February 11, 1993

Docket No. 52-001

Mr. Patrick W. Marriott, Manager
Licensing & Consulting Services
GE Nuclear Energy
175 Curtner Avenue
San Jose, California 95125

Dear Mr. Marriott:

SUBJECT: FORWARDING MEETING SUMMARY DATED JANUARY 27, 1993

Enclosed is the summary of a meeting held between the staff and GE on January 11 through 21, 1993, to discuss inspections, tests, analyses, and acceptance criteria.

Sincerely,

original signed by: Jerry N. Wilson

Thomas H. Boyce, Project Manager
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of Nuclear Reactor Regulation

Enclosure:
Meeting Summary
dtd 01/27/93

cc w/enclosure:
See next page

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GE Nuclear Energy

Docket No. 52-001

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
January 27, 1993

APPLICANT: GE NUCLEAR ENERGY (GE)

DOCKET #: 52-001

SUBJECT: SUMMARY OF MEETING WITH GE TO DISCUSS INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA (ITAAC)

The NRC staff met with the General Electric Company, representatives of the Department of Energy, and representatives of the Nuclear industry from January 11 through January 21, 1993, to discuss the development of ITAAC as part of the certification process.

Enclosure 1 is a list of attendees.

Enclosure 2 contains the draft of an Introduction (Section 1.0) and also includes a set of definitions (Section 1.1) for use in conjunction with the Design Descriptions and ITAAC. This information reflects the technical agreements reached during the meeting, but it is anticipated that additional review by OGC and others may lead to the need for some wording changes.

Enclosure 3 includes a draft of each of the ten example systems discussed during the meeting. These included:

Residual Heat Removal
Nuclear Boiler
Leak Detection and Isolation
Reactor Cooling Water
Reactor Service Water
Reactor Water Cleanup
Standby Liquid Control
Turbine Main Steam
Condenser
Control Room Habitability Area HVAC

In general, these systems were chosen because the designs (and SSAR material) were considered to be relatively complete. The notable exception was the Control Room Habitability Area HVAC system and as a result significant rework is required for this Design Description and ITAAC. It is also recognized that the completion of other activities, such as the PRA insights, may impact the Design Descriptions and ITAAC.

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Approaches for a number of common or generic concerns were also developed. Enclosure 4 includes summaries of the proposed approaches for the following "generic" concerns:

Welding
Environmental Qualification
Seismic Qualification
Verification of Motor-operated Valves
EMI/RFI issues for I&C
Instrument Setpoints
Electrical Independence (Separation)
Piping

In conjunction with the conclusion of the ITAAC meetings, a Senior Management Meeting was held on January 21, 1993 at the GE offices in San Jose to summarize the overall ITAAC review process and other matters related to the ABWR. (For further information, see the minutes of that meeting.) It was decided that the team would develop staff review guidance from the lessons learned during the ITAAC meetings. GE and the industry are expected to also develop their lessons learned. A followup meeting at NRC headquarters is planned for the week of February 1, 1993 to discuss the results of these efforts.

In addition, the need for developing an example of a ITAAC for a building was identified. It was decided to discuss this further during the upcoming meeting.

Original signed by

Conrad E. McCracken, Chief
Plant Systems Branch
Division of Systems Safety
and Analysis

Enclosures:
As stated

Enclosure 1

GE/NRC ITAAC REVIEWS

ATTENDEES

J. LYONS	NRC	S. FRANKS	DOE
H. WALKER	NRC	N. FLETCHER	DOE
W. BURTON	NRC	S. FRANTZ	NEWMAN & HOLTZINGER (GE)
G. THOMAS	NRC	A. P. HEYMER	NUMARC
D. TERAO	NRC	D. WILSON	NIAGARA MOHAWK/EPRI
T. SULLIVAN	NRC	J. REC	ABB-CE
R. LI	NRC	E. WHITAKER	TVA/EPRI
D. THATCHER	NRC	W. L. ZIMMERMANN	AEP
J. STEWART	NRC	D. ANTOLOVIC	W
M. CHIMARMAL	NRC	J. WALDRON	CEI/SBWR
T. POLICH	NRC	J. WHELESS	SOUTHERN CO.
T. BOYCE	NRC	A. J. JAMES	GE
J. WILSON	NRC	R. LOUISON	GE
R. GRAMM	NRC	N. HACKFORD	GE
S. MALUR	NRC	J. CHAMBERS	GE
W. RUSSELL	NRC	J. F. QUIRK	GE
M.FINKELSTEIN	NRC		
C. MCCRACKEN	NRC		

PLUS SUPPORTING GE TECHNICAL PERSONNEL AS NEEDED.

AJJ-4A
1/21/93

GE/NRC ITAAC REVIEWS

SUPPORTING GE TECHNICAL PERSONNEL

SYSTEM/ISSUE	GE ENGINEER(S)
LEAK DETECTION AND ISOLATION SYSTEM	N. G. TOTAH
REACTOR WATER CLEANUP SYSTEM	E. V. NAZARENO
NUCLEAR BOILER SYSTEM	J. K. SAWABE F. E. COOKE
STANDBY LIQUID CONTROL SYSTEM	P. F. BILLIG
REACTOR COOLING WATER SYSTEM	G. E. MILLER
REACTOR SERVICE WATER SYSTEM	G. E. MILLER
RESIDUAL HEAT REMOVAL SYSTEM	W. E. TAFT
TURBINE MAIN STEAM SYSTEM	J. C. BLACK
CONDENSER	J. C. BLACK
CONTROL ROOM HABITABILITY AREA HVAC SYSTEM	M. MUNSON
CONTROL BUILDING SAFETY-RELATED EQUIPMENT AREA HVAC SYSTEM	M. MUNSON
WELDING	L. FINNEY D. SANDUSKY

AJJ-4B
1/21/93

GE/NRC ITAAC REVIEWS

SUPPORTING GE TECHNICAL PERSONNEL

SYSTEM/ISSUE	GE ENGINEER(S)
EQUIPMENT QUALIFICATION (50.49)	N. G. LURIA D. C. RENNELS
SEISMIC/DYNAMIC QUALIFICATION	N. G. LURIA D. C. RENNELS
MOV (89-10 ISSUES)	B. GENETTI G. L. MOORE
EMI/RFI/SWC	B. H. SIMON
I&C ENVIRONMENTAL QUALIFICATION	B. H. SIMON
INSTRUMENT SETPOINTS	A. J. JAMES
SECTION 1 OF THE CERTIFIED DESIGN DOCUMENT	A. J. JAMES S. FRANTZ (LEGAL)
PIPING DESIGN ACCEPTANCE CRITERIA	J. B. KNEPP M. HERZOG E. O. SWAIN
ELECTRICAL SEPARATION	C. F. CHRISTENSEN

AJJ-4C
1/21/93

1.0 Introduction

The purpose of this document is to present the certified design for the Advanced Boiling Water Reactor (ABWR).

The certified design consists of:

(1) A design description, design commitments and associated inspections, tests, analyses and acceptance criteria (ITAAC);

(2) The site parameters upon which the design is based; and

(3) Interface requirements.

The certified design does not include all of the material submitted as part of design certification application. Where a conflict exists between the Standard Safety Analysis Report for the ABWR and the certified design, the certified design is controlling.

This document is structured as follows:

Section 1.1—Definitions.

Section 2.0—A design description, ITAAC and interface requirements for systems, structures and components within the ABWR certified design.

Section 3.0—A design description, ITAAC and interface requirements for systems, structures and components for which the design process and selected design features are being certified.

Section 4.0—Identifies the interface requirements to be met by those portions of the plant for which design certification is not being sought. A COL applicant referencing the ABWR certified design is required to submit Design Descriptions and ITAAC for the site specific design features which meet the ABWR interface requirements.

Section 5.0—Describes the site parameters applicable to the ABWR certified design, such as tornado strength, flood height, and earthquake accelerations.

Appendix A—A legend which defines the symbols used in figures which are part of the design description.

Appendix B—List of Abbreviations.

1.1 Definitions

As used in this document, the following terms are defined:

Acceptance Criteria—The performance, physical condition, or analysis result for a structure, system, or component that demonstrates the design commitment is met.

Analysis—A calculation or mathematical computation and/or engineering evaluation. Engineering evaluation could include, but is not limited to, comparisons with operating experience or design of similar equipment.

As-built—The physical condition of the system, structure, or component following the completion of its installation or construction at its final location at the plant site.

Basic Configuration (Building)—The building arrangement of structural features (e.g., floors, ceiling, walls, columns, and door ways) which are specified in the design description and the relative location of systems or components which are also specified in the building design description. If the building design description includes systems or components, then the basic configuration (system) definition applies to those systems or components within the building.

Basic Configuration (System)—The functional arrangement of structures, divisions and components specified in the design description; and includes and is limited to pressure-boundary welds for ASME Code Class 1, 2 and 3 components; dynamic qualification of seismic Category I mechanical and electrical equipment; environmental qualification of Class 1E electrical equipment; and mechanical qualification of active seismic Category I motor operated valves. Inspections for basic configuration include inspection of the system functional arrangement and inspections limited to the following:

(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 the ASME Code Section III for assuring the quality of pressure boundary welds are met.

(2) Inspections of the results of the tests and/or analyses of the Seismic Category I mechanical and electrical equipment (including associated instrumentation and controls) identified in the design description, including associated anchorages to demonstrate that the as-built condition of such equipment is qualified to withstand the design basis dynamic loads without loss of their safety functions.

(3) Inspection of the result of test and/or analyses of the as-built Class 1E electrical equipment (including connected instrumentation and controls) identified in the design description demonstrates that the Class 1E electrical equipment is qualified to withstand the environmental conditions associated with design basis events without loss of safety function for the time needed to be functional. Such equipment includes the connected electrical equipment (such as cabling, wiring, and terminations) and lubricants necessary to support performance of the safety functions of the components identified in the design description

As-built electrical components (including ~~associated~~ connected instrumentation and controls) 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 event(s): Expected time-dependent temperature and pressure profiles, humidity, chemical effects, radiation, aging, submergence, and synergistic effects which have a significant effect on equipment performance.

Electrical equipment environmental qualification may be demonstrated through 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, testing a similar item of equipment with supporting analysis to show that the equipment to be qualified is acceptable, experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable, or analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

(4) Inspection of the results of tests of active safety related motor-operated valves identified in the design description demonstrate that the MOVs are qualified to perform their safety functions under design differential pressure, system pressure, fluid temperature, ambient temperature, minimum voltage, and minimum and/or maximum stroke times.

Design Commitment—That portion of the design description that is verified by ITAAC.

Design Description—That portion of the ABWR design that is approved by certification.

Division (for electrical systems/equipment)—The designation applied to a given system or set of components that enables the establishment and maintenance of physical, electrical, and functional independence from other redundant sets of components.

Division (for mechanical systems/equipment)—The designation applied to a specific set of components within a system.

Inspect or Inspection—Visual observations, physical examinations, or review of records based on visual observation or physical examination that compares the as-built structure, system, or component condition to design commitments. Examples include walkdowns, configuration checks, measurements of dimensions, and non-destructive examinations. Inspections can be performed in parts or segments over a period of time.

ITAAC—The inspections, tests, analyses, and acceptance criteria that are described in 10CFR 52.97(b) for the structures, systems, and components that are within the scope of the design certification. The ITAAC could apply to multiple systems and components, and they could include a design process for the design of systems and components that will be completed after certification. The ITAAC will be provided in tables with the following three-column format:

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria

The ITAAC tables contain inspections, tests, or analyses (ITA) for each design commitment. The identification of a separate ITA entry for each design commitment shall not be construed to require that separate inspections, tests, or analyses 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.

Simulated Signal—The intentional generation of a signal for testing.

Test—Operation of a structure, system or component, to evaluate its performance or structural integrity.

Enclosure 3

<u>Example Systems</u>

Residual Heat Removal

Nuclear Boiler

Leak Detection and Isolation

Reactor Cooling Water

Reactor Service Water

Reactor Water Cleanup

Standby Liquid Control

Turbine Main Steam

Condenser

Control Room Habitability Area HVAC

2.4 Core Cooling

2.4.1 Residual Heat Removal System

Design Description

The Residual Heat Removal (RHR) System is comprised of three divisionally separate subsystems that perform a variety of functions utilizing the following six basic modes of operation: (1) low pressure core flooder (LPFL), (2) suppression pool cooling, (3) wetwell and drywell spray cooling, (4) shutdown cooling, (5) augmented fuel pool cooling, and (6) alternating current power source (AC) independent water addition. The system configuration of each loop is shown in Figure 2.4.1a, b, c ~~(aligned in the standby mode)~~. The major functions of the various modes of operation include: (1) containment heat removal, (2) reactor decay heat removal, (3) emergency reactor vessel level makeup and (4) augmented fuel pool cooling. In line with its given functions, portions of the system are a part of the Emergency Core Cooling System (ECCS) network and the containment cooling system. Additionally, portions of the RHR System are considered a part of the Reactor Coolant Pressure Boundary (RCPB).

Except for the non-ASME Code components of the AC independent water addition feature (Figure 4.2.1.c), the entire RHR System is designed to safety-related standards, although it performs some non-safey functions. The safety-related modes of operation include: (1) low pressure core flooding, (2) suppression pool cooling, (3) wetwell spray cooling and (4) shutdown cooling. Non-safety-related modes of operation include: (1) drywell spray cooling, (2) AC independent water addition and (3) augmented fuel pool cooling. The RHR System also provides a backup, safety-related fuel pool makeup capability. Ancillary modes of operation include minimum flow bypass and full flow testing.

The ECCS function of the RHR System is performed by the LPFL mode. Following receipt of a loss-of-coolant accident (LOCA) signal ~~(low reactor water level or high drywell pressure, which is also called the RHR initiation signal)~~, the RHR System automatically initiates and operates in the LPFL mode ~~(in a conjunction with the remainder of the ECCS network)~~ to provide emergency makeup to the reactor vessel ~~in order to keep the reactor core cooled~~. The LPFL injection flow for each loop begins when the Reactor Pressure Vessel (RPV) dome pressure is no less than 15.8 kg/cm2 above the drywell pressure. For a design basis large break LOCA (LBLOCA), for RPV dome pressure no less than 2.8 kg/cm2 greater than the drywell pressure, the LPFL injection flow for each loop is 954m3/hr minimum within 36 seconds of receipt of actuation signal ~~with the RHR system initially in the standby mode~~. The LPFL mode is accomplished by all three loops of the RHR System by transferring water from the suppression pool to the RPV, via the RHR heat exchangers. Although the LPFL mode is automatically initiated, it may also be initiated manually. The system will also

INSERT A

automatically revert to the LPFL mode of operation from the test mode, the suppression pool cooling, ~~drywell spray,~~ or wetwell spray modes upon receipt of a LOCA signal. Each RHR loop's RPV injection valve requires a low reactor pressure permissive signal whether being opened manually or automatically in response to a LOCA signal.

The containment heat removal function in the ABWR is performed by the Containment Cooling System, which is comprised of the LPFL, suppression pool cooling, and wetwell and drywell spray cooling modes of the RHR System. Following a LOCA, the energy present within the reactor primary system is dumped either directly to the suppression pool via the Safety Relief Valves (SRVs), or indirectly via the drywell and connecting vents. Subsequently, fission product decay heat continues to add energy to ~~the pool~~ containment. The Containment Cooling System is designed to limit the long-term bulk temperature of the suppression pool, and thus limit the long-term peak temperatures and pressures within the wetwell and drywell regions of the containment to within their analyzed design limits, with only two of the three loops in operation. The cooling requirements of the containment cooling function establish the necessary heat removal capacity for each loop as no less than 88.5 kcal/sec°C, which includes the RHR heat exchanger, the Reactor Building Cooling Water (RCW) system, and the Reactor Service Water (RSW) system referenced to the ultimate heat sink. The heat removal capacity is based on the suppression pool cooling mode with the RHR tube side heat exchanger (Hx) flow rate 954m3/hr minimum.

The LPFL mode, in addition to its primary function of cooling the core, serves to cool the containment. The suppression pool cooling mode is made available in each of the three loops of the RHR System by circulating suppression pool water through the respective RHR heat exchanger and then directly back to the suppression pool. ~~RHR will be initiated in response to high suppression pool temperature.~~ The wetwell and drywell spray modes of RHR are each available in only two of the three subsystems (Loops B and C). These functions are performed by drawing water from the suppression pool and delivering it to a common wetwell spray header and/or a common drywell spray header, both via the associated RHR heat exchanger(s). The wetwell spray flow rate for either loop is no less than 114m3/hr. These containment spray modes of the RHR System can be initiated manually. However, the drywell spray inlet valves can only be opened if there exists high drywell pressure and if the RPV injection valves are fully closed. Wetwell and drywell sprays serve as a augmented method of containment cooling. Wetwell spray also serves to mitigate the consequences of steam bypassing the suppression pool.

The normal operational mode of the RHR System is in the shutdown cooling mode of operation, which is used to remove decay heat from the reactor core. This mode provides the required safety-related capability needed to achieve and maintain a cold shutdown condition. The RHR heat exchanger heat removal

Insert A

If a drywell spray valve is open, the RHR System automatically reverts to allow the LPFL mode in response to the injection valve beginning to open.

capacity requirements in this mode are bounded by containment cooling requirements. Shutdown cooling is initiated manually once the RPV has been depressurized below the system low pressure permissive. In this mode, each loop takes suction from the RPV via its dedicated suction line, pumps the water through its respective heat exchanger tubes at 954m3/hr minimum, and returns the cooled water to the RPV. Two loops (B and C) discharge water back to the RPV via dedicated spargers, while the third loop (A) utilizes the vessel spargers of one of the two feedwater lines (FW-A). The heat removed in the RHR heat exchangers is transported to the ultimate heat sink via the respective division of reactor cooling water and service water. Each shutdown cooling suction valve is interlocked with that loop's suppression pool suction and discharge valves and wetwell and drywell spray valves to prevent draining of the reactor vessel to the suppression pool. Also, to prevent draining of the reactor vessel, each shutdown cooling suction valve is interlocked with, and automatically closes on, low reactor water level.

The augmented fuel pool cooling mode of the RHR System supplements/ replaces the normal fuel pool cooling system during conditions of high heat load. This mode is accomplished manually in one of two ways. When the reactor vessel head is removed, the cavity is flooded and the fuel pool gates are removed, the RHR System cools the fuel pool in the normal shutdown cooling mode. When the fuel pool is otherwise isolated from the reactor cavity, two loops (B and C) of the RHR System can directly cool the pool by taking suction from and discharging back to the ~~normal~~ fuel pool cooling system at 350 m3/hr minimum. This connection also provides for emergency fuel pool makeup capability by supplying a safety-related makeup path to the fuel pool from a safety-related source (i.e., the suppression pool).

One loop (C) of the RHR System also functions in an AC independent water addition mode. This mode provides a means of cross connecting the reactor building fire protection (FP) system header to the RHR System just outside the containment in the absence of the normal ECCS network and independent of the normal safety related AC power distribution network. The connection is accomplished by manually at the valve opening two in-series valves on the cross-connection piping just upstream of its tie-in to the normal RHR piping. Fire protection system water can be directed to either the RPV or the drywell spray sparger by manual opening of the loop C RHR injection valve or the two loop C drywell spray valves. These three valves also have manual hand wheels. The fire water is supplied via the system's Reactor Building distribution header.

Permit

Each loop of the RHR System also has both a minimum flow mode and a full flow test mode to premit pump flow testing during plant operation. The minimum flow mode assures that there is pump flow sufficient to keep the pump cool anytime the pump is running by opening as needed a minimum flow valve that directs flow back to the suppression pool . Upon sensing that there is sufficient

flow in the pump main discharge line, the minimum flow valve is automatically closed. In the full flow test mode, the system is ~~essentially~~ operated in the suppression pool cooling mode, drawing suction from and discharging back to the suppression pool.

The RHR System is comprised of three separate loops or subsystems, each of which includes a pump and a heat exchanger, takes suction from either the RPV or the suppression pool, and directs water back to either the RPV or the suppression pool. Two of the three loops can divert a portion of the suppression pool return flow to a common wetwell spray sparger or direct the entire flow to a common drywell spray sparger. The divisions ~~of subsystems~~ of the RHR System are separated ~~both~~ physically ~~and electrically, as well as being physically located in different areas of the plant~~. Each of the three ~~subsystems~~ divisions is powered from the respective Class 1E division as shown on Figure 2.4.1a, b, c. Cooling water to each division of RHR equipment (heat exchanger as well as pump and motor coolers) is supplied by the respective division of the reactor cooling water (RCW) System. The RHR System also has provisions for containment isolation and reactor containment pressure boundary RCPB pressure isolation.

~~The RHR System will maintain the capability to perform its intended safety-related functions during and following design basis accidents conditions. Except for the non-ASME Code components of the AC independent water addition feature (Figure 4.2.1.c),~~ the RHR System is Seismic Category I and is housed in the Seismic Category I Reactor Building. ~~Components shown in Figure 2.4.1.a,b,c are qualified for environmental conditions except (1) the passive heat exchangers, piping, suppresion pool suction strainers, and containment spray headers and (2) the non-safety-related equipment associated with the AC independent water addition features shown on Figure 2.4.1.c.~~ INSERT AA

~~The RHR pumps are motor-driven centrifugal pumps each capable of supplying at least 954 m 3/hr at greater than or equal to 2.8 kg/cm2 (drywell to RPV)~~. The RHR pump will provide the head/flow when the surface of the fluid is saturated. The pumps are ASME Code Class 2 components. Sufficient NPSH is provided to the pumps for the design conditions of 100°C water, the containment at atmospheric pressure, the suppression pool at its minimum level, and the strainers blocked greater than or equal to 50%. The pumps are interlocked from starting without an open suction path. ~~The RHR pumps are protected from possible pump run out conditions during operation. The RHR heat exchangers are U-tube/shell type. The primary and secondary sides of the heat exchangers are ASME Code Class 2 and 3, respectively. Each loop of the RHR System has its own jockey pump to act as a keep-fill system for that loop's pump discharge piping.~~

~~The RHR System piping and values are ASME Code Class 1 or 2 as shown in Figures 2.4.1a,b,c. The design pressures and temperatures of various system~~

INSERT AA

The safety related electrical equipment shown on Figure 4.2.1-a, Figure 4.2.1-b and Figure 4.2.1-c located inside containment and the reactor building is qualified for a harsh environment.

INSERT B

~~features are given in Table 2.4.1.a.~~ The low pressure portions of the shutdown cooling piping are protected from full reactor pressure by automatic pressure isolation valves that are interlocked with reactor pressure. ~~Additionally, in series inboard and outboard containment/pressure isolation valves in each loop are powered from separate electrical divisions.~~

The RHR System control room alarms, indications and controls allow for monitoring and control during operational conditions. The control room has indication for system flows, temperatures and pressures, as well as valve open/close and pump on/off status, and controls for those instruments and components shown on Figures 2.4.1a, ~~b and c, with the exception of simple check valves (of the check valves shown, only the testable check valves downstream of each loop's RPV injection valve has control and status indication in the control room). Valves shown on Figures 2.4.1a, b, c with box M operators can be operated from the control room.~~
Figure 2.4.1b and Figure 4.2.1c.

RHR System components with status indication and/or control interfaces with the Remote Shutdown System (RSS) are shown on Figures 2.4.1a and Figure 2.4.1b. ~~These mechanical components are designed to meet ASME code requirements as shown below:~~

Table 2.4.1.a

Component	ASME Code Class	Design Conditions Pressure	Design Conditions Temperature
Main pumps	2	$35.2 kg/cm^2g$	182°C
Jockey Pumps	2	$28.8 kg/cm^2g$	182°C
Heat exchangers primary side (reactor)	2	$35.2 kg/cm^2g$	182°C
Heat exchangers secondary side (cooling)	3	$14 kg/cm^2g$	182°C
Piping and components from the RPV out to and including the outboard containment isolation valves	1	$87.9 kg/cm^2g$	302°C
Piping and components open to the containment atmosphere, out to and including the outboard containment isolation valves	2	$3.16 kg/cm^2g$	104°C
Piping and components outside the containment isolation valves and on the suction side of the pump	2	$28.8 kg/cm^2g$	182°C

Insert B

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/cm²g for ~~inter~~ inter system LOCA (ISLOCA) conditions.

Component	ASME Code Class	Design Conditions Pressure	Temperature
Piping and components outside the containment isolation valves and on the discharge side of the pump	2	35.2kg/cm^{2}g	182°C

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.

Table 2.4.1.b: Residual Heat Removal System

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. ~~A~~ The basic configuration of the RHR System is shown in Figures 2.4.1a, 2.4.1b and 2.4.1c.	1. ~~Visual~~ Inspections of the as-built system ~~configuration~~ will be conducted.	1. The as-built ~~configuration of the~~ RHR System ~~as in accordance with~~ conforms with the basic configuration shown in figures 2.4.1a, 2.4.1b and 2.4.1c.
2. The RHR System ~~operates in the LPFL mode to provide emergency core make up to the reactor vessel as follows:~~ provides the following:		
2a. The RHR System operates in the LPFL mode to provide emergency core make up to the reactor vessel as follows.	2a ~~and 2b~~. RHR System functional tests shall be performed on the RHR LPFL mode. Analysis shall be performed to convert the test results to the conditions of the Certified Design Commitment.	2a. The converted RHR flow satisfies the following: 2a. The LPFL injection flow for each loop begins when the RPV dome pressure is greater than or equal to 15.8 kg/cm^2 above the drywell pressure. 2b. The LPFL injection flow for each loop is greater than or equal to 954 m^3/hr, for a dome pressure greater than or equal to 2.8 kg/cm^2 above the drywell pressure.
~~2a. The LPFL injection flow for each loop begins when the RPV dome pressure is greater than or equal to 15.8 kg/cm^2 above the drywell pressure.~~ ~~2b. The LPFL injection flow for each loop is greater than or equal to 954 m^3/hr minimum, for a dome pressure at greater than or equal to 2.8 kg/cm^2 above the drywell pressure.~~		
~~2c. The injection valve opens within 36 sec of receipt of low pressure permissive signal.~~	~~2c. Using simulated low pressure permissive actuation signal, RHR injection valve tests will be performed.~~	~~2c. The injection valve opens within 36 sec of receipt of low pressure permissive signal.~~

Table 2.4.1.b: Residual Heat Removal System

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2b. In the S/P cooling mode, the RHR tube side heat exchanger, (Hx) flow rate is greater than or equal to 954 m^3/hr.	2b. In the S/P cooling mode, RHR System functional tests shall be performed ~~to determine the flow rate through the heat exchanger,~~	2b. In the ~~S/P~~ suppression pool cooling mode, the RHR tube side heat exchanger, (Hx) flow rate is greater than or equal to 954 m^3/hr.
2c. RHR loops, either B or C, can provide greater than or equal to 114m^3/hr flow to the wetwell spray headers.	2c. RHR System functional tests will be performed on loops B and C in the wetwell spray mode.	2c. RHR loop B provides wetwell spray flow greater than or equal to 114 m^3/hr. RHR loop C provides wetwell spray flow greater than or equal to 114 m^3/hr.
2d. 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).	2d. 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.	2d. 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.
2e. In the augmented fuel pool cooling mode, the RHR tube side heat exchanger flow rate is greater than or equal to 350 m^3/hr (heat exchanger heat removal capacity in this mode is bounded by suppression pool cooling requirements).	2e. In the augmented fuel pool 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.	2e. In the augmented fuel pool cooling mode, the RHR tube side heat exchanger flow rate is greater than or equal to 350 m^3/hr. Heat exchanger heat removal capacity in this mode is bounded by suppression pool cooling requirements.

Table 2.4.1.b: Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Inspections, Tests and analyses

ITA

ACC

system

performed

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2f. ~~15.~~ RHR System is designed to permit pump flow during plant operation.	2f. ~~15.~~ Using installed controls, power supplies and other auxiliaries, the pump flow test will be conducted for each RHR loop after system installation. Water will be pumped in the test loop. System head will be equivalent to a pressure differential of 2.8 kg/cm^2 between the RPV and the drywell.	2.f ~~15.~~ Water is pumped at a flow rate of greater than or equal to 954 m^3/hr. ~~Note: This ITAAC is satisfied by the test and analysis used for RHR ITAAC 2b.~~
3. ~~6.~~ The RHR pumps have sufficient NPSH.	3. ~~6.~~ ~~An analysis for NPSH~~ will be ~~prepared~~ based upon as-built ~~data~~. This analysis will ~~be based upon~~ 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.	~~The analyzed NPSH exceeds pump NPSH required by the vendor of the pump.~~ 3. ~~6.~~ The available NPSH exceeds the NPSH required.

Table 2.4.1.b: Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. In the S/P cooling mode, ~~and~~ the effective heat removal capability of each RHR Hx, including the effects of RCW, RSW, and ultimate heat sink is greater than or equal to 88.5 kcal/sec °C.	4. For the S/P cooling mode, ~~and~~ inspections and analysis shall be performed to determine the heat exchanger's effective heat removal capability. (A) ANALYSES METHOD TO BE IN SSAR	4. For the S/P cooling mode, ~~and~~ the effective heat removal capability of each RHR Hx, including the effects of RCW, RSW, and ultimate heat sink is greater than or equal to 88.5 kcal/sec °C.

Table 2.4.1.b: Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
C 5. ~~X.~~ **Initiation Logic** 5. ~~A.~~ Each RHR division initiates in the LPFL mode in response to an automatic RHR initiation (LOCA) signal ~~from the SSLC~~ or to a manual initiation signal. ~~from its respective divisional pushbutton~~.	5. Using simulated initiation signals, tests will be performed of the RHR initiation logic.	5. Upon receipt of a simulated initiation signal (manual or automatic) at the input to the RHR logic, 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 ~~suction~~ valve receives a signal to ~~close~~ OPEN STET d) The wetwell spray valve receives a signal to close (Divisions B&C only)
6 ~~Y.~~ **Isolation Logic** 6a The shutdown cooling inboard and outboard isolation valves automatically close on receipt of an RHR isolation signal. ~~(from LDS)~~.	6a. Using simulated {RHR isolation} signals, tests will be performed of the {shutdown cooling isolation valve} isolation logic.	6a. Upon receipt of a simulated RHR isolation signal the following occurs: a) The shutdown cooling inboard and outboard isolation valves automatically close
~~Y. Isolation Logic~~ B 6b. The injection ~~shutdown cooling inboard and outboard isolation~~ valves automatically close on receipt of a high reactor pressure ~~an RHR isolation~~ signal. ~~(from LDS)~~.	6b. Using a simulated high reactor pressure ~~{RHR isolation}~~ signal~~s~~, tests will be performed of the injection valve ~~{shutdown cooling isolation valve}~~ isolation logic.	6b. Upon receipt of a simulated high reactor pressure ~~RHR~~ isolation signal the following occurs: a) The injection ~~shutdown cooling inboard and outboard isolation~~ valves automatically close.

Table 2.4.1.b: Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

(C) 7. Interlock Logic

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7a. ~~8.~~ If already operating in the test mode, the suppression pool cooling, ~~drywell spray~~ or wetwell spray modes, the RHR System automatically reverts to the LPFL mode in response to a LOCA signal.	7a. ~~8.~~ Using simulated LOCA actuation signals, tests will be performed on the RHR System.	7a. ~~8.~~ Upon receipt of simulated LOCA actuation signal, RHR logic functions to automatically reconfigure the system to the LPFL mode of operation when, operating in the test mode, the suppression pool cooling, ~~drywell spray~~ or wetwell spray modes.
7~~8~~b. If a drywell spray valve is open, the RHR System automatically reverts to allow the LPFL mode in response to the injection valve beginning to open.	7~~8~~b. Using an injection valve opening signal, tests will be performed on the RHR System.	7~~8~~b. With either drywell spray valve open and upon receipt of an injection valve ~~opening~~ NOT FULLY CLOSED signal, RHR logic functions to close the drywell spray valve.
7c. The minimum flow mode assures that there is pump flow sufficient to keep the pump cool anytime the pump is running by opening, as needed, a minimum flow valve that directs flow back to the suppression pool. Upon sensing that there is sufficient flow in the pump main discharge line, the minimum flow valve is automatically closed.	7c. Using simulated [pressure and flow] signals, tests will be performed of the [pump minimum flow valve] interlock logic.	7c. The pump minimum flow valve receives a signal to open when *signals indicative of* the following conditions exist concurrently (for the required time delay period): a) Pump discharge pressure is high b) Pump flow is low The pump minimum flow valve receives a signal to close when *signals indicative of* the following condition exists: a) Pump flow is high

Table 2.4.1.b: Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

(B)

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7d. Each RHR Division's RPV injection valve requires a low pressure permissive signal whether being opened manually or automatically in response to an RHR initiation.	7d. Using simulated reactor pressure signals, tests will be performed of the RHR injection valve permissive logic.	7d. The RPV injection valve is blocked from opening, whether demanded manually or via an RHR initiation signal, if *signals indicative of* the following condition exists: a) Reactor pressure is high
7e. The RHR pumps are interlocked from starting without an open suction path.	7e. Using simulated [valve position] signals, tests will be performed of the [RHR pump start] interlock logic.	7e. The RHR pump is blocked 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)

2.4.1

Table 2.4.1.b: Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
~~CDC~~	injection ~~ITA~~	~~AC~~
7f. The suppression pool return valve and ~~the~~ the wetwell spray valve ~~and the two drywell spray valves~~ are interlocked with the RPV injection valve. ~~(Indirect auto reversion logic)~~	7f. Using simulated ~~valve position~~ signals, tests will be performed of the ~~(RHR valve)~~ suppression pool return valve and the wetwell spray valve interlock logic.	7f. The suppression pool return valve and the wetwell spray valve ~~and the two drywell spray valves~~ are blocked from opening unless *signals indicative of* the following condition exists: a) The RPV injection valve is full closed. ~~or~~ ~~For the drywell spray valves only (b)~~ ~~b) The other in-series drywell spray valve is fully closed~~ The suppression pool return valve and the wetwell spray valve ~~and the two drywell spray valves~~ will automatically close if *signals indicative of* the following condition exists: a) The RPV injection valve is not fully closed

Table 2.4.1.b: Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7g. The in-series drywell spray valves can only be opened simultaneously if there exists high drywell pressure and the RPV injection valve and shutdown cooling suction valve are both fully closed. The drywell spray valves close when injection valve opens.	7g. Using simulated drywell pressure and valve position signals, tests will be performed of the drywell spray valves interlock logic.	7g. The two in-series drywell spray valves are blocked from being open simultaneously unless *signals indicative of* the following conditions exist concurrently: a) pressure is high b) The RPV injection valve is fully closed c) The shutdown cooling suction valve is fully closed The ~~suppression pool return valve, the wetwell spray valve and the two in~~ drywell spray valves will automatically close if *signals indicative of* the following condition exists: a) The RPV injection valve is not fully closed

Table 2.4.1.b: Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. ~~7.~~ Class 1E Loads of the RHR System are powered from Class 1E Divisions, as described in Section 2.4.1.	8. Tests will be performed on the RHR System by providing a test signal in only one Class 1E Division at a time.	8. The test signal exists only in the Class 1E Division under test in the RHR System.
9. ~~8.~~ The RHR System utilizes jockey pumps (one in each loop) to keep the pump discharge line filled.	9. ~~8.~~ Functional tests will be performed on the ability of the jockey pump (one in each loop) to keep its respective RHR pump discharge line full.	9. ~~8.~~ Each jockey pump keeps its respective pump discharge line filled.
10a. ~~9.~~ Control Room ~~features~~ indication and controls provided for RHR System are defined in Section 2.4.1.	10b. ~~9.~~ Inspections will be performed on the Control Room ~~features~~ indication and controls for the RHR system.	10c. ~~9.~~ ~~Features are available~~ indication and controls exist or can be retrieved in the Control Room as defined in Section 2.4.1.

Table 2.4.1.b: Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

indications and/or controls

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10b. ~~3b~~. Remote Shutdown System (RSS) ~~features~~ indications and/or controls provided for the RHR System are as defined in Section 2.4.1.	10b. ~~3b~~. Inspections will be performed on the RSS ~~features~~ indications and/or controls for the RHR System.	10b. ~~3b~~. ~~Features are available~~ indications and/or controls exist on the RSS as defined in Section 2.4.1.
11. ~~15~~. The ASME Code components of the RHR System retain their pressure boundary integrity under internal pressures that will be experienced during service.	11. ~~15~~. 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.	11. ~~15~~. 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.

Table 2.4.1.b: Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. Each mechanical Division of the RHR System (Divisions A, B, C) is physically separated.	12. Inspections of the as-built RHR System will be performed.	12. Each mechanical Division of the RHR System is physically separated from other mechanical Division of RHR System by structural and/or fire boundaries with the exception of primary containment.

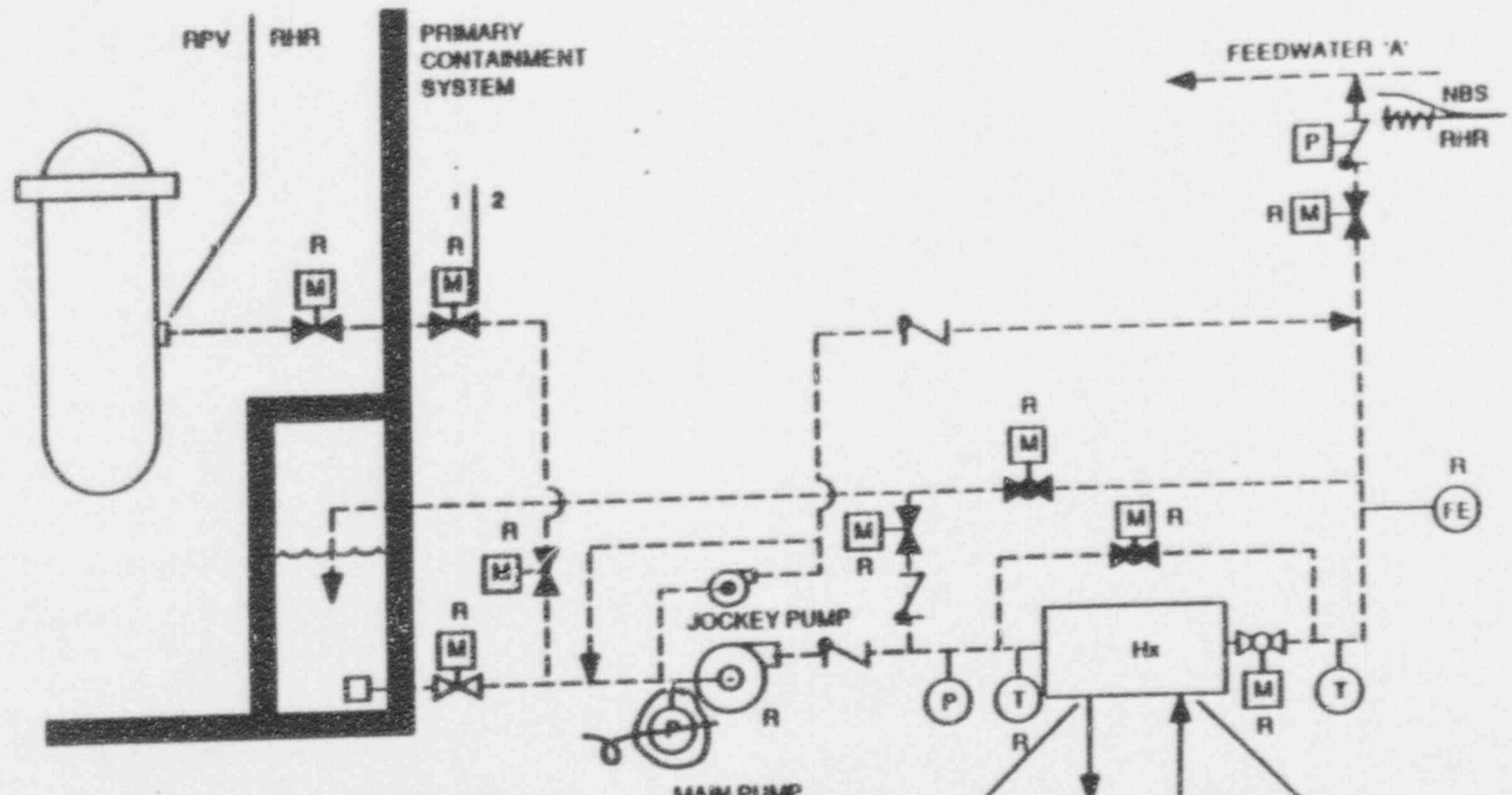
(D)

Table 2.4.1.b: Residual Heat Removal System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
13 ~~22~~. Motor-operated valves (MOV) designated in Section 2.4.1 as having an active safety function, ~~-suppression pool suction valve, - inboard shutdown cooling valve, - outboard shutdown cooling valve, -third shutdown cooling valve, - minimum flow valve, -heat exchanger flow valve, -heat exchanger bypass valve, -suppression pool return valve, -wetwell spray valve, -injection valve, -testable check valve, -upstream drywell spray valve, -downstream drywell spray valve, -FPC return upstream valve, -FPC return downstream valve, -FPC source valve,~~ will open and/or close under differential pressure and fluid flow conditions.	13 ~~22~~. Opening and/or closing tests of installed valves will be conducted under pre-operational differential pressure, fluid flow, and temperature conditions.	13 ~~22~~. Each MOV opens and/or closes. The following valves open and/or close in the following time limits: Valve — Time (sec) Injection — 36 open

* -19-



LEGEND: R= REMOTE SHUTDOWN SYSTEM

ALL ELECTRICAL POWER LOADS FOR THE COMPONENTS SHOWN ON THIS FIGURE ARE POWERED FROM DIVISION I EXCEPT FOR THE OUTBOARD CONTAINMENT ISOLATION VALVE FOR SHUTDOWN COOLING, WHICH IS DIVISION II.

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

- 21

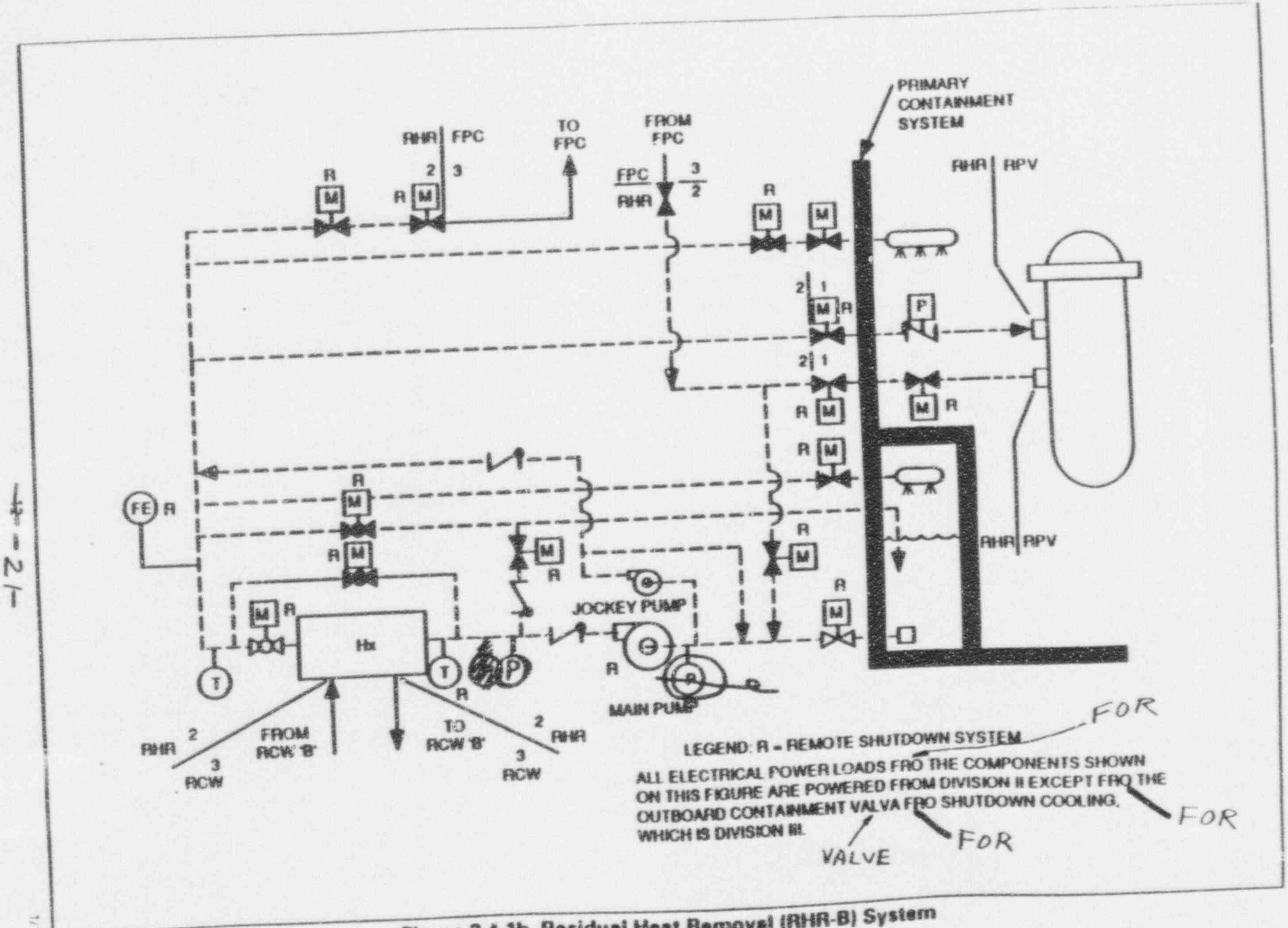
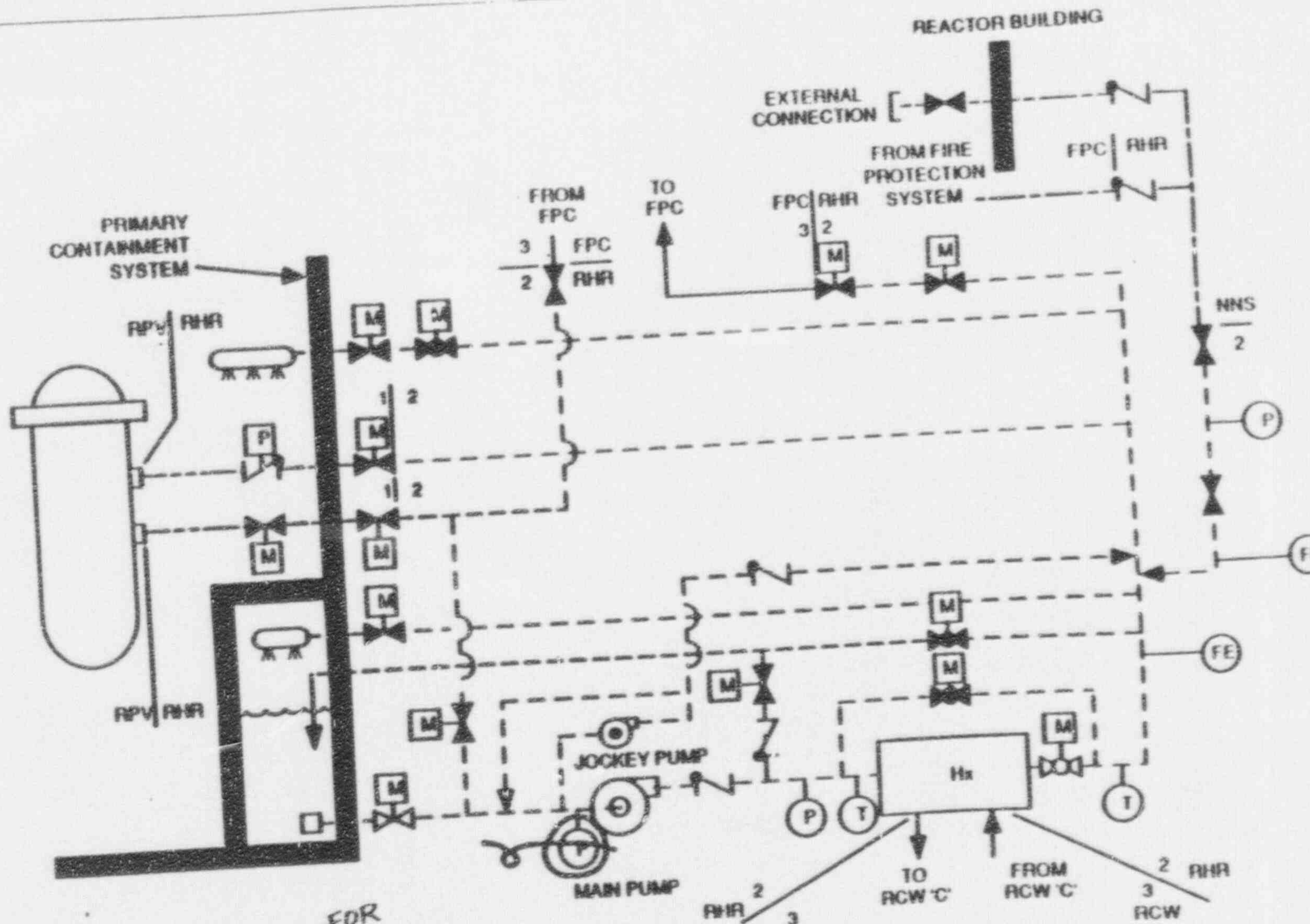


Figure 2.4.1b Residual Heat Removal (RHR-B) System



FOR
ALL ELECTRICAL POWER LOADS FRO THE COMPONENTS SHOWN ON THIS FIGURE ARE POWERED FROM DIVISION III EXCEPT FOR THE OUTBOARD CONTAINMENT ISOLATION VALVE FOR SHUTDOWN COOLING. WHICH IS DIVISION I.

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

1/16/93

RHR

(A) ANALYSIS METHOD TO BE IN SSAR

(B) ITAAC 6B AND 7d NEED TO BE COORDINATED TO ENSURE CONSISTENCY OF TEST FUNCTION

(C) CHECK CONSISTENCY OF INTERLOCK LOGIC AND TESTING IN ITAAC 5 THROUGH 7.

(D) MODIFY FOR CONSISTENCY BASED ON DEFINITION ACCEPTANCE

(E) CONSISTENT TREATMENT OF CODES FOR FIGURES

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) 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.

Within the NBS, the FW lines, the MSLs and the MSL drain lines are located in the drywell and the steam tunnel. With the exception of the instrumentation attached directly to the RPV or NBS piping, the NBS instrumentation is located within the Reactor Building.

See Figures 2.1.2a, 2.1.2b, and 2.1.2c, for the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class.

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 MSLs are classified as Seismic Category I from the reactor pressure vessel out to the seismic interface shown in Figure 2.1.2b.

The MSL drain lines provide a flow path for the Main Steam Isolation Valve (MSIV) leakage during an accident. The pneumatic operated valve in the MSL drain line shown in Figure 2.1.2b opens should either electric power to the valves

actuating solenoid be lost, or pneumatic pressure to the valve be lost. The MSL drain lines from the MSLs to the main condenser are seismically analyzed to withstand the Safe Shutdown Earthquake (SSE).

The total steam 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 (choke) 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 MSL flow limiter also serves as a flow element to monitor the MSL flow. Instruments lines are provided to monitor the pressure at the throat of the MSL flow limiter. The RPV steam dome pressure instrument lines are used to provide the pressure upstream of the MSL flow limiter.

The MSL flow limiters limit the loss of coolant from the RPV following a MSL rupture outside the containment.

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 MSIVs are Y-pattern globe valves. 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.

The MSIV closing speed is equal to or greater than 3 and less than or equal to 4.5 seconds when N_2 or air pressure is admitted to the MSIV actuator. When all the MSIVs are closed, the total 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 pressure absolute.

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.

The FW piping consists of two nominal 550 mm diameter lines from the FW supply header. Isolation of each 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 Motor-Operated (MO) valve, and a seismic interface restraint.

The feedwater piping is classified as Seismic Category I from the MO shutoff valves to the RPV. The ASME Boiler and Pressure Vessel Code Class 2 piping from the Control Rod Drive (CRD) System, the Reactor Water Cleanup (CUW) System, Reactor Core Isolation Cooling (RCIC) System, and the Residual Heat Removal (RHR) System, shown in Figure 2.1.2c is also classified as Seismic Category I.

Safety/Relief Valves

The nuclear pressure relief system consists of Safety/Relief Valves (SRVs) located on the MSLs between the RPV and the first isolation valve, i.e. the inboard MSIV, within the drywell. These valves protect against overpressurization of the nuclear system. 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 line is designed to achieve critical flow conditions through the valve, thus providing flow independence to discharge pipe losses. Each SRV has its own discharge line. The SRV discharge lines terminate at the quenchers located below the surface of the suppression pool.

The SRV discharge lines are classified as Seismic Category I.

The SRVs provide three main protection functions:

(1) Overpressure safety operation: The valves function as safety valves and open to prevent nuclear system overpressurization—they are self-actuating by inlet steam pressure if not already signaled open for relief operation.

Table 2.1.2a 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.

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

For overpressure relief valve operation (power-actuated mode), pressure sensors on the RPV generate a RPV high pressure trip signal which is used to initiate opening the SRVs. The relief (power actuated) mode of operation is initiated when an electrical signal is received at any of the SRV solenoid valves.

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 as part of the Emergency Core Cooling System (ECCS) for events involving breaks in the nuclear system process barrier. Automatic depressurization by the ADS is provided to reduce the reactor pressure during a LOCA in which the High Pressure Core Flooder (HPCF) System and/or the Reactor Core Isolation Cooling (RCIC) System are unable to restore water level.

Eight of the eighteen SRVs are designated as ADS valves and are capable of operating from either ADS logic or safety/relief logic signals. Table 2.1.2a identifies the ADS SRVs.

The ADS consists of redundant trip channels arranged in two divisionally separated logics that control two separate solenoid-operated gas 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 instrumentation and logic is obtained from the Safety System Logic and Control (SSLC) Division I and II.

Sensors 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.

Upon receipt of an RPV low water signal the ADS automatically initiates. 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 of at least one RHR or one HPCF pump running, the ADS function is initiated.

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.

SRVs have individual non-safety-related accumulators. In addition, those with ADS function have separate safety-related accumulators with separate redundant gas power actuators, as shown in Figure 2.1.2d.

The ADS accumulator capacity is sufficient to open the SRV with the drywell pressure at design gauge 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.

Temperature sensors are located on the discharge pipe of the SRVs.

NBS Instrumentation

The purpose of the NBS RPV instrumentation is to monitor and provide control input during plant operation.

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

The safety-related NBS instrumentation is located in separate divisional areas.

Pressure instrumentation ~~detect~~ detects reactor vessel internal pressure from the same instrument lines used for measuring reactor vessel water level.

The RPV coolant temperatures are determined by measuring saturation pressure (which gives the saturation temperature), outlet flow temperature to the Reactor Water Cleanup (CUW) System, and the RPV bottom head drain line temperature (instrumentation in the ~~RWCU~~ CUW System). The reactor vessel outside surface (metal) temperatures are measured at the head flange and the bottom head locations. During plant operation, either reactor steam saturation temperature and/or inlet temperatures of the reactor coolant to the CUW System and the RPV bottom head drain can be used determine the RPV coolant temperature.

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 corresponding liquid temperature) conditions. The water level measurement design is the condensate reference chamber type. Instrument zero for the RPV water level ranges is the top of the active fuel.

The NBS contains the instrument lines to monitor the differential pressure across the RPV pump deck and core support plate. The instrumentation which actually performs these functions is located within the Reciculation Flow Control System.

The NBS also contains the drywell pressure instrumentation used to generate the safety-related high drywell pressure trip LOCA signal. The Reactor Protection System (RPS) utilizes this signal as a scram initiation signal. The Leak Detection and Isolation System (LDS) utilizes this signal to initiate containment isolation. The Emergency Core Cooling Systems (ECCSs) utilizes this signal as a system initiation signal.

The NBS control room alarms, indications and controls allows for monitoring and control during operational conditions. The control room has indication alarms for and/or control of the ADS, RPV water level, RPV pressure, drywell pressure, SRVs, FW line outboard check valves, and RPV metal temperature.

Remote Shutdown System (RSS) Interfaces

NBS components with status indication and/or control interfaces with the RSS are shown on Figure 2.1.2a and 2.1.2e.

Environmental Qualification

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.

Inspections, Tests, Analyses and Acceptance Criteria

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

Table 2.1.2a: Nuclear System Safety/Relief Valve Setpoints

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.

Table 2.1.2b: Nuclear Boiler System

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the NBS is shown in Figures 2.1.2a, b, c, d, and e.	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, and e.
2. Each Main Steam Line (MSL) has a flow limiter located in the RPV MSL outlet nozzle. The throat diameter for the MSL flow limiter is less than or equal to 355 mm.	2. Inspection will be performed on the throat diameter of the MSL flow limiters, which are located within the RPV MSL outlet nozzles.	2. The throat diameter of the MSL flow limiters is less than or equal to 355 mm.
3. The ASME Code components of the NBS System retain their pressure boundary integrity under internal pressures that will be experienced during service.	3. A hydrostatic test of the ASME Code components of the NBS System will be conducted.	3. 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
4. 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.	4. Using as-built dimensions of the steam lines volumetric analysis will be performed to determine the combined main steam line volume.	4. The combined steam line volume is greater than or equal to 113.2 m^3.
5a. Control Room alarms, indications and/or controls provided for NBS are defined in Section 2.1.2.	5a. Inspections will be performed on the Control Room alarms, indications and/or controls for the NBS.	5a. Alarms, Indications and/or controls exist or can be retrieved in the Control Room as defined in Section 2.1.2.
5b. Remote Shutdown System (RSS) indications and/or controls provided for the NBS are defined in Section 2.1.2.	5b. Inspections will be performed on the RSS indications and/or controls for the NBS.	5b. Indications and/or controls exist on the RSS as defined in Section 2.1.2.
6. The Main Steam Isolation Valve (MSIV) closing time is equal to or greater than 3 and less than or equal to 4.5 seconds when N_2 or air is admitted into the valve pneumatic actuator.	6. Tests will be conducted to determine the closure time of the MSIVs.	6. The MSIV closing time is equal to or greater than 3 and less than or equal to 4.5 seconds.

Table 2.1.2b: Nuclear Boiler System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. When all four 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. Leakage tests will be performed to determine the leakage through the closed MSIVs.	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. The SRV spring set pressure and capacities are given in Table 2.1.2a. The opening time for the SRVs from when the pressure exceeds the valve set pressure to when the valve is fully open is less than or equal to 0.3 seconds.	8. Tests and analysis in accordance with the ASME Boiler and Pressure Vessel Code will be performed to determine the spring set pressure, capacity and opening time of each SRV.	8. The SRVs have the capacities and set pressures shown on Table 2 1.2a. 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.
9a. Upon receipt of an RPV low water level signal the ADS logic automatically initiates.	9a. Using simulated signals, tests will be performed of the automatic ADS initiation logic.	9a. I 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 2.1.2b: Nuclear Boiler System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9a. (Continued)	9a. (Continued)	9a. II 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 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.
9b. Upon receipt of a manual initiation signal the ADS logic initiates.	9b. Tests will be performed of the manual ADS initiation logic.	9b. Upon receipt of a manual initiation signal at the input to the ADS initiation logic, concurrent with a signal that represents positive indication of at least one RHR or HFCF pump running, an ADS actuation signal is generated to the associated ADS valve solenoids.

Table 2.1.2b: Nuclear Boiler System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. The SRV ADS accumulators have the capacity to open the SRVs one time with the drywell at the drywell design pressure.	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. Class 1E loads of the NBS are powered from Class 1E Divisions, as described in Section 2.1.2.	11. Tests will be performed for the NBS by providing a test signal in only one Class 1E Division at a time.	11. The test signal exists only in the Class 1E Division under test in the NBS.
12. The MSL drain lines from the MSLs to the main condenser are seismically analyzed to withstand the SSE.	12. An inspection of the stress report containing the dynamic analysis of the piping will be conducted.	12. The existence of a stress report will be confirmed. This report documents that a dynamic seismic analysis has been performed.
13. Springs close the MSIV if pneumatic pressure to the MSIV actuator is lost.	13. Tests will be performed to demonstrate that the MSIV will stroke to the fully closed position upon the loss of pneumatic pressure to the MSIV actuator.	13. The MSIV closes when pneumatic pressure is removed from the MSIV.
14. The pneumatic operated valve in the MSL drain line shown in Figure 2.1.2b opens should either electric power to the valve actuating solenoid be lost, or pneumatic pressure to the valve be lost.	14. Tests will be performed to demonstrate that the MSL drain line valve opens when pneumatic pressure to the valve is lost, or electric power to the valve actuating solenoid is lost.	14. 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.

Table 2.1.2b: Nuclear Boiler System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
15. Motor-operated valves designated in Section 2.1.2 as having an active safety function will open or close under differential pressure and flow conditions.	15. Opening and/or closing tests of installed valves will be conducted under preoperational differential pressure, fluid flow, and temperature conditions.	15. Each MOV opens and/or closes.

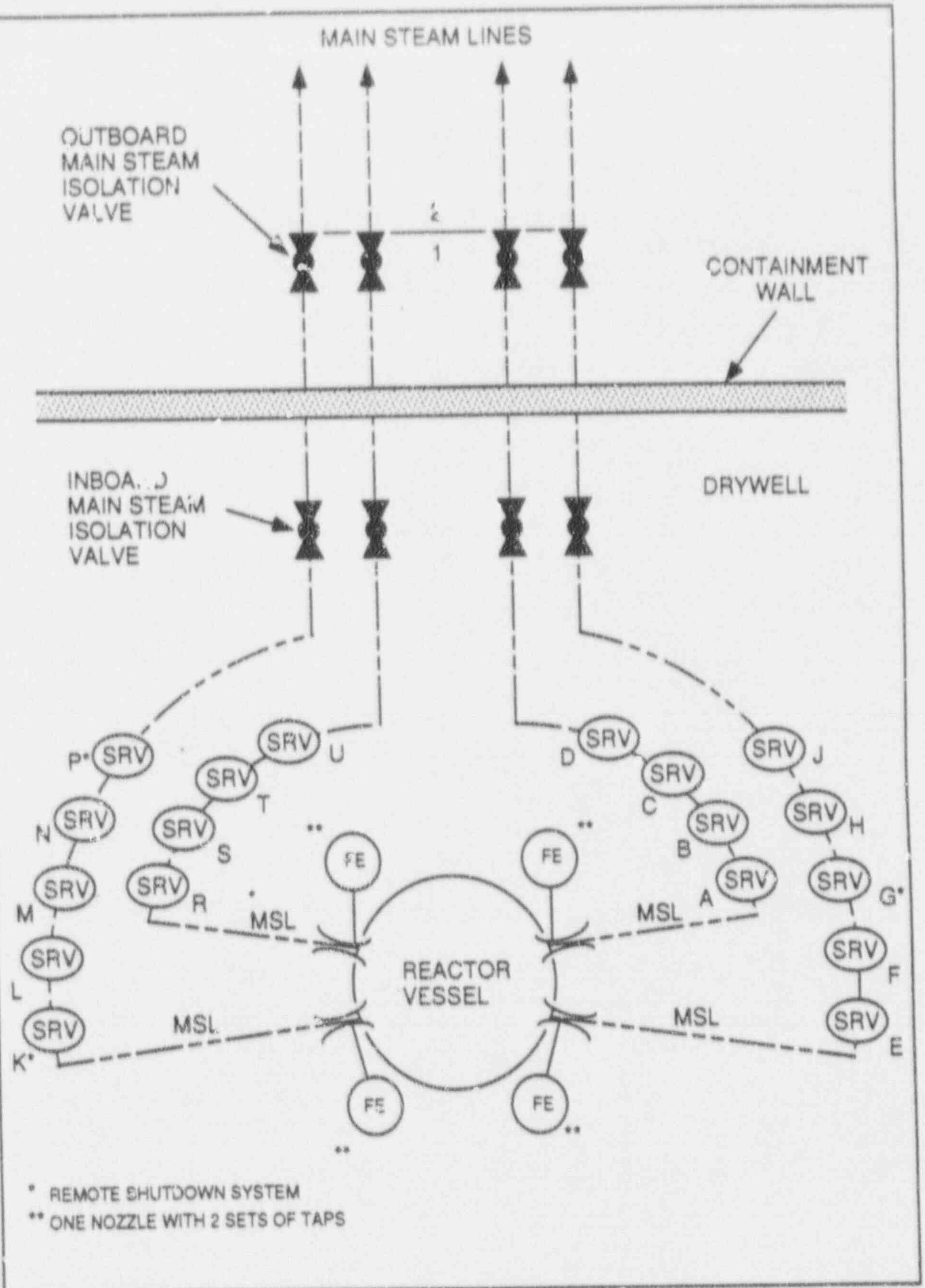
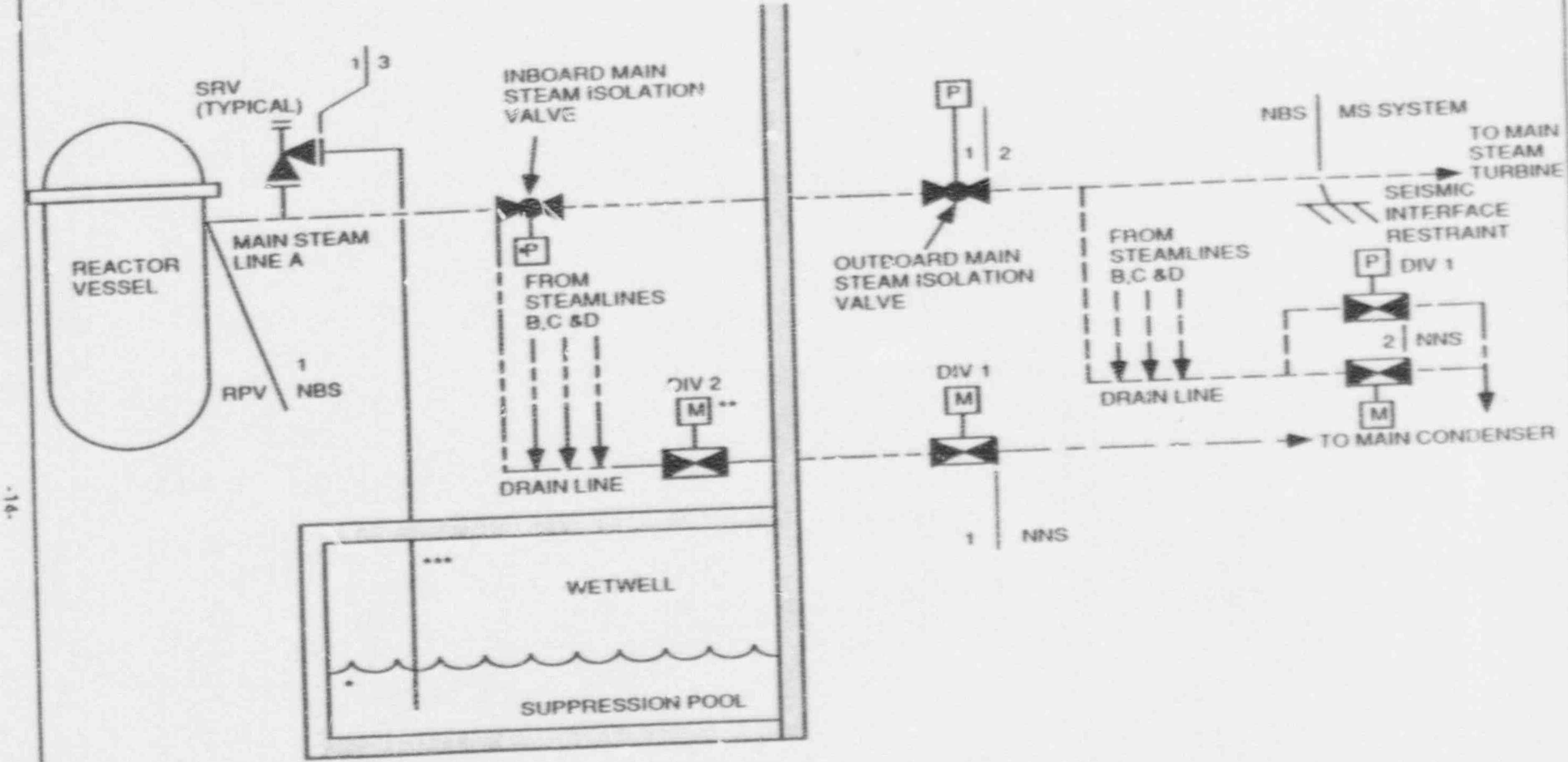


Figure 2.1.2a Safety/Relief Valves and Steamline



* AT MINIMUM WATER LEVEL

** MAY BE PNEUMATIC

*** THE PIPING PRESSURE WELDS IN THE WETWELL AIRSPACE SHALL BE EXAMINED USING ASME CODE CLASS 2 REQUIREMENTS.

Figure 2.1.2b Steamline

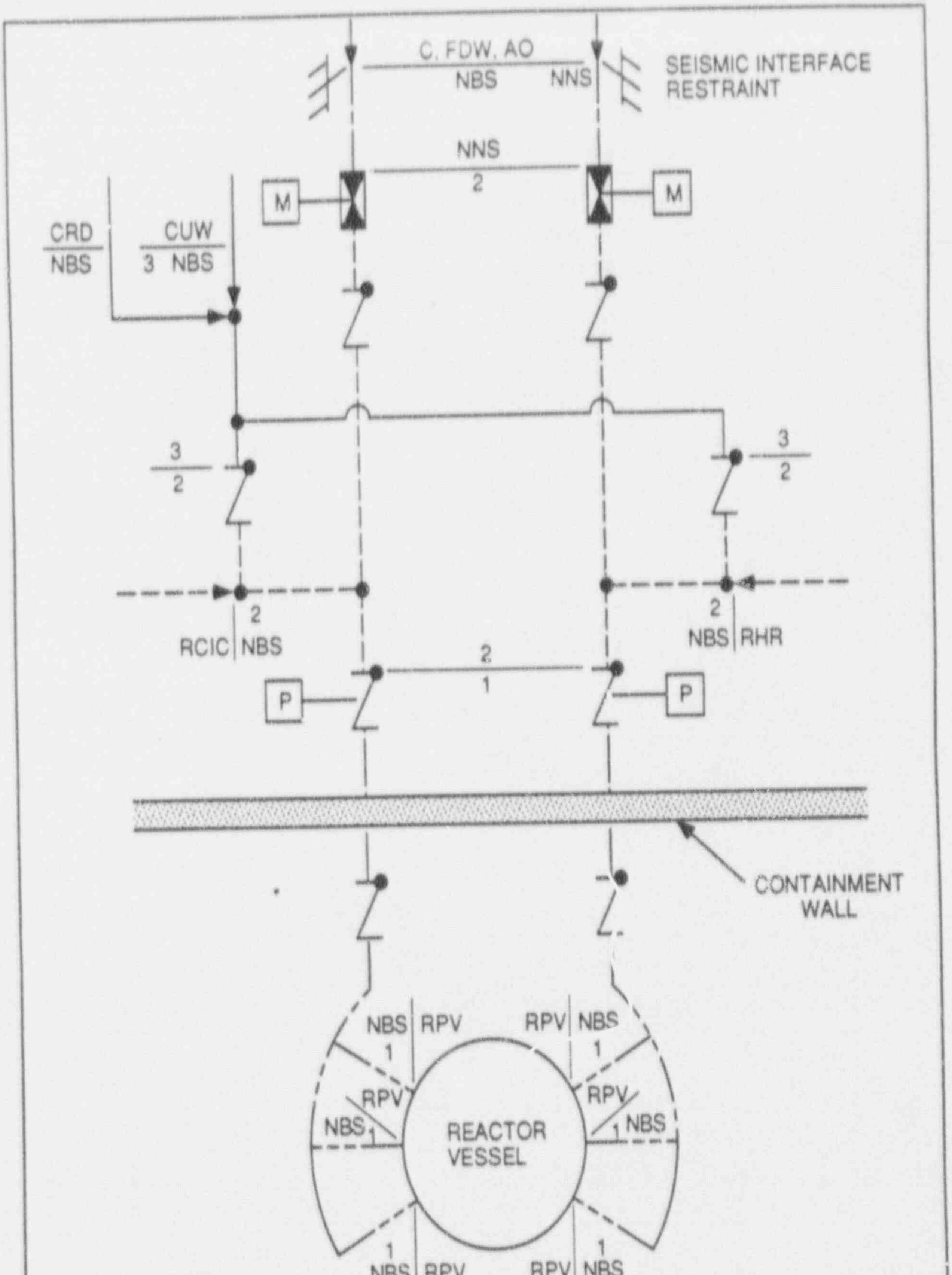


Figure 2.1.2c Feedwater Line

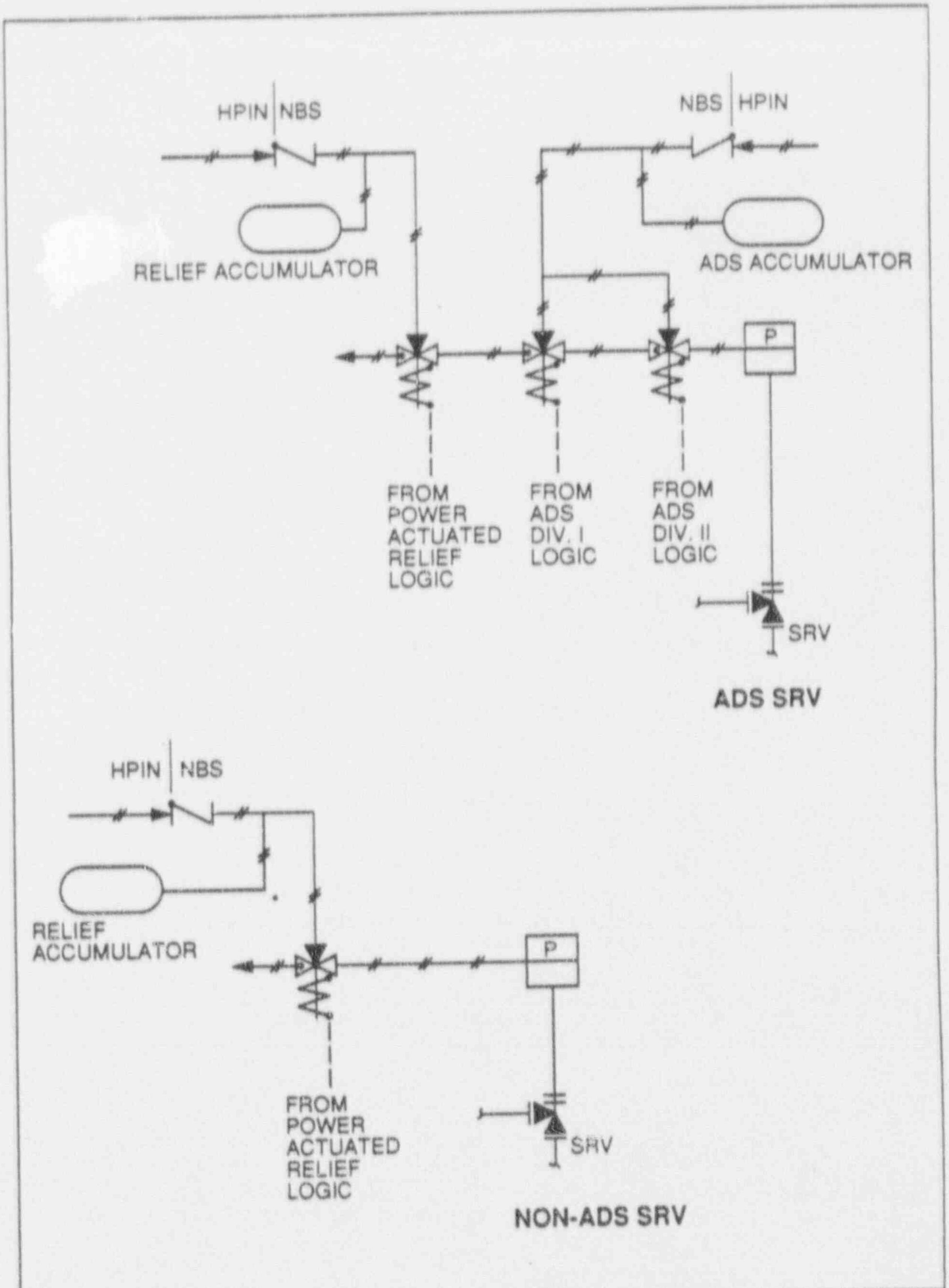


Figure 2.1.2d Safety/Relief Valve Pneumatic Lines

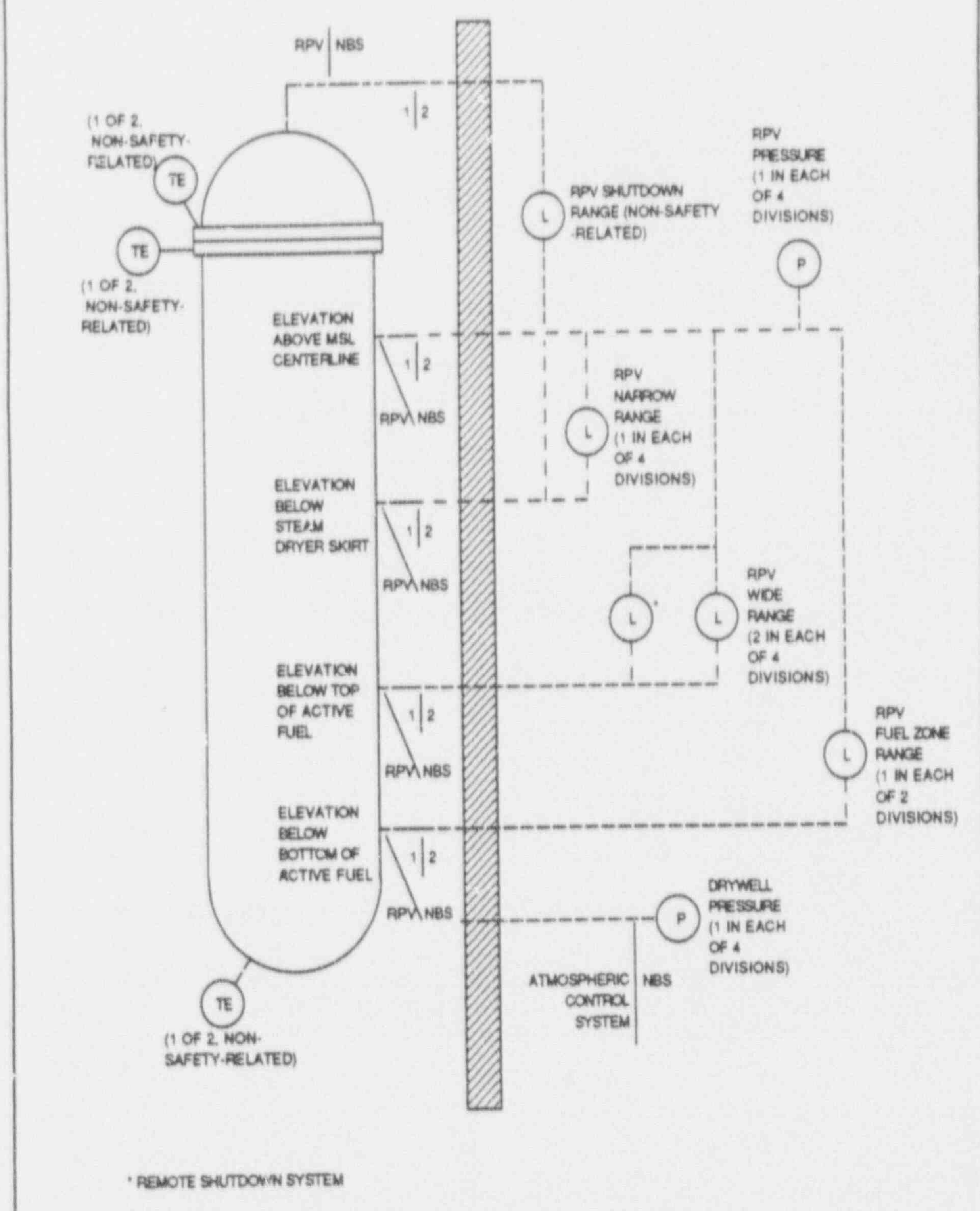


Figure 2.1.2e NBS Drywell Pressure and Reactor Vessel Instrumentation

NUCLEAR BOILER SYSTEM

(A) NEED TO REFERENCE FIGURE 2.1.2 D AND INCLUDE IT. ✓ p3

(B) INSERT BOILER PLATE AND / OR ITAAC TO ADDRESS BWR PRESSURE VESSEL LEVEL PROBLEM. ✓

(C) ADD STD. BOILER PLATE AS SHOWN BY JOHN CHAMBERS (CAPTURE SPECIFIC TIMES FOR TIMERS) ✓

(D) ANALYSIS ALGORITHM TO BE PROVIDED IN TIER 2

(E) ADD BOILER PLATE ✓

~~(F) IDENTIFY EXCEPTIONS TO PHYSICAL SEPERATION~~

(F) STD. ELECTRICAL / I&C SEPERATION ITAAC

(G) CHECK MAIN CONDENSER DESIGN DESCRIPTION FOR INLET PRESSURE AND CONDENSER VACUUM

2.4.3 Leak Detection and Isolation System (LDS)

Design Description

LDS is classified as a safety-related Class 1E control and instrumentation system whose function is to detect and monitor leakage from the reactor coolant pressure boundary (RCPB) 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 LDS include isolation of the main steamlines, the primary and secondary containment, and individual system process lines; activation of the standby gas treatment system; the monitor of leakages inside and outside the primary containment; and indicating the monitored leakage parameters in the control room. The LDS design is fail safe, single failure proof and redundant.

The LDS logic design uses two-out-of-four voting in initiation of each isolation function. Also, the logic is designed to incorporate channel bypass provisions to permit channel test and repair. In the bypass mode, the trip logic utiizes two-out-of-three voting for initiation of the isolation functions.

The LDS safety-related channel measurements are provided as inputs to the safety system and logic control (SSLC) system for signal processing, setpoint comparisons, and generation of the trip signals that initiate the 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 LDS using four instrument channels to monitor leakage:

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

(2) Isolation of the Reactor Water Clean-up (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 equipment areas, or when the Standby Liquid Control System (SLCS) is activated.

(3) Initiation of the Standby Gas Treatment System (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) 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) 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 a high ambient temperature in the RHR loop equipment area.

(8) Isolation of the Reactor Core Isolation Cooling (RCIC) 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 fission products monitor drywell sample and return lines on a signal indicating high drywell pressure or low reactor water level.

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

As shown in Figures 2.4.3, the LDS isolation logic consists of safety related sensors, redundant instrument channels and logic trip units that initiate the automatic isolation functions. Also, separate manual controls in the control room are provided in LDS design for logic reset, MSIV operational control, MSIV closure tests, and for manual isolation.

LDS provides the following control signals to each MSIV pilot solenoid valves:

1. Four divisional control signals are provided to each MSIV solenoids #2 & #3 to open the valve. MSIV closure is automatic on loss of any two divisional signals to both solenoids.

2. Two divisional control signals are provided to each MSIV test solenoid #1 to exercize partial valve closure. Division I or III is used to test close the outboard MSIVs and Division II or IV is used to test close the inboard MSIVs.

Also, 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 two divisional MSIV isolation switches, either Divisions I and IV or II and III.

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 steam line to the RCIC turbine and causes turbine trip. Division I closes the inboard while Division II closes the outboard isolation valves.

Manual reset logic functions are provided at the divisional level to initialize the logic and for logic reset after a isolation has been initiated. Separate reset functions are provided in the LDS logic design for the MSIVs, the RCIC, and the primary and secondary containment isolation circuitry.

The LDS controls and indications are provided in the control room to allow for monitoring and control during operational conditions. The indications in the control room consist of the monitored leakage parameters as defined under the LDS functions.

Each LDS divisional channel is powered from the same divisional power source. Independence is provided between the Class IE divisions, and also between the Class IE divisions and the non-Class IE equipment.

The LDS safety related components and associated hardware are qualified Seismic Category I. The safety related LDS sensors and associated wiring in the reactor and turbine buildings are qualified to operate in a harsh environment.

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 for the Leak Detection and Isolation System.

Table 2.4.3

LEAK DETECTION & ISOLATION SYSTEM

Inspections, Tests, Analyses and Acceptance Criteria

Certified 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 system conforms with the description in Section 2.4.3.
2. LDS monitors and detects leakages from the RCPB, and initiates closure of primary and secondary containment isolation valves.	2. Each LDS instrument channel shall be tested using simulated signal inputs to test the trip condition.	2. Each channel trips.
3. The LDS isolation logic uses four redundant instrument channels to monitor each RCPB leakage parameter. The isolation signal is initiated when any two out of four channels have tripped.	3. The instrument channels of each LDS isolation function shall be tested using simulated signal inputs.	3. Isolation signal is initiated when at least any two out of four channels have tripped.

Table 2.4.3 (CONT'D)

LEAK DETECTION & ISOLATION SYSTEM

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The LDS isolation logic incorporates channel bypass provisions for on line testing and repair. In this mode, the isolation signal is initiated when any two out of three channels have tripped.	4. In channel bypass mode, each LDS logic isolation function shall be tested using simulated signal inputs.	4. Isolation signal is initiated when at least any two out of three channels have tripped.
5. Each MSIV can be subjected to a partial closure test from the control room.	5. Actuate each MSIV test switch to check partial closure of the valve.	5. Each MSIV partially closes and then reopens automatically when its test switch is actuated.
6. LDS provides separate manual controls in the control room for MSIV closure, for closure ~~isolation~~ of the primary and secondary containment isolation valve, and for isolation of the RCIC system.	6. a. Simultaneously actuate two of the four MSIV isolation switches (Div. I & IV or Div. II & III) to close all the MSIVs. Repeat the same test by actuating the other two MSIV isolation switches. b. Actuate each RCIC isolation switch (Div. I or II) to isolate RCIC. c. Actuate each primary and secondary containment isolation switch (Div. I, II & III) to isolate the containment.	6. a. Closure of all the MSIVs occurs only when Divsions I & IV or II & III 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 close only its respective containment isolation valves.

Table 2.4.3 (CONT"D)

LEAK DETECTION & ISOLATION SYSTEM

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. Manual reset controls are provided to perform the manual reset functions as described in Section 2.4.3.	7. Tests will be performed using the LDS reset functions in the control room.	7. The logic circuitry resets for normal operation.
8. Control room indications and controls for this system are defined in Section 2.4.3.	8. Inspections will be performed on the control room indications and controls for this system.	8. Controls exist and indications exist or can be retrieved in the control room as defined in Section 2.4.3.
9. LDS logic design is fail-safe, such that loss of electrical power to one LDS divisional logic channel initiates a channel trip.	9. Tests will be conducted to simulate electrical power failure to each divisional LDS channel.	9. The faulted channel trips.

10 The divisional LDS logic channels and associated sensors are powered from Class IE divisional power.	10 Tests will be performed on the LDS system by providing a test signal in only one Class IE division at a time.	10 The test signal exists only in the Class IE division of the LDS system under test.
11 Independence is provided in the system between Class IE divisions, and between Class IE divisions and non-Class IE equipment.	11 Inspection of the installed LDS Class IE divisions will be performed.	11 Physical separation exists in LDS between Class IE divisions, and between the Class IE divisions and the non-Class IE equipment.

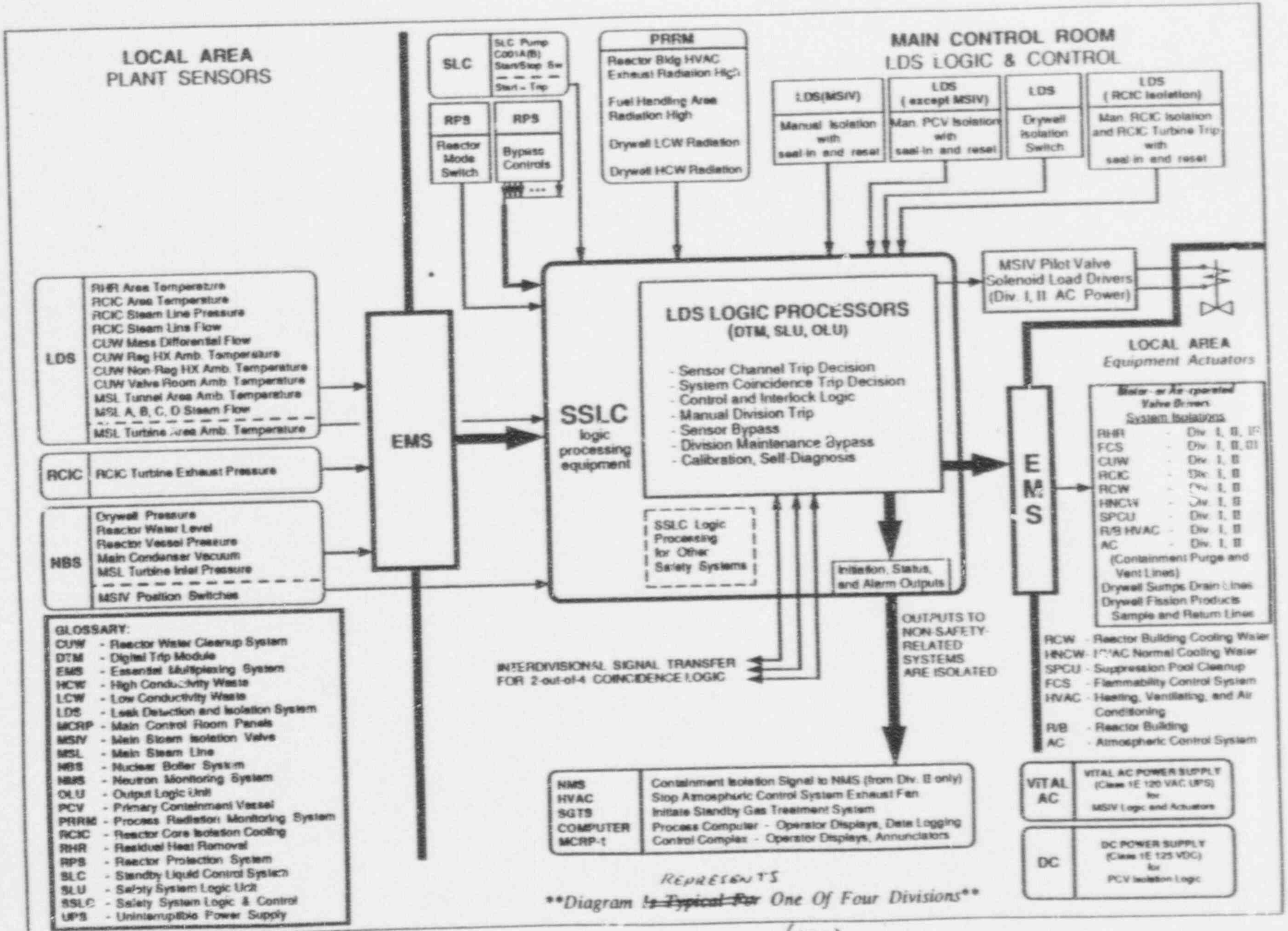


Figure 2.4.3 Leak Detection and Isolation System Interface Diagram

[illegible]

LDS

* MAKE CONFORMING CHANGE TO SECTION 1 TO PICK-UP EQ SEISMIC

(A) ADD A BOILER PLATE FOR CONTROL ROOM CONFIGURATION (MINUS ALARMS) ✓

(B) DEVELOP LOGIC TEST TO DEMONSTRATE FAIL-SAFE PROVISION ON LOSS OF DIVISIONAL POWER ✓

(C) ADD DIVISIONAL POWER SUPPLY BOILER PLATE ✓

(D) ADD PHYSICAL SEPERATION BOILER PLATE ✓
(CHECK ELECTRICAL WRITE-UP)
:IDE DISCUSSIONS

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 ~~separated~~ independent divisions ~~(except for containment isolation valves which are not electrically separated)~~. 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 the Emergency Core Cooling system (ECCS) equipment including the emergency diesel generators (DGs) during a safe reactor shutdown cooling function. RCW system configurations are shown in Figures 2.11.3a, b, and c. All components cooled by the RCW system are parts of other systems and are not part of the RCW 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 ~~RCW subsystem~~ or RCW support system. Each of the three divisions is powered from its respective class 1E division as shown in Figures 2.11.3a, b and c. In case of a failure which disables any one of the three RCW divisions, the other two divisions perform safe reactor shutdown cooling.

A LOCA signal does the following:

(a) starts any standby RCW pumps

(b) opens any closed RCW standby heat exchanger outlet valves

(c) opens all RHR heat exchanger cooling water outlet valves

(d) closes all RCW containment isolation valves ~~(to CUW and HWH heat exchangers)~~

A (e) closes valves to non-safety-related components (to CUW and HWH heat exchangers and RIP Mg Sets), and

(f) overrides any low water level signal which stops operating RCW pumps

(f) opens the RCW water temperature pneumatic control valve (located ~~just~~ downstream of RCW heat exchangers)*

Safety-related valves separate the safety-related portions of the RCW ~~cooled components~~ system from the non-safety-related portions of the system ~~RCW cooled components~~ during a LOCA. The isolation valves to ~~in~~ the non-safety-related RCW system are automatically or remote-manually operated, and their positions are indicated in the main control room.

Each RCW division includes two pumps which circulate 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.

*AND CLOSED THE RCW HTX B/P VALVE

Insert O

The RCW system main control room (MCR) indication and controls allows monitoring and control during operational conditions. The control room has control of motor operated valves (MOVs) and pneumatically operated valves (POVs), the number of RCW pumps and heat exchangers in operation and flow rate to the components being cooled. Main Control Room instrument indication is provided for each RCW division for surge tank level, cooling water radiation level and Residual Heat Removal (RHR) heat exchanger flow rate and temperature water supply temperature and pressure, water radiation level, surge tank water level.

The RCW system components with status indication and/or control interfaces with the Remote Shutdown System (RSS) are shown in Figure 2.11.3a, b and c.

2.11.3a, b and c

The RCW ASME Code classifications for different portions of the system are indicated on Figures 2.11.3a-d. The safety-related portions of the RCW divisions are classified as Seismic Category I and are located in Seismic Category I structures.

Component design parameters are:

C

	Division A/B	Division C
~~Design pressure (kg/cm²a)~~	~~14~~	~~14~~
~~Design temperature (°C)~~	~~70~~	~~70~~
Discharge flow rate lpm/pump)	≥ 21,700	≥ 18,200
~~Pump total head (kg/cm²a)~~	~~≥ 6.8~~	~~≥ 6.5~~
Heat exchanger capacities are each:	≥ 11.5×10^6 kCal/hr	≥ 10.6×10^6 kCal/hr

~~Connections to a radiation monitor are provided in each division to detect radioactive contamination resulting from a tube leak in a heat exchanger.~~

D

2.11.3a, b and c

MATCH BOILERPLATE WORDS

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-c). All safety-related electrical equipment in the reactor building are ~~environmentally~~ qualified for a harsh ~~service~~ ENVIRONMENT. Tables 2.11.3b, c, and d show which equipment receives RCW flow during various plant ~~operation~~ operating and emergency conditions. The tables also indicate how many heat exchangers are in service under each condition.

~~The following cooling loads are in all three RCW divisions: RHR heat exchangers, RHR motor and seal coolers, RHR room coolers and HECW refrigerators. All other cooling loads are in either one or two RCW divisions. For these cooliong loads, changes may be made in the divisional assignment if it can be shown that the cooling capacity margins in all divisions have not been significantly reduced by the changes.~~

Insert

O

~~BOILER PLATE DESIGN DESCRIPTION WORDS FOR~~ RSS + ~~MCR FEATURES ETC.:~~

THE RCW SYSTEM CONTROL ROOM INDICATIONS, ~~ALARMS~~, AND CONTROLS* ALLOW FOR MONITORING AND CONTROL DURING OPERATIONAL CONDITIONS ~~MODES~~. THE CONTROL ROOM HAS INDICATION FOR SYSTEM FLOWS, TEMPERATURES, AND PRESSURES, AS WELL AS ~~[illegible]~~ PNEUMATIC AND MOTOR-OPERATED VALVE OPEN/CLOSE AND PUMP ON/OFF STATUS. ~~CONTROLS FOR THESE COMPONENTS ARE IDENTIFIED IN SECTION [illegible] 2.11.3~~.

THE RCW SYSTEM COMPONENTS WITH STATUS INDICATION AND ~~/OR~~ CONTROL INTERFACES WITH THE REMOTE SHUTDOWN SYSTEM (RSS) ARE IDENTIFIED IN ~~SECTION ____~~.
~~Figures 2.11.3a, b and~~ c.
~~Section~~ 2.11.3.a and b
FIGURES

~~* MODIFY WORDS TO FIT SYSTEM~~

00 C

~~The RCW heat exchanger heat removal capacities include 20 per cent margin above the design heat removal requirements* which are: 2.80 x 10^6 kcal/h for division A, 2.84 x 10^6 kcal/h for division B and 26.4 x 10^6 kcal/h for division C.~~

*for the LOCA condition

During plant operation, RCW flows through equipment which is operating and requires cooling. RCW also flows through safety-related coolers except RHR heat exchangers and EDG coolers, as shown ~~by open or closed valves~~ in Figures 2.11.3a, b and c.

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 MCR signal. Low water level signals in the surge tanks ~~and standpipe~~ do the following (in order of decreasing level):

(a) (low) opens the MUWP makeup water valve

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

(c) (low-low-low) stops any operating RCW pumps

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

The pneumatic valves of safety significance fail as follows: The MUWP ~~majeup~~ makeup water valves fail open ~~and~~ the RCW water temperature control valves fail open, AND THE RCW HTX O/P VALVES FAIL CLOSED

Inspections, Tests, Analyses and Acceptance Criteria

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

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. ~~A~~ The basic configuration ~~for~~ of the RCW system is as shown on Figures 2.11.3a, b and c.	1. ~~Visual~~ Inspections of the as-built system ~~configuration~~ will be conducted.	1. The as-built ~~configuration of the~~ RCW system ~~is in accordance with~~ conforms with the basic configuration shown in Figures 2.11.3a, b and c.
2. The ASME ~~pressure boundary~~ 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.
3a. Control room ~~features~~ ~~alarms~~ Indications and controls provided for RCW system are as defined in Section 2.11.3.	3a. Inspections will be performed on the Control Room ~~features~~ ~~alarms~~, indications and controls for the RCW system.	3a. ~~Features~~ ~~Alarms~~ Indications and controls exist or can be retrieved in the Control Room as defined in Section 2.11.3.
3b. Remote Shutdown System (RSS) ~~features~~ indications and controls provided for the RCW system are as defined in Section 2.11.3.	3b. Inspections will be performed on the RSS ~~features~~ indications and controls for the RCW system.	3b. ~~Features~~ Indications and controls exist on the RSS as defined in Section 2.11.3.
4. ~~Safety-related electrical power loads for~~ Class 1E loads of the RCW System are powered from Class 1E Divisions, as described in Section 2.11.3.	4. Tests will be performed for the components in Section 2.11.3 by checking voltage [illegible] the electrical loads.	4. Safety-related electrical power loads for the RCW System are powered from Class 1E Divisions, as described in Section 2.11.3. (1)
5. The safety-related mechanical portion of ~~each~~ division of the RCW system (~~Loops~~ Divisions A, B, C) is physically separated.	5. ~~Visual~~ Inspections of the as-built RCW system as will be performed.	5. A room outside the control room and primary containment does not contain safety-related mechanical components from more than one loop of the RCW system. (2)
6. The RCW System responses to a LOCA signal are as specified in Section 2.11.3.	6. Using simulated LOCA signals, tests will be performed for the RCW System.	6. Upon receipt of simulated LOCA signals, the response of the RCW System ~~satisfy the Certified Design Commitments~~ are as specified in Section 2.11.3.
7. The RCW pump capacity is as specified in Section 2.11.3.	7. System ~~hydraulic~~ tests will be conducted on the pumps in all ~~the~~ operational modes specified in Tables 2.11.3b, c and d.	7. The pump discharge flow rates ~~satisfy the Certified Design Commitment~~ are as specified in Section 2.11.3.
8. The RCW heat exchanger capacities ~~is as~~ are as specified in Section 2.11.3.	8. A documentation review shall be performed to determine the heat removal capacity of the RCW heat exchangers. see RHR	8. The heat removal capacities of the RCW heat exchangers ~~satisfies the Certified Design Commitment~~ are as specified in Section 2.11.3.

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F

9. The RCW pump flow capacities and the RCW heat exchanger heat removal capacities are as specified in Section 2.11.3.	9. 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 analysis will be performed to estimate the heat removal capacities of the RCW heat exchangers. Inspections and analysis will be performed to estimate the heat removal requirements of the as-built components which are cooled by the RCW System during LOCA conditions.	9. 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.
10. A surge tank with a capacity of at least 16 cubic meters is provided for each RCW division.	10. Inspection and a volume calculation using as-built dimensions will be performed.	10. The capacity of each surge tank~~s~~ is greater than or equal to 16 cubic meters.

11. Motor-operated valves (MOV) designated in Section 2.11.3 as having an active safety-related function will open and/or close under differential pressure and fluid flow conditions.	11. Opening and/or closing tests of installed valves will be conducted under pre-op differential pressure, fluid flow, and temperature conditions.	11. Each MOV opens and/or closes.
SAME 12. The pneumatic valves ~~of~~ HAVING AN ACTIVE SAFETY RELATED safety significance fail in a safe condition as specified in Section 2.11.3. DON'T INCLUDE	OPENING AND/OR CLOSING 12. Tests will be performed. ~~of the pneumatic valves of safety significance.~~	PNEUMATIC VALVE EACH ~~MOV~~ OPENS AND/OR CLOSE 12. The safety significant pneumatic valves fail in a safe position as specified in Section 2.11.3.

(G) 12. SAME AS NUCLEAR BOILER ITAAC 15

Table 2.11.3b: Reactor Building Cooling Water ~~Consumers~~ 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				
RWCU Heat Exchanger	X	X	X	-
Inside Drywell	X	X	X	-
Others (non-safety-related)	X	X	X	X

Notes

(x) = Equipment receives RCW in this mode.

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

* 3

** HECW refrigerator, room coolers (FCP ~~pump~~, RHR RCIC, SGTS, FCS, CAMS) RHR motor, BEARING and seal coolers, AND CAMS COOLER

E

Table 2.11.3c: Reactor Building Cooling Water ~~Consumers~~ 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

(x) = Equipment receives RCW in this mode.

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

* 3

** HECW refrigerator, room coolers (FPC ~~pumps~~, RHR ~~[illegible]~~, SGTS, FCS, CAMS), RHR motor, bearing and seal coolers, AND CAMS COOLER
HPCF

E

Table 2.11.3d: Reactor Building Cooling Water ~~Consumers~~ 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

(x) = Equipment receives RCW in this mode.

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

* 3

** HECW refrigerator; room coolers, motor BEARING coolers, and mechanical seal coolers for RHR and HPCF.

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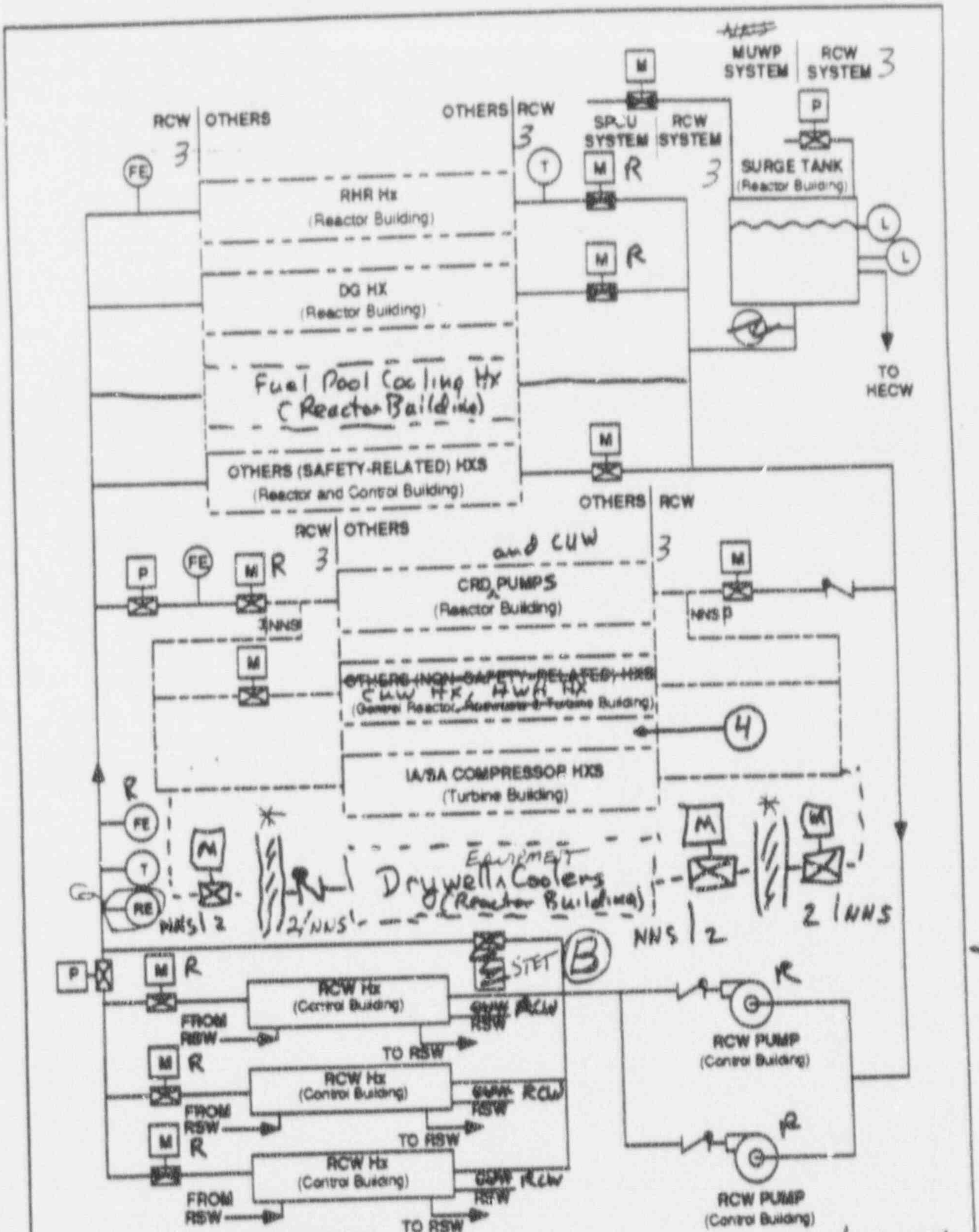

THIS DIVISION IS POWERED FROM CLASS 1E DIVISION #I, except for the containment outboard isolation valves which is powered by Class 1E DIVISION II.

Figure 2.11.3a RCW Division - A

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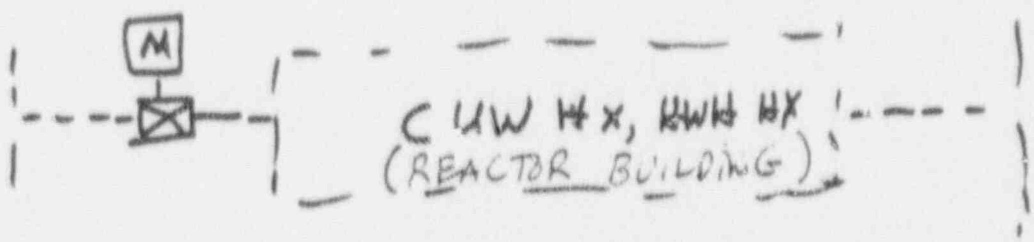


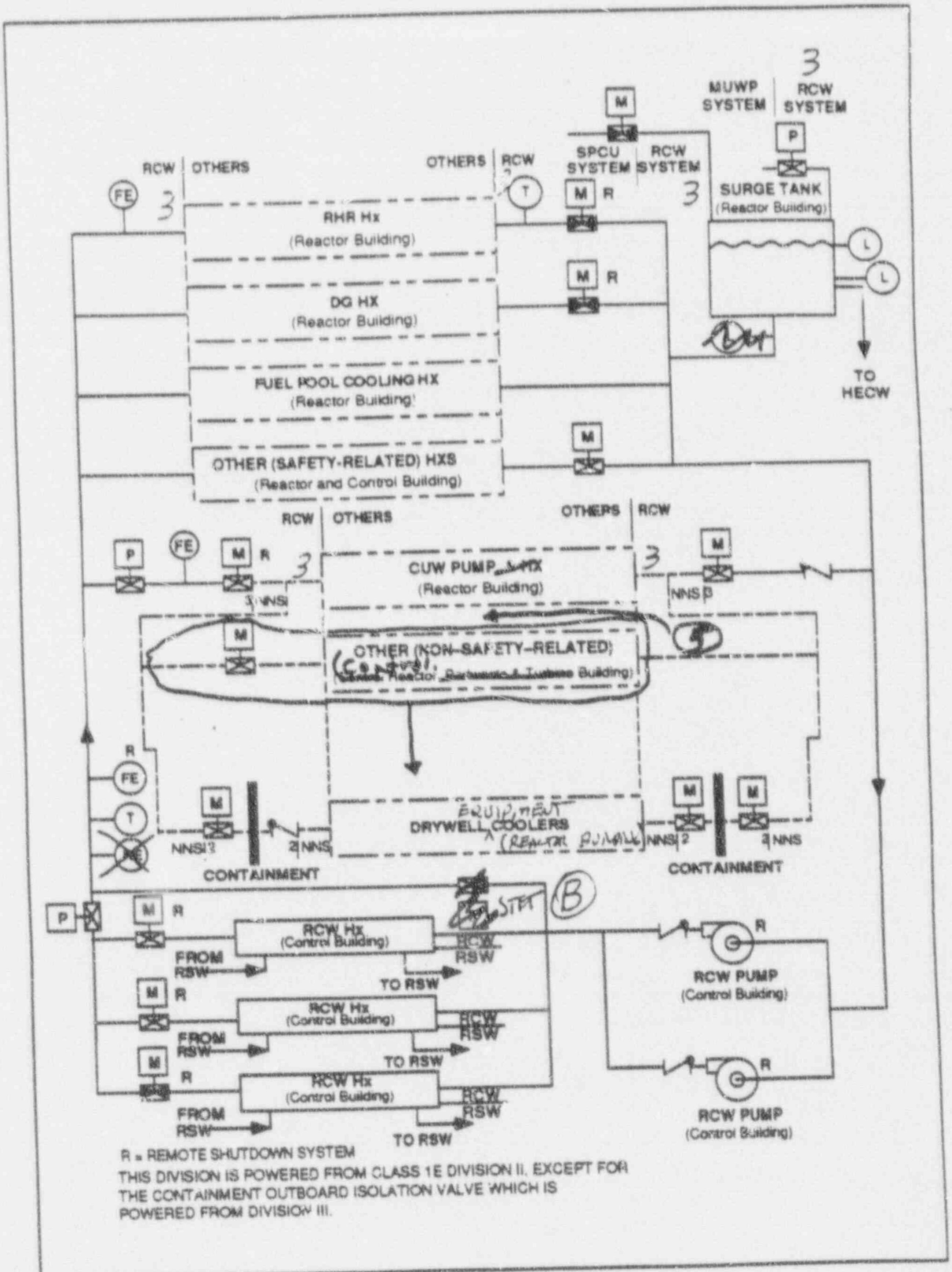
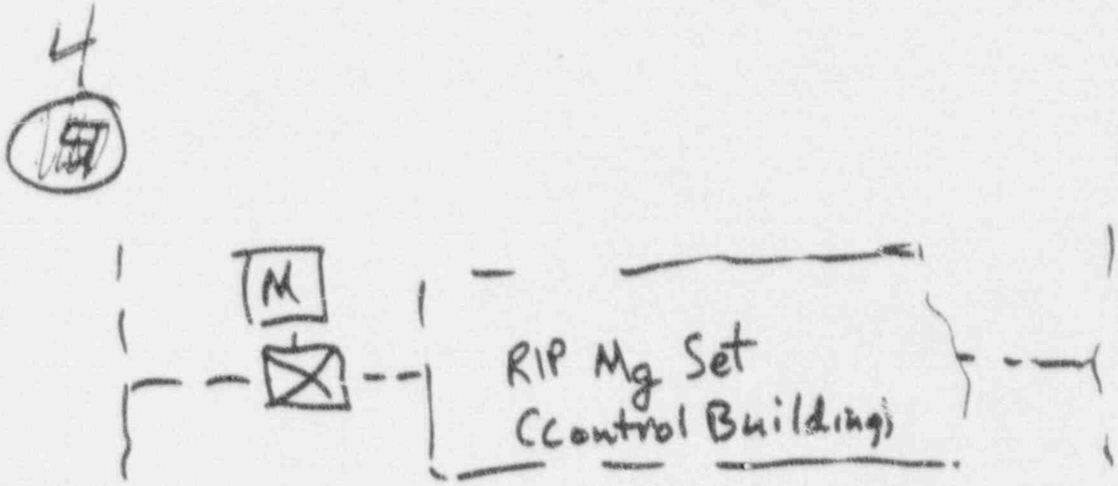


Figure 2.11.3b RCW Division - B


4
M
RIP Mg Set
(Control Building)

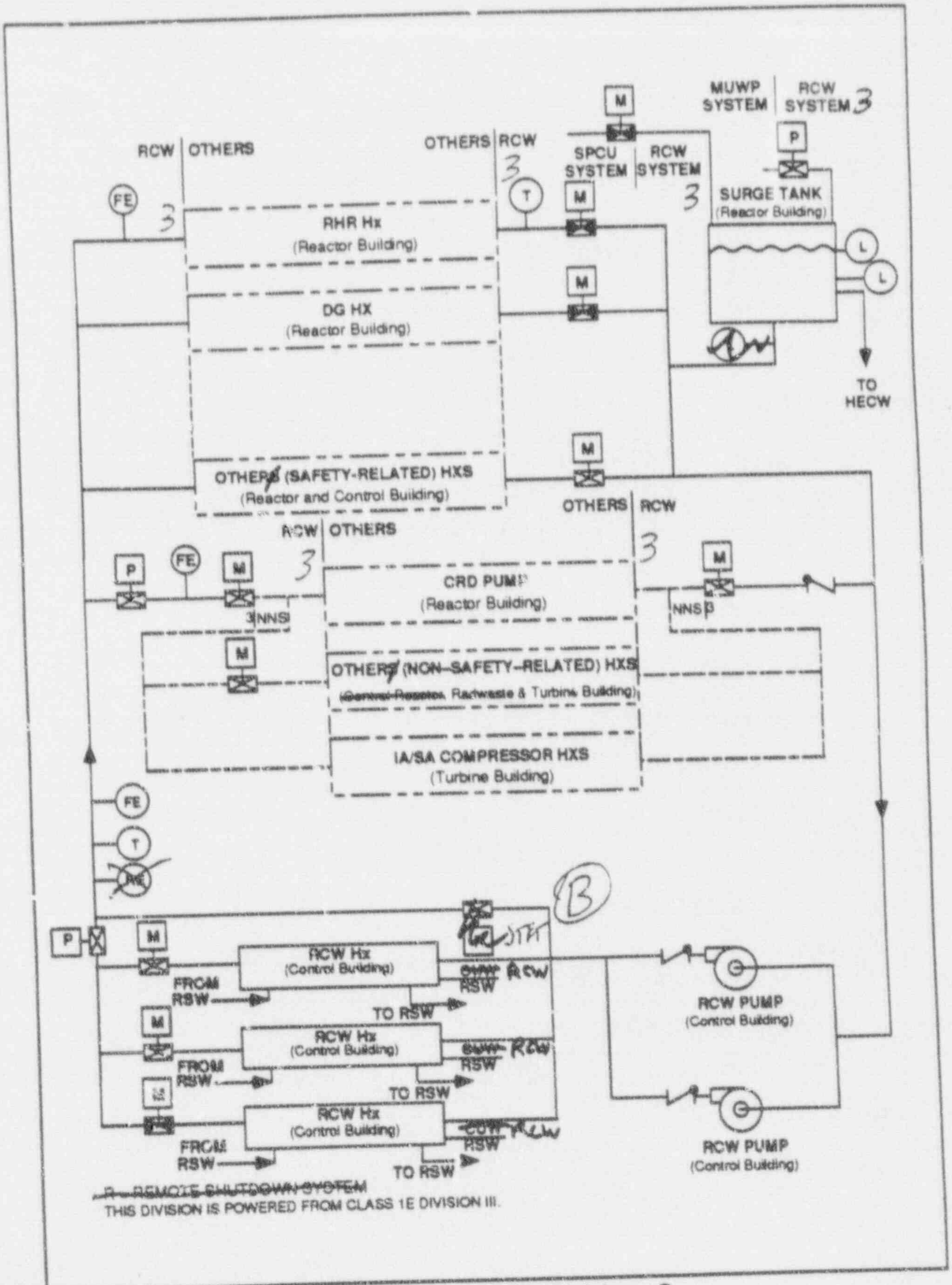


Figure 2.11.3c RCW Division - C

REACTOR BUILDING COOLING WATER

(A) ~~NRC TO VERIFY ACCEPTABILITY GE CONTINUING TO PROVIDE FLOW TO NON-SAFETY COMPONENTS DURING A LOCA (LOW STANDPIPE ISOLATES ALL NON-SAFETY LOADS)~~

(B) GE TO VERIFY ACTUATION ON LOCA AND MAKE TEXT CHANGES TO DESIGN DESCRIPTION AND SSAR (VALVE STAYS IN)

(C) ~~GE TO DETERMINE DESIGN BASIS HEAT REMOVAL CAPACITIES AND SPECIFY (MARGINS DISCUSSION TO BE IN THE SSAR)~~

(D) GE INCLUDE RADIATION MONITOR IN PROCESS RADIATION MONITOR SYSTEM

(E) ~~GE TO VERIFY LOADS ON EACH DIVISION~~

(F) ANALYSIS METHODS TO BE DISCUSSED IN SSAR (CONSISTENT WITH RHR DISCUSSION ON HEAT LOADS)

(G) ~~SAME AS NUCLEAR BOILER ITAAC 15~~

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

Each division is powered by its respective Class 1E Division and is located in a separate room in the Control Building.

The RSW System Control Room indications and controls allow for monitoring and control during operational conditions. The control room has controls and indications for motor-operated valve open/close status. On a LOCA signal, any closed valves for standby heat exchangers will be automatically opened. The RSW System components with status indication and control interfaces with The Remote Shutdown System (RSS) are identified in Figure 2.11.9.

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.

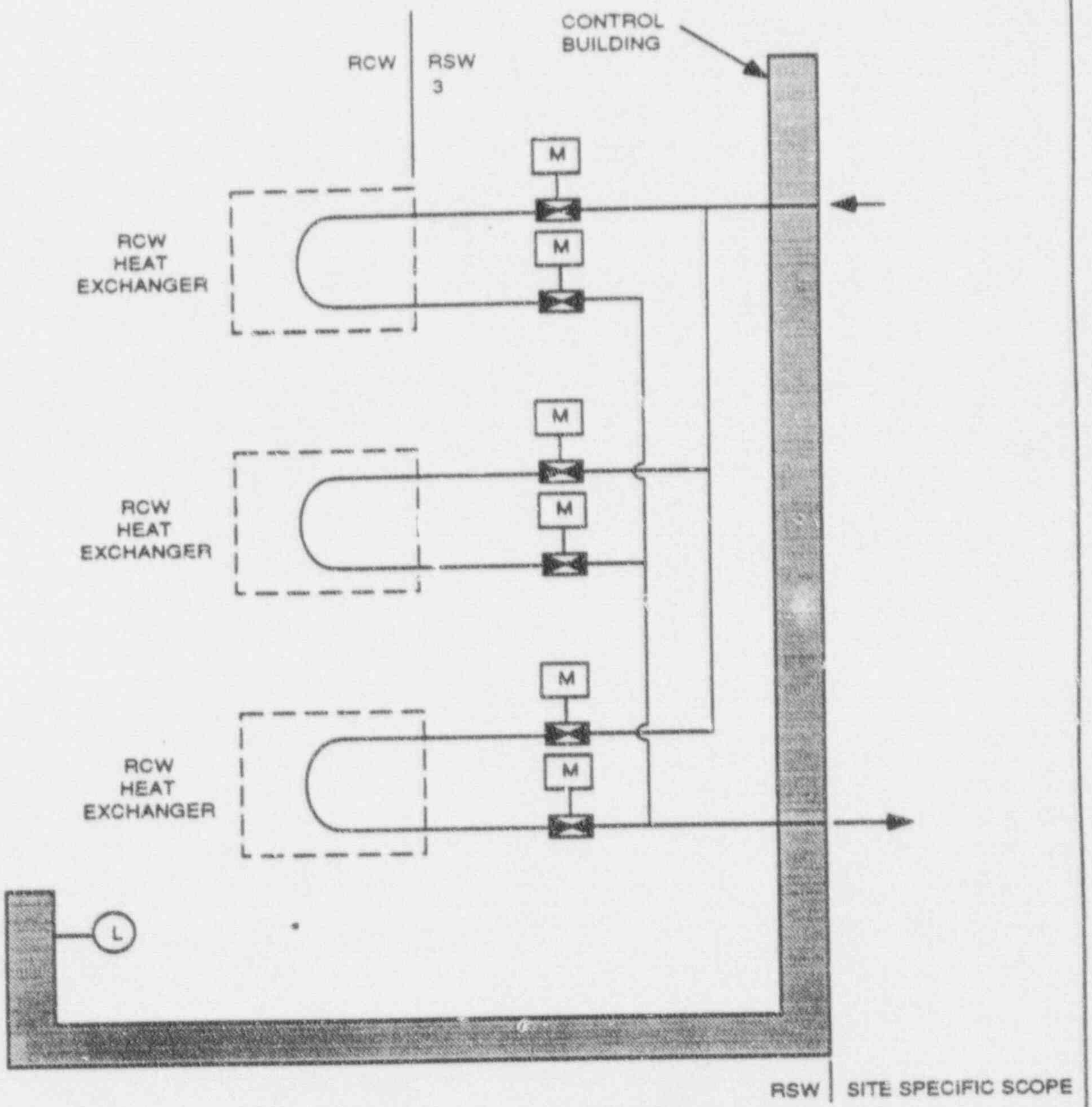
Inspections, Tests, Analyses and Acceptance Criteria

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.

Table 2.11.9: Reactor Service Water System

Inspections, Tests, Analyses and Acceptance Criteria

Certified 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 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 conform with the requirements in the ASME Code, Section III.
3a. Control room indications and controls provided for RSW System are defined in Section 2.11.9.	3a. Inspections will be performed on the control room indications and controls for the RSW System.	3a. Indications and controls exist or can be retrieved in the control room as defined in Section 2.11.9.
3b. Remote Shutdown System (RSS) indications and controls provided for the RSW System are as defined in Section 2.11.9.	3b. Inspections will be performed on the RSS indications and controls for the RSW System.	3b. Indications and controls exist on the RSS as defined in Section 2.11.9.
4. Each mechanical division of the RSW System (Divisions A, B, C) is physically separated.	4. Inspections of the as-built system will be performed.	4. Each mechanical division of the RSW System is physically separated from other mechanical divisions of the RSW System.
5. Any closed standby heat exchanger inlet or outlet valves automatically open upon receipt of a LOCA signal.	5. Using simulated LOCA signals, tests will be performed on standby heat exchanger inlet and outlet valves.	5. Upon receipt of simulated LOCA signals, the standby heat exchanger inlet and outlet valves open.
6. Class 1E loads for the RSW System are powered from Class 1E Divisions, as described in Figure 2.11.9.	6. Tests will be performed on the RSW System by providing a test signal in only one Class 1E Division at a time.	6. The test signal exists only in the Class 1E Division under test in the RSW System.
7. Motor-operated valves designated in Section 2.11.9 as having an active safety function will open and/or close under differential pressure and fluid flow conditions.	7. Opening and /or closing tests of installed valves will be conducted ~~and~~ UNDER preoperational differential pressure, fluid flow, and temperature conditions.	7. Each MOV opens and/or closes.



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

VALVES SHOWN ABOVE IN DIVISIONS A AND B HAVE CONTROLS AND OPEN/CLOSE STATUS INDICATION ON THE REMOTE SHUTDOWN PANEL.

Figure 2.11.9 Reactor Service Water System

4.5 REACTOR SERVICE WATER SYSTEM

Interface Requirements

Interface requirements are in Section 2.11.9.

~~RSW~~ REACTOR SERVICE WATER

(A) CORRESPONDING SSAR CHANGE NEEDED TO ADDRESS MAXIMUM FLOOD HEIGHT ?

(B) WATER LEVEL SENSOR MOVED TO CONTROL ~~ROOM~~ 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

On 1/21/93:

(A) check ASME standardization words for classification of piping.

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 suppression pool. The CUW System also provides processed water to the 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 containment isolation valves (CIV) automatically close upon receipt of an isolation signal from the Leak Detection and Isolation System (LDS).

The suction valves (containment isolation valves) are designed to close against a maximum differential pressure of 87.9 kg/cm^2 within 30 seconds upon receipt of isolation signal. The inboard containment isolation valve is powered from Class 1E Division II AC bus, and the outboard containment isolation valves are fed from Class 1E Division I AC bus.

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 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.

The safety-related electrical equipment (including instrumentation and controls shown in Figure 2.6.1) located in the containment and reactor building is qualified for a harsh environment.

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 designed to ASME Code Class 1 requirements and classified as Seismic Category I. The remainder of the piping system is designed to ASME Code Class 3 requirements and classified as non-Seismic Category I.

The 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.

The CUW System control room indication and controls allows for monitoring and control during operational conditions. The control room has indication for and/or control of the containment isolation valves.

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.

Table 2.6.1: Reactor Water Cleanup System

Inspections, Tests, Analyses and Acceptance Criteria

Certified 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 show in Figure 2.6.1.
2. Motor-operated valves (MOV) designated in Section 2.6.1 as having an active safety-related function will open and/or close under differential pressure and fluid flow conditions.	2. Opening and/or closing tests of installed valves will be conducted under pre-op differential pressure, fluid flow, and temperature conditions.	2. Each MOV opens and/or closes. The following valves open and/or close in the following time limits: Valve / Time / Open/Close Suction line inboard CIV / equal or less than 30 sec. / Close Suction line outboard CIV / equal or less than 30 sec. / Close
3. The ASME Code components of the CUW System retain their pressure boundary integrity under internal pressures that will be experienced during service.	3. A hydrostatic test will be conducted on those Code components of the CUW System required to be hydrostatically tested by the ASME Code.	3. 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.
4. Control room indications and/or controls provided for CUW System are as defined in Section 2.6.1.	4. Inspections will be performed on the control room indications and/or controls for the CUW System.	4. Indications and/or controls exist or can be retrieved in control room as defined in Section 2.6.1.
5. Maximum throat diameter of the CUW suction line flow restrictor is 135 mm.	5. Inspection will be conducted on the CUW suction line flow restrictor throat diameter.	5. Maximum throat diameter of the CUW suction line flow restrictor is 135 mm.
6. 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.	6. Inspection will be conducted on the elevation of the center line of the vessel bottom head drain line tee connection.	6. 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.

Table 2.6.1: Reactor Water Cleanup System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitments	Inspections, Tests, Analyses	Acceptance Criteria
7. Class 1E loads of the CUW System are powered from Class 1E Divisions, as described in Section 2.6.1.	7. Test will be performed on the CUW System by providing a test signal in only one Class 1E Division at a time.	7. The test signal exists only in the Class 1E Division under test in the CUW System.
8. CUW containment isolation valves automatically close upon receipt of CUW isolation signal (from LDS).	8. Using simulated CUW isolation signals, tests will be performed on the (CUW containment isolation valves) isolation logic.	8. Upon receipt of a simulated CUW isolation signal, the CUW containment isolation valves automatically close.

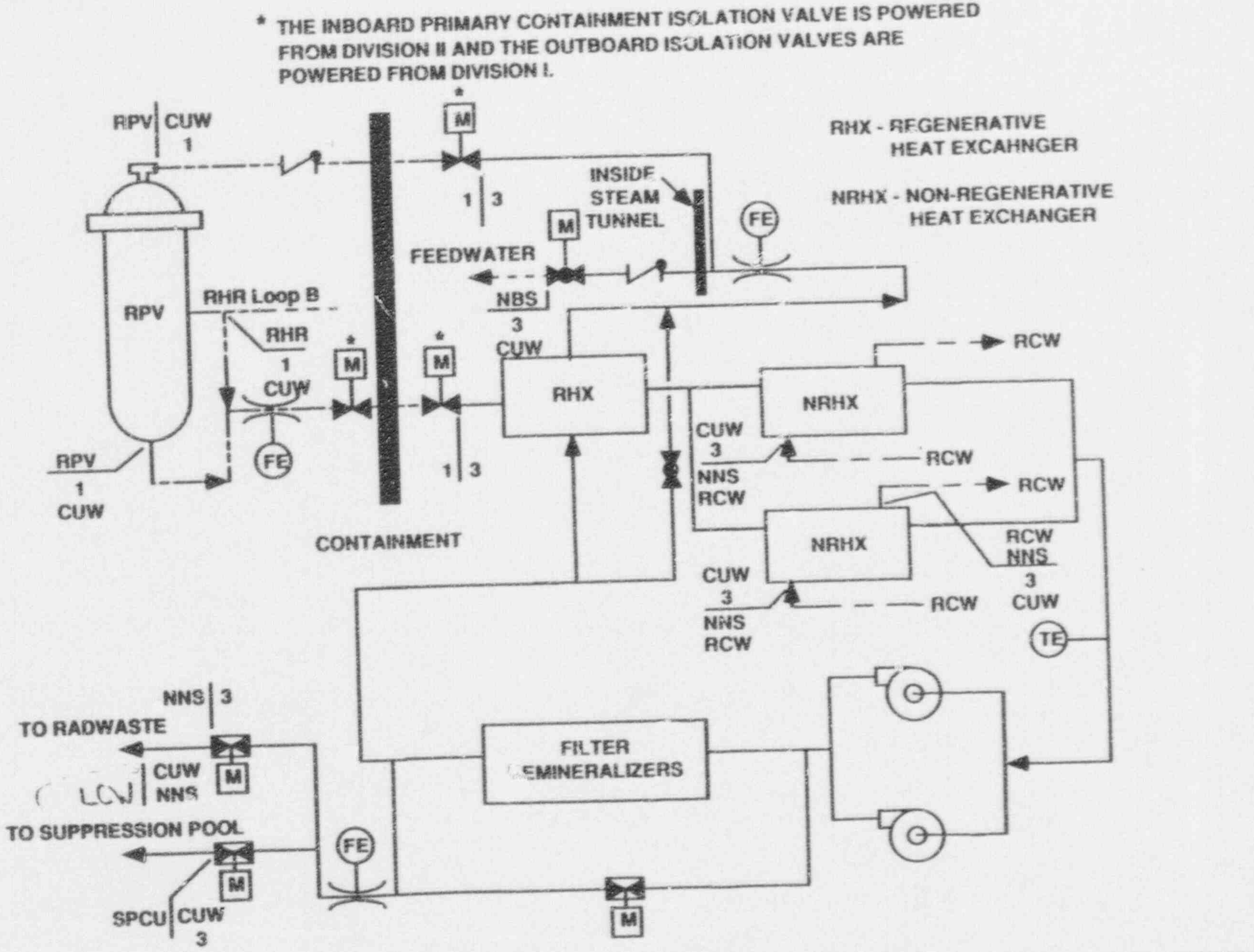


Figure 2.6.1 Reactor Water Cleanup (CUW) System Schematic

REACTOR WATER CLEANUP SYSTEM

(A) GE TO CONFIRM 460 MM IS AT CENTERLINE ?

(B) GE TO MODIFY WORDS (TEXT ALSO) TO REFLECT FLOW RESTRICTOR FUNCTION ✓

(C) GE TO CHECK STANDARDIZATION WORDS FOR CLASSIFICATION OF PIPING ?

2.2.4 Standby Liquid Control System

The Standby Liquid Control (SLC) System injects neutron absorbing poison using a boron solution into the reactor and thus provides back-up reactor shutdown capability independent of the normal reactivity control system based on insertion of control rods into the core. The SLC System is capable of operation over a range of reactor pressure conditions which bound the elevated pressures associated with an anticipated plant transient coupled with a failure to scram (ATWS) when SLC is required to operate.

The SLC System is designed to bring the reactor, at any time in a core cycle, and at design basis conditions, from full power to a subcritical condition, with the reactor in the most reactive xenon-free state, without control rod movement. With the storage tank at minimum level and both pumps operating, the system will inject the minimum required boron solution.

The SLC System as shown in Figure 2.2.4 is located in the Reactor Building which is classified as seismic Category I and consists of a boron solution storage tank, two positive displacement pumps, two motor-operated injection valves which are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged through the 'B' high pressure core flooder (HPCF) subsystem sparger. Key equipment performance requirements are:

(1)	Pump flow (minimum)	22.7 m^3/hr with both pumps running
(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 is designed to 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 are actuated by each division switch:

(1) The specified division injection valve is opened.

(2) The specified division suction valve is opened.

(3) The specified division injection pump is started.

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

The SLC System is automatically initiated during an ATWS. When the SLC System is automatically initiated to inject a liquid neutron absorber into the reactor, both divisions are actuated.

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 average 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 (less than or equal to 455×10^3 kg) plus the mass of water in the RPV shutdown cooling piping (less than or equal to 130×10^3 kg). At least this quantity of boron solution is contained above the pump suction shutoff level in the tank.

The SLC System pumps are capable of producing discharge pressure to inject the solution into the reactor when the reactor is at pressure conditions corresponding to the system relief valve setpoint (109.7 kg/cm2 gauge), which is above peak ATWS pressure.

The SLC System control room indications, alarms, and controls allow for monitoring and control during operational conditions. The control room has indication for pump discharge pressure, storage tank liquid level, injection and suction valve open/close, and pump on/off.

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

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

A test tank and associated piping and valves permit system testing. The tank is supplied with demineralized water which is pumped in a closed loop through either pump or injected into the reactor.

The SLC System is physically separated from and independent of the hydraulic portion of the Control Rod Drive System which performs the scram function.

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.

Table 2.2.4: Standby Liquid Control System

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The performance of the SLCS is based on the following plant parameters: a. Storage Tank pumpable volume is greater than or equal to 23.1 m^3. b. RPV water inventory is less than or equal to 455 x 10^3 kg at normal water level and 20°C. c. RHR shutdown cooling system inventory is less than or equal to 130 x 10^3 kg at 20°C.	1. The as-built dimensions will be used in a volumetric analysis to calculate the volumes listed below: a. Minimum Storage tank pumpable volume b. RPV water inventory at normal water level and 20°C. c. RHR shutdown cooling system water inventory at 20°C.	1. a. Storage tanks pumpable volume is greater than or equal to 23.1 m^3. b. RPV water inventory is less than or equal to 455 x 10^3 kg at 20°C. c. RHR shutdown cooling system inventory is less than or equal to 130x 10^3 kg at 20°C.
2. The basic configuration of the SLC System is as shown in Figure 2.2.4.	2. Inspections of the as-built system will be conducted.	2. The as-built SLC System conforms with the basic configuration shown in Figure 2.2.4.
3a. The SLC System delivers at least 378 liters/minute of solution with both pumps operating against the elevated pressure conditions in the reactor during events involving SLC System initiation.	3a. Using installed controls, power supplies and other auxiliaries, demineralized water will be injected from the storage tank into the RPV with both pumps running against a discharge pressure of greater than or equal to 88.9 kg/cm^2a.	3a. The SLC System injects greater than or equal to 378 liters/minute into the RPV with both pumps running against a discharge pressure of greater than or equal to 88.9 kg/cm^2a.
3b. The SLC System delivers at least 189 liters/minute of solution with either pump operating against the elevated pressure conditions in the reactor during events involving SLC System initiation.	3b. Using installed controls, power supplies and other auxiliaries, demineralized water will be injected from the storage tank into the RPV with one pump running against a discharge pressure of greater than or equal to 88.9 kg/cm^2a.	3b The SLC System injects greater than or equal to 189 liters/minute into the RPV with either pump running against a discharge pressure greater than or equal to 88.9 $kg.cm^2a$.

Table 2.2.4: Standby Liquid Control System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The SLC System is designed to permit pump flow during plant operation.	4. Using installed controls, power supplies and other auxiliaries, the following system tests will be conducted for each SLCS division after system installation: a. Demineralized water will be pumped against a pressure greater than or equal to 88.9 kg/cm^2a in a closed loop on the test tank. b. Demineralized water will be injected from the test tank into the RPV.	4. a. Demineralized water is pumped with a flow rate greater than or equal to 189 liters/minute in the closed loop. b. Demineralized water is injected from the test tank into the RPV.
5. Class 1E loads of the SLC System are powered from Class 1E Divisions, as described in Section 2.2.4.	5. Tests will be performed on the SLC System by providing a test signal in only one Class 1E Division at a time.	5. The test signal exists only in the Class 1E Division under test in the SLC System.
6. The ASME Code components of the SLCS retain their pressure boundary integrity under internal pressures that will be experienced during service.	6. A hydrostatic test will be conducted on those Code components of the SLCS that are required to be hydrostatically tested by the ASME Code.	6. The results of the hydrostatic test of the ASME Code components of the SLCS conform with the requirements in the ASME Code, Section III.
7. Control room alarms, indications, and controls provided for the SLC System are defined in Section 2.2.4.	7. Inspections will be performed on the control room alarms, indications, and controls for the SLC System.	7. Alarms, indications, and controls exist or can be retrieved in the control room as defined in Section 2.2.4.
8. The SLC pumps have sufficient NPSH.	8. Tests will be conducted by injecting demineralized water using both SLCS pumps from the storage tank to the RPV with conditions in the storage tank of low level (down to pump trip level) and a temperature of greater than or equal to 43°C.	8. The available NPSH exceeds the NPSH required when the SLCS injects greater than or equal to 378 liters/minute.
9. The SLCS automatically initiates both divisions upon receipt of an ATWS signal.	9. Using simulated ATWS signals, test will be performed on the SLC System initiation logic.	9. Upon receipt of a simulated ATWS signal, SLCS logic functions to automatically initiate both divisions.
10. The SLCS can be manually initiated with the system initially in the standby mode for each division.	10. Using the SLCS manual initiation switch, SLCS testing will be performed.	10. The SLCS initiates by the SLCS manual initiation switches with the system initially in the standby mode for each division.

Table 2.2.4: Standby Liquid Control System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. The SLCS pump relief valves open when the inlet pressure to the valve equals or exceeds the setpoint (109.7 kg/cm^{2}g).	11. Shop or field tests will be performed to determine the relief valve setpoint.	11. The SLCS pump relief valves open when the inlet pressure to the valve equals or exceeds 109.7 kg/cm^{2}g.
12. Motor-operated valves (MOV) designated in Section 2.2.4 as having an active safety-related function will open and/or close under differential pressure and fluid flow conditions	12. Opening and/or closing tests of installed valves will be conducted under pre-op differential pressure, fluid flow, and temperature conditions.	12. Each MOV opens and/or closes.

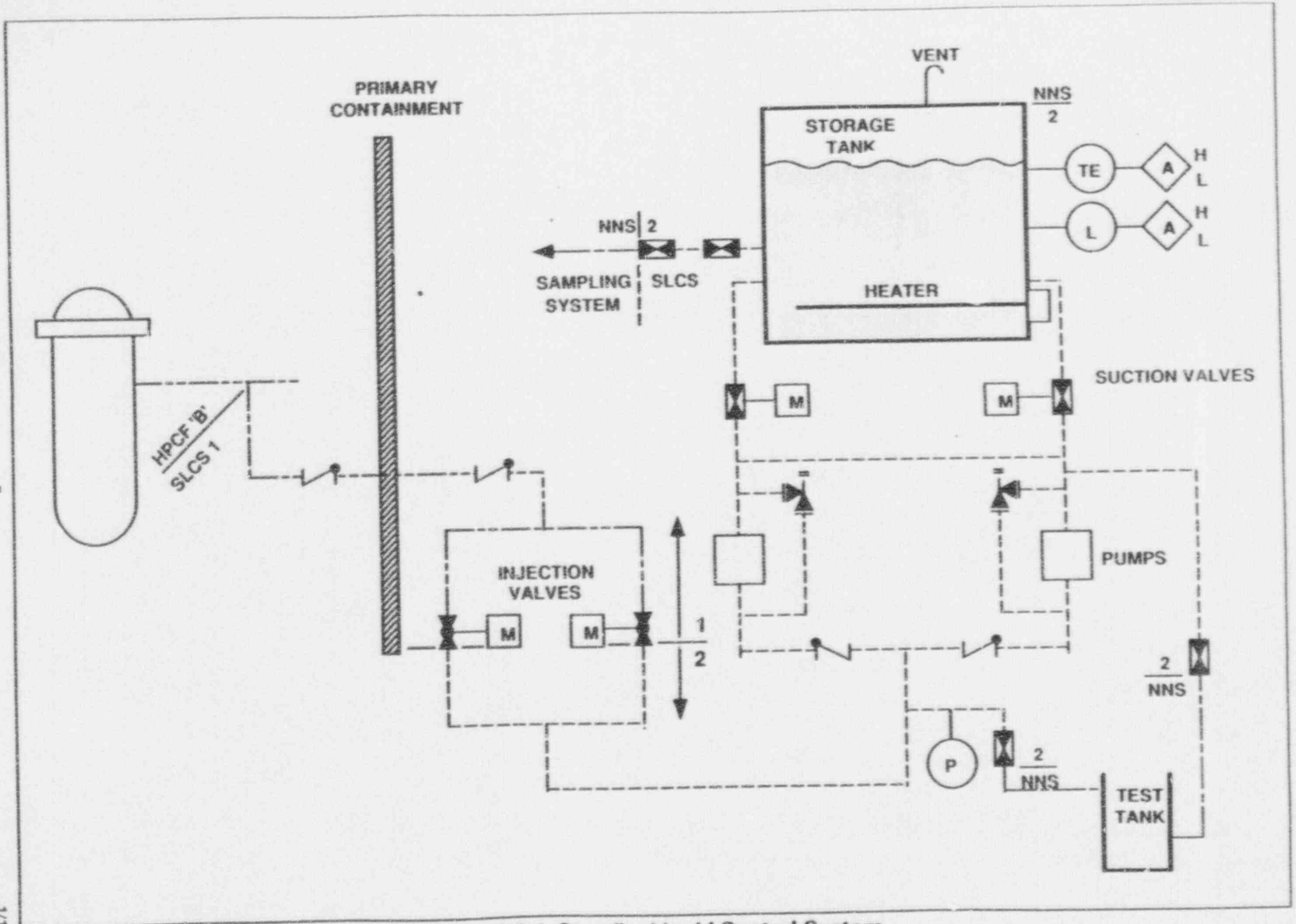


Figure 2.2.4 Standby Liquid Control System

STANDBY LIQUID CONTROL

(A) VERIFY PRE-OP TEST PROCEDURE ~~VERIFIES~~ TESTS ABILITY OF EACH PUMP TO DELIVER CAPACITY WITH OPPOSITE RELIEF VALVE OPEN OR REMOVED.

(B) ADD MOV BOILER PLATE ✓

2.10 Power Cycle

2.10.1 Turbine Main Steam System

Design Description

The 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 and turbine bypass valves. The system includes the steam auxiliary valve(s).

The MS System is designed:

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

(2) to provide a seismically analyzed fission product leakage path to the main condensers.

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

(4) to close the steam auxiliary valve(s) on MSIV isolation signal. These valves fail closed on loss of electrical power to valve actuating solenoid or 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 them to be tested on-line and also supplies steam to the power cycle auxiliaries, as required.

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.

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 sill be undertaken for the MS System.

Table 2.10.1: Main Steam System

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration for the MS System is as shown in Figure 2.10.1.	1. Construction records will be reviewed and visual inspections will be conducted for the configuration of the MS System	1. The as-built configuration of the MS System is in accordance with the description in Figure 2.10.1.
2. MS piping from the seismic interface restraint to the main stop, main turbine bypass and the steam auxiliary valve(s) is analyzed to demonstrates structural integrity under SSE loading conditions.	2. Perform a seismic analysis of the as-built MS piping.	2 The results of the seismic analysis show that the MS piping can withstand a safe shutdown earthquake without loss of structural integrity.
3. Upon receipt of a Main Steam Isolation Valve (MSIV) closure signal, the Steam Auxiliary (SA) valve(s) closes.	3. Tests ~~Testing of the as-built MS system~~ will be performed using simulated MSIV closure signal~~ to verify the SA valves(s) close~~.	3. The SA valve(s) closes following receipt of a simulated MSIV closure signal.
4. The pneumatically operated steam auxiliary valve(s) in the main steam system closes 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 one steam auxiliary valve(s).	4. Steam auxiliary valve(s) closes.
5. The ASME Code components of the MS retain their pressure boundary integrity under internal pressures that will be experienced during service.	5. A hydrostatic test will be conducted on those Code components of the MS System required to be hydrostatically tested by the ASME Code.	5. 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.

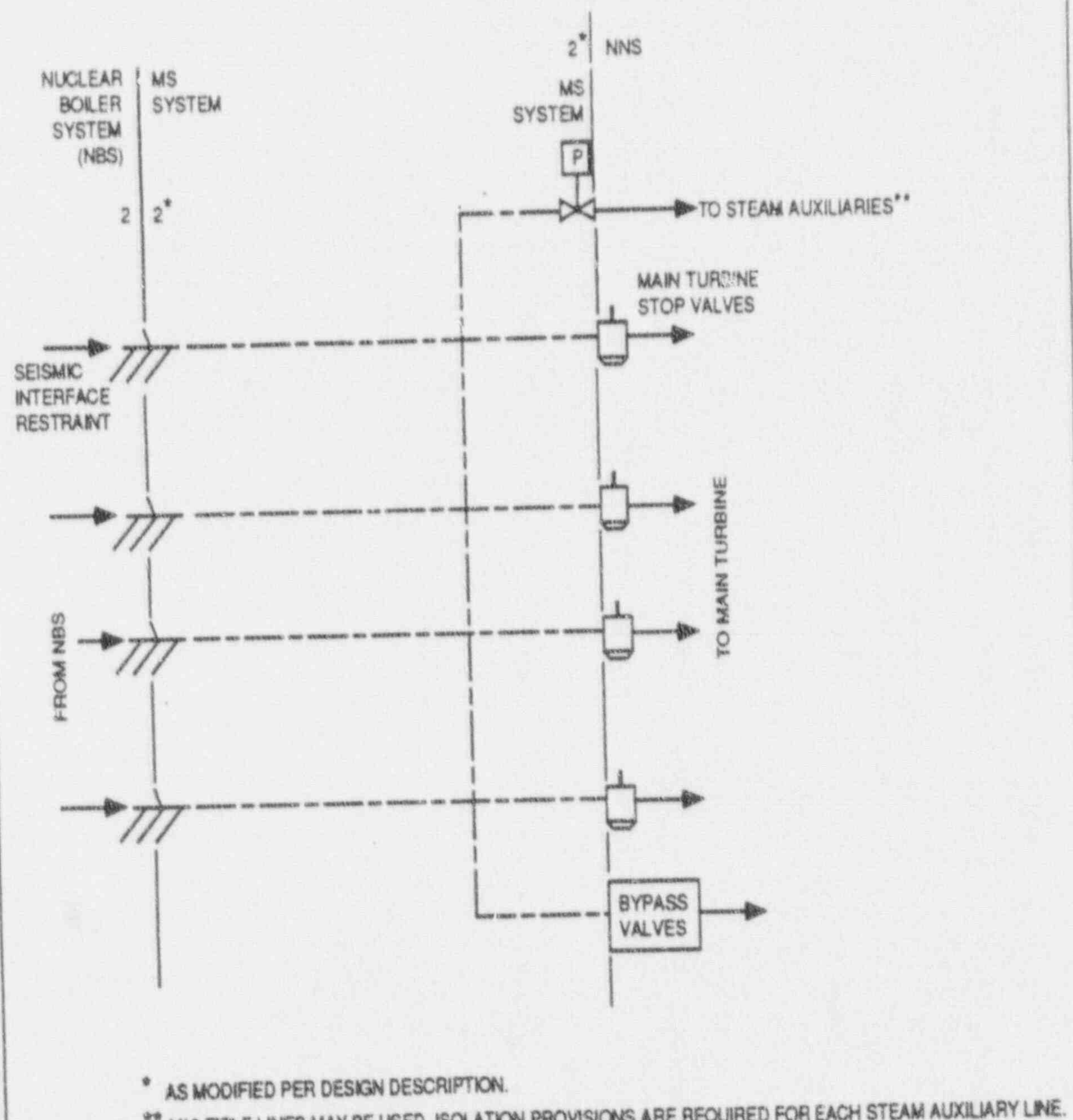


Figure 2.10.1 Turbine Main Steam System

TURBINE MAIN STEAM

A UPDATE SAR TO SHOW STEAM AUXILIARIES VALVE NOW IN SYSTEM ?

~~B ADD BOILER PLATE ON CONFIGURATION~~ ✓

~~C ADD TO DESIGN DESCRIPTION~~ ✓

~~D ADD HYDRO BOILER PLATE~~ ✓

~~E INCLUDE ITAAC AND DESCRIPTION FOR AUX. STEAM VALVE(S) INCLUDING LOGIC TO CLOSE ON MSIV ISOLATION SIGNAL~~ ✓

F. Revised SSAR – Description of Main Steam Line .. (attached draft) V:

these are not available, accepted industry or engineering practice is followed.

3.2.5.3 Main Steam Line Leakage Rate

The ABWR main steam leakage path utilizes the large volume and surface area in the main steam piping, bypass line, and condenser to hold up and plate out the release of fission products following postulated core damage. In this manner, the main steam piping, bypass line, and condenser are used to mitigate the consequences of an accident and are required to remain functional during and after an SSE.

The main steam lines and all branch lines 2-1/2 inches in diameter and larger, up to and including the first valve (including lines and valve supports) are designed by the use of an appropriate dynamic seismic system analysis to withstand the safe shutdown earthquake (SSE) design loads in combination with other appropriate loads, within the limits specified. The mathematical model for the dynamic seismic analyses of the main steam lines and branch line piping includes the turbine stop valves and piping to the turbine casing and the turbine bypass valves and piping to the condenser. The dynamic input loads for design of the main steam lines in the reactor building and the control building are derived from a time history model analysis or an equivalent method as described in Section 3.7.

Figure 3.2-1 depicts the classification requirements for the main steam leakage path as described below.

(1) Main steam piping from the reactor pressure vessel up to and including the outboard isolation valve is classified as QG A (SC 1) and seismic Category I.

(2) Main steam piping beyond the outboard isolation valve up to the seismic interface restraint and connecting branch lines up to the first normally closed valve is classified as QG B (SC 2) and seismic Category I.

(3) The main steamline from the seismic interface restraint up to but not including the turbine stop valve (including branch lines to the first normally closed valve) is classified as QG B and inspected in accordance with applicable portions of American Society of Mechanical Engineers (ASME) Section XI. This portion of the steamline is classified as non-seismic Category I and analyzed using a dynamic seismic analysis method to demonstrate its structural integrity under SSE loading conditions. However, all pertinent QA requirements of Appendix B, 10CFR Part 50 are applicable to ensure that the quality of the piping material is commensurate with its importance to safety during normal operational, transient, and accident conditions.

The seismic interface restraint provides a structural barrier between the seismic Category I portion of the main steamline in the reactor building and the non-seismic Category I portions of the main steamline in the turbine building. The seismic interface restraint is located inside the seismic Category I building. The classification of the main steamline in the turbine building as non-seismic Category I is consistent with the classification of the turbine building.

At the interface between seismic and non-seismic Category I main steam piping system, the seismic Category I dynamic analyses will be extended to either the first anchor point in the non-seismic system or to a sufficient distance in the non-seismic system so as to not degrade the validity of the seismic Category I analysis.

(4) To ensure the integrity of the remainder of main steamline leakage path, the following requirements are met:

(a) the main steam piping between the turbine stop valve and the turbine inlet, the turbine bypass line from the bypass valve to the condenser, and the main steam drain line from the first valve to the condenser are not required to be classified as safety-related nor as seismic Category I, but are analyzed using a dynamic seismic analysis to demonstrate their structural integrity under SSE loading conditions, and

(b) the condenser anchorage is seismically analyzed to demonstrate that it is capable of sustaining the SSE loading conditions without failure.

A plant-specific walkdown of non-seismically designed systems, structures, and components overhead, adjacent to, and attached to the main steam leakage path (i.e., the main steam piping, the bypass line, and the main condenser) shall be conducted to confirm by inspection that the as-built main steam piping, bypass lines to the condenser, and the main condenser are not compromised by non-seismically designed systems, structures and components.

3.2.6 Quality Assurance

Structures, systems, and components that perform nuclear safety-related functions conform to the quality assurance requirement of 10CFR50 Appendix B as shown in Table 3.2-1 under the heading, "Quality Assurance Requirements," and in Table 3.2-2. Some NNS structures, systems, and components meet the same requirements as noted on Table 3.2-1. The Quality Assurance Program is described in Chapter 17.

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
B2 Nuclear Boiler System (Continued)						
4. Piping including supports main steamline (MSL) and feedwater (FW) line up to and including the outermost isolation valve	1	C,SC	A	B	I	
5. Piping including supports-						
(a) MSL (including branch lines to first valve) from outermost isolation valve up to and including seismic interface restraint	2	SC	B	B	I	(r)
(b) FW (including branch lines to first valve) from outermost isolation valve to and including the shutoff valve	2	SC	B	B	I	(ee)
6. Piping including supports-MSL (including branch lines to first valve) from the seismic interface restraint up to but not including the turbine stop valve and turbine bypass valve	N	SC,T	B	E	---	(r)
7. Piping beyond FW shutoff valve	N	SC	D	---	---	(ee)
8. Deleted						
9. Deleted						
10. Pipe whip restraint - MSL/FW if needed	3	SC,C	---	B	---	(dd)
11. Piping including supports-other within outermost isolation valves						
a. RPV head vent	1	C	A	B	I	(g)
b. Main steam drains	1	C,SC	A	B	I	(g)
12. Piping including supports-other beyond outermost isolation or shutoff valves						

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
B2 Nuclear Boiler System (Continued)						
a. RPV head vent beyond shutoff valves	N	C	C	E	---	
b. Main steam drains to first valve	N	SC,T	B	B	I	(r)
c. Main steam drains beyond first valve	N	SC,T	D	---	---	(r)

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
B2 Nuclear Boiler System (Continued)						
13. Piping including supports-instrumentation up to and beyond outermost isolation valves	2/N	C,SC	B/D	B/E	I/---	(g)
14. Safety/relief valves	1	C	A	B	I	
15. Valves - MSL and FW isolation valves, and other FW valves within containment	1	C,M	A	B	I	
16. Valves - FW, other beyond outermost isolation valves up to and including shutoff valves	2	SC	B	B	I	(ee)
17. Valves - within outermost isolation valves						
a. RPV head vent1	C	A	B	I	(g)	
b. Main steam drains	1	C,SC	A	B	I	(g)
18. Valves, other						
a. RPV head vent	3	C	C	B	I	
b. 1st Main steam drain valve	N	SC	B	B	I	(r)
c. other main steam drain valves	N	SC	D	---	---	(r)
19. Deleted						
20. Mechanical modules-instrumentation with safety-related function	3	C,SC	---	B	I	
21. Electrical modules with safety-related function	3	C,SC,X	---	B	I	(i)
22. Cable with safety-related function	3	C,SC,X	---	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
K1 Radwaste System						
1. Drain piping including supports and valves - radioactive	N	ALL (except RZ,X)	D	E	--	(p)
2. Drain piping including supports and valves - nonradioactive	N	ALL	D	E	---	(p)
3. Piping and valves - containment isolation	2	C,SC	B	B	I	
4. Piping including supports and valves forming part of containment boundary	N	C,SC	B	B	I	
5. Pressure vessels including supports	N	W	---	E	---	(p)
6. Atmospheric tanks including supports	N	C,SC,H, T,W	---	E	---	(p)
7. 0-15 PSIG Tanks and supports	N	W	---	E	---	(p)
8. Heat exchangers and supports	N	C,SC,W	---	E	---	(p)
9. Piping including supports and valves	N	C,SC,H, T,W	---	E	---	(p)
10. Other mechanical and electrical modules	N	ALL	--- E	---	(p)	
11. ECCS equipment room sump backflow protection check valves	3	SC	C	B	I	
N1 Turbine Main Steam System						
1. Deleted (See B2.5)						

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
N1	Turbine Main Steam System (Continued)						
	2. Deleted (See B2.6)						
N2	Condensate, Feedwater and Condensate Air Extraction System						
	1. Main feedwater line (MFL) including supports from second isolation valve branch lines and components and including to outboard shutoff valves	N	SC	B	B	I	
	2. Feedwater system components beyond outboard shutoff valve	N	T	D	E	—	
N3	Heater, Drain and Vent System	N	T	—	E	—	
N4	Condensate Purification System	N	T	—	E	—	
N5	Condensate Filter Facility	N	T	—	E	—	
N6	Condensate Demineralizer	N	T	—	E	—	
N7	Main Turbine	N	T	—	E	—	
N8	Turbine Control System						
	1. Turbine stop valve, turbine bypass valves, and the main steam leads from the turbine stop valve to the turbine casing	N	T	D	—	—	(l)(n)(o)(r)

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
N9 Turbine Gland Steam System	N	T	D	E	---	
N10 Turbine Lubricating Oil System	N	T	---	E	---	
N11 Moisture Separator Heater	N	T	---	E	---	
N12 Extraction System	N	T	---	E	---	
N13 Turbine Bypass System						
1. Turbine bypass piping including supports up to the condenser	N	T	D	---	---	(r)
N14 Reactor Feedwater Pump Driver	N	T	---	E	---	
N15 Turbine Auxiliary Steam System	N	T	---	E	---	
N16 Generator	N	T	---	E	---	
N17 Hydrogen Gas Cooling System	N	T	---	E	---	
N18 Generator Cooling System	N	T	---	E	---	
N19 Generator Sealing Oil System	N	T	---	E	---	
N20 Exciter	N	T	---	E	---	
N21 Main Condenser	N	T	---	E	---	
N22 Offgas System	N	T	---	E	---	
N23 Circulating Water System	N	T	D	E	---	
N24 Condenser Cleanup Facility	N	T	---	E	---	

NOTES (Continued)

n. All cast pressure-retaining parts of a size and configuration for which volumetric methods are effective are examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards is used as an alternate to radiographic methods. Examination procedures and acceptance standards are at least equivalent to those defined in Paragraph 136.4, Nonboiler External Piping, ANSI B31.1.

o. The following qualifications are met with respect to the certification requirements:

1. The manufacturer of the turbine stop valves, turbine control valves, turbine bypass valves, and main steam leads from turbine control valve to turbine casing utilizes quality control procedures equivalent to those defined in GE Publication GEZ-4982A, General Electric Large Steam Turbine Generator Quality Control Program.

2. A certification obtained from the manufacturer of these valves and steam loads demonstrates that the quality control program as defined has been accomplished.

The following requirements shall be met in addition to the Quality Group D requirements:

1. All longitudinal and circumferential butt weld joints shall be radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrate examination may be substituted. Examination procedures and acceptance standards shall be at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4 in ANSI B31.1.

2. All fillet and socket welds shall be examined by either magnetic particle or liquid penetrate methods. All structural attachment welds to pressure retaining materials shall be examined by either magnetic particle or liquid penetrate methods. Examination procedures and acceptance standards shall be at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4 in ANSI B31.1

3. All inspection records shall be maintained for the life of the plant. These records shall include data pertaining to qualification of inspection personnel, examination procedures, and examination results.

p. A quality assurance program meeting the guidance of Regulatory Guide 1.143 will be applied during design and construction.

q. Detailed seismic design criteria for the offgas system are provided in Subsection 11.3.4.8.

r. See Section 3.2.5.3.

NOTES (Continued)

s. The recirculation motor cooling system (RMCS) is classified Quality Group B and Safety Class 2 which is consistent with the requirements of 10CFR50.55a. The RMCS, which is part of the reactor coolant pressure boundary (RCPB) meets 10CFR50.55a (c)(2). Postulated failure of the RMCS piping cannot cause a loss of reactor coolant in excess of normal makeup (CRD return or RCIC flow), and the RMCS is not an engineered safety feature. Thus, in the event of a postulated failure of the RMCS piping during normal operation, the reactor can be shutdown and cooled down in an orderly manner, and reactor coolant makeup can be provided by a normal make up system (e.g., CRD return or RCIC system). Thus, per 10CFR50.55a(c)(2), the RMCS need not be classified Quality Group A or Safety Class 3, however, the system is designed and constructed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Class 1 criteria as specified in Subsection 3.9.3.1.4 and Figure 5.4-4.

t. A quality assurance program for the Fire Protection System meeting the guidance of Branch Technical Position CMEB 9.5-1 (NUREG-0800), is applied.

u. Special seismic qualification and quality assurance requirements are applied.

v. See Reg Guide 1.143, paragraph C.5 for the offgas vault seismic requirements.

w. The condensate storage tank will be designed, fabricated, and tested to meet the intent of API Standard API 650. In addition, the specification for this tank will require: (1) 100% surface examination of the side wall to bottom joint and (2) 100% volumetric examination of the side wall weld joints.

x. The cranes are designed to hold up their loads and to maintain their positions over the units under conditions of SSE.

y. All off-engine components are constructed to the extent possible to the ASME Code, Section III, Class 3.

z. Components associated with safety-related function (e.g., isolation) are safety-related.

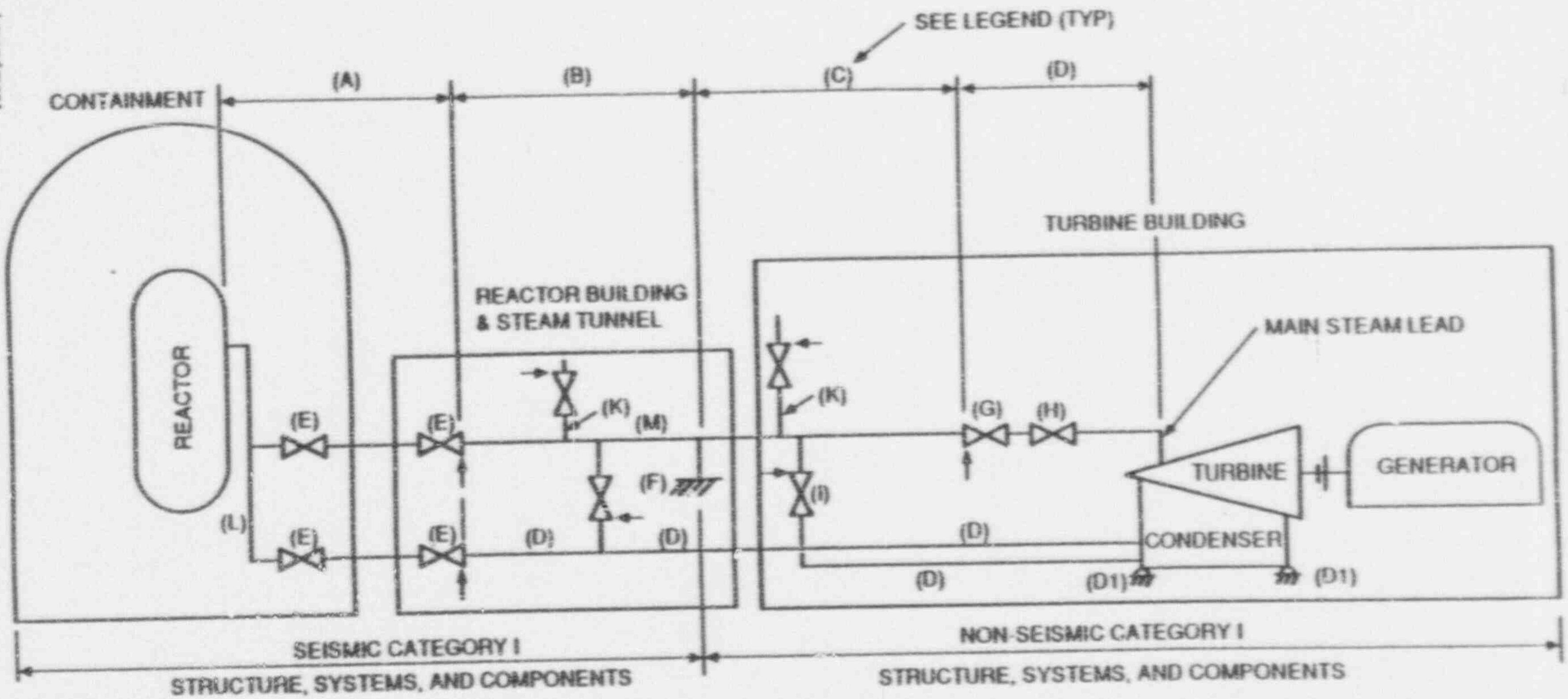
aa. Structures which support or house safety-related mechanical or electrical components are safety-related.

bb. All quality assurance requirements shall be applied to ensure that the design, construction and testing requirements are met.

cc. A quality assurance program, which meets or exceeds the guidance of Generic Letter 85-06, is applied to all non-safety related ATWS equipment.

dd. The need for pipe whip restraints on the MSL/FW piping will be determined by a "leak-before-break" evaluation.

ee. Figure 3.2-2 depicts the classification requirements for the Feedwater System. At the interface between seismic and non-seismic Category I feedwater piping system, the seismic Category I dynamic analyses will be extended to either the first anchor point in the non-seismic system or to sufficient distance in the non-seismic system so as to not degrade the validity of the seismic Category I analysis.



LEGEND:	
A. QUALITY GROUP A	G. TURBINE STOP VALVE
B. QUALITY GROUP B	H. TURBINE CONTROL VALVE
C. QUALITY GROUP B, NON-SEISMIC CATEGORY I, DYNAMICALLY ANALYZED	I. TURBINE BYPASS VALVE
D. QUALITY GROUP D, NON-SEISMIC CATEGORY I, DYNAMICALLY ANALYZED	J. MAIN STEAM LEAD
D1. NON-SEISMIC CATEGORY I, DYNAMICALLY ANALYZED	K. BRANCH LINE
E. ISOLATION VALVE	L. DRAIN LINE
F. SEISMIC INTERFACE RESTRAINT	M. STEAM LINE

Figure 3.2-1 QUALITY GROUP AND SEISMIC CATEGORY CLASSIFICATION APPLICABLE TO POWER CONVERSION SYSTEM

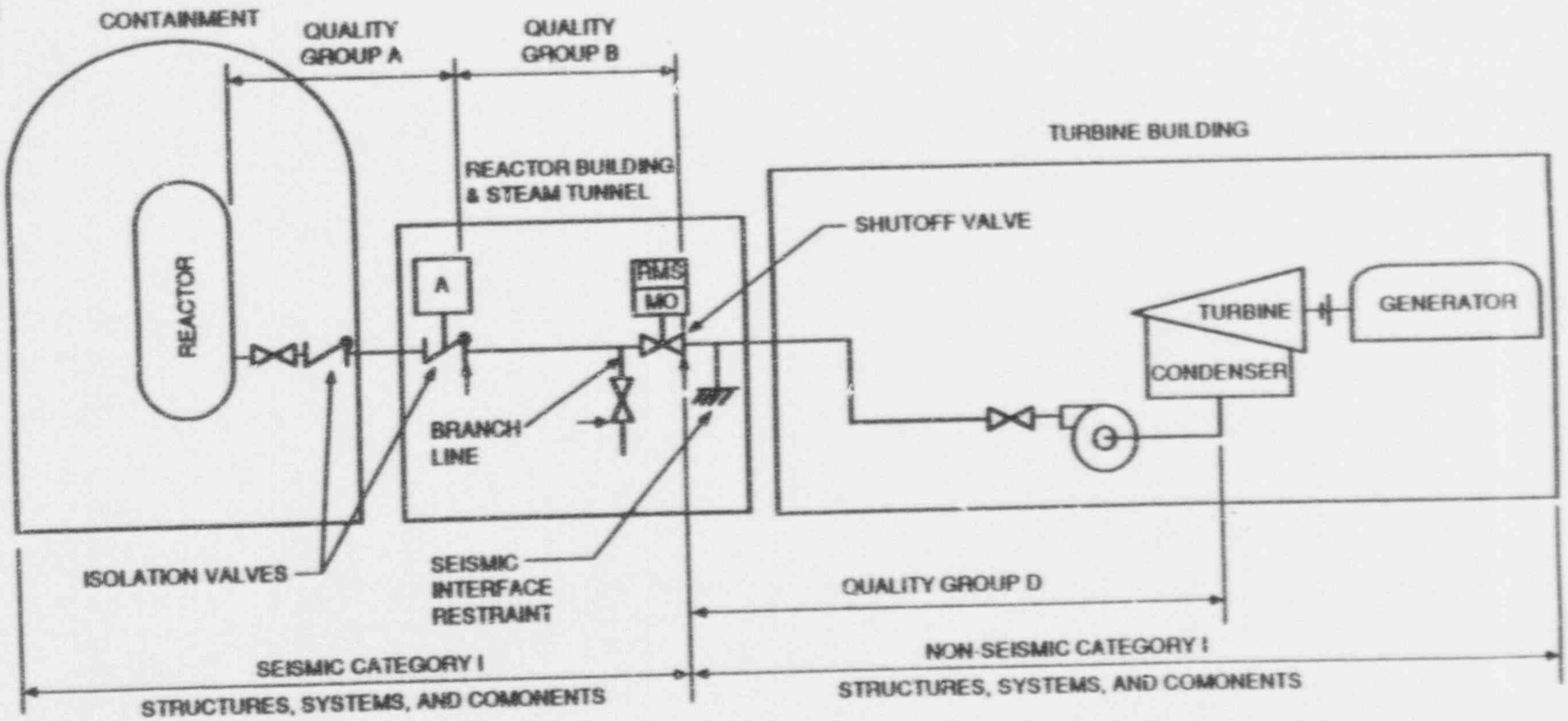


Figure 3.2-2 QUALITY GROUP AND SEISMIC CATEGORY CLASSIFICATION APPLICABLE TO FEEDWATER SYSTEM

2.10.21 Main Condenser

Design Description

The main condenser is classified as non-safety-related and non-seismic Category I. It is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the Turbine Bypass (TB) System. The main condenser is also a collection point for other steam cycle drains/vents.
AND

The main condenser hotwell provides a holdup volume for MSIV fission product leakage. The supports and anchors for the main condenser are designed to withstand a safe shutdown earthquake (SSE).

The main condenser tubes are made from corrosion resistant material.

The main condenser is located in the Turbine Building.

Since the main condenser operates at a vacuum, leakage is into the shell side of the main condenser. Tube side or circulating water inleakage is detected by measuring the conductivity of sample water extracted beneath the tube bundles. In addition, conductivity is monitored at the discharge of the condensate pumps and alarms provided in the main control room.

A signal is provided to the Leak Detection and Isolation System (LDS) System on loss of vacuum.

Condenser pressure indicators are located above the design basis flood level.

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.

A. [illegible]

WHY WOULD THIS BE IN ITAAC?

Table 2.10.21: Main Condenser

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Main condenser supports and anchors are designed to withstand the safe shutdown earthquake.	1. Perform analysis of the ability or the as-built condenser supports and anchors to withstand the safe shutdown earthquake.	1. The results of the analyses show that the as-built main condenser supports and anchors are able to withstand the safe shutdown earthquake.

2.) CR FEATURES

PRESS. SENSORS ABOVE FLOOD LEVEL SHOULD BE VERIFIED
IN TO ITAAC

2.15.5 Heating, Ventilating and Air Conditioning Systems

Design description, basic configuration figure and table for the Inspection, Test, Analyses and Acceptance Criteria are provided for the Control Room Habitability Area, Ventilating and Air Conditioning (HVAC) System.

Design Description

Control Room Habitability Area HVAC System

The Control Room Habitability Area (CRHA) HVAC System controls the thermal, radiological and pressurel environments. The system consists of two Safety-related Divisions classified as Seismic Category I which are physically separated and electrically independent.

The basic configuration for the Control/Room Habitability Area HVAC System is shown in Figure 2.15.5a.

The HVAC equipment, plenums, ducts and dampers outside the habitability area have leak-tight design features enabling the CRHA HVAC System to maintain at least 3.2 mm water gauge positive differential pressure between the habitability area and adjacent Control Building areas.

The temperature is controlled within a range of 10°C to 29°C, and a relative humidity (RH) within a range of 10% to 60%.

The CRHA HVAC System's Control Room indication and controls allow for monitoring and control during operational conditions. The Control Room has indication for and control of temperature and humidity. A control room pressure signal positions an automatic damper in the exhaust fan discharge duct to maintain a positive pressure in the Control Room. Manual control of each motorized damper, fan and emergency filtration unit is accomplished with remote manual switches and indicating lights in the Control Room. A flow device in the emergency filtration unit discharge duct automatically starts the redundant Control Room Habitability Area HVAC System on loss of air flow.

When the radiation monitor in the operating outdoor air intake detects airborne contamination, an isolation signal is generated to close the normal outdoor air intake damper, open the emergency outdoor air intake damper, stop the normal exhaust fans, start the emergency air filtration unit to decrease the contamination before air is supplied to the Control Room Habitability Area and maintain a positive pressure relative to adjacent spaces of the Control Building. The redundant train is connected to another outdoor air intake separated from the or greater than or equal to 50 m. Each emergency air filtration unit treats both indoor recirculated air and outdoor air to maintain a positive pressure with not more than 1300 m^3 per hour of filtered outdoor air. Leakage shall be less

than that required to meet the personnel dose limits of Section 3.7, Radiation Protection.

The Safety-related electrical equipment including instrumen. d controls located in the Control Building outside the Habitability Area is qualified for a harsh environment.

When the Products of Combustion (POC) monitor in the outdoor air intake detects smoke, a signal will initiate the recirculation mode by isolating dampers, stopping the exhaust fans and closing the exhaust dampers.

Table 2.15.5a: Control Room Habitability Area HVAC System
Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. A basic configuration for the Control Room Habitability Area HVAC System is as shown in Figure 2.15.5a.	1. Inspection of the system will be conducted.	1. The as-built Control Room Habitability Area HVAC System conforms with the basic configuration shown in Figure 2.15.5a.
2. The Control Room Habitability Area is maintained at a positive pressure greater than or equal to 3.2 mm water gauge relative to the atmosphere and other areas of the Control Building.	2. The Control Room Habitability Area HVAC System will be tested in all modes of operation.	2. The Control Room Habitability Area is maintained at a positive pressure of at least 3.2 mm water gauge relative to atmosphere and other areas of the Control Building.
3. Control Room indicators and controls provided for the Control Room Habitability Area HVAC SystemS are as defined in Section 2.15.5.	3. Inspections will be performed on the Control Room indicators and controls for the Control Room Habitability Area HVAC System.	3. Indicators and controls exist or can be retrieved in the Control Room, as defined in Section 2.15.5.
4. Class 1E loads for the Control Room Habitability Area HVAC System are powered from Class 1E Divisions, as described in Section 2.15.5	4. Tests will be performed on the Control Room Habitability Area HVAC System by providing a test signal in only on class 1E division at a time.	4. The test signal exists only in the safety-related electrical power loads for Class 1E Division under test in the Control Room Habitability Area HVAC System.
5. Each mechanical Division of the Control Room Habitability Area HVAC System is physically separated.	5. Inspection of the as-built system will be performed.	5. Each mechanical Division of the Control Room Habitability Area HVAC System is physically separated from the other mechanical Divisions of the Control Room Habitability Area HVAC System.
6. On detection of smoke at the outdoor intake, the outdoor air intake dampers shall close, recirculation dampers shall open, and the exhaust fans and exhaust dampers isolate.	6. Tests with simulated outdoor smoke signal will be performed with the Control Room Habitability Area HVAC System in the recirculation mode.	6. The Control Room Habitability Area System HVAC is in the recirculation mode when smoke is present outside the outdoor air intakes of the Control Room Habitability Area HVAC System.
7. Two outdoor air intakes of the Control Room Habitability Area HAVC System are at least 50 m apart.	7. Inspect the distance between the outdoor air intakes of the Control Room Habitability Area System.	7. The control Room Habitability Area HVAC System outdoor air intakes are 50 m apart.

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. A low flow condition during operation of the Emergency Filter Unit in the emergency mode will cause the redundant Control Room Habitability Area HVAC System to start in the emergency mode.	8. Test the automatic transfer from the Control Room Habitability Area HVAC Division operating in the emergency mode with a simulated low flow signal to the standby Control Room Habitability Area HVAC Division in the emergency mode.	8. The standby Control Room Habitability Area HVAC Division starts in the emergency mode when a low flow condition exists in the operating Control Room Habitability Area HVAC Division in the emergency mode.

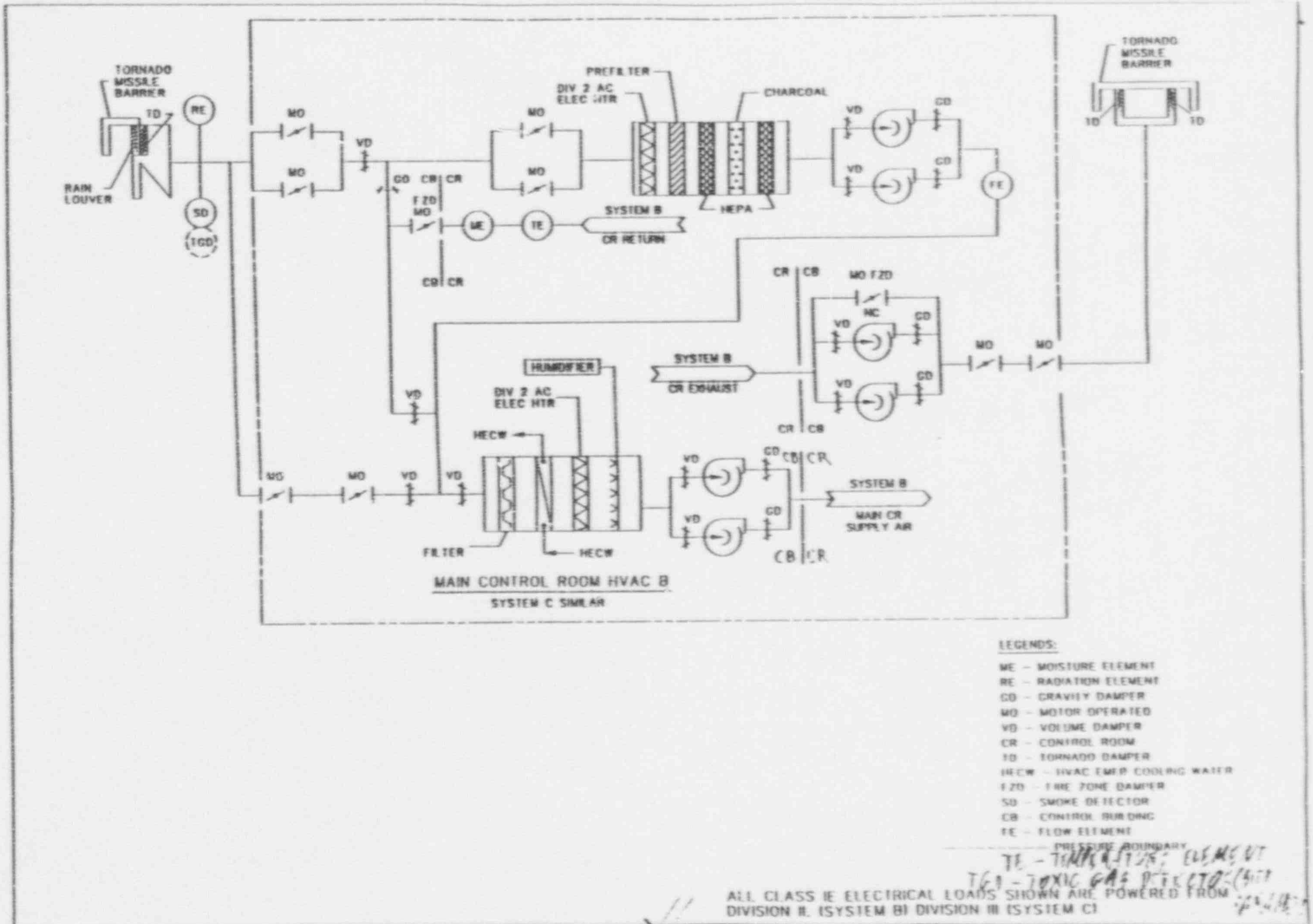


Figure 2.15.5a Control Room ~~and~~ Habitability Area HVAC System

(A) VERIFY 1300^3/HR IS CORRECT DESIGN BASIS NUMBER. ALSO, CHECK USE OF SECTION 3.7 RADIATION PROTECTION

(B) ~~-INCLUDE BOUNDS OF CONTROL ROOM~~
~~-PRESSURE BOUNDS~~
~~-USE STD BOILER PLATE FOR POWER SUPPLIES~~ STET

(C) MODIFY FOR CONSISTENCY BASED ON DEFINITION ACCEPTANCE

(D) TOXIC GASSES OUT OF SCOPE FOR ABWR. MUST BE ADDRESSED IN SITE PARAMETERS ?

~~(E) RECIRC MODE NEEDED FOR SMOKE IN TEXT AND ITAAC~~

~~(F) ITAAC FOR 50 M SEPERATION OF INTAKES~~

(G) ITAAC FOR AUTO START OF OTHER DIVISION ON EMERGENCY FILTRATION LOW FLOW (UPDATE SSAR)

Enclosure 4

<u>Generic Concern Summaries</u>

1. Welding

2. Environmental Qualifications

   Seismic Qualification

3. Verification of MOV capabilities

4. Electromagnetic Interference (EMI/RFI)

   Instrument Setpoints

5. Electrical Independence (Separation)

6. Piping

1. Summary of Welding

The staff and GE reached agreement on the approach for verifying welding. The verification will be performed as a part of an ITAAC basic configuration check in each system. ASME Code Class 1, pressure-boundary welds will be inspected to ensure the physical quality of the welds. The supporting SSAR material (attached in draft form) will specify the codes/standards and the acceptance criteria for evaluating the quality of welds for ASME Boiler and Pressure Vessel Code components and supports, non-ASME pressure retaining piping, structural and building steel, electrical cable tray and conduit supports, HVAC supports, and refueling cavity and spent fuel pool liners. GE will decide later in which sections of the SSAR these welding design criteria and acceptance criteria will appear.

METHODS

Welding and ~~Weld~~ Acceptance Criteria

~~The requirements listed below are considered by the staff to be essential in controlling welding activities. Any change will require the COL applicant to submit the changes to the NRC staff for review and approval prior to use.~~

OR SECTION VIII AS APPLICABLE,

ASME Code Welding

FOR PRESSURE BOUNDARY AND CORE SUPPORT STRUCTURES

Welding activities shall be performed in accordance with the requirements of Section III of the ASME Code. The required nondestructive examination and acceptance criteria is stated in Table 1. Component supports shall be fabricated in accordance with the requirements of Subsection NF of Section III of the ASME Code. ~~except that the visual weld acceptance criteria shall be the Nuclear Construction Issue Group (NCIG) standard NCIG-01, "Visual Weld Acceptance Criteria for Structural Welding of Nuclear Power Plants," Revision 2.~~

AND EXAMINED

Welding of non-ASME pressure retaining Piping

Welding activities involving non-ASME pressure retaining piping shall be accomplished in accordance with written procedures and shall meet the requirements of the ANSI B31.1, Code. The weld acceptance criteria shall be as defined for the applicable nondestructive examination method described in ANSI B31.1 Code

Welding of Structural and Building Steel

Welding activities shall be accomplished in accordance with written procedures and shall meet the requirements of the American Institute of Steel Construction (AISC) Manual of Steel Construction. The visual acceptance criteria shall be as defined in NCIG-01, Revision 2.

INSERT

Welding of Electrical Cable Tray and Conduit Supports

Welding activities shall be accomplished in accordance with the ~~American Welding Society (AWS)~~ Structural Welding Code, D1.1 The weld visual acceptance criteria shall be as defined in NCIG-01, Revision 2.

AWS STRUCTHRAL WELDING CODE D1.1 AND

Welding of Heating Ventilating and Air Conditioning Supports

Welding activities shall be accomplished in accordance with the ~~American Welding Society (AWS)~~ Structural Welding Code, D1.1 The weld visual acceptance criteria shall be as defined in NCIG-01, Revision 2.

AWS STRUCTURAL WELDING CODE D1.1 AND

Welding of Refuel Cavity and Spent Fuel Pool Liners

WHERE ACCESSIBLE

Welding activities shall be accomplished in accordance with the ~~American Welding Society (AWS)~~ Structural Welding Code, D1.1 The welded seams of the liner plates shall be spot radiographed, liquid penetrant and vacuum box examined after fabrication to ensure that the liner do not leak. The acceptance criteria for these examination shall meet the acceptance criteria stated in subsection NE-5200 of Section III of the ASME Code.

INSERT - AMERICAN WELDING SOCIETY (AWS) STRUCTURAL WELDING CODE D1.1 AND NUCLEAR CONSTRUCTION ISSUE GROUP (NCIG) STANDARD ~~NCIG-01~~, "VISUAL WELD ACCEPTANCE CRITERIA FOR STRUCTURAL WELDING AT NUCLEAR POWER PLANTS,

TABLE 1
Welding Activities and Weld Examination Requirements for
ASME Code, Section III Welds

Class 1 Components (1)(2)(3)

Component	Weld Type	NDE Requirements
Vessel	Category A (Longitudinal)	RT plus MT or PT
Vessel, Pipe, Pump, Valve	Category B (Circumferential)	RT plus MT or PT
Pipe, Pump, Valve	Butt weld Fillet and socket welds	RT plus MT or PT MT or PT
Vessels (6)	Category C and similar welds Partial penetration and fillet welds	RT plus MT or PT. RT must be multiple exposure MT or PT on all accessible surfaces
Vessels (6) & Branched Connections	Category D a) Butt welds, all b) Corner welded nozzles c) Corner welded branch and piping connection exceeding 4" nominal diameter d) Corner welds branch and piping 4" and less e) Weld buildup deposits at openings f) Partial penetration g) Oblique full penetration branch and piping connections	 RT plus MT or PT RT plus MT or PT RT plus MT or PT MT or PT UT plus a, b, c above if connected to nozzle or pipe MT or PT progressive and final surface RT or UT plus MT or PT. In addition, UT of weld, fusion zone, and parent metal beneath attachment surface.
General	Fillet, partial penetration, socket welds	MT or PT
General	Structural attachment welds	MT or PT
Special welds	1) Specially designed seals 2) Weld metal cladding 3) Hard surfacing a) Valves 4" or less 4) Tube-tube sheet welds 5) Brazed joints	MT or PT PT PT None PT VT

Class 2 Components (1)(2)(4)

Component	Weld Type	NDE Requirements
Vessel	Category A (Longitudinal) a) Either of the members exceeds 3/16 inch b) Each member 3/16 inch or less	RT MT, PT, or RT
Pipe, Pump, Valve	Longitudinal	RT
Vessel	Category B (Circumferential) a) Either of the members exceeds 3/16 in. b) Each member 3/16" or less	RT MT, PT, or RT
Pipe, Pump and Valve	Circumferential a) Butt welds b) Fillet and partial penetration	RT MT or PT
Vessel (6) and similar joints in other components	Category C a) Corner joints, either of the members exceeds 3/16" of thickness b) Each member 3/16" or less c) Partial penetration and fillet welds	RT MT, PT, or RT MT or PT
Vessel (6) and similar welds in other components	Category D a) Full penetration joints when either members exceed 3/16" of thickness b) Full penetration corner joints when either member exceeds 3/16" c) Both members 3/16" or less d) Partial penetration and fillet weld joints	RT MT or PT MT or PT MT or PT
Branch Con. and Nozzles in pipe, valve, pump	a) Nominal size exceed 4" b) Nominal size 4" or smaller	RT MT or PT (external and accessible internal surfaces)

Class 2 Components (Cont'd)(1)(2)(4)

Component	Weld Type	NDE Requirements
Vessels designed to NC-3200	· Cat. A	RT
	· Cat. B	RT
	· Cat. C, Butt weld	RT
	· Cat. C, Full penetration corner	UT or RT
	· Cat. C, Partial penetration corner and fillet welds	MT or PT both sides
	· Cat. D, Full penetration (6)	RT
	· Cat. D, Partial penetration	MT or PT both sides
	Fillet, Partial Penetration, socket, and structural attachment welds	MT or PT
Special Welds	a) specially designed seals	MT or PT
	b) weld metal cladding	MT or PT
	c) hard surfacing	PT
	d) hard surfacing for valves with inlet connection 4" nominal pipe size or less	None
	e) tube-tube sheet welds	PT
	f) Brazed joints	VT
Storage Tanks (Atmospheric)	a) side joints	RT
	b) roof and roof-to-sidewall	VT
	c) bottom joints	vacuum box testing of at least 3 psi
	d) bottom to sidewall	vacuum box + MT or PT
	e) Nozzle to tank side	MT or PT
	f) Nozzle to roof	VT
	g) Joints in nozzles	RT
	h) others	Similar welds in vessels
Storage Tanks (0-15 psi)	a) sidewall	RT
	b) roof	RT
	c) roof-to-sidewall	RT if not possible, MT or PT
	d) bottom & bottom-to-side	vacuum box method + MT or PT
	e) nozzle tank	MT or PT
	f) joints to nozzles	RT
	g) others	same as similar vessel joints

Class 3 Components (1)(2)(5)

Component	Weld Type	NDE Requirements
Vessels	Category A (Longitudinal) 1. a) Thickness exceeding the limits of Table ND. 5211.2-1	RT
	b) Welds based on joint efficiency permitted by ND.3351.1	RT
	c) butt welds in nozzles attached to vessels in a or b above	RT
	2. Welds not included in 1 above	Spot RT each 50 ft of weld. additional RT to cover each welders work.
	3. Nonferrous vessels exceeding 3/8 inch	RT
Pump, Valve, Pipe	· pipes greater than 2 in. size · pumps & valves greater than 2 in.	RT, MT, or PT according to the product form
Vessel	Category B (Circumferential) 1. a) Thickness exceeds Table ND.5211.2 for Ferrous metals	RT
	b) thickness exceeds 3/8 in for nonferrous metals	RT
	c) joint efficiency according to ND.3352.1(a)	RT
	d) attachments to vessels and exceeds nominal pipe size 10" or thickness 1 1/8 in.	RT
	2. welds not involved in 1 above	PT 6 in. long sections - the intersections of Cat. A welds
pipe, pump and valve	Greater than 2" nominal pipe size	RT, PT, or MT
Vessel	Category C: 1. a) Thickness exceeds Table ND-5211.2 or ND-5211.3	RT
	b) Attachments exceed 10 inch NPS or 1 1/8 inch wall thickness	RT
	2. Welds not involved in 1 or 2 above	Spot RT to cover each welders work
Pipe, Pump, Valves	Greater than 2" nominal pipe size	RT, PT, or MT

Class 3 Components (Cont'd)(1)(2)(5)

Vessel	Category D: 1. Full penetration butt welds designed for joint efficiency per ND.3352.1(a)	RT
	2. In nozzles or communicating chambers attached to vessels or heads requiring full RT	RT
	3. Welds not covered by 1 and 2 above	Spot RT to cover each welders work
Pipe, Pump and Valve	Greater than 2" nominal pipe size	RT, PT, or MT
Special Welds	a) weld metal cladding	PT
	b) hard surfacing	PT
	(i) hard surfacing for valves with inlet connection 4" nominal pipe size or less	none
	c) tube-tube sheet welds	PT
	d) Brazed joints	VT
Storage Tanks (Atmospheric)	a) sidewall joints	Same as Category A or B vessel joints
	b) roof and roof-to-sidewall	VT
	c) bottom joints	vacuum box testing of at least 3 psi, or PT or MT plus VT during pressure test
	d) bottom to sidewall	Same as bottom joints
	e) Nozzle to tank side	MT or PT
	f) Nozzle to roof	VT
	g) Joints in nozzles ex. roof nozzles	MT or PT
	h) others	Similar welds in vessels
Storage Tanks (0-15 psi)	a) sidewall	Same as Category A or B vessel joints
	b) roof	Same as Category A vessel joints
	c) roof-to-sidewall	Same as above if possible. or MT or PT
	d) bottom & bottom-to-side	Vacuum box testing at least 3 psi, or PT ~~or~~ OR MT plus VT during pressure test
	e) nozzle to tank	MT or PT
	f) joints in nozzles	MT or PT
	g) others	same as similar vessel joints

Containment Vessel (1)(2)(6)

Component	Weld Type	NDE Requirements
Containment	Category A, Butt Welds (Long'l)	RT
Containment	Category B, Butt Welds (Circ.)	RT
Containment	Category C, Butt weld	RT
Containment	Category C, Nonbutt Welds	UT or MT or PT
Containment	Category D, Butt Welds	RT
Containment	Category D, Nonbutt Welds	UT or MT or PT
Containment	Structural attachment welds a) Butt Welds b) Nonbutt Welds	 RT UT or MT or PT
Special welds	Weld Metal Cladding	PT

Components Supports (1)(2)(7)

Component	Weld Type	NDE Requirements
Class 1 Supports	Primary Member, Full Penetration Butt Welds All other welds Secondary Member Welds	RT MT or PT VT
Class 2 and MC Supports	Primary Member, Full Penetration Butt Welds Partial Penetration or fillet welds throat greater than 1" All other Welds Secondary Member Welds	RT MT or PT VT VT
Class 3 Supports	Primary Member, Groove or throat greater than 1" All other welds Secondary Member Weld	MT or PT VT VT
Special Requirements, All Classes	Welds Transmitting Loads in the Through Thickness Direction in Members Greater than 1"	UT base metal beneath the weld

CORE SUPPORT STRUCTURES
(1)(2)(8)

Component	Weld Type	NDE Req'ment
Core Support Structures (Provide direct support or restraint of the fuel, etc, under normal operating conditions).	Category A, longitudinal butt welds Category B, circumferential butt welds Category C, flange to shell welds Category D, nozzle to shell welds Category E, beam end connections to other structures.	*
	Repair welds under 3/8 inch or 10% deep.	MT or PT
	Repair welds over 3/8 inch or 10% deep.	MT or PT plus RT or UT
Internal Structures (Can be any other structure within the reactor vessel). Nonmandatory.	Same as above	Same
Temporary Attachments (Removed before operation)	All	MT or PT

* Examination may be by any technique or certain combinations of techniques, from simple VT to MT or PT plus RT or UT. Quality factor n and fatigue factor f are dependent on the technique(s) selected, in accordance with Table NG-3352-1.

NOTES:

1) The required confirmation that facility welding activities are in compliance with the certified design commitments will include the following third party verifications:

   a. Facility welding specifications and procedures meet the applicable ASME Code requirements
   b. Facility welding activities are performed in accordance with the applicable ASME Code requirements
   c. Welding activities related records are prepared, evaluated and maintained in accordance with the ASME Code requirements
   d. Welding processes used to weld dissimilar base metal and welding filler metal combinations are compatible for the intended applications
   e. The facility has established procedures for qualifications of welders and welding operators in accordance with the applicable ASME Code requirements
   f. Approved procedures are available and used for preheating and post heating of welds, and those procedures meet the applicable requirements of the ASME Code
   g. Completed welds are examined in accordance with the applicable examination method required by the ASME Code

2) Radiographic film will be reviewed and accepted by the COL applicant's nondestructive examination (NDE), Level III examiner prior to final acceptance

3) The NDE requirements for Class 1 components will be as stated in subarticle NB-5300 of Section III of the ASME Code

4) The NDE requirements for Class 2 components will be as stated in subarticle NC-5300 of Section III of the ASME Code

5) The NDE requirements for Class 3 components will be as stated in subarticle ND-5300 of Section III of the ASME Code

6) The NDE requirements for containment vessels will be as stated in subarticle NE-5300 of Section III of the ASME Code

7) The NDE requirements for component supports will be as stated in subarticle NF-5300 of Section III of the ASME Code

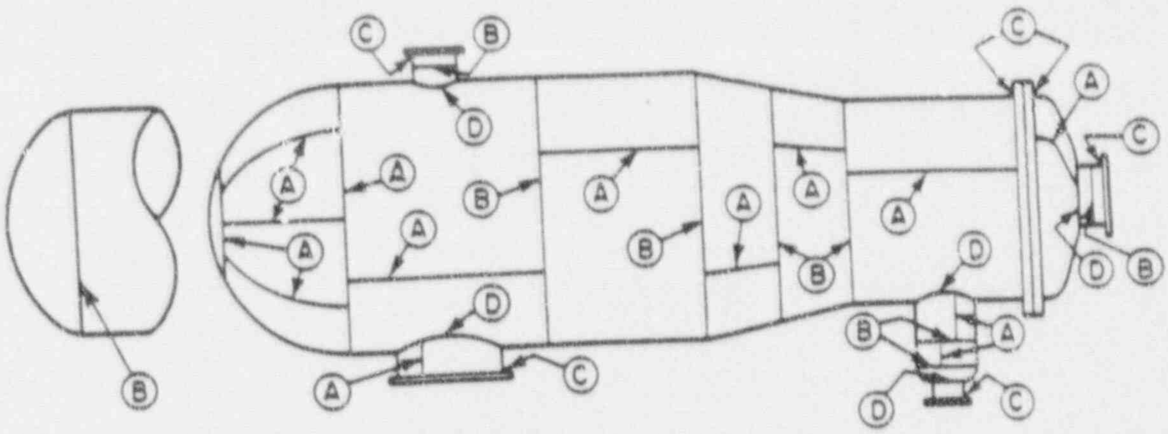
8)

~~8)~~ 9 For corner joints UT may be used instead of RT. For Type 2 full penetration corner weld joints, if RT is used, the fusion zone, and parent metal beneath the attachment surface shall be UT examined after welding.

LEGEND:

RT - Radiographic Examination; UT - Ultrasonic Examination; MT - Magnetic Particle Examination; LP - Liquid Penetrant Examination; VT - Visual Examination

The NDE requirements for Core Support structures will be as stated in subarticle NG-5300 of Section III of the ASME Code.



~~Fig. NB-3351-1~~ Welded joint locations typical of categories A, B, C, and D

~~Fig. NB-3352-1 typical butt joints~~

2. Summary of Seismic Qualification and Environmental Qualification

The staff and GE reached agreement on the approach to be used for verifying (1) the seismic qualification of mechanical and electrical equipment and (2) the environmental qualification of electrical equipment. The verification will be performed as part of an ITAAC basic configuration check in each system. The supporting SSAR material has been compiled from GE report NEDE-24326-1-P and is under review by GE for its proprietary nature.

3. Summary of MOVs

The staff and GE reached an agreement on Tier 2 description for MOVs that covers design and qualification as well as pre-operational testing commitments (see attached). We also reached an agreement on Tier 1 description of this issue. The agreement was that MOVs will be included in the basic configuration ITAAC for qualification testing. For each system there will also be a MOV ITAAC in the three column format (see attached). The staff also indicated the need for an ITAAC for check valves. Tier 2 descriptions for check valves must also be written.

Although it was concluded that other valves and pumps would not need further Tier 1 treatment, Tier 2 (the SSAR) does not contain sufficient information on the design and qualification and pre-operational testing of other types of valves as well as pumps. The staff will be working with GE to reach agreement on the information that will be added to the ABWR SSAR.

Table

(System Name Used Here (Motor Operated Valves))

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptence Criteria
1. Motor-operated valves (MOV) designated in Section ________ as having an active safety-related function will open and/or close under differential pressure and fluid flow conditions	1. Opening and/or closing tests of installed valvas will be conducted under preoperational differential pressure, fluid flow, and temperature conditions.	1. Each MOV opens and/or closes. The following valves open and/or close in the following time limits: **Valve** — **Time (sec)** _____ — _____ open _____ close _____ — _____ open _____ close

ldin lealer

14A.6 METHOD FOR MEETING EQUIPMENT ITAAC

14A.6.1 Motor Operated Valves

Design and Qualification

For each motor-operated valve assembly (MOV) with an active safety related function, the design basis and required operating conditions (including testing) will be established.

The licensee will establish the following design and qualification requirements and will provide acceptance criteria for these requirements. By testing each size, type, and model the licensee will determine the torque and thrust (as applicable to the type of MOV) requirements to operate the MOV and will ensure the adequacy of the torque and thrust that the motor-operator can deliver under design (design basis and required operating) conditions. These design conditions include fluid flow, differential pressure (including pipe break), system pressure, fluid temperature, ambient temperature, minimum voltage, and minimum and maximum stroke time requirements. The licensee will ensure that the structural capability limits of the individual parts of the MOV will not be exceeded under design conditions. The licensee will demonstrate by testing each size, type, and model that torque and thrust requirements from in-situ testing conditions can be extrapolated to design conditions. The licensee will ensure that the valve specified for each application is not susceptible to pressure locking and thermal binding.

Preoperational Testing

The licensee will test each MOV in the open and close directions under static and maximum achievable conditions using diagnostic equipment that measures torque and thrust (as applicable to the type of MOV), and motor parameters. The licensee will test the MOV under various differential pressure and flow conditions and perform a sufficient number of tests to reliably extrapolate the torque and thrust requirements to its design conditions. The licensee will determine the torque and thrust requirements to close the valve for the position at which there is diagnostic indication of hard seat contact. The licensee will extrapolate the torque and thrust requirements (including diagnostic equipment inaccuracy) from the test to design conditions for such parameters as differential pressure, fluid flow, under voltage and temperature. For the point of control switch trip, the licensee will determine any loss in torque produced by the actuator and thrust delivered to the stem for increasing differential pressure and flow conditions (referred to as load sensitive behavior). The licensee will compare the design torque and thrust requirements to the control switch trip torque and thrust subtracting margin for load sensitive behavior, control switch repeatability, and degradation. The licensee will measure the total thrust and torque delivered by the MOV under static and dynamic conditions (including diagnostic equipment inaccuracy and control switch repeatability) to compare to the allowable structural capability limits for the individual parts of the MOV. The licensee will test for proper control room position indication of the MOV.

The parameters and acceptance criteria for demonstrating that the above functional performance requirements have been met are as follows.

(a) As required by the safety function: the valve must fully open; the valve must fully close with diagnostic indication of hard seat contact.

(b) The control switch settings must provide adequate margin to achieve design requirements including consideration of diagnostic equipment inaccuracy, control switch repeatability, load sensitive behavior, and margin for degradation.

(c) The motor output capability at degraded voltage must equal or exceed the control switch setting including consideration of diagnostic equipment inaccuracy, control switch repeatability, load sensitive behavior and margin for degradation.

(d) The maximum torque and thrust (as applicable for the type of MOV) achieved by the MOV including diagnostic equipment inaccuracy and control switch repeatability must not exceed the allowable structural capability limits for the individual parts of the MOV.

(e) The remote position indication testing must verify that proper disk position is indicated in the control room.

(f) Stroke-time measurements taken during valve opening and closing must meet minimum and maximum stroke-time requirements.

4. Summary of EMI/RFI, Setpoints, EQ for I&C

The structure of ITAAC and DAC affecting development, testing, and installation of digital, safety-related, instrumentation and control (I&C) equipment will be as follows:

1. The concept of several generic ITAAC or DAC is eliminated.

2. I&C development is viewed as an integrated process involving both hardware and software aspects simultaneously, instead of the previous emphasis on a separate software process without specific attention to hardware performance. Hardware aspects will specifically incorporate EQ (including seismic qualification) and EMI/RFI issues.

3. GE will provide one Instrumentation and Control DAC describing the process of integrated software and hardware development.

4. The DAC process will describe capturing hardware performance requirements with regard to EQ and EMI/RFI in the hardware and software design specification.

5. Hardware testing of the integrated hardware/software equipment will be specifically addressed by the DAC in the description of the V&V plan and overall plan.

6. The integrated hardware/software process will include verification of accuracy of instrument loops in the installed safety-related systems. When sensors are processed through digital I&C logic, this involves both sensor accuracy and analog-to-digital conversion accuracy. ITAAC statements will be developed to address accuracy based on a non-proprietary version of GE's setpoint methodology that GE will submit. These statements will be part of DAC because normal operating parameters that determine alarm and control thresholds may not be established before the various process systems are installed. To ensure proper identification of Class 1E sensors that are to be included in accuracy determination, individual system or building design descriptions (figures or text) shall indicate these sensors.

5. Summary of Electrical Independence (Separation)

GE and the staff decided to treat the electrical independence (e.g. fluid system components powered from independent Class 1E electrical divisions) in the fluid system ITAAC by a "standard" type ITAAC entry. See individual fluid system ITAAC. The overall treatment of electrical independence including separation is to be treated in the Electrical System and I&C ITAAC. See attached "standard language for independence for electrical and I&C systems."

Standard Language for ~~Electrical~~ Independence for Electrical and I&C Systems (includi the instrument lines)

January 15, 1993

CDC

x. Independence is provided between Class 1E Divisions, and between Class 1E Divisions and non-Class 1E equipment, in the ___________ System.

ITA

x.1. Tests will be performed in the ______________ System by providing a test signal in only one Class 1E Division at a time.

AC

x.1. The test signal exists only in the Class 1E division under test in the ________ System.

ITA

x.2. Inspection of the as-installed Class 1E Divisions in the ______ System will be performed.

AC

x.2. Physical separation exists between Class 1E Divisions in the _______ System. Physical separation exists between Class 1E Divisions and non-Class 1E equipment in the __________ System.

6. Summary of Piping Design Acceptance Criteria (DAC)

The staff and GE reached agreement on the resolution of comments from the industry/NUMARC and the NRC's Greybeard Committee on the generic piping ITAAC (also referred to as Piping DAC). The piping design description has been substantially expanded to include the certified design commitments. The ITAAC have been reduced in number to consolidate those design commitments that are implicit in the ASME Code requirements and to eliminate some design criteria that were deemed not appropriate for the ITAAC treatment. Proposed additions and changes to the SSAR were provided by GE to support the piping DAC/ITAAC changes. See the attached information.

3.3 PIPING DESIGN

3.3.1 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 components are designed and constructed in accordance with the ASME Code requirements identified in the individual system Design Descriptions.

Piping systems are 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 components which are subjected to severe thermal transients, the effects of these transients shall be included in the design.

The Feedwater lines shall be designed for thermal stratification loads.

¶ Piping systems will 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 minimize 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 are 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 are within their allowable stress limits, (2) pipe whip restraints and jet shield designs are capable of mitigating pipe break loads, and (3) loads on safety-related systems, structures and components are within their design loads limits. 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.

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 (3.3.1).

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 GENERIC PIPING DESIGN

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The piping system is 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.	Inspections of ASME Code required documents will be conducted.	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.	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.	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 3.3.1.	An inspection of the as-built piping and supports will be performed. A reconciliation analysis using the as-designed and as-built information will be performed.	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.

The results of the data analyses, vibration amplitudes, natural frequencies, and mode shapes are then compared to those obtained from the theoretical analysis.

Such comparisons provide the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained from previous vibration tests has been utilized in the generation of the dynamic models for seismic and loss of coolant accident (LOCA) analyses for this plant. The models used for this plant are similar to those used for the vibration analysis of earlier prototype BWR plants.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

This section delineates the criteria for selection and definition of design limits and loading combination associated with normal operation, postulated accidents, and specified seismic and other reactor building vibration (RBV) events for the design of safety-related ASME Code components (except containment components which are discussed in Section 3.8).

This section discusses the ASME Class 1, 2, and 3 equipment and associated pressure retaining parts and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. A discussion of major equipment is included on a component-by-component basis to provide examples. Design transients and dynamic loading for ASME Class 1, 2, and 3 equipment are covered in Subsection 3.9.1.1. Seismic-related loads and dynamic analyses are discussed in Section 3.7. The suppression pool-related RBV loads are described in Appendix 3B. Table 3.9-2 presents the combinations of dynamic events to be considered for the design and analysis of all ABWR ASME Code Class 1, 2, and 3 components, component supports, core support structures and equipment. Specific loading combinations considered for evaluation of each specific equipment are derived from Table 3.9-2 and are contained in the design specifications and/or design reports of the respective equipment. See Subsection 3.9.7.4 for COL license information requirements.

Thermal stratification of fluids in a piping system is one of the specific operating conditions that is included in the loads and load combinations that are contained in the piping design specifications and design reports. It is known stratification can occur in the feedwater piping during plant startup and when the plant is in hot standby conditions following scram (see Subsection 3.9.2.1.3). If, during design or startup, evidence of thermal stratification is detected in any other piping system, then stratification will be evaluated. If it cannot be shown that the stresses in the pipe are low and that movement due to bowing is acceptable, then stratification will be treated as a design load. In general, if temperature differences between the top and bottom of the pipe are less than 50°F, it may be assumed design specification and stress reports need not be revised to include stratification.

The design life for the ABWR Standard Plant is 60 years. A 60 year design life is a requirement for all major plant components with reasonable expectation of meeting this design life. However, all plant operational components and equipment except the reactor vessel are designed to be replaceable, design life not withstanding. The design life requirement allows for refurbishment and repair, as appropriate, to assure the design life of the overall plant is achieved. In effect, essentially all piping systems, components and equipment are designed for a 60 year design life. Many of these components are classified as ASME Class 2 or 3 or Quality Group D. In the event any non-Class 1 components are subjected to cyclic loadings, including operating vibration loads and thermal transient effects, of a magnitude and/or duration so severe that the 60 year design life can not be assured by required Code calculations, COL applicants ~~will~~ identify these components and either provide an appropriate analysis to demonstrate the required design life or provide designs to mitigate the magnitude or duration of the cyclic loads. Components excluded from this requirement are (1) tees where mixing of hot and cold fluids occurs and thermal sleeves have been provided in accordance with the P&IDs, (2) components, such as the quencher, for which a fatigue analysis has already been performed, providing the com-

Severe thermal transients that will be evaluated for possible effect on plant life are temperature rate changes faster than 1500°F/Hour when the total fluid temperature change is greater than 50°F.

3.9.7 COL License Information

3.9.7.1 Reactor Internals Vibration Analysis, Measurement and Inspection Program

The first COL applicant will provide, at the time of application, the results of the vibration assessment program for the ABWR prototype internals. These results will include the following information specified in Regulatory Guide 1.20.

R. G. 1.20	Subject
C.2.1	Vibration Analysis Program
C.2.2	Vibration Measurement Program
C.2.3	Inspection Program
C.2.4	Documentation of Results

NRC review and approval of the above information on the first COL applicant's docket will complete the vibration assessment program requirements for prototype reactor internals.

In addition to the information tabulated above, the first COL applicant will provide the information on the schedules in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals.

Subsequent COL applicants need only provide the information on the schedules in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals. (See Subsection 3.9.2.4).

3.9.7.2 ASME Class 2 or 3 or Quality Group D Components with 60 Year Design Life

COL applicants will identify ASME Class 2 or 3 or Quality Group D components that are subjected to cyclic loadings, including operating vibration loads and thermal transients effects, of a magnitude and/or duration so severe the 60 year design life can not be assured by required Code calculations and, if similar designs have not already been evaluated, either provide an appropriate analysis to demonstrate the required design life or provide designs to mitigate the magnitude or duration of the cyclic loads. (See Subsection 3.9.3.1.)

3.9.7.3 Pump and Valve Inservice Testing Program

COL applicants will provide a plan for the detailed pump and valve inservice testing and inspection program. This plan will

(1) Include baseline pre-service testing to support the periodic in-service testing of the components required by technical specifications. Provisions are included to disassemble and inspect the pump, check valves, and MOVs within the Code and safety-related classification as necessary, depending on test results. (See Subsections 3.9.6, 3.9.6.1, 3.9.6.2.1 and 3.9.6.2.2)

(2) Provide a study to determine the optimal frequency for valve stroking during inservice testing. (See Subsection 3.9.6.2.2)

(3) Address the concerns and issues identified in Generic Letter 89-10; specifically the method of assessment of the loads, the method of sizing the actuators, and the setting of the torque and limit switches. (See Subsection 3.9.6.2.2)

3.9.7.4 Audit of Design Specification and Design Reports

COL applicants will make available to the NRC staff design specification and design reports required by ASME Code for vessels, pumps, valves and piping systems for the purpose of audit. (See Subsection 3.9.3.1)

insert Att. A

3.9.8 References

1. *BWR Fuel Channel Mechanical Design and Deflection*, NEDE-21354-P, September 1976.

2. *BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings*, NEDE-21175-P, November 1976.

3. NEDE-24057-P (Class III) and NEDE-24057 (Class I) Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants.

Attachment A

3.9.7.5 ASME Class 1,2 and 3 Piping System Clearance Requirements

ASME Class 1,2 and 3 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 COL licensee shall verify that the maximum calculated piping system deflections under service conditions do not exceed the minimum clearances between the piping system and nearby structures, systems, or components. The COL licensee shall document in the certified design stress report that the clearance requirements have been met.

3.9.7.6 As-Built Reconciliation Analysis For ASME Class 1,2 and 3 Piping Systems

For ASME Class 1,2 and 3 piping systems, the COL licensee shall reconcile the as-built piping system with the as-designed piping system. The COL licensee will perform an as-built inspection of the pipe routing, location and orientation, the location, size, clearances and orientation of piping supports, and the location and weight of pipe mounted equipment. This inspection will be performed by reviewing the as-built drawings containing verification stamps, and by performing a visual inspection of the installed piping system. The piping configuration and component location, size, and orientation shall be within the tolerances specified in the certified as-built piping Stress Report. The tolerances to be used for reconciliation of the as-built piping system with the as-designed piping system are provided in the EPRI report, "Guidelines for Piping System Reconciliation (NCIG-05, Revision 1)," NP-5639 dated May 1988. A reconciliation analysis using the as-built and as-designed information shall be performed. The certified as-built Stress Report shall document the results of the as-built reconciliation analysis.

HZH

3.6.4 See attach. B

3.6.~~4~~5 COL License Information

3.6.~~4~~5.1 Details of Pipe Break Analysis Results and Protection Methods

The following shall be provided by the COL applicant (See Subsection 3.6.2.5):

(1) A summary of the dynamic analyses applicable to high-energy piping systems in accordance with Subsection 3.6.2.5 of Regulatory Guide 1.70. This shall include:

(a) Sketches of applicable piping systems showing the location, size and orientation of postulated pipe breaks and the location of pipe whip restraints and jet impingement barriers.

(b) A summary of the data developed to select postulated break locations including calculated stress intensities, cumulative usage factors and stress ranges as delineated in BTP MEB 3-1.

(2) For failure in the moderate-energy piping systems listed in Table 3.6-6, descriptions showing how safety-related systems are protected from the resulting jets, flooding and other adverse environmental effects.

(3) Identification of protective measures provided against the effects of postulated pipe failures for protection of each of the systems listed in Tables 3.6-1 and 3.6-2.

(4) The details of how the MSIV functional capability is protected against the effects of postulated pipe failures.

(5) Typical examples, if any, where protection for safety-related systems and components against the dynamic effects of pipe failures include their enclosure in suitably designed structures or compartments (including any additional drainage system or equipment environmental qualification needs).

(6) The details of how the feedwater line check and feedwater isolation valves functional capabilities are protected against the effects of postulated pipe failures.

(7) see attach B

Att. B

3.6.4 As-Built Inspection of High Energy Pipe Break Mitigation Features

An as-built inspection of the high energy pipe break mitigation features shall be performed. The as-built inspection shall confirm that systems, structures and components, that are required to be functional during and following an SSE, are protected against the dynamic effects associated with high energy pipe breaks. An as-built inspection of pipe whip restraints, jet shields, structural barriers and physical separation distances shall be performed.

For pipe whip restraints and jet shields, the location, orientation, size and clearances to allow for thermal expansion shall be inspected. The locations of structures, identified as a pipe break mitigation feature, shall be inspected. Where physical separation is considered to be a pipe break mitigation feature, the assumed separation distance shall be confirmed during the inspection.

3.6.5 COL License Information

3.6.5.1 Details of Pipe Break Analysis Results and Protection Methods

(7) An inspection of the as-built high energy pipe break mitigation features shall be performed. The pipe break analysis report or leak-before-break report shall document the results of the as-built inspection of the high energy pipe break mitigation features. (See Subsection 3.6.4, for a summary of the as-built inspection requirements.)