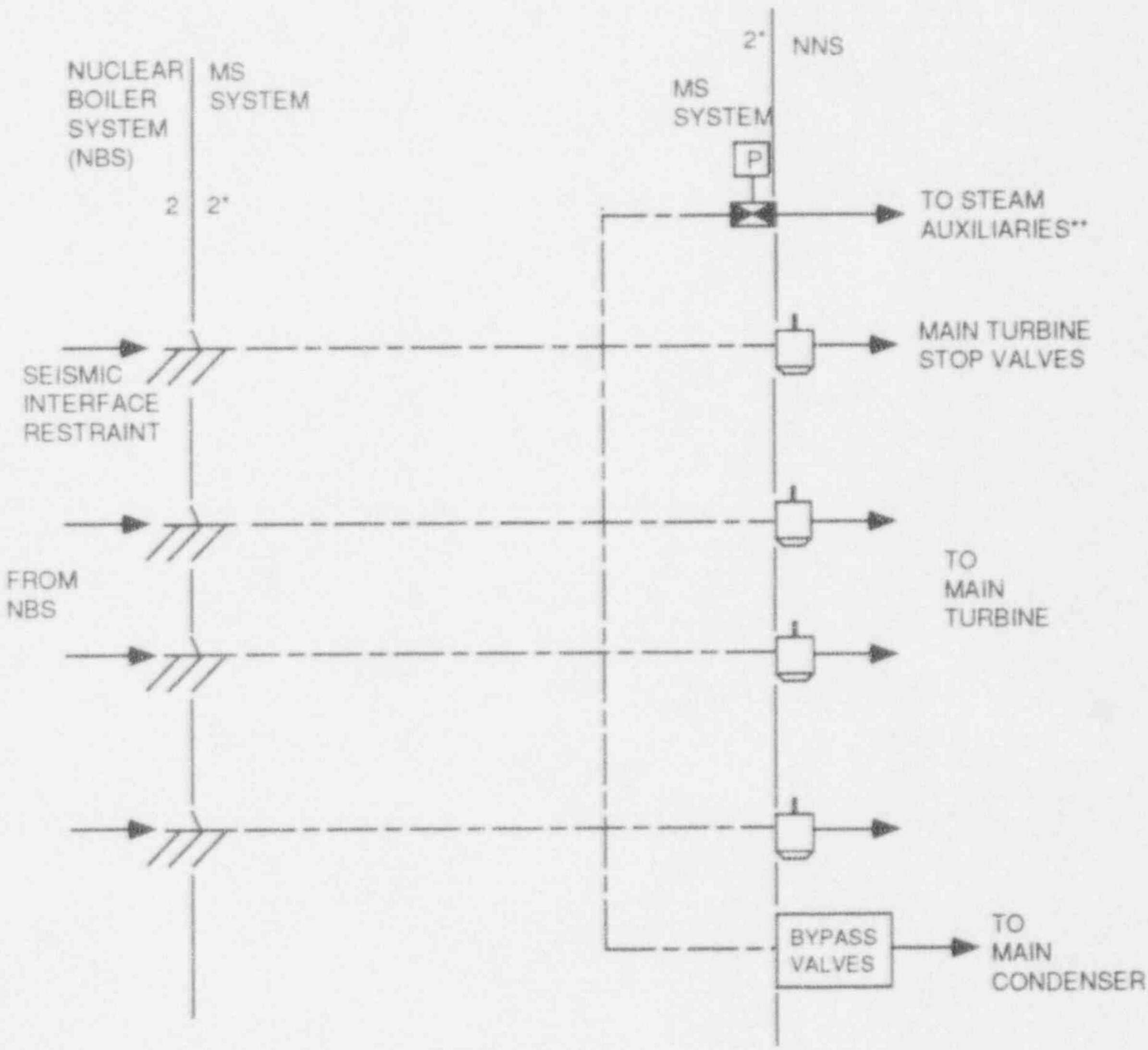
The MS System is analyzed, fabricated and examined to ASME Code Class 2 requirements, classified as non-Seismic Category I, and subject to pertinent QA requirements of Appendix B, 10CFR Part 50. Inservice inspection shall be performed in accordance with ASME Section XI requirements for Code Class 2 piping. ASME authorized nuclear inspector and ASME Code stamping is not required.

MS piping from the seismic interface restraint to the main stop, main turbine bypass and the steam auxiliary valve(s) is analyzed to demonstrate structural integrity under safe shutdown earthquake (SSE) loading conditions.

The MS System is located in the steam tunnel and Turbine Building.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.10.1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the MS System.



* AS MODIFIED PER DESIGN DESCRIPTION.
** MULTIPLE LINES MAY BE USED. ISOLATION PROVISIONS ARE REQUIRED FOR EACH STEAM AUXILIARY LINE.

Figure 2.10.1 Turbine Main Steam System

Table 2.10.1 Turbine Main Steam System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the MS System is as shown on Figure 2.10.1.	1. Inspections of the as-built system will be conducted.	1. The as-built MS System conforms with the basic configuration shown in Figure 2.10.1.
2. The ASME Code components of the MS 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 MS System required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME Code components of the MS System conform with the requirements in the ASME Code, Section III.
3. Upon receipt of an MSIV closure signal, the SA valve(s) close(s).	3. Using simulated MSIV closure signals, tests will be performed on the SA valve(s).	3. The SA valve(s) close(s) following receipt of a simulated MSIV closure signal.
4. The SA valve(s) fail(s) closed on loss of electrical power to the valve actuating solenoid or on loss of pneumatic pressure. The pneumatically operated SA valve(s) close(s) when either electrical power to the valve actuating solenoid is lost or pneumatic pressure to the valve(s) is lost.	4. Test will be performed on SA valve(s).	4. The SA valve(s) close(s) on loss of electrical power to the valve actuating solenoid or on lose of pneumatic pressure.
5. MS piping from the seismic interface restraint to the main stop, main turbine bypass and the SA valve(s) is analyzed to demonstrate structural integrity under SSE loading conditions.	5. A seismic analysis of the as-built MS piping will be performed.	5. An analysis report exists which concludes that the as-built MS piping can withstand a safe shutdown earthquake without loss of structural integrity.

SYSTEM: 2.10.1 Turbine Main Steam

PUNCH LIST ITEM: A. Update SAR to show steam auxiliaries valve now in system.

GE DISPOSITION: The SAR has been modified to show these auxiliaries.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.10.1 Turbine Main Steam

PUNCH LIST ITEM: B. Add boilerplate on configuration.

GE DISPOSITION: A configuration ITAAC has been added to 2.10.1.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.10.1 Turbine Main Steam

PUNCH LIST ITEM: C. Add to design description.

GE DISPOSITION: This punch list item was completed at the time of the January 1993 GE/NRC review of this system. A paragraph discussing the design bases for the main steam lines in the Turbine Building is included in the design description and addressed in the ITAAC for entry 2.10.1.

SYSTEM: 2.10.1 Turbine Main Steam

PUNCH LIST ITEM: D. Add hydro boilerplate.

GE DISPOSITION: Hydrostatic test entry added.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.10.1 Turbine Main Steam

PUNCH LIST ITEM: E. Include ITAAC and descriptions for auxiliary steam valve(s) including logic to close on MSIV isolation signal.

GE DISPOSITION: Complete.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.10.1 Turbine Main Steam

PUNCH LIST ITEM: F. Revised SSAR - Description of Main Steam Line. (See attached draft on next page.)

GE DISPOSITION: Complete.

2.10.21 Main Condenser

Design Description

The main condenser condenses and deaerates the exhaust steam from the main turbine and provides a heat sink for the Turbine Bypass (TB) System. The main condenser is also a collection point for other steam cycle drains and vents.

The main condenser hotwell provides a holdup volume for main steam isolation valve (MSIV) fission product leakage.

The main condenser is classified as non-safety-related and non-seismic Category I. The supports and anchors for the main condenser are designed to withstand a safe shutdown earthquake (SSE).

The main condenser is located in the Turbine Building (T/B).

The main condenser tubes are made from corrosion resistant material. The main condenser operates at a vacuum; consequently, leakage is into the shell side of the main condenser. Circulating water leakage from the tubes to the condenser is detected by measuring the conductivity of sample water extracted beneath the tube bundles. In addition, a conductivity monitor is located at the discharge of the condensate pumps and alarms are provided in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.10.21 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the main condenser.

Table 2.10.21 Main Condenser

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The supports and anchors for the main condenser are designed to withstand a safe shutdown earthquake.	1. An analysis of the ability of the as-built condenser supports and anchors to withstand a safe shutdown earthquake will be performed.	1. An analysis report exists which concludes that the as-built main condenser supports and anchors are able to withstand a safe shutdown earthquake.
2. A conductivity monitor is located at the discharge of the condensate pumps.	2. The as-built system will be inspected.	2. A conductivity monitor exists at the discharge of the condensate pumps.
3. Main control room alarms provided for the main condenser are as defined in Section 2.10.21.	3. Inspections will be performed on the main control room alarms for the main condenser.	3. Alarms exist in the main control room as defined in Section 2.10.21.

2.11.3 Reactor Building Cooling Water System

Design Description

The Reactor Building Cooling Water (RCW) System distributes cooling water through three physically separated and electrically independent divisions. The system removes heat from plant auxiliaries and transfers it to the Ultimate Heat Sink (UHS) via the Reactor Service Water (RSW) System. The RCW System removes heat from emergency core cooling equipment including the emergency diesel generators (DGs) during a safe reactor shutdown cooling function. RCW System configurations are shown in Figures 2.11.3a, 2.11.3b, and 2.11.3c. All components cooled by the RCW System are parts of other systems and are not part of the RCW System. Each RCW division includes two pumps which circulate cooling water through the equipment cooled by the RCW System and through three heat exchangers which transfer the RCW heat to the UHS via the RSW System.

The RCW System performs safe reactor shutdown cooling function following a loss-of-coolant accident/loss-of-offsite power (LOCA/LOOP), assuming a single active failure in any mechanical or electrical division or RCW support system. In case of a failure which disables any one of the three RCW divisions, the other two divisions perform safe reactor shutdown cooling.

Tables 2.11.3a, 2.11.3b, and 2.11.3c show which equipment receives RCW flow during various plant operating and emergency conditions. The tables also indicate how many heat exchangers are in service under each condition.

The RCW System responses to a LOCA signal are the following:

(1) Starts any standby RCW pumps,

(2) Opens any closed standby RCW heat exchanger outlet valves,

(3) Opens all Residual Heat Removal (RHR) System heat exchanger cooling water outlet valves,

(4) Closes all RCW containment isolation valves,

(5) Closes valves to non-safety-related components (to Reactor Water Cleanup System (CUW) and Hot Water Heating (HWH) System heat exchangers and reactor internal pump (RIP) MG sets),

(6) Opens the RCW water temperature pneumatic control valves (located downstream of RCW heat exchangers) and closes the RCW heat exchanger bypass valves.

Except for Instrument Air (IA) System, Service Air (SA) System, Control Rod Drive (CRD) System pump oil coolers, and CUW System pump coolers, these valves close during a LOCA. Safety-related valves separate the safety-related portions of the RCW System from the non-safety-related portions of the system. The isolation valves to the non-safety-related RCW System are automatically or remote-manually operated, and their positions are indicated in the main control room.

Component design parameters are:

	Division A/B	Division C
Discharge flow rate (lpm/pump)	$\geq$ 21,700	$\geq$ 18,200
Heat exchanger design basis heat removal capacities are each:	$\geq 11.4 \times 10^6$ kcal/hr	$\geq 10.6 \times 10^6$ kcal/hr

These values include a 20% margin above the minimum required for design basis accident conditions. Consequently, plant operation is acceptable with heat exchanger capacities of greater than or equal to 80% of these values.

Figures 2.11.3a, 2.11.3b, and 2.11.3c show the ASME Code Class for the RCW System piping and components. The safety-related portions of the RCW divisions are classified as Seismic Category I.

The RCW pumps and heat exchangers are located in the lower floors of the Control Building. The equipment cooled by the RCW divisions are located in the Control Building, Reactor Building, Turbine Building, and Radwaste Building, (Figures 2.11.3a, 2.11.3b, and 2.11.3c).

Each of the three RCW divisions is powered from its respective Class 1E division as shown in Figures 2.11.3a, 2.11.3b, and 2.11.3c. The safety-related portion of each mechanical division of the RCW System (divisions A, B, C) is physically separated from the other divisions.

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

(1) Parameter displays for instruments shown on Figures 2.11.3a, 2.11.3b, and 2.11.3c.

(2) Controls and status displays for the RCW active safety-related components shown on Figures 2.11.3a, 2.11.3b, and 2.11.3c.

The RCW System components with displays and control interfaces with the Remote Shutdown System (RSS) are identified in Figures 2.11.3a and 2.11.3b.

The safety-related electrical equipment shown on Figures 2.11.3a, 2.11.3b, and 2.11.3c, located in the Reactor Building, is qualified for a harsh environment.

The RCW motor-operated valves shown on Figures 2.11.3a, 2.11.3b, and 2.11.3c all have safety-related functions and perform these functions under differential pressure, fluid flow, and temperature conditions.

A separate surge tank of at least $16m^3$ is provided for each RCW division. Each surge tank is shared with the corresponding division of the HVAC Emergency Cooling Water (HECW) System. Makeup water is provided for the surge tank by the Makeup Water (Purified) (MUWP) System by an automatic or main control room signal. Low water level signals in the surge tanks do the following (in order of decreasing level):

(1) (low) opens the MUWP makeup water valve,

(2) (low-low) closes the pneumatic and motor-operated valves which stop flow to the non-safety-related components.

The Suppression Pool Cleanup (SPCU) System provides a backup surge tank water supply.

The pneumatic-operated valves shown in Figures 2.11.3a, 2.11.3b, and 2.11.3c fail as follows in the event that either electric power to the valve-actuating solenoid is lost or pneumatic pressure to the valve is lost: RCW makeup valves to the MUWP fail open, RCW water temperature control valves fail open, RCW heat exchanger bypass valves fail closed, and the safety-related/non-safety-related separation valve fails closed.

Inspections, Tests, Analyses and Acceptance Criteria

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

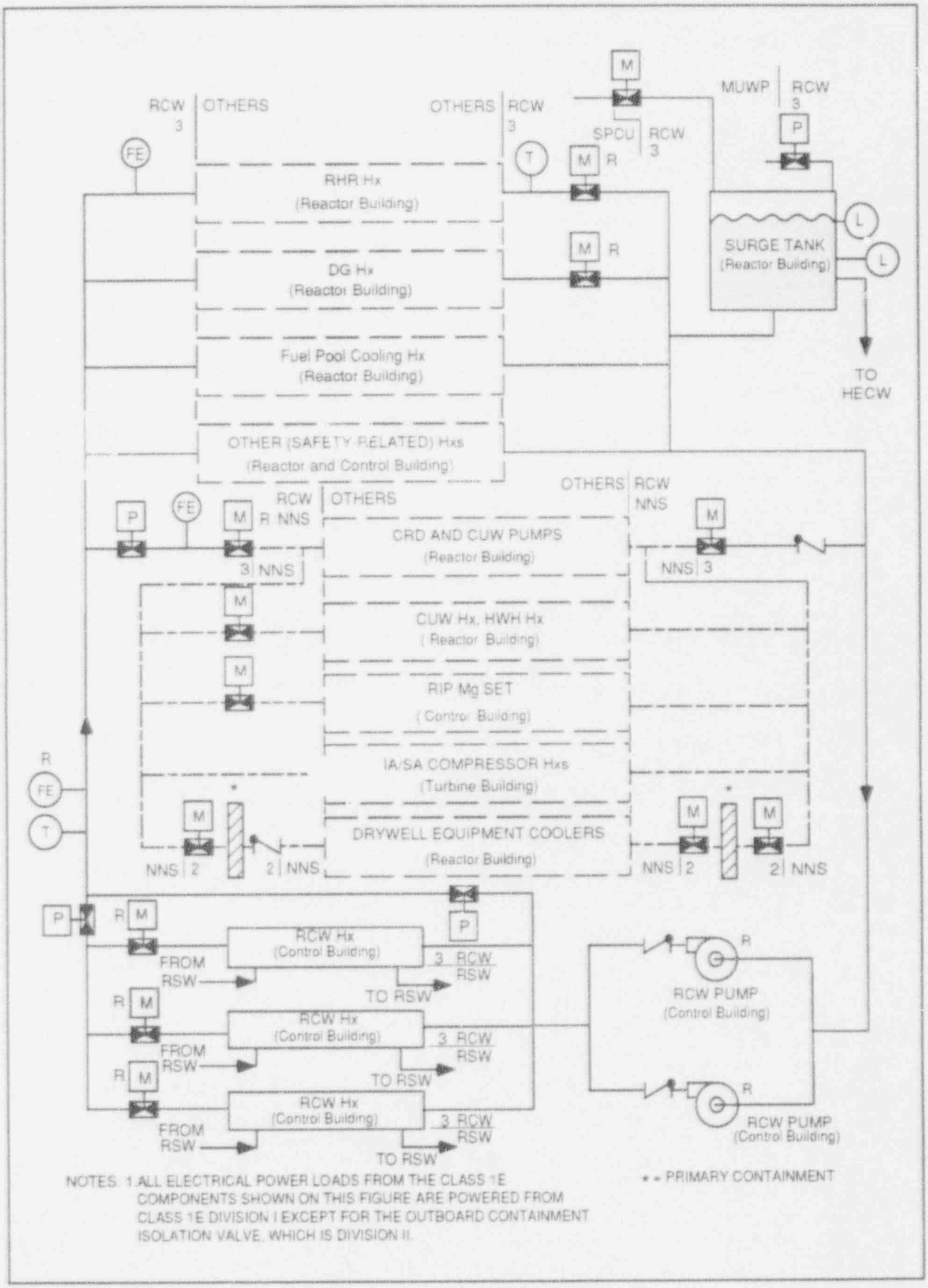


Figure 2.11.3a RCW Division - A

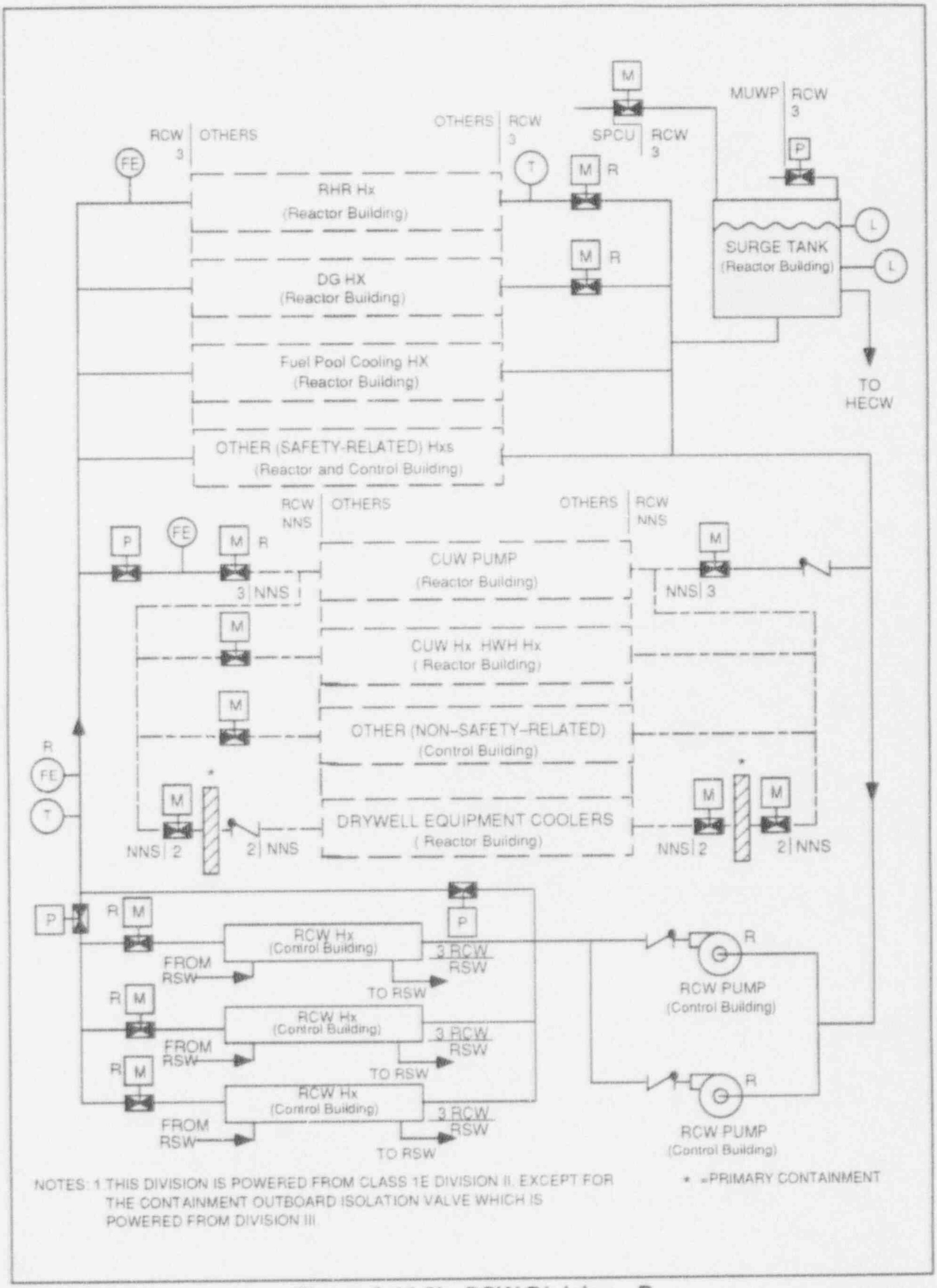
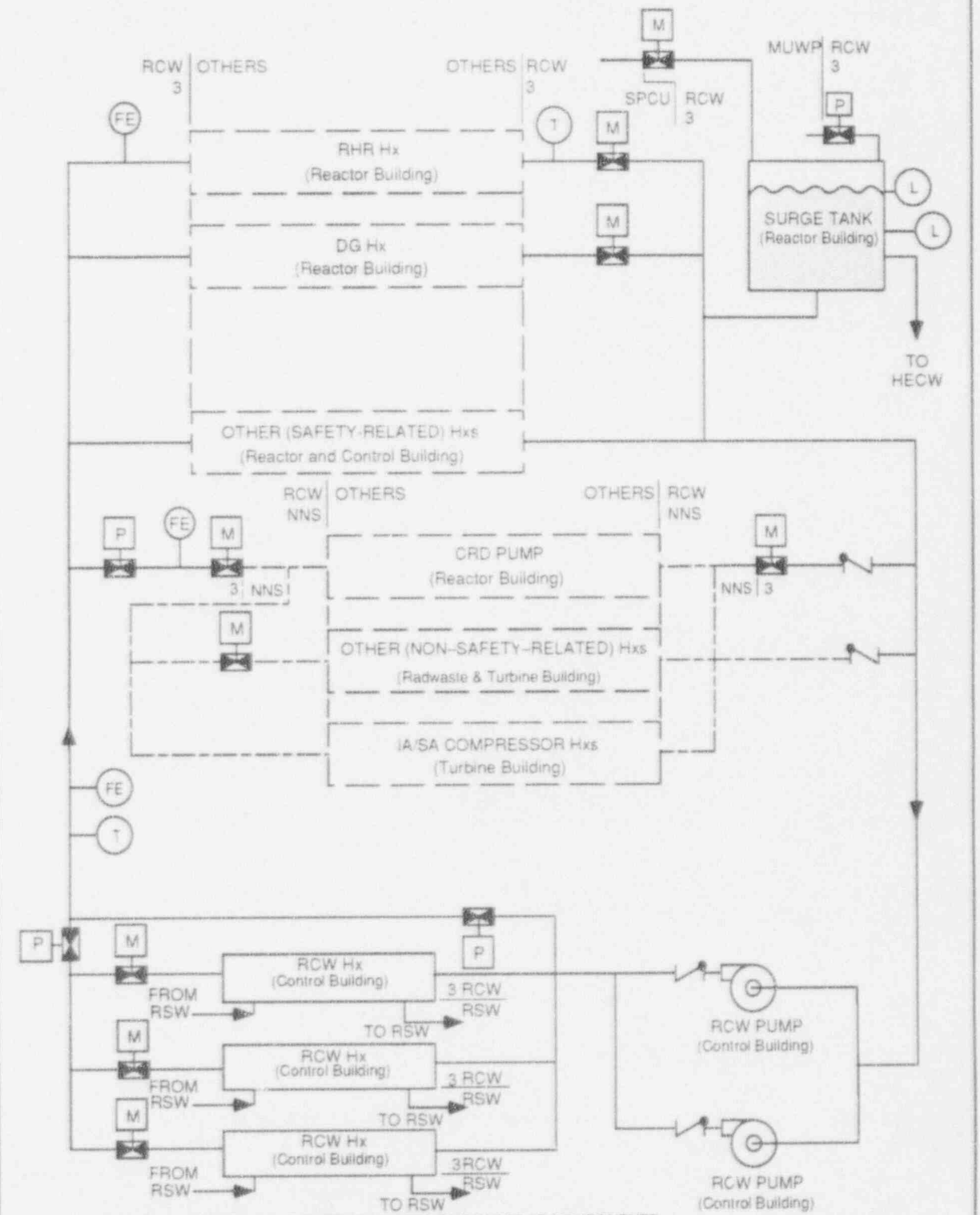


Figure 2.11.3b RCW Division - B



NOTES: 1. ALL ELECTRICAL POWER LOADS FOR THE CLASS 1E COMPONENTS SHOWN ON THIS FIGURE ARE POWERED FROM CLASS 1E DIVISION III.

Figure 2.11.3c RCW Division - C

Table 2.11.3a: Reactor Building Cooling Water Cooling Loads Division A

Operating Mode/ Components*	Normal Operating Conditions	Shutdown	Hot Standby (loss of AC Power)	Emergency (LOCA)
RCW/RSW Heat Exchangers In Service	2	3	3	3
SAFETY-RELATED				
Emergency Diesel Generator A	-	-	X	X
RHR Heat Exchanger A	-	X	X	X
FPC Heat Exchanger A	X	X	X	X
Others (safety-related)**	X	X	X	X
NON-SAFETY-RELATED				
CUW Heat Exchanger	X	X	X	-
Inside Drywell	X	X	X	-
Others (non-safety-related)	X	X	X	X

Notes:

* = Some of these cooling loads are serviced by only one or two RCW divisions. These components may be reassigned to other RCW divisions if redundancy and divisional alignment of supported and supporting systems is maintained and the design basis cooling capacity of the RCW divisions is assured.

** = HECW refrigerator, room coolers (FPC, RHR, RCIC, SGTS, CAMS) RHR motor bearing and seal coolers, and CAMS cooler.

x = Equipment receives RCW in this mode.

\- = Equipment does not receive RCW in this mode.

Table 2.11.3b: Reactor Building Cooling Water Cooling Loads Division B

Operating Mode/ Components*	Normal Operating Conditions	Shutdown	Hot Standby (loss of AC Power)	Emergency (LOCA)
RCW/RSW Heat Exchangers In Service	2	3	3	3
SAFETY-RELATED				
Emergency Diesel Generator B	-	-	X	X
RHR Heat Exchanger B	-	X	X	X
FPC Heat Exchanger B	X	X	X	X
Others (safety-related)**	X	X	X	X
NON-SAFETY-RELATED				
RWCU Heat Exchanger	X	X	X	-
Inside Drywell	X	X	X	-
Others (non-safety-related)	X	X	X	X

Notes:

* = Some of these cooling loads are serviced by only one or two RCW divisions. These components may be reassigned to other RCW divisions if redundancy and divisional alignment of supported and supporting systems is maintained and the design basis cooling capacity of the RCW divisions is assured.

** = HECW refrigerators, room coolers (FPC, RHR, HPCF, SGTS, FCS, CAMS) RHR motor bearing and seal coolers, and CAMS cooler.

x = Equipment receives RCW in this mode.

\- = Equipment does not receive RCW in this mode.

Table 2.11.3c: Reactor Building Cooling Water Cooling Loads Division C

Operating Mode/ Components*	Normal Operating Conditions	Shutdown	Hot Standby (loss of AC Power)	Emergency (LOCA)
RCW/RSW Heat Exchangers In Service	2	3	3	3
SAFETY-RELATED				
Emergency Diesel Generator B	-	-	X	X
RHR Heat Exchanger B	-	X	X	X
Others (safety-related)**	X	X	X	X
NON-SAFETY-RELATED				
Others (Non-safety-related)	X	X	X	X

Notes:

* = Some of these cooling loads are serviced by only one or two RCW divisions. These components may be reassigned to other RCW divisions if redundancy and divisional alignment of supported and supporting systems is maintained and the design basis cooling capacity of the RCW divisions is assured.

** = HECW refrigerators; FCS room coolers, room coolers, motor bearing coolers, and mechanical seal coolers for RHR and HPCF.

x = Equipment receives RCW in this mode.

\- = Equipment does not receive RCW in this mode.

Table 2.11.3d Reactor Building Cooling Water (RCW) System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the RCW system is as shown on Figures 2.11.3a, 2.11.3b and 2.11.3c.	1. Inspections of the as-built system will be conducted.	1. The as-built RCW system conforms with the basic configuration shown in Figures 2.11.3a, 2.11.3b and 2.11.3c.
2. The ASME Code components of the RCW 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 RCW system required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME code components of the RCW system conform with the requirements in the ASME Code, Section III.
3. The RCW System responses to a LOCA signal are as specified in Section 2.11.3.	3. Using simulated LOCA signals, tests will be performed for the RCW System.	3. Upon receipt of simulated LOCA signals, the responses of the RCW System are as specified in Section 2.11.3.
4. The RCW pump flow capacities and the RCW heat exchanger heat removal capacities are as specified in Section 2.11.3.	4. An analysis of the as-built RCW System will be performed. Tests will be performed of the flow capacities of the installed RCW pumps. Inspections and analyses will be performed to estimate the heat removal capacities of the RCW heat exchangers. Inspections and analyses will be performed to estimate the heat removal requirements of the as-built components which are cooled by the RCW System during LOCA conditions.	4. The estimated heat removal capacities of the as-built RCW System divisions exceed the estimated heat removal requirements of the components cooled by the RCW System divisions during LOCA conditions.
5. In the RCW System, independence is provided between Class 1E Divisions, and between Class 1E Divisions and non-Class 1E equipment.	5a. Tests will be performed on the RCW System by providing a test signal in only one Class 1E division at a time. 5b. Inspections of the as-installed Class 1E Divisions in the RCW System will be performed.	5a. The test signal exists only in the Class 1E division under test in the RCW System. 5b. Physical separation exists between Class 1E Divisions in the RCW System. Physical separation exists between Class 1E Divisions and non-Class 1E equipment.
6. The safety-related portion of each mechanical division of the RCW system (Divisions A, B,C) is physically separated from the other divisions.	6. Inspections of the as-built RCW System will be performed.	6. The safety-related portions of each mechanical division of the RCW System is physically separated from the safety-related portions of the other mechanical divisions of the RCW System.

Table 2.11.3d Reactor Building Cooling Water (RCW) 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 RCW system are as defined in Section 2.11.3.	7. Inspections will be performed on the main control room displays and controls for the RCW system.	7. Displays and controls exist or can be retrieved in the main control room as defined in Section 2.11.3.
8. Remote Shutdown System (RSS) displays and controls provided for the RCW system are as defined in Section 2.11.3.	8. Inspections will be performed on the RSS displays and controls for the RCW system.	8. Displays and controls exist on the RSS as defined in Section 2.11.3.
9. Motor-operated valves (MOV) designated in Section 2.11.3 as having an active safety-related function will open and/or close under differential pressures, fluid flow, and temperature conditions.	9. Opening and/or closing tests of installed valves will be conducted under pre-operational differential pressure, fluid flow, and temperature conditions.	9. Each MOV opens and/or closes upon receipt of an actuation signal.
10. The pneumatic-operated valves shown in Figures 2.11.3a, 2.11.3b, and 2.11.3c fail as follows in the event that either electric power to the valve actuating solenoid is lost or pneumatic pressure to the valve is lost: MUWP makeup valves fail open, RCW water temperature control valves fail open, RCW heat exchanger bypass valves fail closed, and the safety-related/non-safety-related separation valve fails closed.	10. Tests will be performed on the as-built valves by initiating loss of pneumatic pressure and power to the actuating solenoids.	10. The pneumatic actuated valves listed below fail as desired when either electric power to the valve actuating solenoid is lost or pneumatic pressure to the valve is lost: MUWP makeup water valves fail open, RCW water temperature control valves fail open, RCW heat exchanger bypass valves fail closed, and the safety-related/non-safety-related separation valve fails closed.
11. A surge tank with a capacity of greater than or equal to $16m^3$ is provided for each RCW division.	11. Inspection and a volume calculation using as-built dimensions will be performed.	11. The capacity of the surge tanks is greater than or equal to $16\ m^3$.
12. A low surge tank water level signal closes the pneumatic and motor-operated valves which stop flow to the non-safety-related components.	12. Tests will be performed on the as-built equipment.	12. The pneumatic and motor-operated valves which stop flow to the non-safety-related components close upon receipt of a low surge tank water level signal.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.11.3 Reactor Building Cooling Water

PUNCH LIST ITEM: A. NRC to verify acceptability of continuing to provide flow to non-safety component during a LOCA. (Low standpipe isolates all non-safety loads.)

GE DISPOSITION: Not a GE action item; NRC input required.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.11.3 Reactor Building Cooling Water

PUNCH LIST ITEM: B. GE to verify actuation on LOCA and make text changes to Design Description and SSAR (Valve stays in).

GE DISPOSITION: This issue refers to the RCW heat exchanger bypass valve (F009A, B and C) which is descriped in the RCW ITAAC design description figures and ITAACs.

This valve is shown in the SSAR on Figure 9.2-1, Sheet 1 marked by an arrow and its operation during LOCA is shown on Figure 3.7-7, Sheet 18. The electro-pneumatic converter reduces cooling water temperature by increasing the opening of F006 and closing F009. The LOCA signal fully opens F006 and fully closes F009.

A revision of the P&ID, IBD and the SSAR text is being prepared in which some valves (including F009A, B and C) are being renumbered.

SYSTEM: 2.11.3 Reactor Building Cooling Water

PUNCH LIST ITEM: C. GE to determine design basis heat removal capacities and so specify. (Margins discussion to be in the SSAR.)

GE DISPOSITION: The design basis heat removal capacities of the RCW heat exchangers are provided in Table 9.2-4D. The margin is discussed in Section 9.2.11.4, third paragraph. The overall heat removal capacity is addressed in the Tier 1 entry.

SYSTEM: 2.11.3 Reactor Building Cooling Water

PUNCH LIST ITEM: D. GE include radiation monitor in process radiation monitor system.

GE DISPOSITION: The RCW radiation monitor is discussed in Section 11.5.2.2.6 and Table 11.5-3 of the SSAR. This confirms that the process radiation monitoring system includes the necessary RCW radiation monitor, and GE believes no other action is required.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.11.3 Reactor Building Cooling Water

PUNCH LIST ITEM: E. GE to verify loads on each division.

GE DISPOSITION: The loads in each division are provided in Figures 9.2-4A, -4B, and -4C in the SSAR.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.11.3 Reactor Building Cooling Water

PUNCH LIST ITEM: G. Same as Nuclear Boiler System ITAAC 15.

GE DISPOSITION: This issue refers to the failure mode of pneumatic valves. This material has been added to the 2.11.3 design description and addressed in an ITAAC aimed at confirming failure mode.

2.11.9 Reactor Service Water System

Design Description

The Reactor Service Water (RSW) System removes heat from the Reactor Building Cooling Water (RCW) System and rejects this heat to the Ultimate Heat Sink (UHS). The portions of the RSW System that are in the Control Building are within the Certified Design. Those portions of the RSW System that are outside the Control Building are not in the Certified Design.

The RSW System provides cooling water flow to either two or three of the RCW System heat exchangers in each division. On a loss-of-coolant accident (LOCA) signal, any closed valves for standby heat exchangers are automatically opened and cooling flow is provided to all these heat exchangers in each division.

For each division of RSW, the heat exchanger inlet and outlet valves close upon receipt of a signal indicating Control Building flooding in that division.

The RSW System is classified as Seismic Category I and ASME Code Section III, Class 3 and consists of three separate safety-related divisions.

Each of the three RSW divisions is powered by its respective Class 1E Division. Each mechnical divison of the RCW system (Divisions A, B, C) is physically separated from the other divisions.

The RSW System has the following main control room (MCR) displays and controls: Control and status displays for the valves shown on Figure 2.11.9. The RSW System components with status displays and control interfaces with the Remote Shutdown System (RSS) are identified in Figure 2.11.9.

The motor-operated valves (MOVs) shown on Figure 2.11.9 all have active safety-related functions and open and close under differential pressure, fluid flow, and temperature conditions.

Interface Requirements

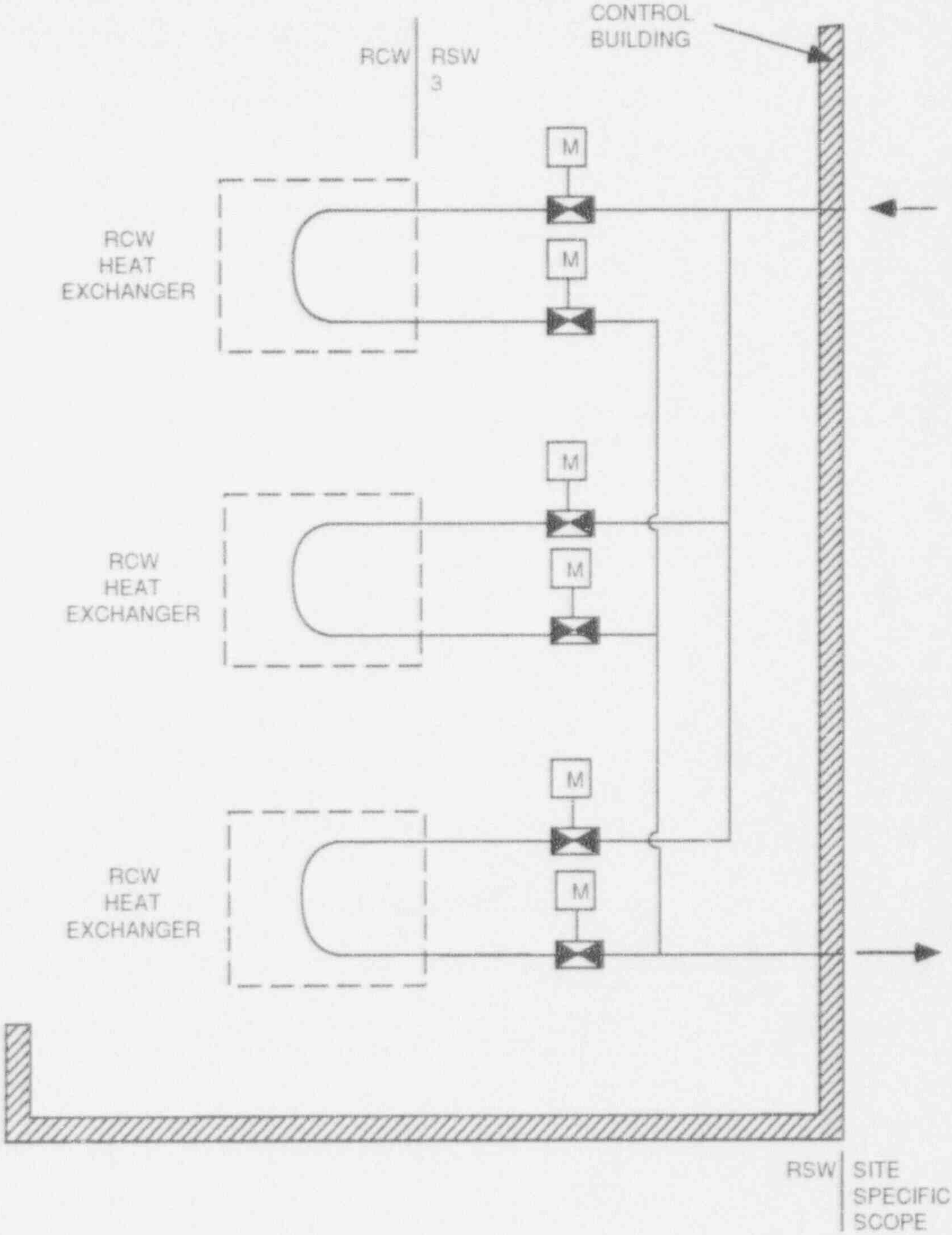
The portions of the RSW System which are not part of the Certified Design shall meet the following requirements:

- Design features shall be provided to limit the maximum flood height to 5.0 meters in each RCW heat exchanger room.

- The design shall have three divisions which are physically separated. Each division shall be powered by its respective Class 1E Division. Each division shall be capable of removing the design heat capacity (as specified in Section 2.11.3) of the RCW heat exchangers in its division.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.11.9 provides a definition of the inspections, tests, and/or analyses together with associated acceptance criteria for the portions of the RSW System within the Certified Design.



NOTES:

1. THIS FIGURE SHOWS ONE OF THREE IDENTICAL DIVISION. ALL ELECTRICAL POWER LOADS FOR THE COMPONENT IN DIVISIONS A, B, AND C ARE POWERED FROM DIVISIONS I, II, AND III, RESPECTIVELY.

2. VALVES SHOWN ABOVE IN DIVISION A AND B HAVE CONTROLS AND OPEN/CLOSE STATUS DISPLAY ON THE REMOTE SHUTDOWN PANEL.

Figure 2.11.9 Reactor Service Water System

Table 2.11.9 Reactor Service Water System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the RSW System is as shown on Figure 2.11.9.	1. Inspections of the as-built system will be conducted.	1. The as-built RSW System conforms with the basic configuration shown in Figure 2.11.9.
2. The ASME Code components of the RSW 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 RSW System required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME Code components of the RSW System conform with the requirements in the ASME Code, Section III.
3. On a LOCA signal, any closed valves for standby heat exchangers are automatically opened	3. Using simulated LOCA signals, tests will be performed on standby heat exchanger inlet and outlet valves.	3. Upon receipt of simulated LOCA signals, the standby heat exchanger inlet and outlet valves open.
4. For each division of RSW, the heat exchanger inlet and outlet valves close upon receipt of a signal indicating Control Building flooding in that division.	4. Using simulated signals, tests will be conducted on the heat exchanger inlet and outlet valves.	4. The heat exchanger inlet and outlet valves close upon receipt of a signal indicating Control Building flooding in that division.
5. In the RSW System, independence is provided between Class 1E Divisions, and between Class 1E Divisions and non-Class 1E equipment.	5a. Tests will be performed on the RSW System by providing a test signal in only one Class 1E Division at a time. 5b. Inspections of the as-installed Class 1E Divisions in the RSW System will be performed.	5a. The test signal exists only in the Class 1E Division under test in the RSW System. 5b. Physical separation exists between Class 1E Divisions in the RSW System. Physical separation exists between Class 1E Divisions and non-Class 1E equipment.
6. Each mechanical division of the RSW System (Divisions A, B, C) is physically separated.	6. Inspections of the as-built system will be performed.	6. Each mechanical division of the RSW System is physically separated from other mechanical divisions of the RSW System by structural and/or fire barriers.
7. MCR displays and controls provided for RSW System are defined in Section 2.11.9.	7. Inspections will be performed on the MCR displays and controls for the RSW System.	7. Displays and controls exist or can be retrieved in the MCR as defined in Section 2.11.9.
8. RSS displays and controls provided for the RSW System are as defined in Section 2.11.9.	8. Inspections will be performed on the RSS displays and controls for the RSW System.	8. Indications and controls exist on the RSS as defined in Section 2.11.9.

Table 2.11.9 Reactor Service Water System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. MOVs designated in Section 2.11.9 as having an active safety-related function will open and close under differential pressure, fluid flow, and temperature conditions.	9. Opening and closing tests of installed valves will be conducted under preoperational differential pressure, fluid flow, and temperature conditions.	9. Each MOV opens and closes.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.11.9 Reactor Service Water

PUNCH LIST ITEM: A. Corresponding SSAR change needed to address maximum flood height.

On 1/21/93:
A. Check ASME standardization words for classification of piping.

GE DISPOSITION: A: **The SSAR Section 3.4.1.1.2.2, which discusses Control Building flooding events, is being revised and will include a more detailed discussion of the maximum flood height that can occur.**

A: The text of 2.11.9 discusses ASME code classification for the piping in the RSW; it is not quite the standardized wording used in other systems.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.11.9 Reactor Service Water

PUNCH LIST ITEM: B. Water level sensor moved to Control Building ITAAC. (Needs to be picked up.) Add corresponding change to SSAR.

On 1/17/93:
B. Water level sensor moved to Control Building ITAAC. Add corresponding change to SSAR.

GE DISPOSITION: B: The Control Building Tier 1 entry will address the issue of water level sensors in the Control Building basement. This issue will be addressed in both Tier 1 and the SSAR sections of control building material.

B: See above response.

2.12 Station Electrical

2.12.1 Electrical Power Distribution System

Design Description

The AC Electrical Power Distribution (EPD) System consists of the transmission system (TS), the plant switching stations, the Main Power Transformer (MPT), the Unit Auxiliary Transformers (UAT), the Reserve Auxiliary Transformer(s) (RAT), the plant main generator (PMG) output circuit breaker, the medium voltage metal-clad (M/C) switchgear, the low voltage power center (P/C) switchgear, and the motor control centers (MCCs). The distribution system also includes the power, instrumentation and control cables and bus ducts to the distribution system loads, and the protection equipment provided to protect the distribution system equipment. The EPD System within the scope of the Certified Design starts at the low voltage terminals of the MPT and the low voltage terminals of the RAT(s) and ends at the distribution system loads. Interface requirements for the TS, plant switching stations, MPT, and RAT(s) are specified below.

The plant EPD System can be supplied power from multiple offsite power sources; these are two independent transmission lines from the TS, the PMG, and the combustion turbine generator (CTG). In addition, the EPD System can be supplied from three onsite Class 1E Standby Power Sources (Emergency Diesel Generators (DGs)). The Class 1E portion of the EPD System is shown in Figure 2.12.1.

During plant power operation, the PMG supplies power through the PMG output circuit breaker to the TS, through the MPT, and to the UATs. When the PMG output circuit breaker is open, power is backfed from the TS through the MPT to the UATs.

The UATs can supply power to the non-Class 1E load groups of medium voltage M/C power generation (PG) and plant investment protection (PIP) switchgear, and to the three Class 1E divisions (Division I, II, and III) of medium voltage M/C switchgear.

The RAT(s) can supply power to the non-Class 1E load groups of medium voltage M/C PG and PIP switchgear, and to the three Class 1E divisions (Division I, II, and III) of medium voltage M/C switchgear.

The UATs are sized to supply their load requirements, during design operating modes, of their respective Class 1E divisions and non-Class 1E load groups. UATs are separated from the RAT(s) by a minimum of 15.24 meters. In addition, UATs are provided with their own oil pit, drain, fire deluge system, grounding, and lightning protection system.

The PMG, its output circuit breaker, and UAT output power feeders are separated from the RAT(s) output power feeders by a minimum of 15.24 meters, or by walls or floors. The PMG, its output circuit breaker, and UAT instrumentation and control circuits, outside the main control room (MCR), are separated from the RAT(s) instrumentation and control circuits by a minimum of 15.24 meters, or by walls or floors, except at the non-Class 1E DC power sources where they are routed in separate raceways. UATs instrumentation and control circuits, inside the MCR, are separated from the RAT(s) instrumentation and control circuits by routing the circuits in separate raceways. Separation between the UATs and RAT(s) output power feeders, and instrumentation and control circuits is maintained at the M/C switchgear by routing the circuits to the opposite ends of the switchgear. Redundant instrumentation and control power for the UATs is supplied from separate, non-Class 1E DC power systems.

The MPT switching station instrumentation and control circuits, from the switchyard(s) to the MCR, are separated from the RAT(s) switching station instrumentation and control circuits by a minimum of 15.24 meters, or by walls or floors. Separation is maintained, inside the MCR, by routing instrumentation and control circuits in separate raceways.

Non-Class 1E load groups of medium voltage M/C switchgear are supplied power from a UAT with an alternate power supply from a RAT. In addition, the non-Class 1E medium voltage M/C switchgear can be supplied power from the CTG.

Class 1E medium voltage M/C switchgear are supplied power directly (not through any bus supplying non-Class 1E loads) from either a UAT or a RAT. Class 1E medium voltage M/C switchgear can also be supplied power from their own dedicated Class 1E EDG or from the non-Class 1E CTG.

The medium voltage M/C switchgear and low voltage P/C switchgear, with their respective transformers, and the low voltage MCCs are sized to supply their load requirements. M/C and P/C switchgear, and MCCs are rated to withstand fault currents for the time required to clear the fault from the power source. The PMG output circuit breaker, and power feeder and load circuit breakers for the M/C and P/C switchgear, and MCCs are sized to supply their load requirements and are rated to interrupt fault currents.

EPD System interrupting devices (circuit breakers and fuses) are coordinated so that the circuit interrupter closest to the fault opens before other devices.

Instrumentation and control power for the Class 1E divisional medium voltage M/C switchgear and low voltage P/C switchgear is supplied from the Class 1E DC power system in the same division.

The PMG output circuit breaker is equipped with redundant trip devices which are supplied from separate, non-Class 1E DC power systems.

EDP System cables and bus ducts are sized to supply their load requirements and are rated to withstand fault currents for the time required to clear the fault from the power source.

For the EPD System, Class 1E power is supplied by three independent Class 1E divisions. Independence is maintained between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.

Non-Class 1E loads that are connected to a Class 1E power source are identified and treated as Associated Class 1E circuits, or are connected to only a single Class 1E division through a Class 1E circuit breaker with Zone Selective Interlocks.

Class 1E medium voltage M/C switchgear and low voltage P/C switchgear and MCCs are identified according to their Class 1E division. Class 1E divisional M/C and P/C switchgear and MCCs are located in Seismic Category I structures, and in their respective divisional electrical equipment rooms or fire areas.

Class 1E EPD System cables and raceways are identified according to their Class 1E division. Class 1E divisional cables are routed in Seismic Category I structures and in their respective divisional raceways.

Harmonic Distortion waveforms do not prevent Class 1E equipment from performing their safety functions.

The EPD System design ensures that the operating voltage supplied at the terminals of the Class 1E utilization equipment is within the utilization equipment's voltage tolerance limits.

An electrical grounding system is provided for (1) instrumentation, control, and computer systems, (2) electrical equipment (switchgear, distribution panels, and motors) and mechanical equipment (fuel and chemical tanks). Lightning protection systems are provided for buildings and for structures and transformers located outside of the buildings. Each grounding system and lightning protection system is separately grounded to the plant ground grid.

PMG and EPD System displays and controls are provided in the MCR. Displays provided for the PMG consist of PMG output voltage, amperes, watts, vars, and frequency. Displays provided for the EPD System consist of medium voltage M/C switchgear bus voltages, feeder and load amperes, and circuit breaker positions. Controls are provided for the PMG output circuit breaker, medium voltage M/C switchgear feeder circuit breakers, load circuit breakers from the medium voltage M/C switchgear to their respective low voltage P/C switchgear, and the low voltage feeder circuit breakers to the low voltage P/C switchgear.

EPD System displays and controls are provided at the Remote Shutdown System (RSS). Displays provided consist of bus voltages for the Class 1E Divisions I and II medium voltage M/C switchgear. Controls are provided for the UAT, RAT(s), CTG, and EDG Class 1E feeder circuit breakers to the Class 1E Divisions I and II medium voltage M/C switchgear and the load circuit breakers from the Class 1E Division I and II medium voltage M/C switchgear to their respective low voltage P/C switchgear, and the low voltage feeder circuit breakers to the Class 1E Division I and II low voltage P/C switchgear.

Interface Requirements

The portions of the EPD System which are not part of the Certified Design shall meet the following requirements.

The TS voltage variation shall be no more than plus or minus 10 percent of its nominal value during periods of steady state operation.

The TS frequency variation shall be within plus or minus 2 Hertz of 60 Hertz during recoverable periods of system instability.

Independent switching stations shall be connected to different transmission lines. Transmission lines shall be separated by a minimum of 15.24 meters.

The MPT and RAT(s) shall be connected to independent switching stations. Switching stations and their circuit breakers shall be sized to supply their load requirements and rated to interrupt fault currents. Switching stations shall be separated by a minimum of 15.24 meters.

The MPT shall be separated from the RAT(s) by a minimum of 15.24 meters.

The MPT shall be sized to supply its load requirements, during designed operating modes, of its respective Class 1E divisions and non-Class 1E load groups. The MPT shall be provided with its own oil pit, drain, fire deluge system, grounding, and lightning protection system.

The MPT output power feeders shall be separated from the RAT(s) output power feeders by a minimum of 15.24 meters. The MPT instrumentation and control circuits shall be separated from the RAT(s) instrumentation and control circuits by a minimum of 15.24 meters, except at the DC power sources, where they shall be routed in sperate raceways.

The RAT(s) shall be sized to supply its load requirements, during all designed operating modes, of its respective Class 1E divisions and non-Class 1E load groups. The RAT(s) shall be provided with its own oil pit, drain, fire deluge system, grounding, and lightning protection system.

Switching station protection circuits shall be redundant, physically separated, and supplied power from separate switching station DC power sources.

Switching station circuit breakers shall be provided with redundant trip devices and supplied power from separate switching station DC power sources.

Redundant instrumentation and control circuits for the MPT and RAT(s) shall be supplied power from separate, non-Class 1E DC power sources.

Final selection of the MPT and RAT(s) impedance shall be compatible with the interrupting capability of the plant's offsite and onsite circuit interrupting devices.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.12.1 provides a definition of the inspections, tests, and/or analyses together with associated acceptance criteria which will be undertaken for the Electrical Power Distribution (EPD) System.

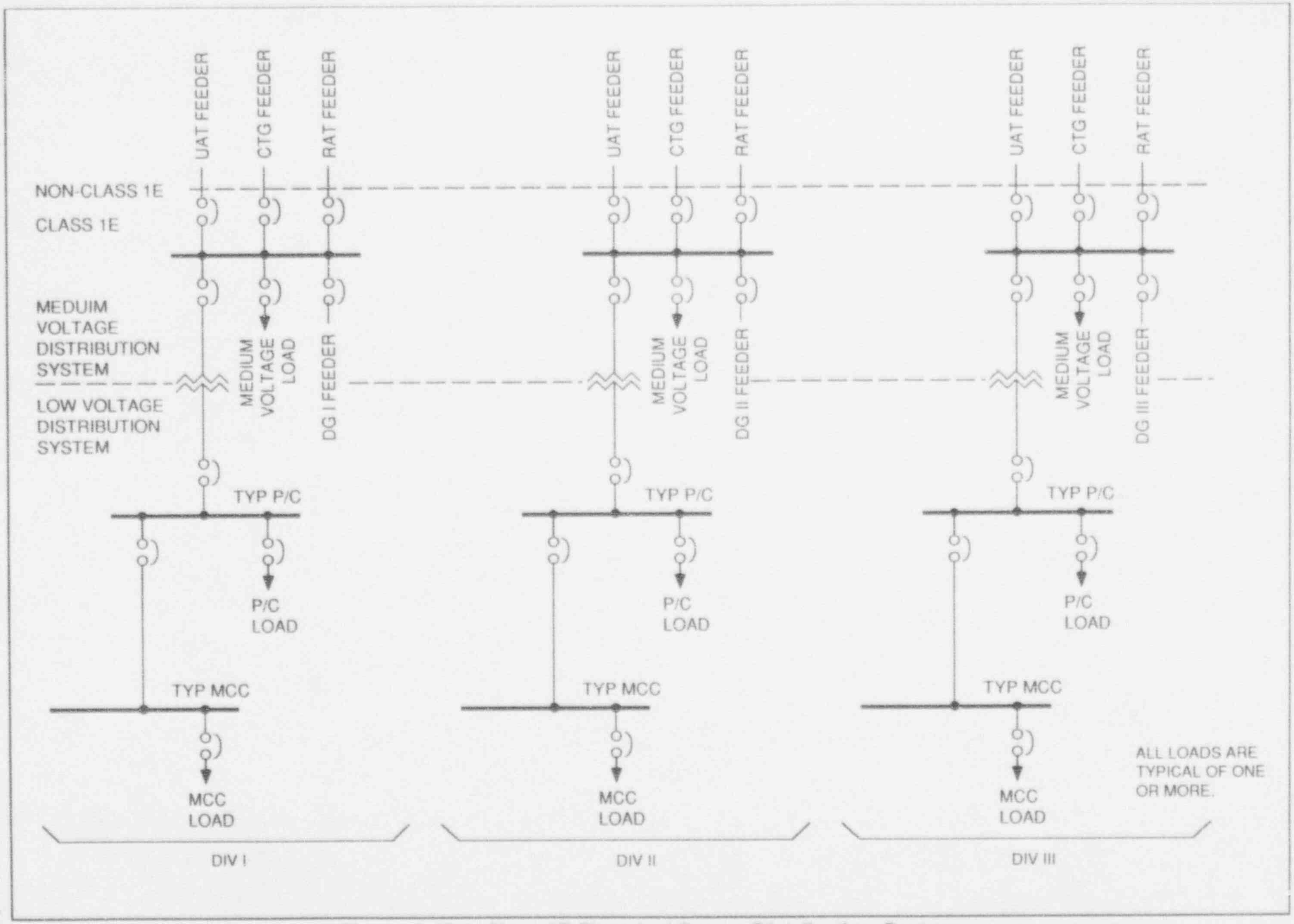


Figure 2.12.1 Class 1E Electrical Power Distribution System

Table 2.12.1 Electric Power Distribution System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration for the EPD System is described in Section 2.12.1.	1. Inspection of the as-built EPD System configuration will be performed.	1. The as-built EPD System configuration is in accordance with Section 2.12.1.
2. UATs are sized to supply their load requirements, during design operating modes, of their respective Class 1E divisions and non-Class 1E load groups.	2. Load Analyses for as-built UATs will be performed.	2. Load Analyses for as-built UATs exist and conclude that UAT capacity, as determined by its nameplate rating, exceeds its analyzed load requirements, during design operating modes, for its Class 1E division and non-Class 1E load group.
3. UATs are separated from the RAT(s).	3. Inspection of as-built UATs installation will be performed.	3. As-built UATs are separated from the RAT(s) by a minimum of 15.24 meters.
4. UATs are provided with their own oil pit, drain, fire deluge system, grounding, and lightning protection systems.	4. Inspection of as-built UATs installation will be performed.	4. As-built UATs are provided with their own oil pit, drain, fire deluge system, grounding, and lightning protection systems.
5. UATs output power feeders, and instrumentation and control circuits are separated from the RAT(s) output power feeders, and instrumentation and control circuits.	5. Inspection of the as-built UATs and RAT(s) output power feeders, and instrumentation and control circuits will be performed.	5. As-built UATs power feeders are separated from the RAT(s) power feeders by a minimum of 15.24 meters, or by walls or floors. As-built UATs instrumentation and control circuits, outside the MCR room, are separated from the RAT(s) instrumentation and control circuits by a minimum of 15.24 meters, or by walls or floors, except at the non-Class 1E DC power sources, where they are routed in separate raceways. UATs instrumentation and control circuits, inside the MCR, are separated from the RAT(s) instrumentation and control circuits by routing the circuits in separate raceways. UATs and RAT(s) power feeders, and instrumentation and control circuits are routed to opposite ends of the medium voltage M/C switchgear.

Table 2.12.1 Electric Power Distribution System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. Redundant instrumentation and control power for the UATs is supplied from separate non-Class 1E DC power systems.	6. Tests will be performed by providing a test signal in only one circuit at a time.	6. A test signal exists in only the circuit under test.
7. The MPT switching station instrumentation and control circuits are separated from the RAT(s) switching station instrumentation and control circuits.	7. Inspection of the as-built MPT and RAT(s) switching station instrumentation and control circuits will be performed.	7. As built MPT switching station instrumentation and control circuits, from the switchyard(s) to the MCR, are separated from the RAT(s) switching station instrumentation and control circuits by a minimum of 15.24 meters, or by walls or floors. MPT switching station instrumentation and control circuits, inside the MCR, are separated from the RAT(s) switching station instrumentation and control circuits by routing the circuits in separate raceways.
8. Medium voltage M/C switchgear, low voltage P/C switchgear, with their respective transformers, and MCCs, and their respective feeder and load circuit breakers are sized to supply their load requirements.	8a. Load Analyses for the as-built M/C and P/C switchgear, with their respective transformers, MCCs, and their respective feeder and load circuit breakers will be performed.	8a. Load Analyses for as-built M/C and P/C switchgear, with their respective transformers, MCCs, and their respective feeder and load circuit breakers exist and conclude that the capacities of Class 1E switchgear, P/C transformers, MCCs, and their respective feeder and load circuit breakers, as determined by their nameplate ratings, exceed their analyzed load requirements.
	8b. Load tests of the Class 1E M/C and P/C switchgear and MCCs and their respective load circuit breakers will be performed by operating connected Class 1E loads in the ranges of 9% to 10% above and 9% to 10% below design voltage.	8b. Connected Class 1E loads operate in the ranges of 9% to 10% above and 9% to 10% below design voltage

Table 2.12.1 Electric Power Distribution System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9a. Medium voltage M/C switchgear, low voltage P/C switchgear and MCCs, and their respective feeder and load circuit breakers are rated to withstand fault currents for the time required to clear the fault from its power source.	9a. Short Circuit Analyses to determine fault currents for the as-built EPD System will be performed.	9a. Short Circuit Analyses for the as-built EPD System exist and conclude that the Class 1E switchgear and MCCs, and their respective feeder and load circuit breaker's current capacities exceed their analyzed fault currents for the time required, as determined by the Breaker Coordination Analyses, to clear the fault from its power source.
9b. The PMG output circuit breaker, medium voltage M/C switchgear, low voltage P/C switchgear and MCC feeder and load circuit breakers are rated to interrupt fault currents.	9b. Short Circuit Analyses to determine fault currents for the as-built EPD System will be performed.	9b. Short Circuit Analyses for the as-built EPD System exist and conclude that the analyzed fault currents do not exceed the PMG output circuit breaker, and M/C, P/C, and MCC feeder and load circuit breakers interrupt capacities, as determined by their nameplating ratings.
10. EPD System interrupting devices (circuit breakers and fuses) are coordinated so that the circuit interrupter closest to the fault opens before other devices.	10. Breaker Coordination Analyses for the as-built EPD System will be performed.	10. Breaker Coordination Analyses for the as-built EPD System exist and conclude that the analyzed circuit interrupter closest to the fault will open before other devices.
11. Instrumentation and control power for Class 1E divisional medium voltage M/C switchgear and low voltage P/C switchgear is supplied from the Class 1E DC power system in the same division.	11. Tests will be performed by providing a test signal in only one Class 1E division at a time.	11. A test signal exists in only the Class 1E division under test.
12. The PMG output circuit breaker is equipped with redundant trip devices which are supplied from separat,e non-Class 1E DC power systems.	12. Tests will be performed by providing a test signal in only one circuit at a time.	12. A test signal exists in only the circuit under test.

Table 2.12.1 Electric Power Distribution System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
13. EPD System cables and bus ducts are sized to supply their load requirements.	13. Load Analyses for the as-built EPD System cables and bus ducts will be performed.	13. Load Analyses for the as-built EPD System exist and conclude that cable and bus duct capacities, as determined by cable and bus duct ratings, exceed their analyzed load requirements.
14. EPD System cables and bus ducts are rated to withstand fault currents for the time required to clear the fault from the power source.	14. Short Circuit Analyses to determine fault currents for the as-built EPD System will be performed.	14. Short Circuit Analyses for the as-built EPD System exist and conclude that cables and bus ducts will withstand the analyzed fault currents for the time required, as determined by the Breaker Coordination Analyses, to clear the faults from their power sources.
15. Independence is maintained between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	15a. Tests will be performed by providing a test signal in only one Class 1E division at a time.	15a. A test signal exists in only the Class 1E division under test.
	15b. Inspection of the as-built Class 1E divisions will be performed.	15b. Physical separation exists between as-built Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.
16. Non-Class 1E loads that are connected to a Class 1E power source are identified and are treated as Associated Class 1E circuits or are connected to only a single Class 1E division.	16a. Inspection of the Class 1E power sources will be performed.	16a. Non-Class 1E loads treated as Associated Class 1E are identified.
	16b Tests will be performed by providing a test signal in only one Class 1E division at a time.	16b. Non-Class 1E loads, not identified and treated as Associated Class 1E, receive a test signal from only one Class 1E division.
17. Class 1E divisional medium voltage M/C switchgear and low voltage P/C switchgear and MCCs are identified according to their Class 1E division.	17. Inspection of the as-built Class 1E divisional M/C and P/C switchgear and MCCs will be performed.	17. As-built Class 1E M/C and P/C switchgear, and MCCs are identified according to their Class 1E division.

Table 2.12.1 Electric Power Distribution System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
18. Class 1E divisional medium voltage M/C switchgear and low voltage P/C switchgear and MCCs are located in seismic Category I structures and in their respective divisional electrical equipment rooms or fire areas.	18. Inspection of the as-built Class 1E divisional M/C and P/C switchgear and MCCs will be performed.	18. As-built Class 1E divisional M/C and P/C switchgear, and MCCs are located in seismic Category I structures and in their respective divisional electrical equipment rooms or fire areas.
19. Class 1E EPD System cables and raceways are identified according to their Class 1E division.	19. Inspection of the as-built Class 1E EPD System divisional cables and raceways will be performed.	19. As-built Class 1E EPD System cables and raceways are identified according to their Class 1E division.
20. Class 1E divisional cables are routed in seismic Category I structures and in their respective divisional raceways.	20. Inspection of the as-built Class 1E EPD System divisional cables and raceways will be performed.	20. As-built Class 1E divisional cables are routed in seismic Category I structures and in their respective divisional raceways.
21. Harmonic Distortion waveforms do not prevent Class 1E equipment from performing their safety functions.	21. Harmonic Distortion Analyses on the as-built EPD System will be performed.	21. Harmonic distortion waveforms do not exceed 5 percent voltage distortion on the Class 1E EPD System.
22. EPD System design ensures that the operating voltage supplied at the terminals of the Class 1E utilization equipment is within the utilization equipment's voltage tolerance limits.	22. Voltage Drop Analyses on the as-built EPD System will be performed.	22. Voltage Drop Analyses for the as-built EPD System exist and conclude that the analyzed operating voltage supplied at the terminals of the Class 1E utilization equipment is within the utilization equipment's voltage tolerance limits, as determined by their nameplate ratings.
23. An electrical grounding system is provided for (1) instrumentation, control, and computer systems, (2) electrical equipment (switchgear, distribution panels, and motors) and mechanical equipment (fuel and chemical tanks). Lightning protection systems are provided for buildings and for structures and transformers located outside of the buildings. Each grounding system and lightning protection system is separately grounded to the plant ground grid.	23. Inspection of the as-built plant Grounding and Lightning Protection Systems will be performed.	23. The as-built instrumentation, control, and computer grounding system, electrical equipment and mechanical equipment grounding system, and lightning protection systems provided for buildings and for structures and transformers located outside of the buildings are separately grounded to the plant ground grid.

Table 2.12.1 Electric Power Distribution System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
24. MCR displays and controls for the EPD System are as defined in Section 2.12.1.	24. Inspection on the as-built MCR displays and controls will be performed.	24. Displays and controls exist or can be retrieved in the MCR as defined in Section 2.12.1.
25. RSS displays and controls for the EPD System are as defined in Section 2.12.1.	25. Inspection on the as-built RSS displays and controls will be performed.	25. Displays and controls exist or can be retrieved on the RSS as defined in Section 2.12.1.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: 1. Confirm there are legitimate Part 52 reasons why the MPT is site-specific. (Put it in-scope if no good reason exists.)

GE DISPOSITION: GE review of the MPT have led to the conclusion that there are multiple reasons why this equipment is site-specific. The major reason is the various grid voltage levels to which the certified ABWR design could be connected. It is noteworthy that this review led to the conclusion that the UAT and RAT should also be made site-specific. The revised ABWR Tier 1 entry 2.12.1 is now structured on the assumption that these transformers are all site-specific, and GE has provided interfaces that must be met by this equipment. This issue has been discussed informally with the NRC, and GE believes there is concurrence with this approach.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: 2. Confirm that 2.12.1 still covers the other systems ID as being part of 2.12.1.

GE DISPOSITION: To the extent that the systems need to be discussed in Tier 1, Item 2.12.1 covers the following ABWR systems.

2.12.2 - Unit Auxiliary Transformer
2.12.5 - Metal-Clad Switchgear
2.12.6 - Power Center
2.12.7 - Motor Control Center
2.12.8 - Raceway System
2.12.9 - Grounding Wire
2.13.1 - Reserve Auxiliary Transformer
4.2 - Offsite Power System

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: 3. Revise the scope section to Group 1E/non-1E.

GE DISPOSITION: Complete. The design description has been rearranged to group the Class 1E and non-Class 1E equipment.

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: 4. Identify system as AC.

GE DISPOSITION: System is now identified as AC.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: 5. Use RSW 2.11.9 as guidance for interface treatment.

GE DISPOSITION: The revised version of 2.12.1 has been structured to include a separate entry at the end of the design description to cover interface requirements for equipment not within the scope of the certified design.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: 6. Define (more precisely) system boundaries in scope paragraph.

GE DISPOSITION: Complete. The design description now clearly identifies the boundaries of this system.

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: 7. Add MCR and RSS boilerplate.

GE DISPOSITION: The revised entry 2.12.1 now includes standardized entry for Main Control Room and Remote Shutdown System interfaces.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: 8. GE to confirm (assess the extent to which) the I&C Tier 1 entry covers
- software needed by EPDS
- setpoints for EPDS items

GE DISPOSITION: The proposed GE I&C Tier 1 material will cover these issues for the EPDS (safety-related).

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: * 9. Review ITAAC Items 2, 3, 9, 10, 11, 15, 16, 20 which currently have non-specific acceptance criteria and revise per the R/B structural analysis report precedent. Add SAR material to define report scope (OK to use applicable IEEE standards).

GE DISPOSITION: a) The revised 2.12.1 ITAAC that addresses these issues is now based on the recommended approach, i.e., an Acceptance Criteria which states that an anlysis exists which concludes that (...the equipment operates correctly...).

b) SAR modifications are under way and will be submitted to the NRC staff.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: 10. Delete all "visual."

GE DISPOSITION: The revised entry for 2.12.1 does not use the word "visual."

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: 11. Change trip coils to trip devices.

GE DISPOSITION: Implemented.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: 12. Clean up No. 8 ITAAC by putting two entries in ITA and A/C columns to reflect two DC entries.

GE DISPOSITION: Complete. This item (non-Class 1E equipment connected to Class 1E buses) now has the recommended two entries in the ITA and AC columns.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: 13. For ITAAC No. 18, add switchgear after medium and fix DD. (Ditto low voltage.)

GE DISPOSITION: Complete. This was a minor editorial item.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: * 14. For ITAAC No. 20:

a) Write up SAR entries on voltage drop commitments (source to load).
b) GE to determine standard testing practice today.
c) Reconvene GE/NRC review.

GE DISPOSITION:

a) SAR changes are in process. The issue is being discussed by GE and NRC personnel on a periodic basis.
b) Complete. The conclusion is that new testing will be required of the COL holder.
c) Being handled by ongoing GE/NRC interactions.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.12.1 Emergency Power Distribution System

PUNCH LIST ITEM: 15. Link 15 and 20 and the breaker coordination/fault analysis items.

a) Write up SAR entries on voltage drop commitments (source to load).
b) GE to determine standard testing practice today.
c) Reconvene GE/NRC review.

GE DISPOSITION: After further review of this item, GE concluded that the proposed linkage of Item 15 (Cable and Bus Duct Sizing) and Item 20 (EPDS Voltage Drop Analyses) is not appropriate. This is because Item 20 is system-wide and depends upon all EPDS components, not just the cables and buses addressed in Item 15. Consequently, GE has left these items as separate ITAAC entries.

2.15.10 Reactor Building

Design Description

The Reactor Building (R/B) is a structure which houses and provides protection and support for, the reactor primary systems, the primary containment and much of the plant safety-related equipment. Figures 2.15.10a through 2.15.10o show the basic configuration and scope of the R/B*.

The R/B is constructed of reinforced concrete and structural steel with a steel frame and reinforced concrete roof. The R/B encloses the primary containment. The R/B slabs and fuel pool girders are integrated with the reinforced concrete containment vessel (RCCV). The R/B slabs are supported by columns and beams to carry vertical loads to the basemat and transfer horizontal loads to either the RCCV or the R/B shear walls. The R/B, together with the RCCV and the reactor pedestal, are supported by a common basemat. Inside the RCCV, the basemat is considered part of the Primary Containment System (PCS); outside the RCCV, the basemat is part of the R/B. The top of the R/B basemat is located 25m ± 1m below the finished grade elevation.

The R/B is divided into three separate divisional areas for mechanical and electrical equipment and four divisional areas for instrumentation racks. Inter-divisional boundaries have the following features:

(1) Inter-divisional 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 0.6 meters thick or greater.

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

The R/B flooding that results from component failures in any of the R/B divisions does not prevent safe shutdown of the reactor. The basement floor is the collection location point for all floods. The building configuration at this elevation is such that even for a flooding event involving release of the suppression pool water into the R/B, no more than one division of safety-related equipment is affected. Except for the basement area, all safety-related electrical, instrumentation and control equipment is located at least 20 cm above the floor surface.

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* The overall building dimensions provided in Figures 2.15.10a through 2.15.10o are provided for information only and are not intended to be part of the certified ABWR information.

The R/B is protected against external flood. The following design features are provided:

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

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

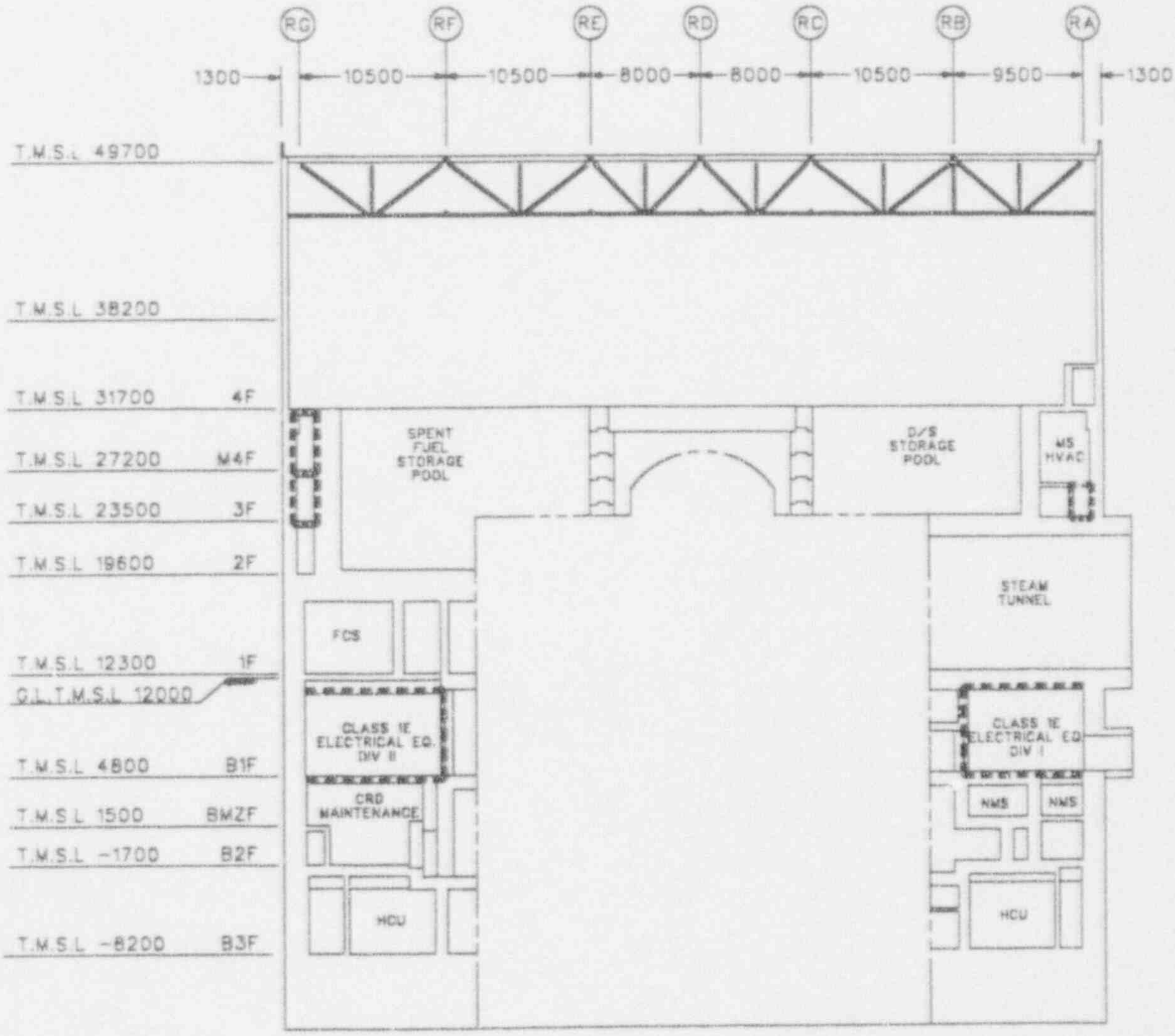
The R/B is protected against the pressurization effects associated with postulated rupture of pipes containing high-energy fluid that occur in subcompartments of the R/B.

The R/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:

- Natural phenomena including wind, floods, tornados, earthquakes, rain and snow.
- Internal events including fires, floods, pipe breaks and missiles.
- Normal plant operation including live loads, dead loads, temperature effects and building vibration loads.

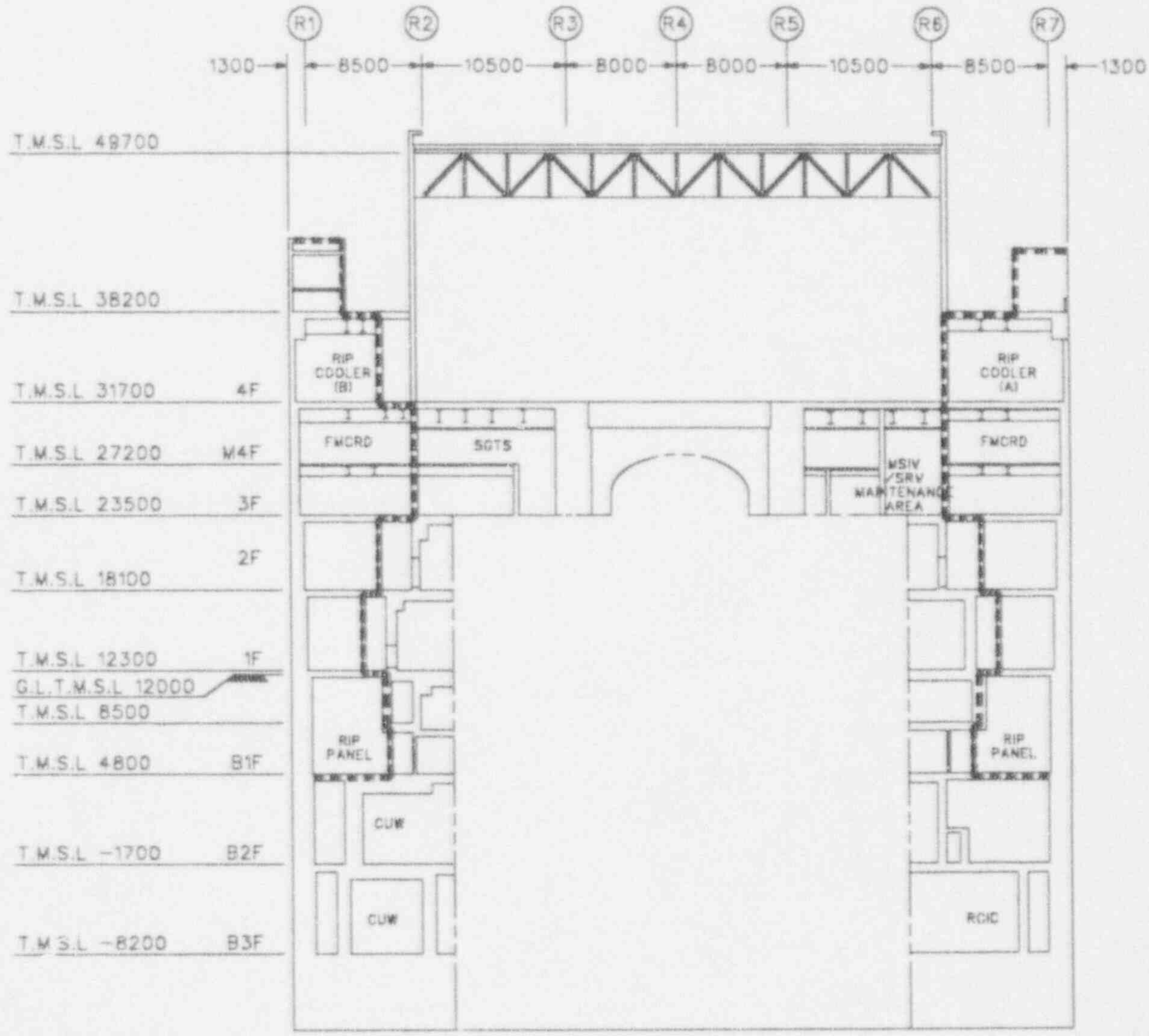
Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.10 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the R/B.



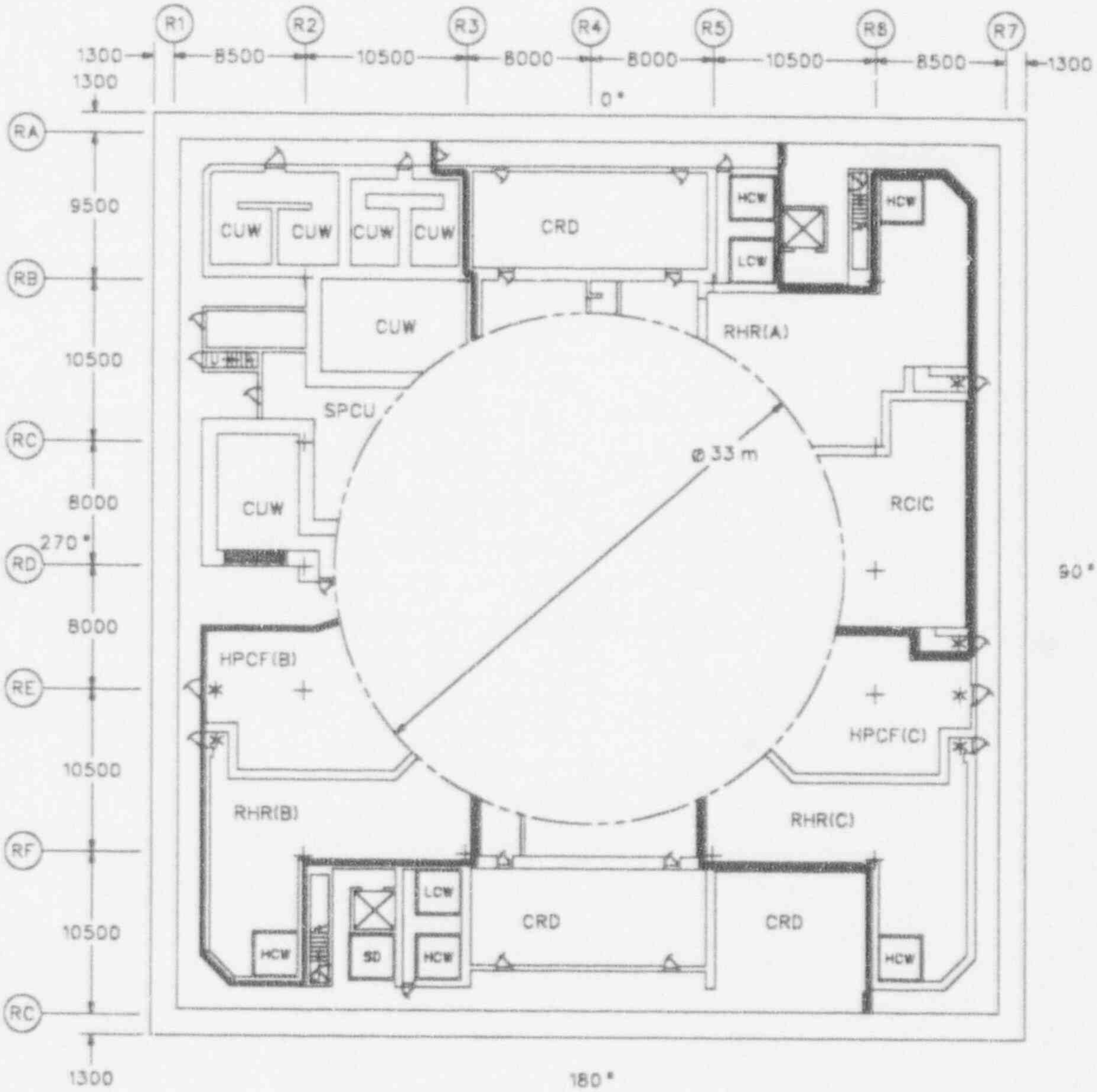
NOTES:
1. DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

Figure 2.15.10a Reactor Building Arrangement—Section A-A



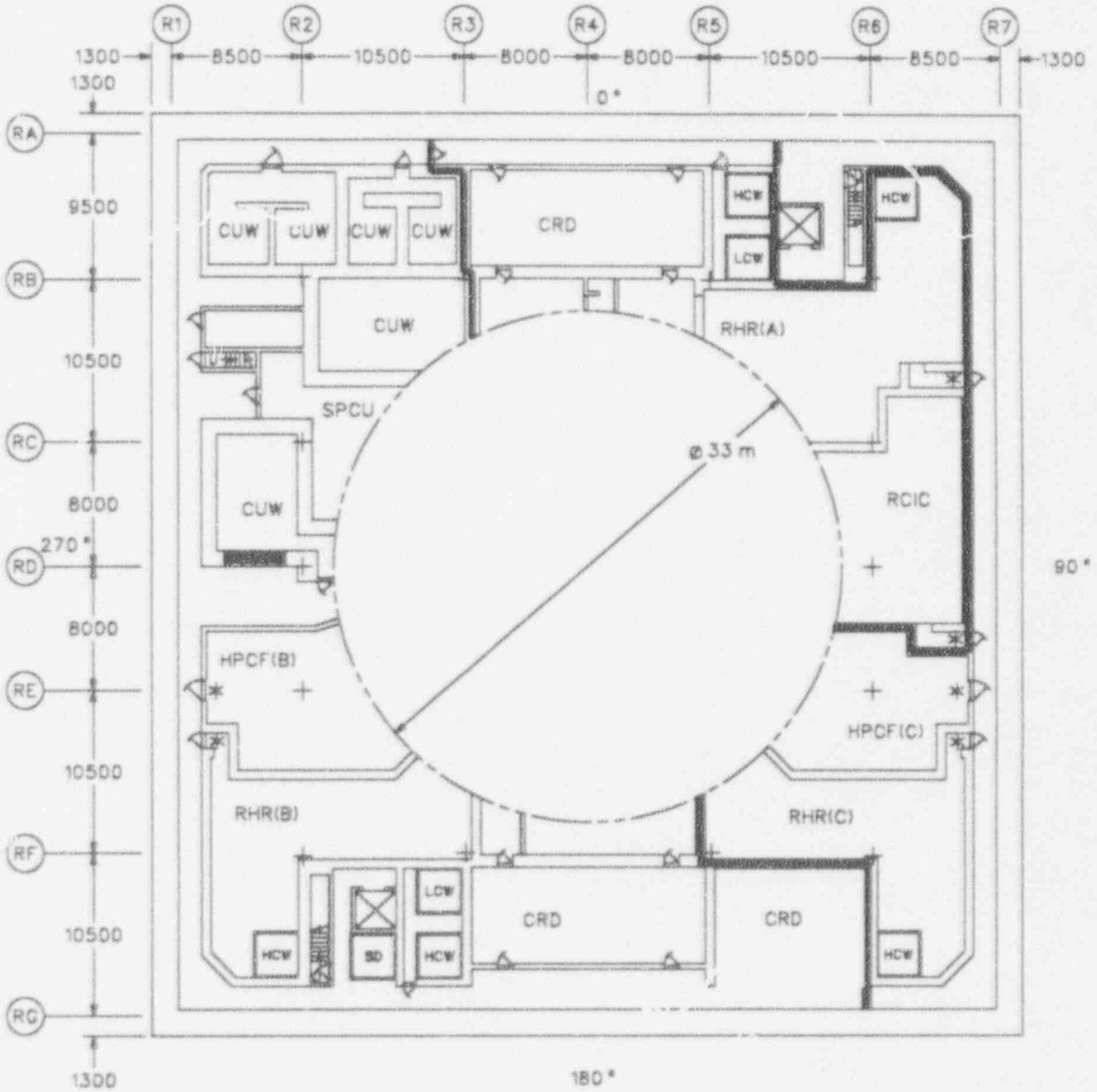
NOTES:
1. DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

Figure 2.15.10b Reactor Building Arrangement—Section B-B



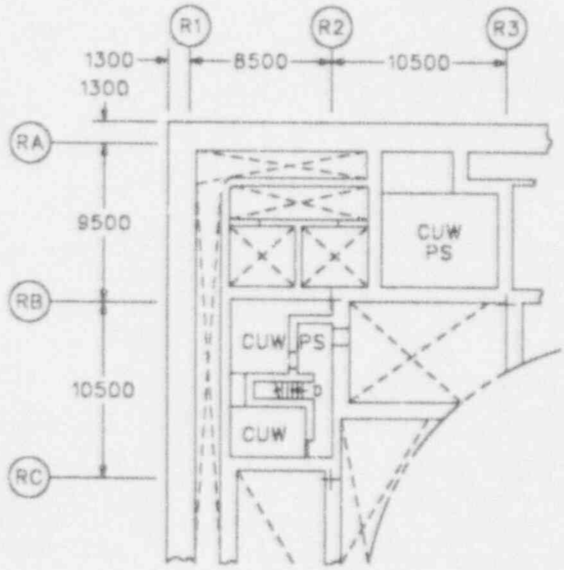
NOTES:
1. DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

Figure 2.15.10c Reactor Building Arrangement Floor B3F with Divisional Boundary for Flood—Elevation –8200 mm



NOTES:
1. DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

Figure 2.15.10d Reactor Building Arrangement Floor B3F with Divisional Boundary for Fire—Elevation –8200 mm



NOTES:
1. DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

Figure 2.15.10e Reactor Building Arrangement Floor BM3F—Elevation –5100 mm

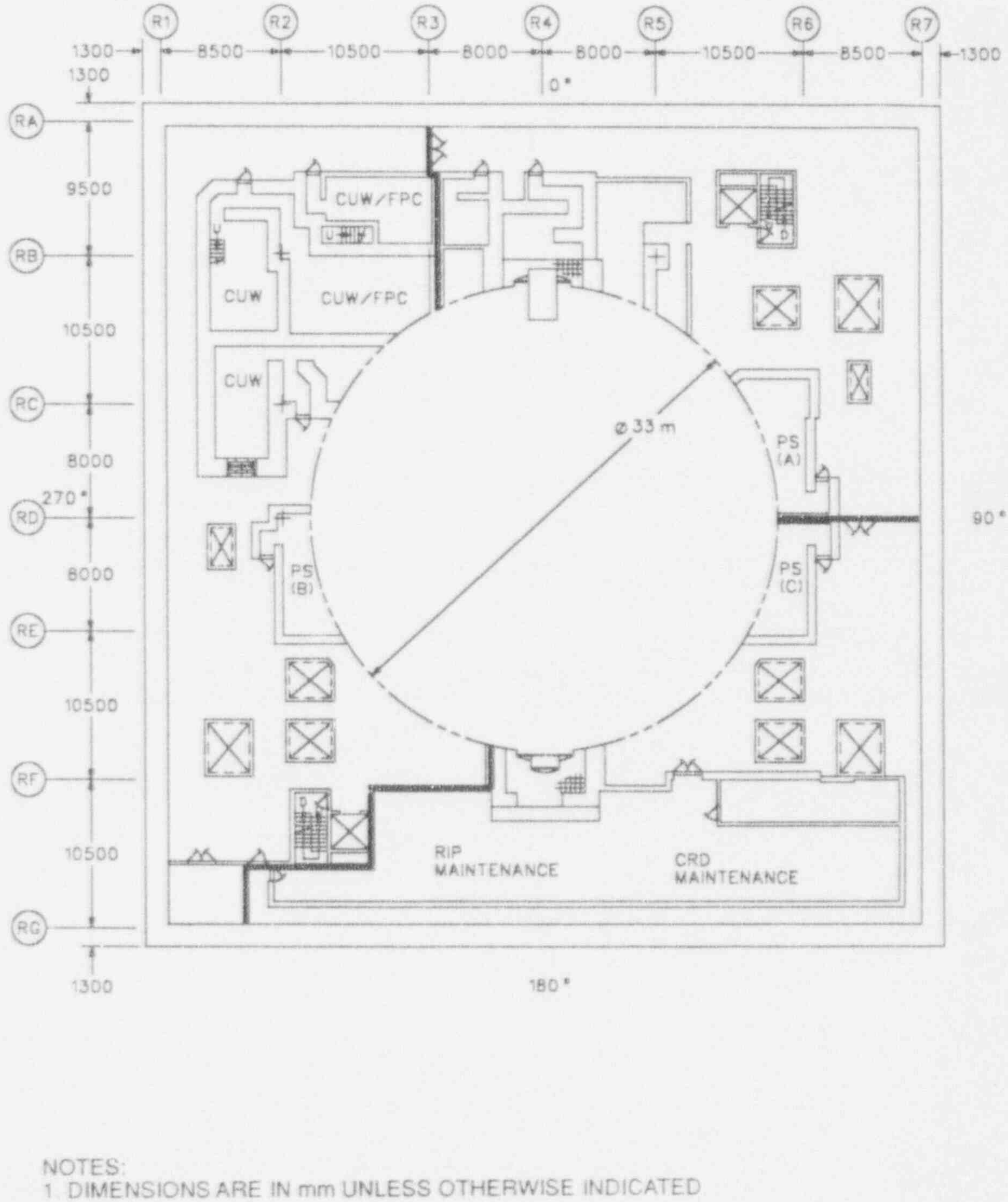
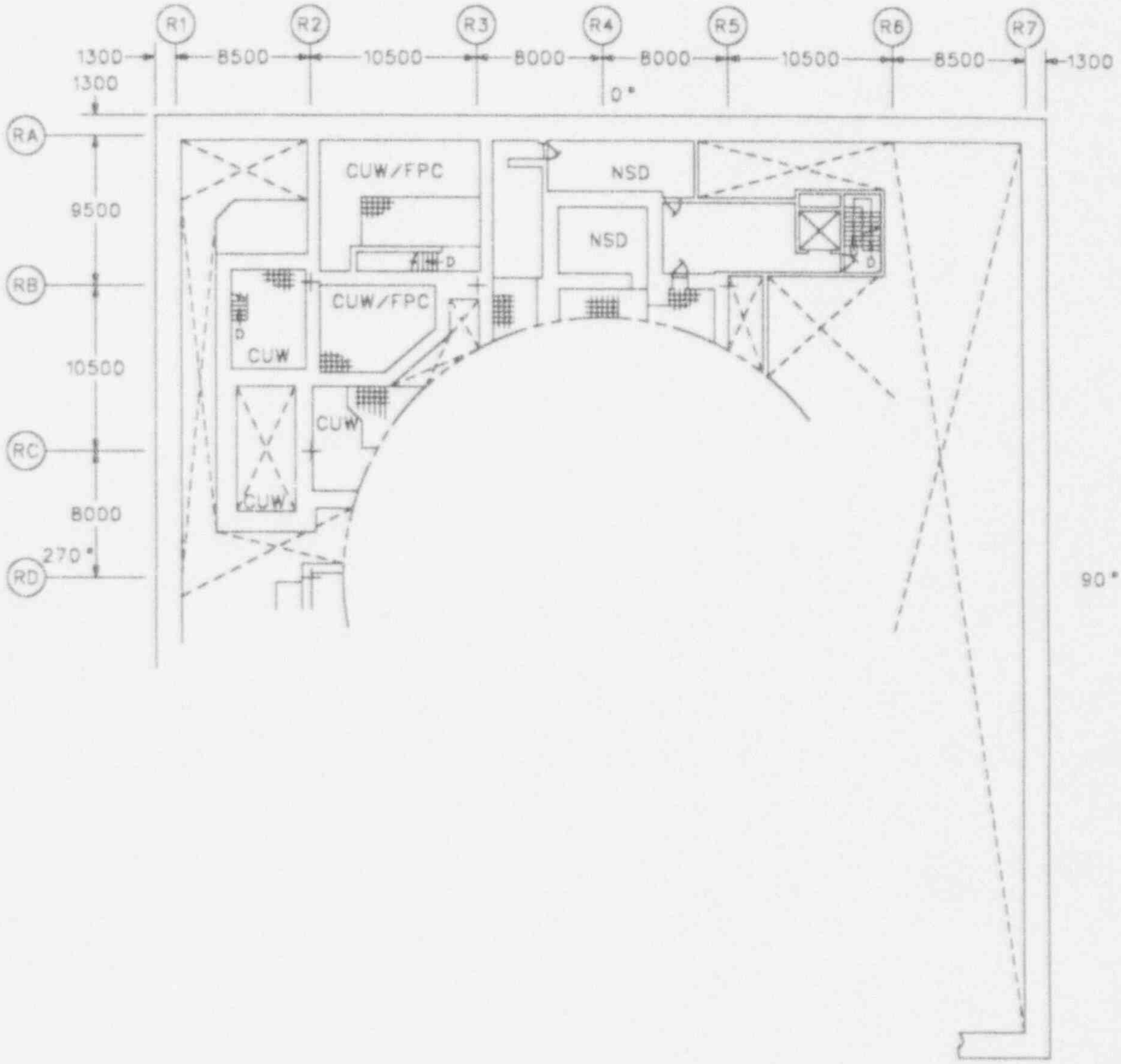
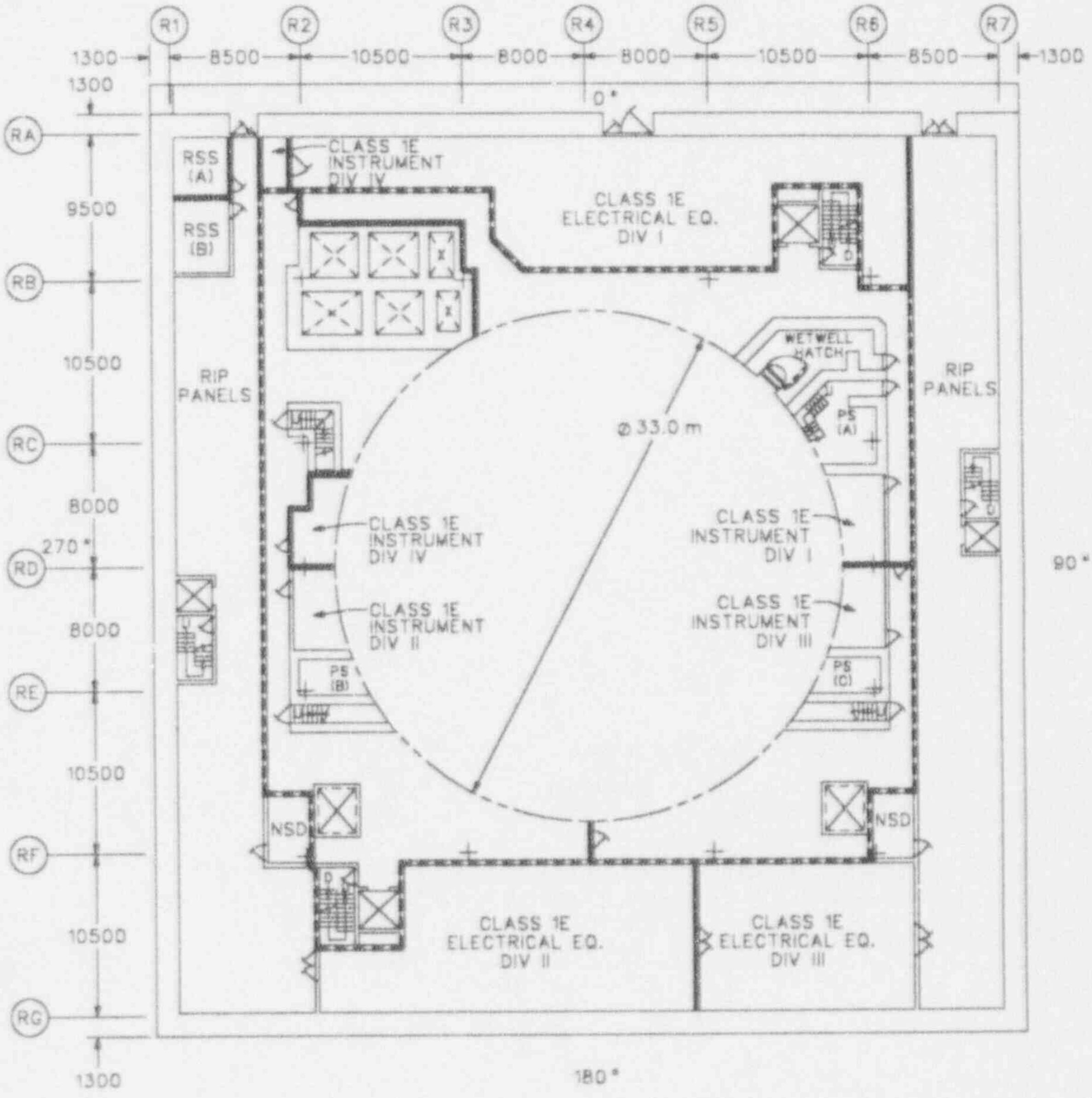


Figure 2.15.10f Reactor Building Arrangement Floor B2F—Elevation –1700 mm



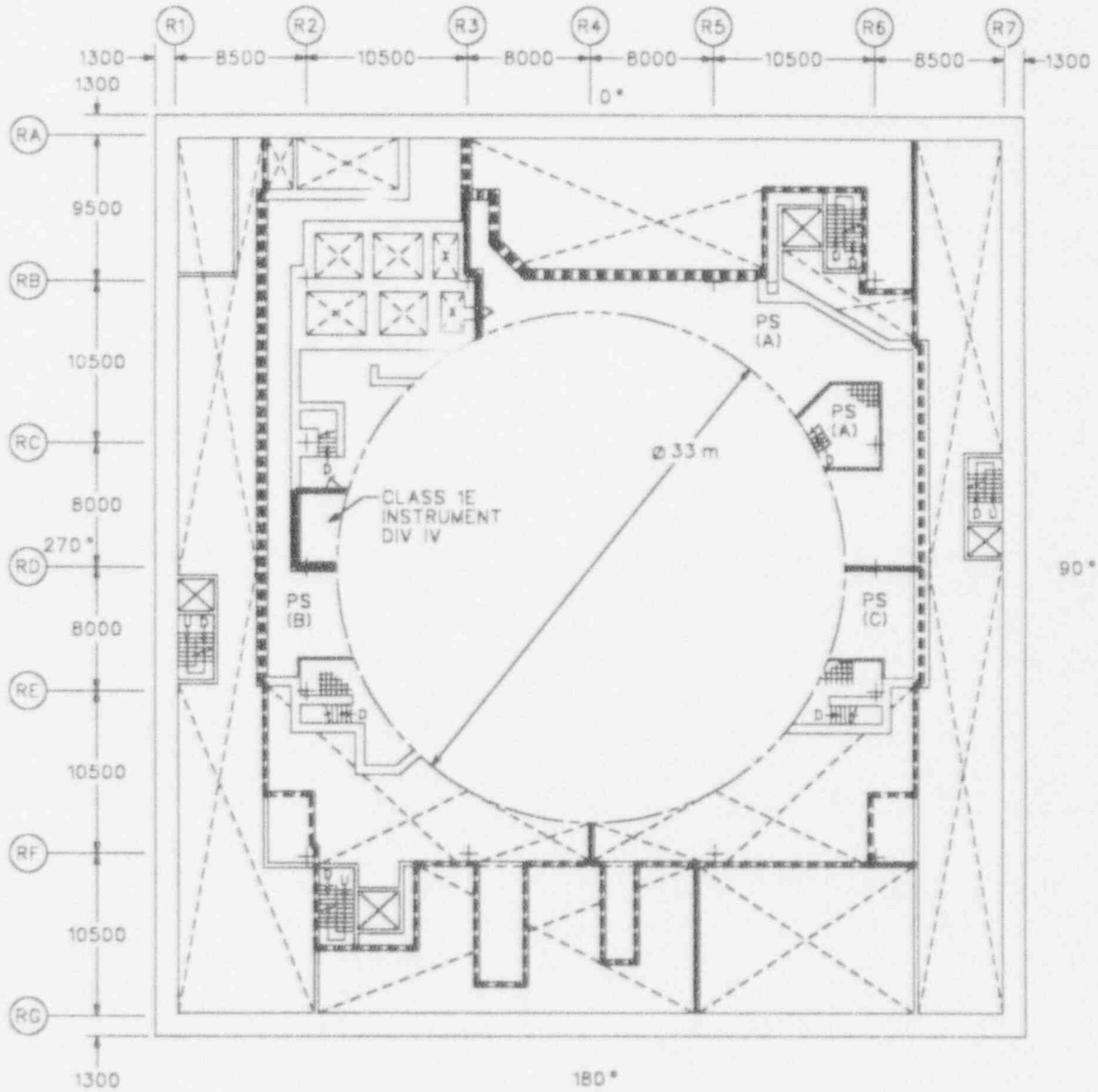
NOTES:
1. DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

Figure 2.15.10g Reactor Building Arrangement BM2F—Elevation 1500 mm



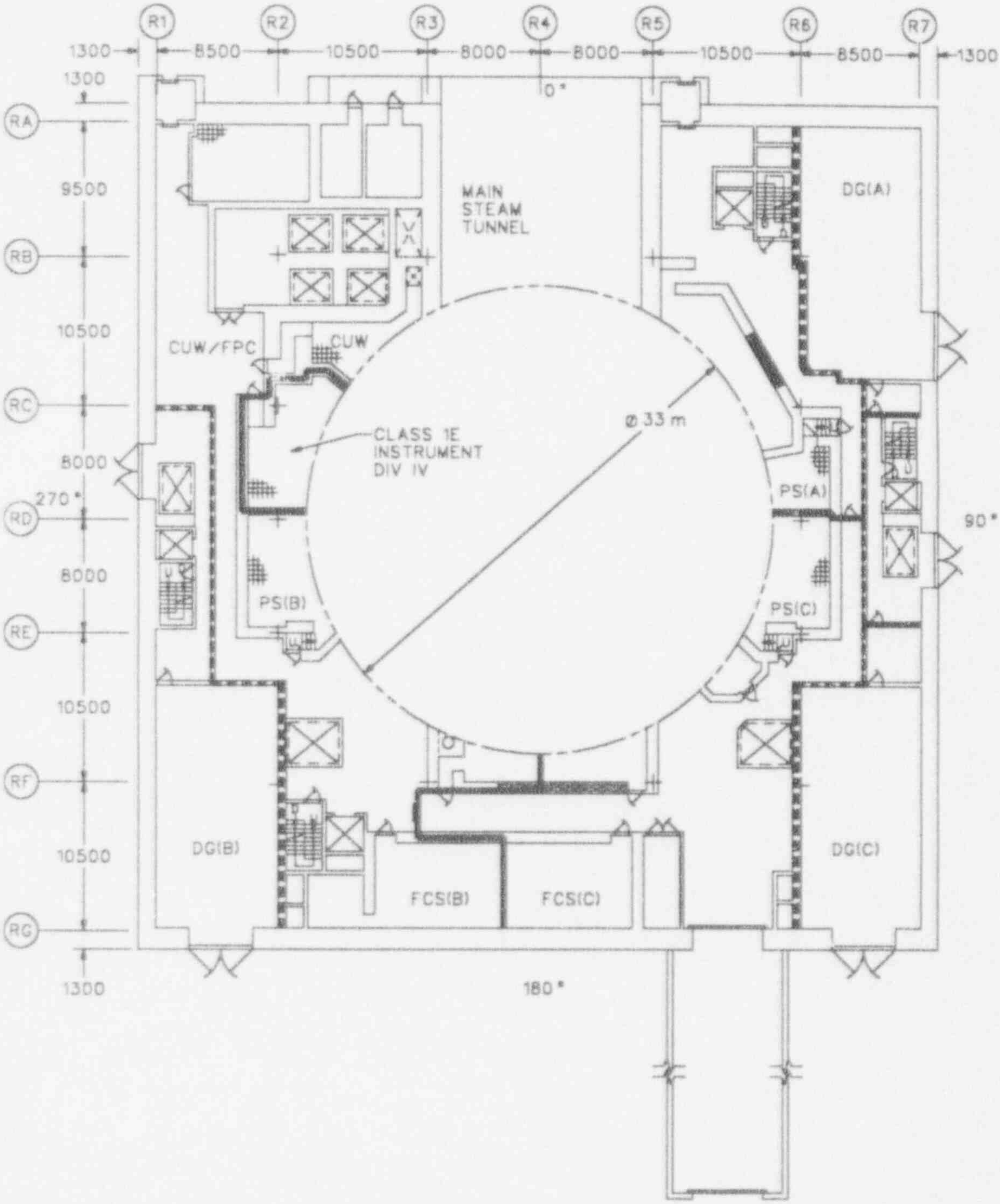
NOTES:
1. DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

Figure 2.15.10h Reactor Building Arrangement Floor B1F—Elevation 4800 mm



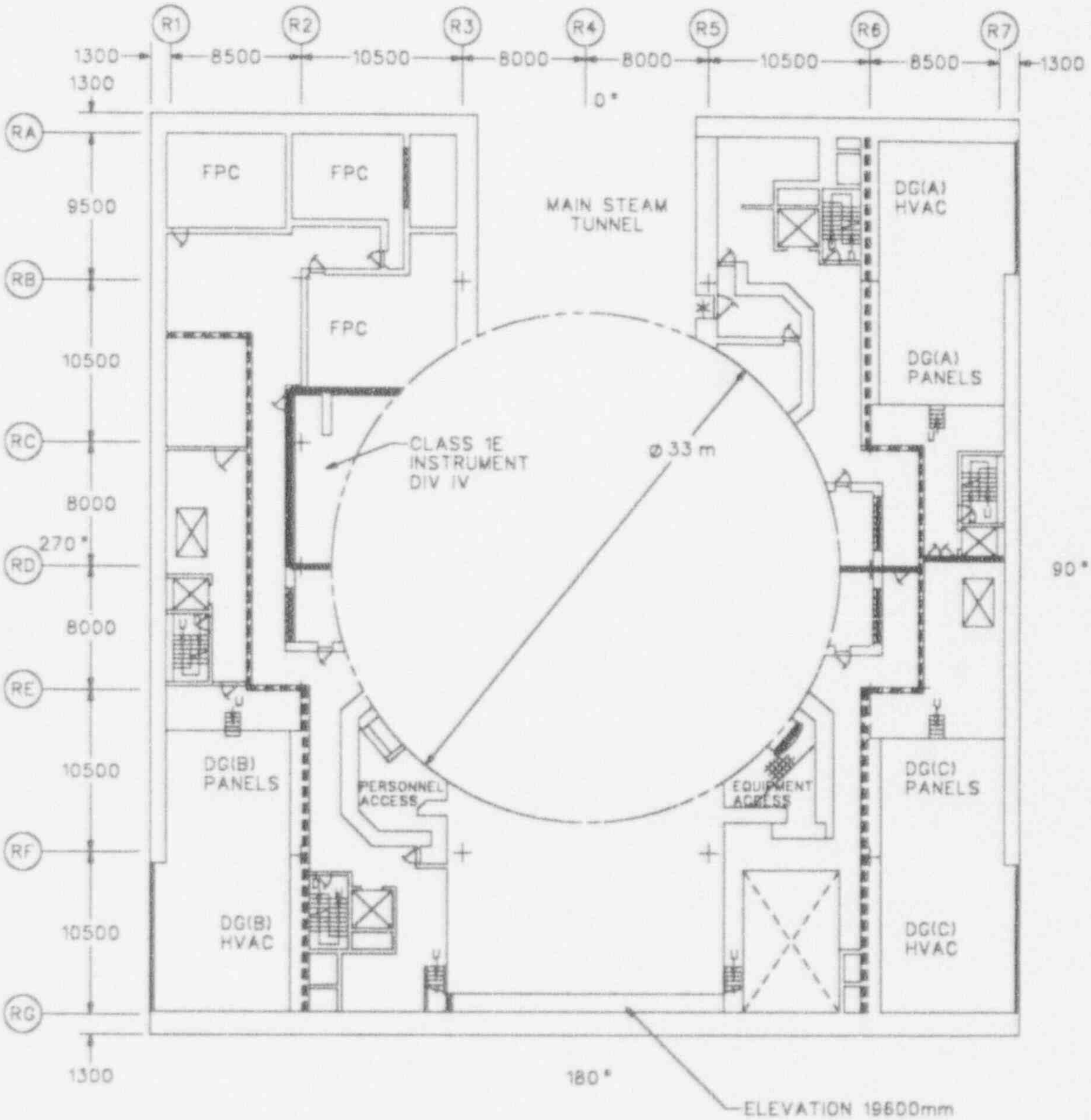
NOTES:
1. DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

Figure 2.15.10i Reactor Building Arrangement Floor BM1F—Elevation 8500 mm



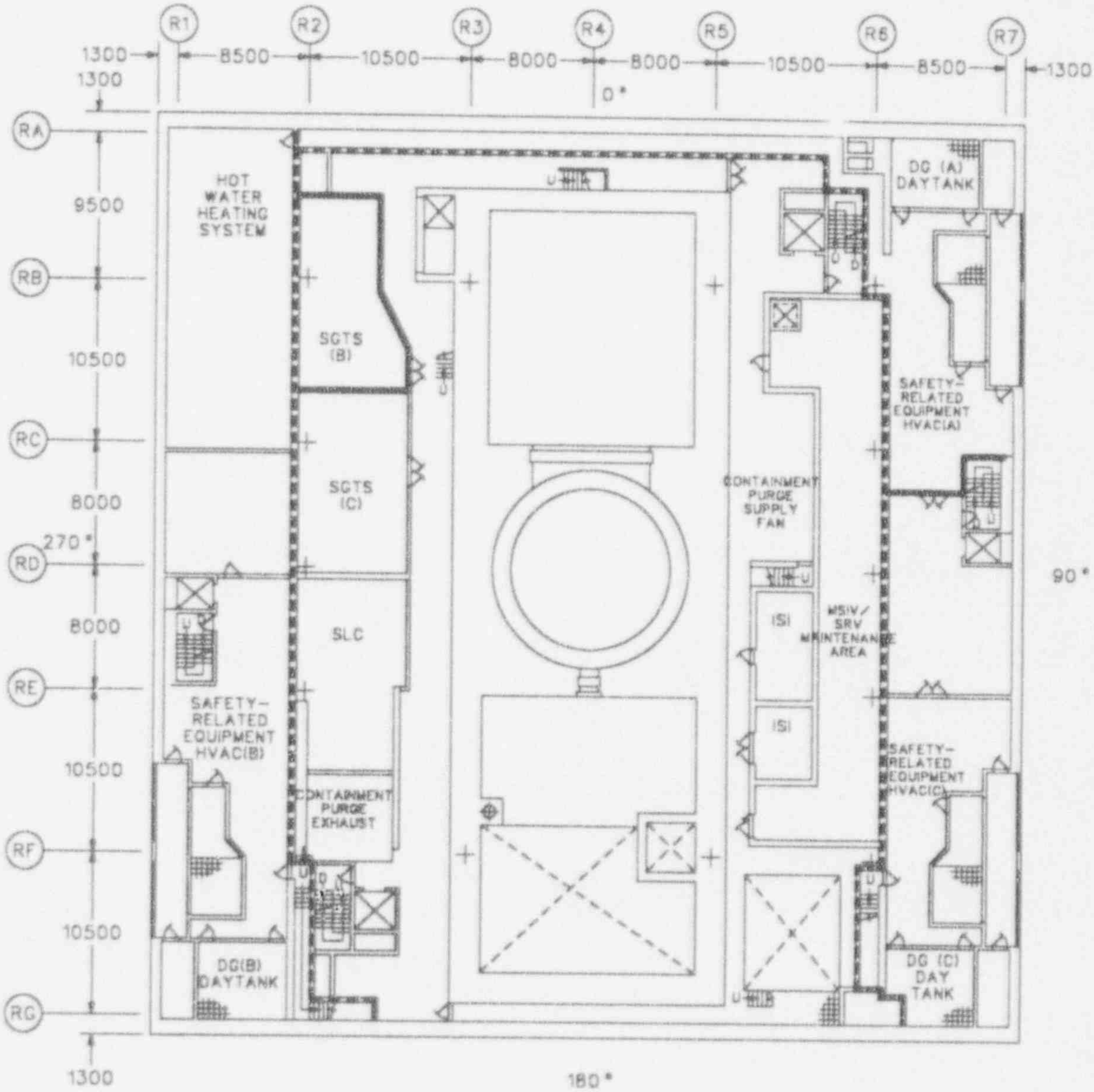
NOTES:
1. DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

Figure 2.15.10j Reactor Building Arrangement Floor 1F—Elevation 12300 mm



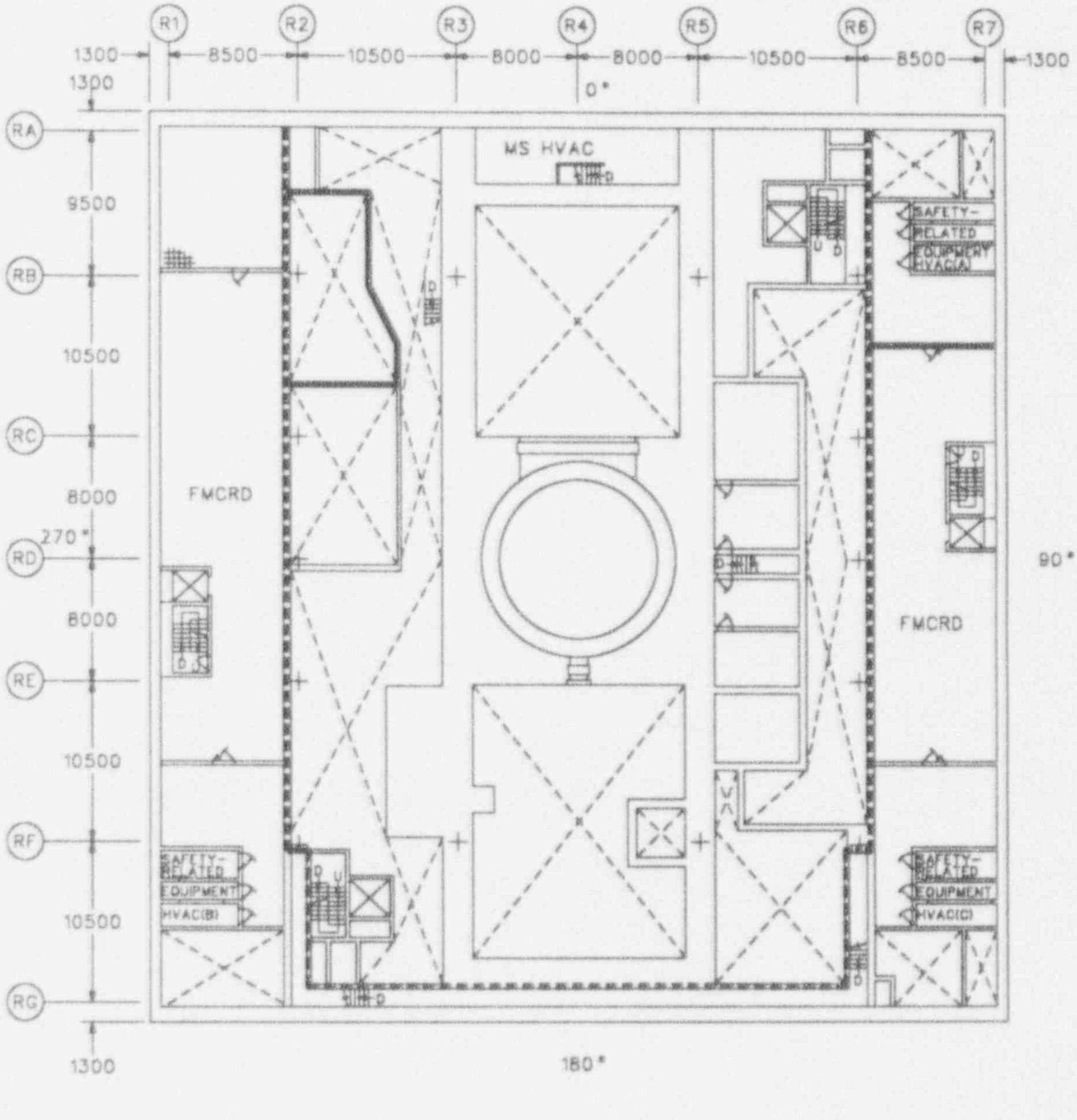
NOTES:
1. DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

Figure 2.15.10k Reactor Building Arrangement Floor 2F—Elevation 18100 mm



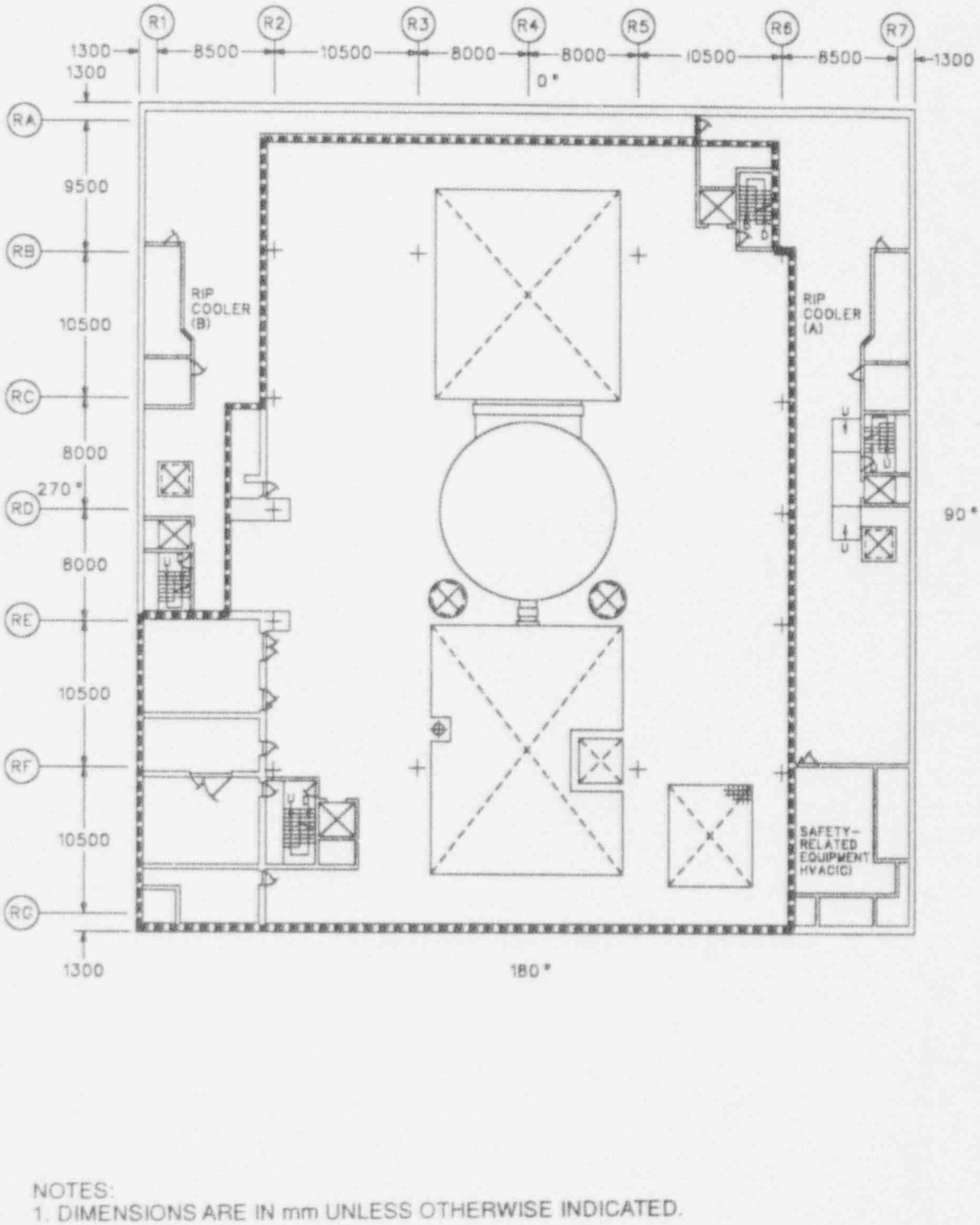
NOTES:
1. DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

Figure 2.15.10l Reactor Building Arrangement Floor 3F—Elevation 23500 mm



NOTES:
1. DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

Figure 2.15.10m Reactor Building Arrangement Floor M4F—Elevation 27200 mm



NOTES:
1. DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

Figure 2.15.10n Reactor Building Arrangement Floor 4F—Elevation 31700 mm

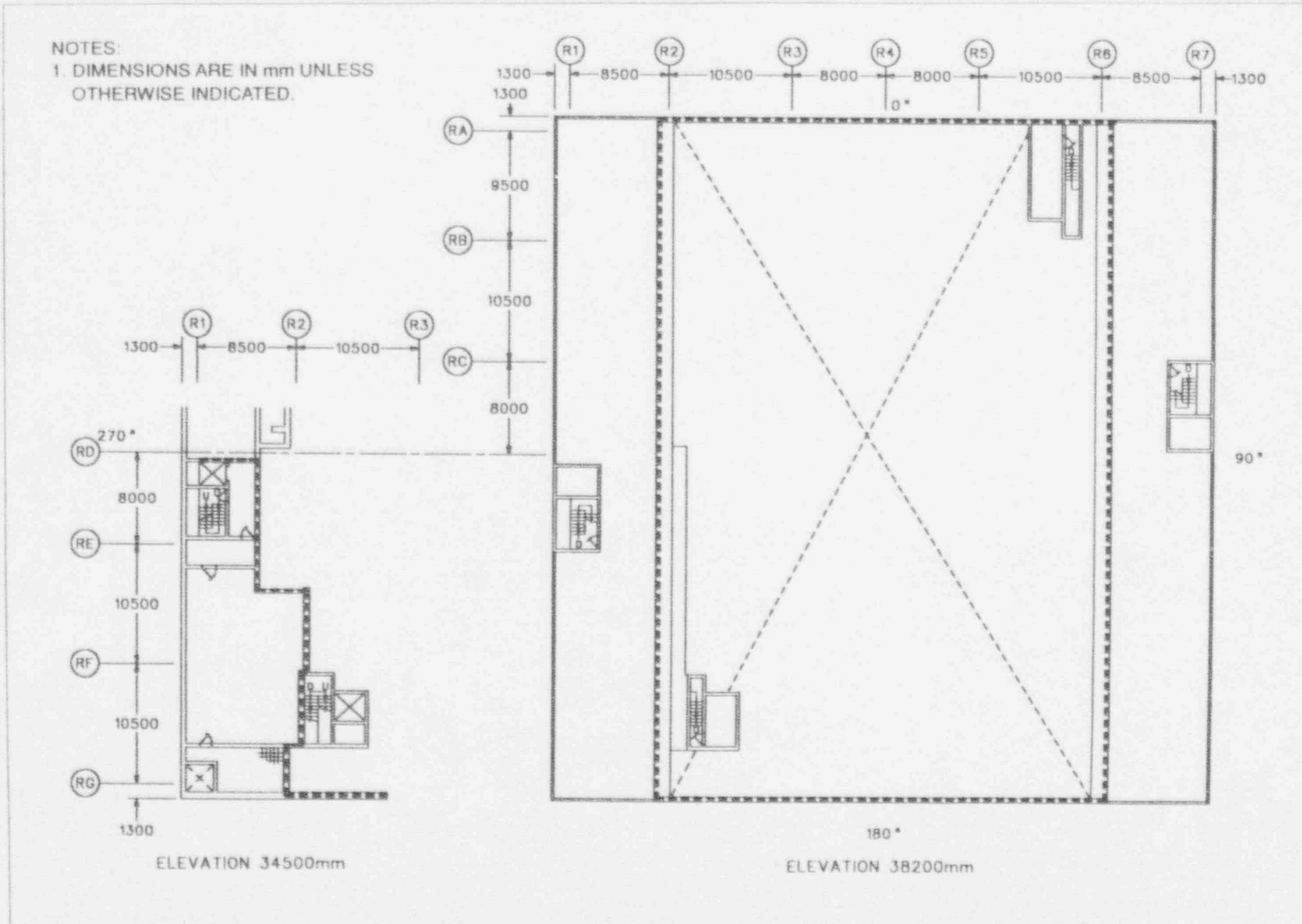


Figure 2.15.10o Reactor Building Arrangement—Elevations 34500 mm and 38200 mm

Table 2.15.10 Reactor Building

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the R/B is shown on Figures 2.15.10a through 2.15.10o.	1. Inspections of the as-built structure will be conducted.	1. The as-built R/B conforms with the basic configuration shown in Figures 2.15.10a through 2.15.10o.
2. The top of the R/B basemat is located 25m±1m below the finished grade elevation.	2. Inspections of the as-built structure will be conducted.	2. The top of the R/B basemat is located 25m±1m below the finished grade elevation.
3. Inter-divisional walls, floors, doors and penetrations in the R/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 R/B has divisional areas with walls, watertight doors and sills as shown on Figures 2.15.10a through 2.15.10o.	4. Inspections of the as-built walls, watertight doors and sills will be conducted.	4. The as-built R/B has walls, watertight doors and sills as shown on Figures 2.15.10a through 2.15.10o.
5. Main control room displays provided for the R/B are as defined in Section 2.15.10.	5. Inspections will beperformed on the main control room displays for the R/B.	5. Displays exist or can be retrieved in the main control room as defined in Section 2.15.10.
6. A flooding event involving release of suppression pool water does not affect more than one division of safety-related equipment.	6. Inspections will be conducted of the divisional boundaries shown on Figure 2.15.10c.	6. Penetrations (except for watertight doors) in the divisional walls are at least 2.5 m above the floor level of -8200mm.
7. Except for the basemat area, safety-related electrical, 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 area, safety-related electrical, instrumentation, and control equipment is located at least 20 cm above the floor surface.
8. The R/B is protected against external floods by having: a) External walls below flood level that are equal to or greater than 0.6m and b) Providing penetrations in the external walls below flood level with flood protection features.	8. Inspections of the as-built structure will be conducted.	8. a) External walls below flood level are equal to or greater than 0.6 meters thick. b) Penetrations in the external walls below flood level are provided with flood protection features.

Table 2.15.10 Reactor Building (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. The R/B is able to withstand the structural design basis loads as defined in Section 2.15.10.	9. A structural analysis will be performed which reconciles the as-built data with structural design basis as defined in Section 2.15.10.	9. A structural analysis report exists which concludes that the as-built R/B is able to withstand the structural design basis loads as defined in Section 2.15.10.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 1. Protection from internal floods due to drain system divisional separation addressed in Drain System ITAAC.

GE DISPOSITION: GE is currently updating the drain system description in the SSAR. When this is complete, a Tier 1 design description and ITAAC table will be developed and included in the ABWR design certification material as Item 2.9.2. Divisional separation of the drains will be addressed.

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 2. Fire detection and suppression addressed in Fire Protection System ITAAC.

GE DISPOSITION: All ABWR fire protection and suppression features that merit Tier 1 treatment are addressed in entry 2.14.6 Fire Protection System.

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 3. Secondary containment leakage control addressed in SGTS ITAAC (2.14.4).

GE DISPOSITION: The SGTS ITAAC addressed secondary containment leakage control.

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 4. Fire dampers and other smoke control features addressed in HVAC ITAAC (2.15.5).

GE DISPOSITION: The HVAC ITAAC addressed these issues.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 5. Load drop considerations addressed by Cranes and Hoists ITAAC (2.15.3).

GE DISPOSITION: Load drop considerations will be addressed in the Cranes and Hoists Tier 1 entry 2.15.3. The approach will be procedural in that acceptable load paths will be defined and load movements constrained within thse acceptable paths.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 6. Add to SSAR that water-tight doors have indication in control room.

GE DISPOSITION: This material has been added to the SSAR.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 7. Add ramps/sills to drawings for flood protection.

GE DISPOSITION: As a result of a review of the reactor building flood protection features, GE has concluded that no ramps and sills are now required to provide adequate equipment protection and flooding characteristics in the reactor building. Consequently, no ramps or sills have been added to the drawings in either the SSAR or Tier 1.

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 8. Add to SSAR how wall thickness will be measured.

GE DISPOSITION: This material will be added to the SSAR.

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 9. Add to SSAR scope of structural reconciliation and description of structural analysis report.

GE DISPOSITION: A description of the structural analysis report will be included in the SSAR.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 10. GE will revisit internal flooding ITAAC.

GE DISPOSITION: GE has completed a review of the reactor building flood analysis. GE believes the flooding event ITAAC as originally defined is still valid, and this item will be included in the revised reactor building entry.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 11. GE will check that GSIs properly characterize water-proofing of external walls.

GE DISPOSITION: GE has been unable to identify the GSI relating to waterproofing of external walls. This issue is being pursued with staff and when an understanding is in place, GE will provide a disposition.

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 12. Add to SSAR flood protection features for penetrations.

GE DISPOSITION: This material will be added to the SSAR.

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 13. Add to Design Description and ITAAC Building embedment for Seismic Category I buildings.

GE DISPOSITION: The revised reactor building Tier 1 material includes an ITAAC entry addressing embedment depth.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 14. GE to resolve the definition of radiation levels for EQ. Resolution: Tie to radiation DAC. [Methods only -- no values.]

GE DISPOSITION: Following the March GE/NRC review of building ITAAC, it was concluded:

1. Development of radiation field for equipment qualification should not be included in the radiation DAC.
2. Methods for developing radiation field for equipment qualification should not be included in the SSAR.
3. The SSAR should be modified to include specific numerical values of radiation levels for equipment qualification as applied to safety-related equipment.
4. Tier 1 should remain silent on numerical values of radiation for equipment qualification.

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 15. Check SAR to confirm the Reactor Building water-tight door sensors are called out in the Reactor Building section -- not the Security section.

GE DISPOSITION: It has been confirmed that the need for water-tight door sensors in the reactor building is called out in the reactor building section of the SAR and not in the security section.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 16. GE to assess the need for an ITAAC/Tier 1 on the stack.

GE DISPOSITION: GE has reviewed this issue and concluded that no Tier 1/ITAAC entries are required for the stack on the reactor building. The bases for this conclusion are:

1. Radiological evaluations of design basis events do not utilize a elevated stack release.
2. The stack is not safety-related.
3. There is a COL action item to provide a stack that is consistent with the site-specific evaluation of radiological consequences.

In summary, GE does not intend to include Tier 1 ITAAC entries for the reactor building stack.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.15.10 Reactor Building

PUNCH LIST ITEM: 17. The CB Tier 1 must pick up the issue of no connections between the steam tunnel and the CB.

GE DISPOSITION: This issue is addressed in the revised version of the Control Building ITAAC/Tier 1.

2.15.11 Turbine Building

Design Description

The Turbine Building (T/B) consists of a non-safety-related turbine and electrical building that is located adjacent to the safety-related Seismic Category I Control Building. The T/B houses the main turbine generator and other power conversion cycle equipment and auxiliaries. With the exception of instrumentation associated with monitoring of condenser pressure, turbine first-stage pressure, turbine control valve oil pressure and stop valve position, there is no safety-related equipment in the T/B. The electrical building houses various plant support systems and equipment such as non-divisional switchgear and chillers.

Flood conditions in the T/B are prevented from propagating into the Control Building (C/B) via the Service Building. This is achieved by locating the access from the T/B to the S/B at or above grade level and providing a flood control doorway at the access location.

The T/B is not classified as a Seismic Category I structure. However, the building is designed to not collapse under seismic loads corresponding to the safe shutdown earthquake (SSE) ground acceleration.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.11 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria which will be undertaken for the Turbine Building.

Table 2.15.11 Turbine Building

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the T/B is described in Section 2.15.11.	1. Inspections of the as-built structure will be conducted.	1. The as-built T/B conforms with the basic configuration described in Section 2.15.11.
2. The T/B does not collapse under seismic loads.	2. A seismic analysis of the T/B will be performed	2. A structural analysis report exists which concludes building collapse does not occur.

SYSTEM: 2.15.11 Turbine Building

PUNCH LIST ITEM: 1. Condenser pit flooding detectors are covered in Circulating Water System ITAAC.

GE DISPOSITION: ITAAC entry 2.10.23 Circulating Water System identifies that condenser pit flood detectors will trip the circulating water in the event of a flooding condition.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.15.11 Turbine Building

PUNCH LIST ITEM: 2. Turbine orientation is addressed in Turbine ITAAC.

GE DISPOSITION: Tier 1 entry 2.10.7 Main Turbine addresses the issue of turbine orientation.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.15.11 Turbine Building

PUNCH LIST ITEM: 2. Turbine orientation is addressed in Turbine ITAAC.

GE DISPOSITION: Tier 1 entry 2.10.7 Main Turbine addresses the issue of turbine orientation.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.15.11 Turbine Building

PUNCH LIST ITEM: 3. Add description of seismic analysis and structural analysis report to SSAR.

GE DISPOSITION: This item will be included at the next revision of the SSAR.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 2.15.11 Turbine Building

PUNCH LIST ITEM: 4. Add separation between Control Building and Turbine Building to SSAR.

GE DISPOSITION: This item will be included in the next revision of the SSAR.

3.3 Piping Design

Design Description

Piping associated with hydraulic and pneumatic systems is categorized as either nuclear safety-related (i.e., Seismic Category I) or non-nuclear safety (NNS) related (i.e., non-Seismic Category I). The piping shall be designed for a design life of 60 years. Piping systems that must remain functional during and following a safe shutdown earthquake (SSE) are designated as Seismic Category I and are further classified as ASME Code Class 1, 2 or 3. Unless otherwise specified in this description, piping systems means nuclear safety-related piping systems. Piping systems and their components are designed and constructed in accordance with the ASME Code requirements identified in the individual system Design Descriptions.

Piping systems shall be designed to meet their ASME Code class and Seismic Category requirements. The ASME Code Class 1, 2 and 3 piping systems shall be designed to retain their pressure integrity and functional capability under internal design and operating pressures and design basis loads. Piping stresses due to static and dynamic loads shall be combined and calculated in accordance with the ASME Code and shall be shown to be less than the ASME Code allowables for each service level.

For ASME Code Class 1 piping systems, a fatigue analysis shall be performed in accordance with the ASME Code Class 1 piping requirements. Environmental effects shall be included in the fatigue analysis. The Class 1 piping fatigue analysis shall show that the ASME Code Class 1 piping fatigue requirements have been met.

For ASME Code Class 2 and 3 piping systems, piping stress ranges due to thermal expansion shall be calculated in accordance with the ASME Code Class 2 and 3 piping requirements. The piping stress analysis shall show that the ASME Code Class 2 and 3 piping thermal expansion stress range requirements have been met. For the ASME Code Class 2 and 3 piping systems and their components which will be subjected to severe thermal transients, the effects of these transients shall be included in the design.

Feedwater lines shall be designed for thermal stratification loads.

Piping systems shall be designed to minimize the effects of erosion/corrosion.

For those piping systems using ferritic materials as permitted by the design specification, the ferritic materials shall not be susceptible to brittle fracture under the expected service conditions.

For those piping systems using austenitic stainless steel materials as permitted by the design specification, the stainless steel piping material and fabrication process shall be selected to reduce the possibility of cracking during service. Chemical, fabrication, handling, welding, and examination requirements that minimize cracking shall be met.

Piping system supports shall be designed to meet the requirements of ASME Code Subsection NF.

For piping systems, the pipe applied loads on attached equipment shall be calculated and shown to be less than the equipment allowable loads.

Analytical methods and load combinations used for analysis of piping systems shall be referenced or specified in the ASME Code Certified Stress Report. Piping systems and their supports shall be mathematically modeled to provide results for piping system frequencies up to the analysis cut-off frequency. Computer programs used for piping system dynamic analysis shall be benchmarked.

Systems, structures and components that shall be required to be functional during and following an SSE, shall be protected against the dynamic effects associated with postulated high energy pipe breaks. The Pipe Break Analyses Report shall specify the criteria used to postulate breaks and the analytical methods used to perform the pipe break analysis. For postulated pipe breaks, the Pipe Break Analysis Report shall confirm: (1) piping stresses in the containment penetration area shall be within their allowable stress limits, (2) pipe whip restraints and jet shield designs shall be capable of mitigating pipe break loads, and (3) loads on safety-related systems, structures and components shall be within their design loads limits. Piping systems that shall be qualified for leak-before-break design may exclude design features to mitigate the dynamic effects from postulated high energy pipe breaks.

Piping systems shall be designed to provide clearance from structures, systems, and components where necessary for the accomplishment of the structure, system, or component's safety function as specified in the respective structure or system Design Description.

The as-built piping shall be reconciled with the piping design required by this section.

Inspections, Tests, Analyses and Acceptance Criteria

Table 3.3 provides a definition of the inspections, tests, analyses, and associated acceptance criteria, which will be performed for ABWR nuclear safety-related and NNS related piping systems as specified in each system's Design Description. Table 3.3 may be completed on an individual system basis.

Table 3.3 Piping Design

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The piping system shall be designed to meet its ASME Code Class and Seismic Category requirements. The ASME Code Class 1, 2, and 3 piping system shall be designed to retain its pressure integrity and functional capability under internal design and operating pressures and design basis loads. Piping and piping components shall be designed to show compliance with the requirements of ASME Code, Section III.	1. Inspections of ASME Code required documents will be conducted.	1. An ASME Code Certified Stress Report exists.
2. Systems, structures, and components, that are required to be functional during and following an SSE, shall be protected against the dynamic effects associated with postulated high energy pipe breaks. Piping systems that are qualified for leak-before-break design may exclude design features to mitigate the dynamic effects from postulated high energy pipe breaks.	2. Inspections of the Pipe Break Analysis Report, or Leak-Before-Break Report, will be conducted. An inspection of the as-built high energy pipe break mitigation features will be performed.	2. A Pipe Break Analysis Report or Leak-Before-Break Report exists. This report includes documentation of the results of inspections of high energy pipe break mitigation features.
3. The as-built piping shall be reconciled with the piping design required in Section 3.3.	3. A reconciliation analysis using the as-designed and as-built information will be performed.	3. An as-built stress report exists. For ASME Code Class piping, the as-built stress report includes the ASME Code Certified Stress Report and documentation of the results of the as-built reconciliation analysis.

3.4 Instrumentation and Control

The following are the proposed 3.4 entries on:

Instrument Setpoint Methodology

Equipment Qualification

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 which 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 which 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)

Instrumentation and control equipment that is safety-related is qualified for the full range of environmental conditions that will exist up to and including the time the equipment has finished performing its safety-related function.

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 by one of the following means:

a. 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.

b. Testing of a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable.

c. Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

d. Analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

Inspections, Tests, Analyses and Acceptance Criteria

Table 3.4 provides a definition of the inspections, tests and analyses, together with associated acceptance criteria, which will be used to:

a. Assure procedures are defined and implemented for establishing instrument setpoints in a structured, disciplined manner.

b. Assure that an equipment qualification program has been established.

Table 3.4: Instrumentation and Control

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Setpoint Methodology		
10. 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.	10. Inspections will be performed of the setpoint methodology plan used to determine, document, install, and maintain instrument setpoints.	10. The setpoint methodology plan is in place. The plan generates requirements for: • Documentation of data, assumptions, and methods used in the bases for selection of trip setpoints. • 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. • The methods used for combining uncertainties. • Use of written procedures for preoperational testing and tests performed to satisfy the Technical Specifications. • Documented evaluation for equivalent or better performance of replacement instrumentation which is not identical to the original equipment.

Table 3.4: Instrumentation and Control

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Setpoint Methodology		
11. Procedures to establish setpoints allow for different approaches based on different types of instrumentation and the associated sensed dynamic processes.	11. Inspections will be performed of the methodologies used to determine, document, install, and maintain instrument setpoints.	11. The setpoint methodology process provides for different procedures depending on the type of information available about the instruments and processes involved.

Table 3.4: Instrumentation and Control

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 during the time to completion. 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. These conditions include: a. number and /or duration of equipment functional and test cycles/events b. process fluid conditions (where applicable) c. voltage, frequency, load, and other electrical characteristics of the equipment d. dynamic loads associated with seismic events e. containment response to hydrodynamic conditions f. system transients and other vibration inducing events g. pressure, temperature, humidity h. chemical and radiation environments i. aging j. submergence (if any)	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 temperature, pressure, humidity, radiation, electromagnetic compatibility, chemicals, and submergence and environmental conditions defined in 10 CFR 50.49 for electrical items. Also included is consideration of synergistic effects and margins for unquantified uncertainty. c. One (or a combination) of the following testing methods used to qualify the equipment:

Table 3.4: Instrumentation and Control

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment *Equipment Qualification*	Inspections, Tests, Analyses	Acceptance Criteria
12. (continued)	12. (continued)	12. (continued) (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. d. Documented results of the qualification that show the equipment: (1) is qualified for its application, and

Table 3.4: Instrumentation and Control

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment *Equipment Qualification*	Inspections, Tests, Analyses	Acceptance Criteria
12. (continued)	12. (continued)	12. (continued) (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.
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.

4.2 Offsite Power System

Interface Requirements

Covered in Section 2.12.1.

4.5 Reactor Service Water System

Interface Requirements

Covered in Section 2.11.9.

5.0 Site Parameters

Table 5.0 and Figures 5.0a and 5.0b provide a definition of the site parameters used as the basis for the certified design.

Table 5.0 ABWR Site Parameters

Maximum Ground Water Level:
61.0 cm below grade

Maximum Flood (or Tsunami) Level:
30.5 cm below grade

Precipitation (for Roof Design):
- Maximum rainfall rate: 49.3 cm/hr[3]
- Maximum snow load: 0.024 kg/cm^2

Design Temperatures:
- Ambient
  1% Exceedance Values
  Maximum: 37.8°C dry bulb/25°C wet bulb (coincident), 26.6°C wet bulb (non-coincident)
  Minimum: -23.3°C
  0% Exceedance Values (Historical Limit)
  Maximum: 46.10°C dry bulb/26.7°C wet bulb (coincident), 27.2°C wet bulb (non-coincident)
  Minimum: -40°C

Extreme Wind: Basic Wind Speed:
177 km/hr[1]/209 km/hr[2]

Tornado
- Maximum pressure drop: 0.141 kg/cm^2d
- Missile spectra: Spectrum I[4]

Soil Properties:
- Minimum static bearing capacity: 7.32 kg/cm^2
- Minimum shear wave velocity: 305 m/sec[5]
- Liquification potential: None underneath structures, systems and components resulting from the site specific SSE

Seismology:
- SSE response spectra: see Figs. 5.0a and 5.0b[6]

(1) 50-year recurrence interval; value to be utilized for design of non-safety-related structures only.

(2) 100-year recurrence interval; value to be utilized for design for safety-related structures only.

(3) Maximum value for 1 hour over 2.6 km^2 probable maximum precipitation (PMP) with ratio of 5 minutes to 1 hour PMP of 0.32. Maximum short term rate: 15.7cm/5 min.

(4) Spectrum I missiles consist of a massive high kinetic energy missile which deforms on impact, a rigid missile to test penetration resistance, and a small rigid missile of a size sufficient to just pass through any openings in protective barriers. These missiles are an 1800 kg automobile, a 125 kg, 20 cm nominal diameter armor piercing artillery shell, and a 2.54 cm solid steel sphere, all impacting at 35% of the maximum horizontal windspeed of the design basis tornado. The first two missiles are assumed to impact at normal incidence, the last to impinge upon barrier openings in the most damaging directions.

(5) This is the minimum shear wave velocity at low strains after the soil property uncertainties have been applied.

(6) Free-field, at plant grade elevation.

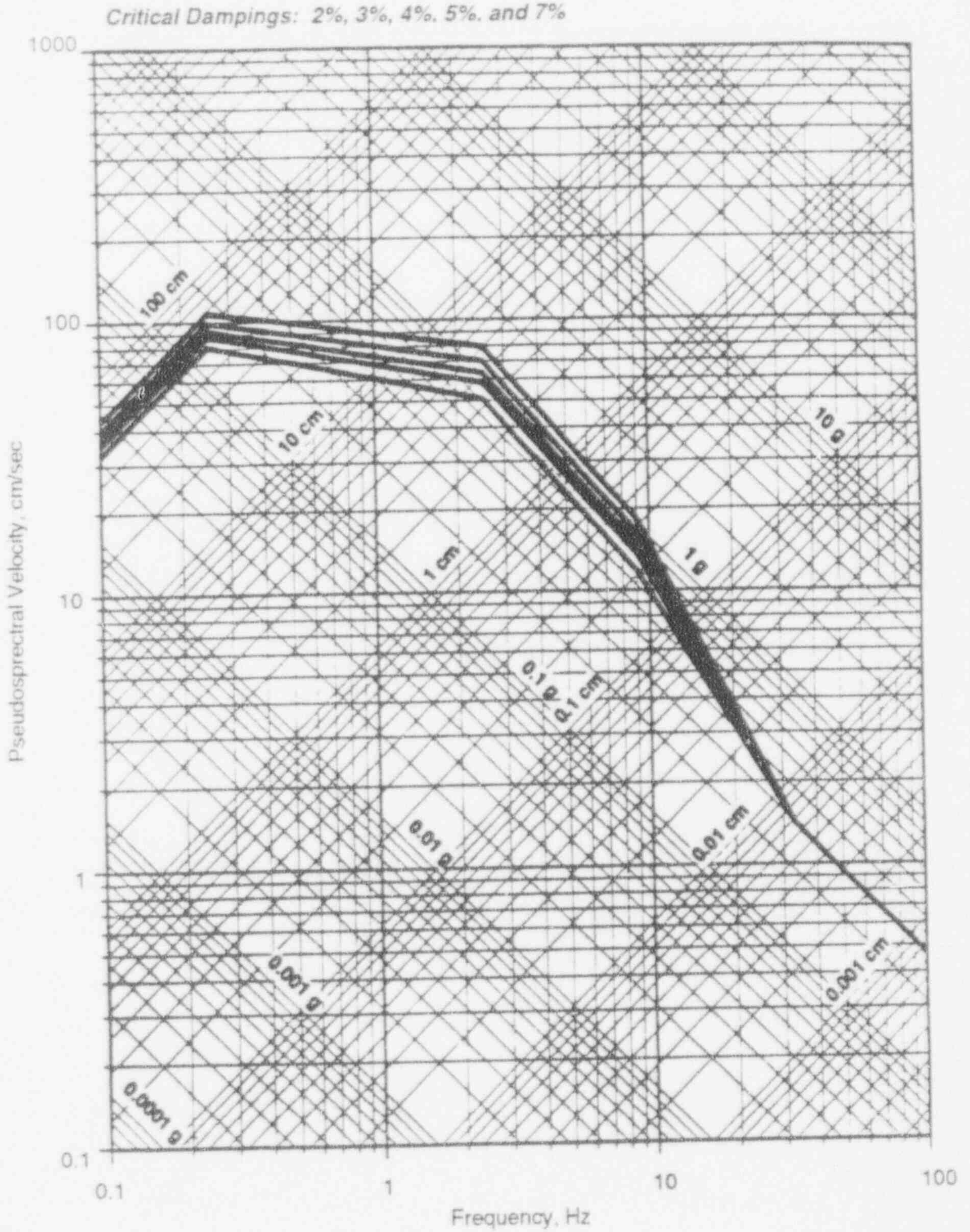


Figure 5.0a Horizontal Safe Shutdown Earthquake Design Spectra

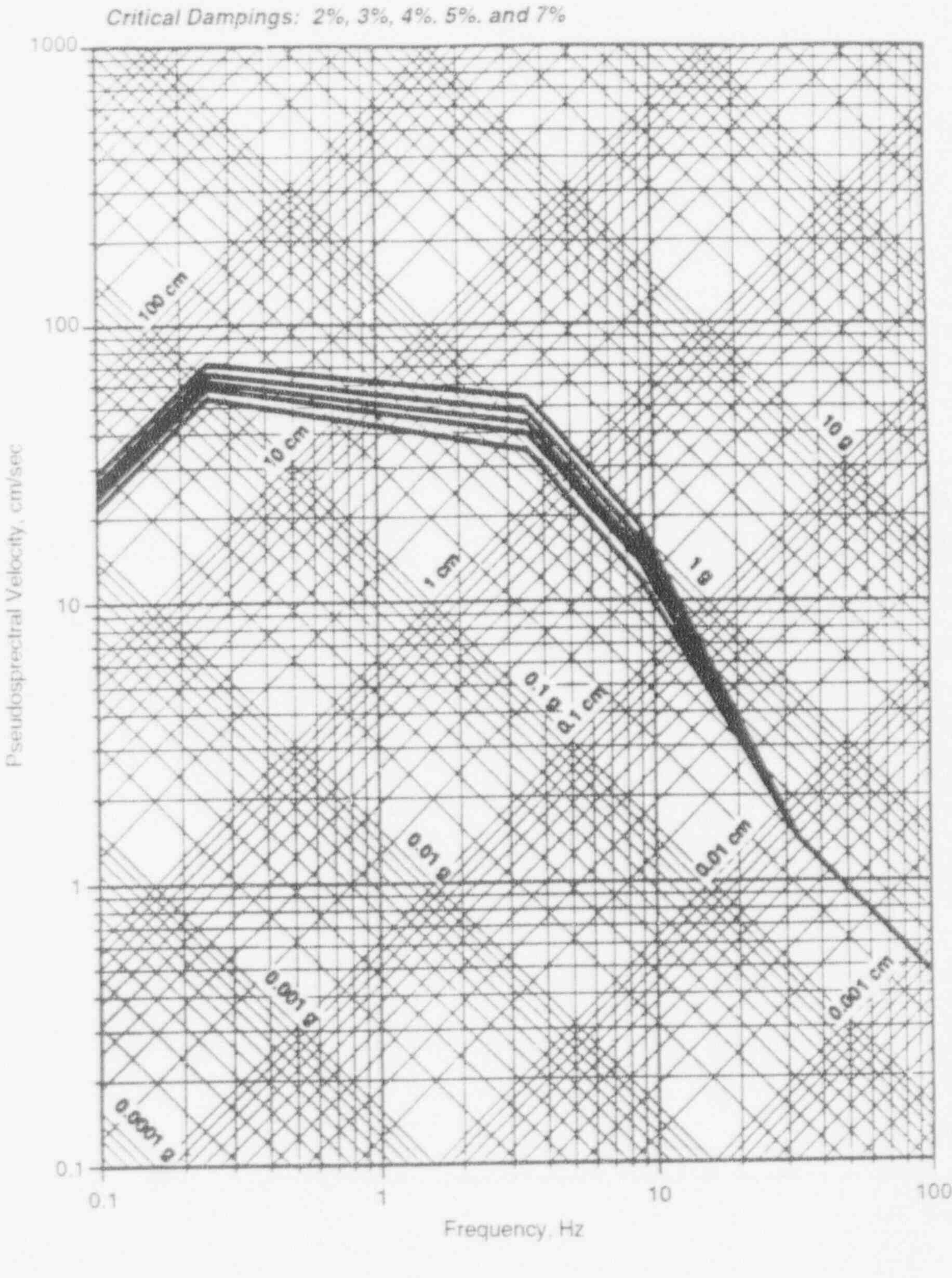


Figure 5.0b Vertical Safe Shutdown Earthquake Design Spectra

Application of Atmospheric χ/Q Values to ITAAC

Purpose: To discuss the potential listing of χ/Q values in the ABWR Tier 1 Site Parameters section. (Section 5.0)

Recommendation: χ/Q values should not be included in the ABWR Tier 1 Site Parameters section.

Discussion: Two specific reasons are discussed below as to why χ/Q values should not be included in the ABWR Tier 1 material.

- χ/Q values are a derived time integrated physical parameter used to simulate atmospheric dispersion. As such they are used as a multiplier on releases to the atmosphere to determine either the airborne concentration or ground contamination for dose model input. As such they are not the final point in the analysis but an intermediate step. The final result of any calculation involving χ/Q values is the dose commitment. Since the calculation of concentration or dose commitment is a requirement of 10CFR20; 10CFR50, Appendix I; and 10CFR100, it is not necessary for the NRC to establish another requirement (on χ/Q values), since that requirement is superfluous to the existing regulations.

- It is not clear how a χ/Q value could be defined in a Tier 1 in a way which would be both useful and not confusing. There are three primary types of χ/Q values which are established and used in the ABWR safety evaluations: (1) annual average, (2) short term design basis accident, and (3) 30 day design basis accident values. Each of the three types of χ/Q values will be discussed below.

  1. The site annual average χ/Q value is a number which is used to evaluate annual exposure to the public for normal releases. It is used for determination of compliance with 10CFR20 for airborne concentrations and 10CFR50, Appendix I for offsite dose. In the ABWR SSAR the number 2.0×10^{-6} sec/m^3 is used to generate the concentration and dose tables in Chapter 12. This value is however not a maximum but was based upon a statistical analysis for the standard design assuming specific values for plant parameters such as HVAC flow, outside air temperature, stack diameter, etc. It is possible to back calculate a χ/Q value which would result in the maximum dose or concentration whichever is larger and this value could be considered for Tier 1 treatment. However, it is not clear whether such a value would be usable. This is because given a specific site, the ABWR design could be and most probably would be changed within limits such that a single number is not usable. As an example, the ABWR standard design employs a single plant stack on the reactor building to which contaminated air spaces in the reactor building, turbine building, and radwaste building are piped for release. However, the site topography might best be served by a design using two stacks, one on the reactor building and a second on the radwaste building for instance. Such a design would be a minor perturbation on the total plant design but would

clearly not be amenable to a single annual average χ/Q value since to design both stacks to the same value would potentially mean an overdesign in the case the effluent release from the radwaste stack which is a small source of the plant effluents. The evaluation of compliance to the above regulations is a complex interaction of the source, the release characteristics, and the local topography in addition to the meteorology which cannot be presented by a single number and as can be clearly demonstrated in this example not easily adaptable to a limiting Tier 1 application.

2. The evaluation of short term χ/Q values for use in design basis analysis is clearly the most easily definable parameter consistent. The χ/Q values are calculated to the site boundary with Regulatory Guide 1.145 in accordance with standard procedures assuming a zero height of release plume over a two hour period. A single number is produced which is directly comparable to an equivalent number given for each of the accidents in Chapter 15 of the SSAR and in fact each of the design basis accidents in Chapter 15 provides the maximum acceptable χ/Q value for that accident for licensing purposes. Therefore all the COL need do is compare the site 2-hour χ/Q value to the maximum value found in Chapter 15 for determination of compliance with 10CFR100 site boundary dose limitations.

3. The 30 day design basis accident χ/Q value is however another situation not easily amenable to such a simple solution. The 30 day χ/Q values are typically evaluated for time periods of 0-8 hours, 8hours to 1 day, 1 day to 4 days, and 4days to 30 days. These values which are based upon a zero height release and evaluated to the site low population zone boundary are used as multipliers on the releases to determine the offsite dose. The problem is in trying to develop bounding χ/Q values for these four time periods. It is not clear how to apportion what fraction of the total dose to each time period such that the total integrated dose meets the regulatory requirements ? As an example the table below apportions one quarter of the dose to each time period, yet in all examples given, the site χ/Q values exceed the limiting χ/Q values yet meet the regulatory requirement which in this case is normalized to one (Total).

Time Period	0-8hrs	8hr-1day	1-4days	4-30 days	Total
Limiting χ/Q	0.25	0.25	0.25	0.25	1.0
Site 1	0.30	0.20	0.40	0.10	1.0
Site 2	0.10	0.10	0.10	0.70	1.0
Site 3	0.60	0.20	0.10	0.10	1.0

Again the question of how to build time integrated χ/Q values is not really a question of what the values themselves are but what is the final integrated dose which is a requirement of 10CFR100 in this case. The χ/Q values are only a computational representation of a complex meteorological interaction and should not be the end point in any licensing evaluation.

This short list above does not include either control room χ/Q values or any meteorological evaluations necessary for PRA analysis. Control room calculations are extremely dependent on micro-climatology and the interactions of the site design with the site topography and not amenable to easy resolution. The argument on the 30 day design basis accident χ/Q values given above is also applicable to the control room calculations. With respect to PRA evaluations, this type of analysis requires not χ/Q values but hourly meteorological data of the form of wind direction, wind speed, precipitation, and stability assessments which would certainly be beyond the scope of Tier 1 treatment since the generation of an hourly meteorological data base which would be backward calculable from the NRC Safety Goals Policy statement is physically impossible.

Summary: In summary, GE does not believe it is either necessary or appropriate to include χ/Q values in the Site Parameters section of the ABWR Tier 1 material. The basis for this position are:

1. There is not clearly definable set of standardized χ/Q values that a specific site must meet in order for the plant/site combination to be in compliance with offsite dose limits.

2. Acceptable χ/Q values are dependent upon the detailed design of the plant.

3. The SSAR contains the full compliment of commitments for compliance with regulatory requirements applicable to site radiological issues.

(93/abwr003.r)

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
<u>FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS</u>

SYSTEM: 5.0 Site Parameters

PUNCH LIST ITEM: 1. Delete from Tier 1 if GE confirms that these values are not used for any safety-related analyses or safety-significant equipment sizing.

GE DISPOSITION: GE has concluded that this data should remain in the 5.0 Site Parameters entry. This data is used for HVAC design.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 5.0 Site Parameters

PUNCH LIST ITEM: 2. GE to determine if CHI/Q values are legitimate site parameters and add to table if they are.

GE DISPOSITION: GE has reviewed this issue and concluded that CHI/Q values should not be included in the Tier 1 site parameters table. The bases for this conclusion are provided on the material attached to the revised Table 5.0.

SYSTEM: 5.0 Site Parameters

PUNCH LIST ITEM: 3. Rewrite DD.

GE DISPOSITION: Complete. The Design Description is now limited to a simple statement that the attached table summarizes the site parameters for the certified design.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 5.0 Site Parameters

PUNCH LIST ITEM: 4. Add back in liquefaction (see markup).

GE DISPOSITION: Complete.

SYSTEM: 5.0 Site Parameters

PUNCH LIST ITEM: 5. Add toxic gas item to CR Habitability System as an interface item.

GE DISPOSITION: The Revised Control Room Habitability System HVAC entry will include this as an interface item.

SYSTEM: 5.0 Site Parameters

PUNCH LIST ITEM: 6. In SSAR, fix the incorrect value of tornado pressure rate.

GE DISPOSITION: The SSAR is being corrected.

SYSTEM: 5.0 Site Parameters

PUNCH LIST ITEM: 7. Fix units on response spectra figures.

GE DISPOSITION: Completed.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 5.0 Site Parameters

PUNCH LIST ITEM: 8. Fix SSAR on tornado probability. Just say what design basis tornado conditions are -- delete ("destroy," to use Russell's word) any discussion of recurrence intervals.

GE DISPOSITION: The SSAR is being modified to reflect this agreement.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 5.0 Site Parameters

PUNCH LIST ITEM: 9. Confirm that the tornado missile design basis (in SAR) bounds a general aviation plane crash. (It is not clear what we do if this is not confirmed.)

GE DISPOSITION: The ABWR design is based on the assumption that a general aviation plane crash is of sufficiently low probability (10^{-7}) that it need not be considered in the plant design basis. In other words, this punch list item was not confirmed. Since this issue has not been identified by the NRC staff as a SAR Open Item, GE believes no further action is required.

ABWR DESIGN CERTIFICATION: DISPOSITION OF PUNCH LIST ITEMS
FROM THE JANUARY AND MARCH 1993 GE/NRC TIER 1/ITAAC REVIEWS

SYSTEM: 5.0 Site Parameters

PUNCH LIST ITEM: 10. GE to evaluate if CHI/Q data is appropriate for Tier 1 site parameter treatment.

GE DISPOSITION: Duplicate of issue in punch list item number 2.

APPENDIX A Legend for Figures

For a number of the systems presented in Section 2, figures depicting the Basic Configuration of the systems have been provided to help facilitate the Design Description. For I&C systems, the figures represent a diagram of significant aspects of the logic of the system. For other systems and buildings, these figures represent a functional diagram, representation, or illustration of design-related information. Unless otherwise specified explicitly, these figures are not necessarily indicative of the scale, location, dimensions, shape, or spatial relationships of as-built structures, systems, and components. In particular, the as-built attributes of structures, systems and components may vary from the attributes depicted on these figures, provided that those safety functions discussed in the Design Description are not adversely affected.

The figures contain information that uses the following conventions:

Mechanical Equipment

Line classification:

		Figure Designation
ASME Code Class 1		1
ASME Code Class 2		2
ASME Code Class 3		3
Non-ASME Code		NNS

Classification/System Boundaries:

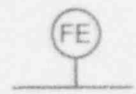
The following is a self-explanatory example of how Code class change and system boundary are identified on the figures:

RPV | NBS 1 | 2

Other line type:

Instrumentation:

Differential pressure indicator	dP
Flow element	FE
Level detector	L
Moisture element	ME
Pressure element	P
Display and/or control interface with RSS	R
Radiation element	RE
Level controller	LC
Speed detector	S
Temperature element	T
Vibration detector	V

Instrumentation should be shown as: 

Equipment:

Gate valve		Plug or Ball valve	
Globe valve		Butterfly valve	
Three way valve		Damper	
Check valve		Vacuum breaker	VB
Valve type not specified		Strainer	S
Relief valve		Filter	F
Annunciator (H=high, L=low)		Flow restrictor	
Solenoid		Water trap	TR

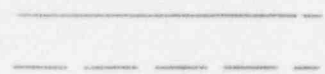
Note: Valves shown do not denote either open or closed position.

Valve Operators:

Motor	M
Pneumatic	P

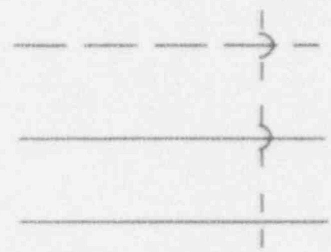
Electrical Equipment

Cable or conduit

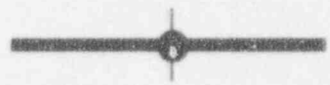


Cable connection

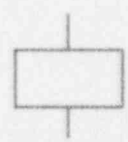
Cables not connected



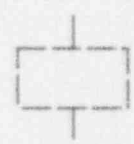
Connection to bus



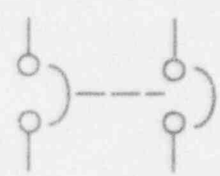
Main power breaker



Racked out breaker



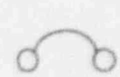
Interlocked circuit breakers



Disconnect device

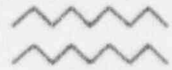
Phone set

Patching cord



Circuit breaker

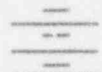
Key lock interlock

Transformer 

Main Generator 

Combustion Turbine Generator 

Diesel Generator (* Number indicates division or non-divisional if absent) 

Battery 

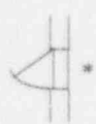
Note: Devices shown do not denote either open or closed position.

Building

Typical floor designation: B3F-Basement, 3rd floor
M4F-Mezzanine, 4th floor

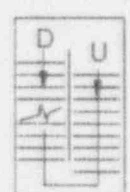
Door (Note 1) 

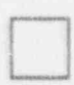
Hatch 

Watertight door 

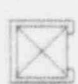
Removable block wall 

Sliding door 

Stairway 

Sump pit 

Grating Floor 

Elevator 

Divisional Barrier (Note 2)

Opening		Secondary containment barrier	
Column line identifier (for information only)	RA	Column line intersection	+

NOTES:

1. Swing of door can be either way.
2. Divisional and secondary containment barriers are fire barriers unless specified otherwise

Control and Instrumentation

Cables:	Fiber-optic	—— – —— –
	Metallic	————