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ASEA BROWN BOVERI

April 30, 1992
LD-92-060

Docket No. 52-002

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

Subject: System 80+™ Tier 1 Design Descriptions and ITAAC's

Reference: Letter LD-92-038 from C. B. Brinkman to D. M. Crutchfield, dated March 25, 1992

Dear Sirs:

Enclosed are ten (10) example Tier 1 design descriptions and associated ITAAC's for the System 80+™ Standard Design.

ABB Combustion Engineering (ABB-CE) has benefitted from the sample ITAAC's which the NRC and the ITAAC lead design have been developing. We consider the enclosure to be draft Tier 1 material because revisions are expected as System 80+ design certification proceeds. We look forward to staff feedback on these initial ITAAC submittals and to agreeing with the staff on the format and content of the Tier 1 design descriptions and ITAAC which are most appropriate for the System 80+ design.

We request a meeting with the staff by May 12, 1992 to discuss the enclosed design descriptions and ITAAC's. We have assembled a multi-disciplinary team which is proceeding immediately with the preparation of the next set of ITAAC's. In order for us to meet the common objective of submitting the bulk of the Tier 1 design descriptions and the associated ITAAC before the draft Safety Evaluation issues in September, it is necessary that we receive very prompt feedback from the staff on the adequacy of the attached material.

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ABB Combustion Engineering Nuclear Power

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This submittal fulfills the commitment in the reference letter to supply a set of about 10 representative ITAAC by April 30th. Please contact me or Mr. George Hess of my staff at (203) 285-5218 if you have any questions on the enclosed material.

Very truly yours,

COMBUSTION ENGINEERING, INC.

C. B. Brinkman

C. B. Brinkman
Acting Director
Nuclear Systems Licensing

cbb/mls

Enclosures: As Stated

cc: T. Wambach (NRC)
T. Boyce (NRC)
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1.3.2 DESIGN FOR THE PROTECTION OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS AGAINST DYNAMIC EFFECTS OF PIPE BREAK AND LEAK BEFORE BREAK

Design Description

Vital structures, components, equipment and systems necessary for the successful operation of the safe shutdown systems are protected from the damaging effects of postulated pipe breaks not eliminated by leak-before-break evaluations. Designs which protect these items consider the consequences of pipe whip, water-spray, jet impingement, flooding, compartment pressurization, and environmental conditions.

Essential systems protected are those that are needed to safely shut down the reactor or mitigate the consequences of a pipe break for a given postulated piping failure. In addition, systems are protected such that no break violates the following criteria:

1. A postulated pipe break which is not a LOCA will not cause a LOCA.

2. The postulated pipe break will not cause unacceptable consequential reactor coolant steam or feedwater line damage.

3. The function of safety systems required to perform protective actions to mitigate the consequences of the postulated break will be maintained.

4. The ability to place the plant in a safe shutdown condition will be maintained.

Protection of vital equipment is achieved primarily by separation of redundant safe shutdown systems and by separation of high-energy pipe lines from safe shutdown systems. This redundancy and separation results in a design which requires very few special protective features such as whip restraints and jet deflectors to ensure safe shutdown capability following a postulated high-energy line break. In general, layout of the facilities follows a multi-step process to ensure adequate separation.

1. Safety-related systems are located away from most high-energy piping.

2. Redundant safety systems and subsystems are located in separate compartments.

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3. As necessary, specific components are enclosed to maintain the redundancy required for those systems that must function as a consequence of specific piping failure events.

Protection requirements are met through the protection afforded by the walls, columns, floors, abutments, and foundations in many cases. Where adequate protection does not already exist due to separation, additional barriers, deflectors, or shields are provided as necessary to meet the functional protection requirements. These additional defenses are designed to withstand the combined effects of the postulated failure plus normal operating loads plus earthquake loading.

Where protection requirements are not met through existing separation, barriers, or shields, piping restraints are provided as necessary to meet those requirements.

Restraints are not provided when it can be shown that the pipe break would not cause unacceptable damage to essential systems or components.

Analyses of postulated pipe break events are performed to identify those safety-related systems and components that provide protective actions required to mitigate, to acceptable limits, the consequences of the postulated pipe break events. In conducting these facility response analyses, the following are among the criteria used to establish the integrity of systems and components necessary for safe shutdown and maintenance of the shutdown condition:

1. Each high- or mode[illegible]ergy fluid system pipe failure is considered separately as a singl[illegible]ed initiating event occurring during normal plant conditions.

2. Offsite power is assumed to be unavailable if an automatic turbine generator trip or automatic reactor trip is a direct result of a postulated piping failure.

3. Piping systems containing high energy fluids are designed so that the effects of a single postulated pipe break cannot, in turn, cause failures of other pipes or components with unacceptable consequences.

The design criteria define acceptable types of isolation for safety-related elements and for high-energy lines from similar elements of the redundant trains. Separation is

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accomplished by.

1. Routing the two groups through separate compartments, or

2. Physically separating the two groups by a specified minimum distance, or

3. Separating the two groups by structural barriers.

The design criteria assure that a postulated failure of a high energy line or a safety-related element cannot take more than one safety-related train out of service.

Postulated pipe ruptures are considered in all plant piping systems not eliminated by leak before break evaluations, and the associated potential for damage to required systems and components is evaluated on the basis of the energy in the system. Each postulated rupture is considered separately as a single postulated initiating event. For each postulated break, an evaluation is made of the effects of pipe whip, jet impingement, compartment pressurization, environmental conditions, and flooding. These evaluations of the required systems and components demonstrate that the protection requirements above are met.

A leak-before-break evaluation is performed for Class 1 piping with a diameter of ten inches or greater (i.e., the reactor coolant system (RCS) main loop piping, surge line shutdown cooling and safety injection lines) and for the main steam line inside containment in order to eliminate the dynamic effects of pipe rupture from the design basis. The evaluation is intended to meet the requirements of 10 CFR 50, Appendix A, General Design Criterion (GDC) 4. The evaluation is performed using the guidelines of NUREG 1061, Vol. 3. Piping of this sort is designed to be not particularly susceptible to failure from the effects of corrosion, water hammer or low- and high-cycle fatigue, or degradation or failure of the piping from indirect causes. In addition, a leak detection system as recommended by Regulatory Guide 1.45 capable of detecting a leakage rate of less than 1.0 gpm from the primary system is included in the System 80+ plant design.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 1.3.2-1 specifies the inspections, tests, analyses and associated acceptance criteria for evaluating design for the protection of structures, components, equipment and systems against dynamic effects of pipe break and leak before break.

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TABLE 1.3.2-1

DESIGN FOR THE PROTECTION OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS AGAINST DYNAMIC EFFECTS OF PIPE BREAK AND LEAK BEFORE BREAK

Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Analytical methods for the dynamic and static analysis of piping systems and the corresponding component stress analysis shall be specified in a certified design specification for each piping system. The dynamic analysis of piping systems shall use a suitable dynamic method, such as time history or response spectrum method, or an equivalent static load method. For the applied method, the key analysis parameters shall be addressed.	1. Review of the certified design specification and the certified stress report will be conducted to confirm that the piping was designed and analyzed in compliance with ASME Code, Section III requirements.	1. Methods shall be in compliance with the requirements of the ASME Code, Section III.

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TABLE 1.3.2-1 (Continued)

DESIGN FOR THE PROTECTION OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS AGAINST DYNAMIC EFFECTS OF PIPE BREAK AND LEAK BEFORE BREAK
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. All ASME Code Safety Class 1, 2, and 3 piping systems which are essential for safe shutdown shall be designed to assure that they will maintain sufficient dimensional stability to perform their required function following application of all loads to which they will be subjected during postulated events requiring their safety function.	2. An inspection of the certified stress report will be conducted to assure that none of the stresses or deflections of the piping system exceed values which could lead to large reductions in the cross-sectional flow area.	2. ASME Code, Section III limits that protect the piping and pipe supports against primary stress failures shall be compared with allowable values that preclude impairment of functional capability. In no case shall stresses exceed values allowed for Service Level D in ASME Code, Section III.

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TABLE 1.3.2-1 (Continued)

DESIGN FOR THE PROTECTION OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS AGAINST DYNAMIC EFFECTS OF PIPE BREAK AND LEAK BEFORE BREAK

Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. Essential piping systems, including required pipe whip restraints, will be designed to protect against the dynamic effects associated with the postulated rupture of high energy and moderate energy fluid systems. A pipe break analysis report will be generated to confirm that the piping system is acceptable for all postulated breaks. Piping systems that are qualified for the optional leak-before-break design approach (i.e., RCS main loop, surge line shutdown cooling, safety injection lines, and the main steam line inside containment) may exclude design against the dynamic effects from the postulation of breaks in high energy piping.	3. Inspections of ASME Code, Section III required documents and the pipe break analysis report will be conducted to confirm that the piping system was designed and analyzed in compliance with requirements that assure postulated pipe breaks will not unduly impact the safety of the plant.	3. The essential functions of structures, systems, and components shall not be precluded by the postulated pipe breaks. For those components required for safe shutdown, limits to meet the ASME Code requirements for faulted conditions and limits to ensure required operability shall be met.

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TABLE 1.3.2-1 (Continued)

DESIGN FOR THE PROTECTION OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS AGAINST DYNAMIC EFFECTS OF PIPE BREAK AND LEAK BEFORE BREAK
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The piping, its appurtenances, and its supports, will satisfy the ASME Class, Seismic Category, and Quality Group requirements commensurate with its classification.	4. Inspections will be conducted of ASME Code required documents and the Code stamp on the components.	4. Existence of ASME Code required documents and the Code stamps on the components shall be reviewed to confirm that the piping and components have been designed, analyzed, fabricated, and examined in accordance with the applicable requirements.
5. Redundant safety systems and subsystems will be located in separate compartments.	5. Visual inspection of plant redundant safety systems and subsystems verifies they are located in separate regions.	5. Visual inspection report shall be prepared to provide confirmation.

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1.6.3 ANNULUS VENTILATION SYSTEM

Design Description

The Annulus Ventilation System (AVS) is designed as an engineered safety feature and is credited in analyzing design basis accidents. The Annulus Ventilation System reduces the concentration of radioactivity in the annulus atmosphere between the primary containment and the secondary containment by filtration, holdup (decay) and recirculation before the air is released to the atmosphere.

The Annulus Ventilation System (AVS) takes the air from above the primary containment dome, filters it and discharges a portion of the air close to the annulus floor and a portion of the air to the atmosphere. Two redundant filtration trains complete with fans, filters, dampers, ductwork, supports and control systems are provided (see Figure 1.6.3-1). Each train is capable of maintaining the air flow within specified limits at maximum and minimum filter pressure drops. The dampers modulate exhaust air as required to maintain the negative pressure greater than 0.25 inches of water gage within the annulus.

Each filter train consists of a moisture eliminator, prefilter, electric heater, carbon absorber and a HEPA filter before and after the carbon absorber. Each train is sized to remove the fission products released to the annulus following any of the postulated accidents. Failure of AVS to perform the intended function will be detected by a unit vent radiation monitor, which monitors the activity level of the system effluent.

Electrical and control component separation is maintained between the redundant trains to meet single active failure criteria although the ducting inside the annulus is shared. Components of the Annulus Ventilation System (AVS) are designed to withstand the post-accident pressure and temperature transients. Components of the AVS are designed as Seismic Category I equipment.

Each train is normally not operating and is activated by a containment spray actuation signal. Each train is powered by Class 1E power and backup power from the Emergency Diesel Generator. Indication of damper position and fan operating status is provided in the control room. High temperature conditions for each absorber bed and high and low differential pressures across filter beds are alarmed in the control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 1.6.3 provides the inspections, tests and/or analyses and associated acceptance criteria.

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TABLE 1.6.3-1

ANNULUS VENTILATION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The AVS is capable of reducing the concentration of radioactivity in the annulus to an acceptable level consistent with safety analysis.	1. Documented records reviews and field test results evaluation will be conducted to verify specified parameters.	1. The filter efficiencies and fan capacity meet the following requirements.
a) Filter Efficiencies: 95% (Elemental and organic iodine) 99% (Particulate)	a) DOP test will be conducted to measure filter efficiencies.	a) 95% (Elemental and Organic iodine) 99% (Particulate)
b) Fan Capacity: < 18000 CFM	b) Fan capacity will be tested in the straight portion of the duct either upstream or downstream of the fan.	b) < 18,000 CFM
2. A simplified system configuration is shown in Figure 1.6.3-1.	2. Inspections of installation records together with plant walkdowns will be conducted to confirm that the installed equipment is in compliance with the design configuration defined in Figure 1.6.3-1.	2. The system configuration is in accordance with Figure 1.6.3-1.

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TABLE 1.6.3-1 (Continued)

ANNULUS VENTILATION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The essential fans, dampers, interlocks and alarms can be supplied power from class 1E power with backup power from Emergency Diesel Generator.	3. The system tests will be conducted after installation to confirm that the electrical power supply configurations are in compliance with design commitments.	3. The installed equipment can be powered from Class 1E power or from the Emergency Diesel Generator.
4. The AVS is designed to Seismic Category I Requirements.	4. Evaluation of de[illegible] characteristics and construction records will be performed to evaluate conformance to design requirements.	4. The essential components of the AVS meet Seismic Category I requirements.
5. Essential AVS Components are protected from malfunctions caused by floods, tornados, internal missiles, pipe breaks and whip jet impingement and interactions with non-seismic systems.	5. Evaluation of as-constructed AVS components will be performed against design requirements.	5. The essential AVS components are protected from the identified hazards.
6. Each train is activated by a Containment Spray Actuation Signal.	6. Field tests will be conducted to verify this actuation of each train upon receiving a Containment Spray Actuation Signal.	6. Each train is activated by a Containment Spray Actuation Signal.

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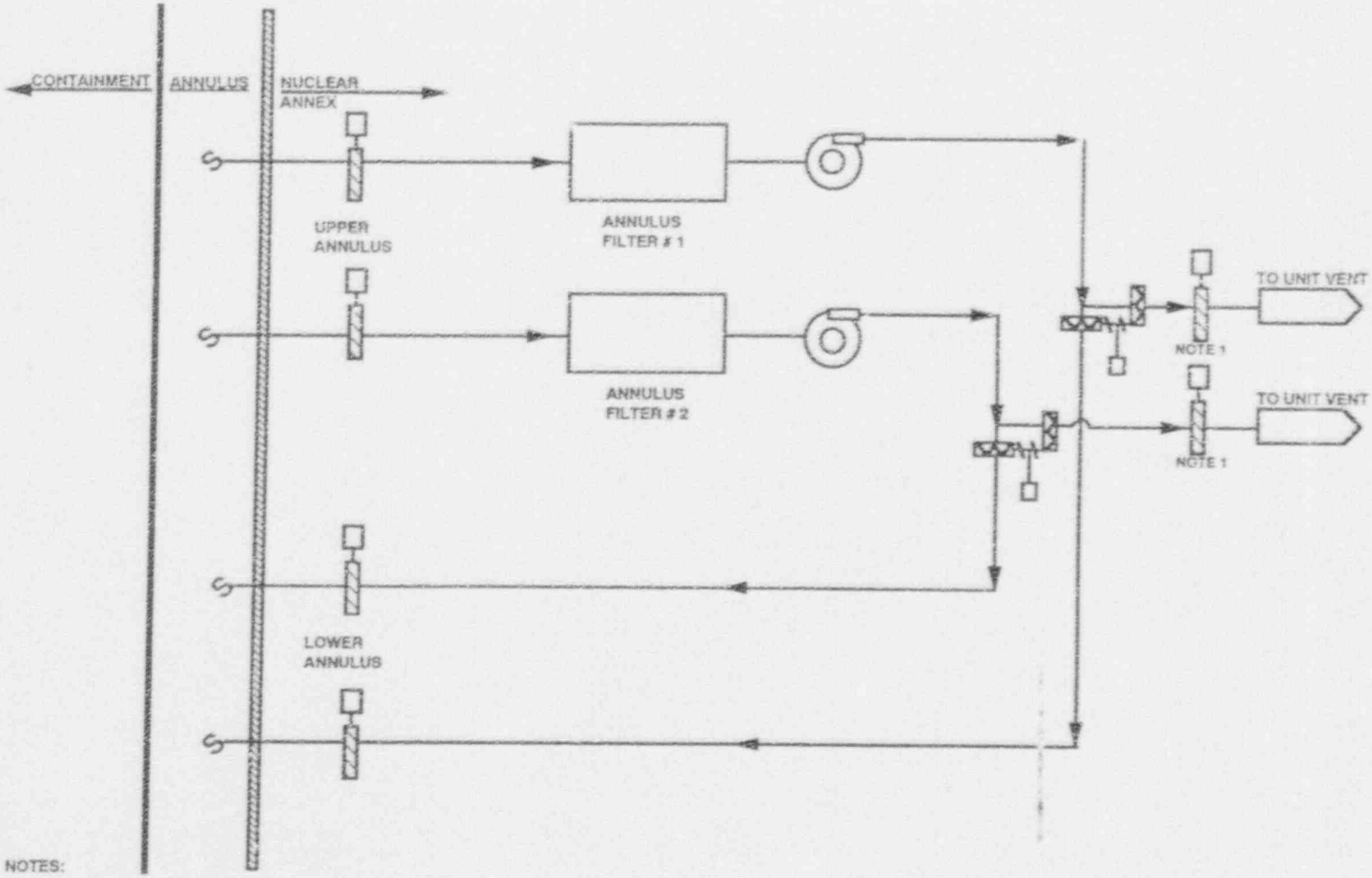

NOTES:

1. THIS DAMPER IS FOR TORNADO PROTECTION AND FOR ISOLATION WHEN FAN IS OFF

FIGURE 1.6.3
ANNULUS VENTILATION SYSTEM

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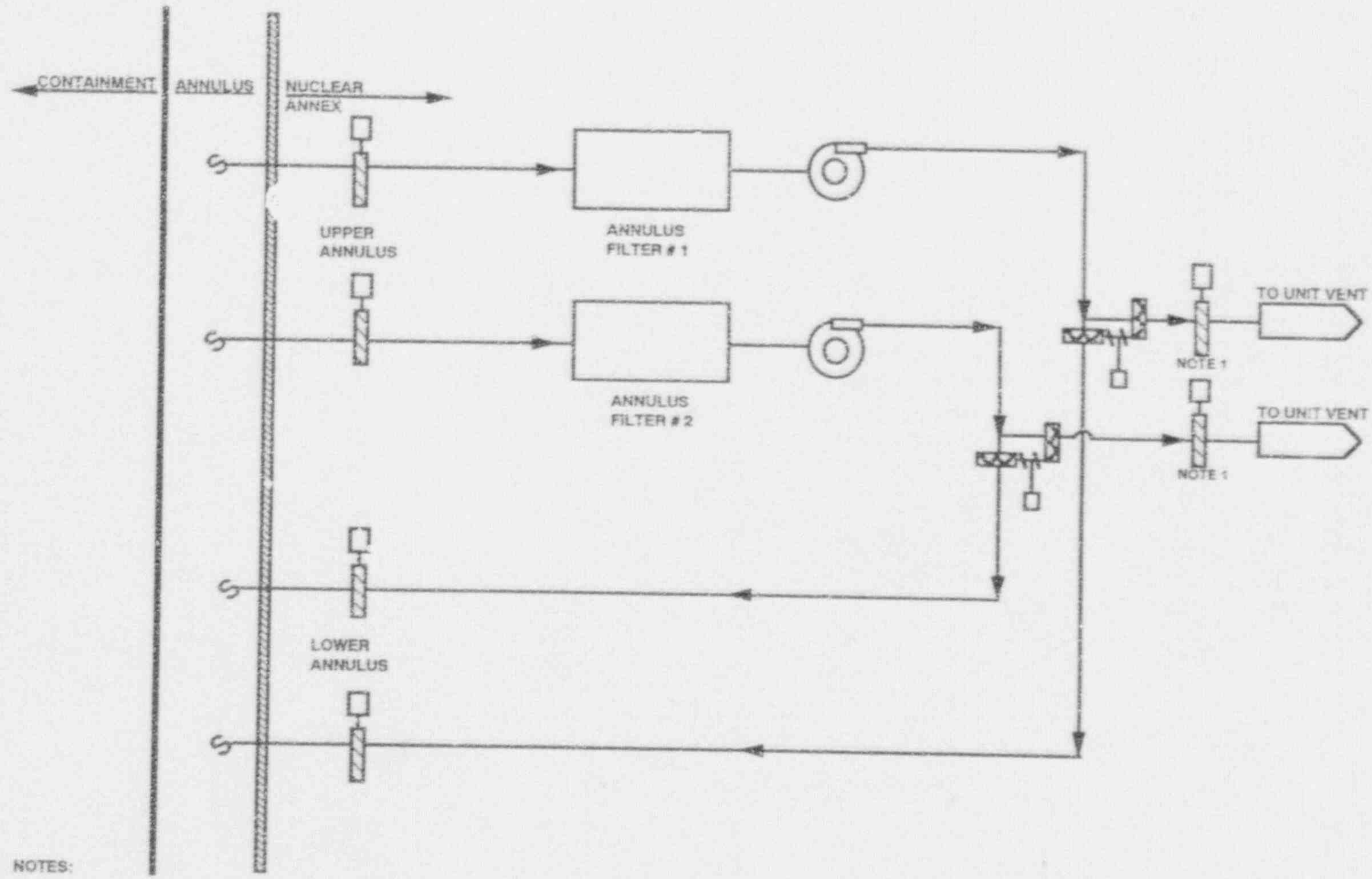

NOTES:

1. THIS DAMPER IS FOR TORNADO PROTECTION AND FOR ISOLATION WHEN FAN IS OFF

FIGURE 1.6.3
ANNULUS VENTILATION SYSTEM

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also be manually initiated from the Main Control Room.

The SITs, which contain borated water pressurized by a nitrogen cover gas, constitute a passive injection system. No operator action or electrical signal is required for operation. Each tank is connected to a DVI nozzle by a separate line containing two check valves which isolate the tank from the RCS during normal operation. When RCS pressure falls below SIT pressure, the check valves open, discharging the contents of the tank into the reactor vessel. A remotely operated isolation valve in each SIT discharge line is administratively controlled open to assure SIT injection when needed. To further assure SIT availability, each SIT isolation valve receives an open signal upon an SIAS.

The SITs are located inside the containment to minimize their distance from the RCS, but outside the biological shield to protect them from RCS-generated missiles.

Vent valves are connected to each SIT. Venting may be required to lower SIT pressure to shutdown cooling entry pressure following a LOCA. During normal operation, the vent valves are locked closed and power is removed.

The shutoff head and flow rates of the SI pumps were selected to insure that adequate flow is delivered to the reactor vessel to cool the core during LOCAs and non-LOCA design basis events. SI pump capacity is also sufficient to remove decay heat during feed and bleed operations. The SIS is designed to provide net positive suction head (NPSH) greater than the pump vendor's required NPSH for all expected fluid temperature conditions during SIS operation. Each SIS pump has a minimum flow recirculation line to the IRWST to ensure sufficient pump flow during operation against shutoff pressure.

The SIS components and instrumentation can be powered from the plant turbine generator (onsite power), and/or plant startup power source (offsite power), or the emergency generators (emergency power). Power connections are through at least two independent buses. One independent electrical bus supplies power to two SI pumps and associated valves. A second independent electrical bus supplies power to the remaining two SI pumps and associated valves.

Power to the SIS hot leg injection valves is designed such that a single electrical failure cannot cause spurious initiation of hot leg injection flow, nor will a single electrical failure prevent initiation of flow through at least one hot leg injection line.

SIS components required for injection of borated water into the reactor vessel are classified Seismic Category I. Mechanical components meet ASME Boiler and Pressure Vessel Code Section III requirements as follows: IRWST, SI pumps, SITs, piping and valves up to the second check valve from the RCS are Class 2; piping and valves from (and including) the second check valve from the RCS are Class 1.

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SIS components and instrumentation which must operate following a design basis event are designed, built, and qualified to operate in the post-event environment in the compartment where the component or instrument is located.

Adequate physical separation is provided between the redundant piping paths and containment penetrations of the SIS to prevent a failure of one train from affecting other trains. Cabling associated with redundant channels of Class 1E circuits for the SIS is physically separated to preserve redundance and prevent a single failure from causing multiple channel malfunctions or interactions between channels.

The SIS permits periodic inspection of important components such as injection nozzles, piping, pumps, and valves, and periodic functional testing, including the full operational sequence that brings the system into operation.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 1.6.5-1 specifies the inspections, tests, analyses and associated acceptance criteria for the SIS.

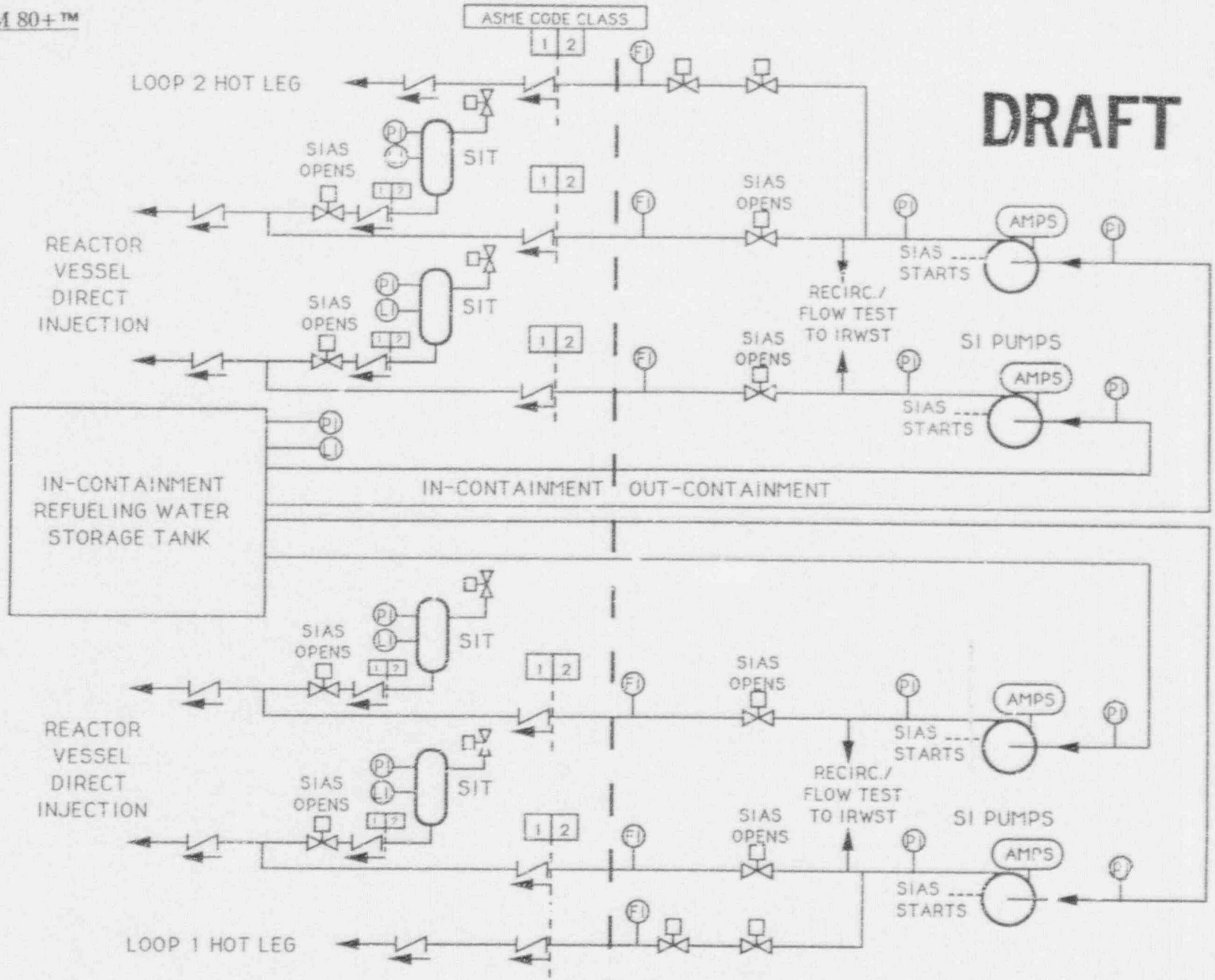

FIGURE 1.6.5-1
SAFETY INJECTION SYSTEM

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TABLE 1.6.5-1

SAFETY INJECTION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The configuration of the SIS includes, as a minimum, the flow paths and components shown in Figure 1.6.5-1.	1. Perform walkdown inspections of the as-built SIS configuration.	1. The as-built SIS configuration includes the flow paths and components shown in Figure 1.6.5-1.
2. SIS mechanical equipment is built in accordance with ASME Code, Section III, requirements.	2. Inspect procurement and installation records to verify components were manufactured per the relevant ASME requirements.	2. SIS equipment has appropriate ASME, Section III, Class 1, 2, or 3 code stamp.
3. All SIS components required for injecting water into the reactor vessel are classified Seismic Category I, and are qualified for the environment in the locations where they are installed.	3. See Generic Seismic and Environmental Qualification inspections, tests, and analyses.	3. See Generic Seismic and Environmental Qualification acceptance criteria.

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TABLE 1.6.5-1 (Continued)

SAFETY INJECTION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Test, Analyses	Acceptance Criteria
4. SIS pumps, valves, and controls necessary for injecting water into the reactor vessel can be powered from the normal, standby, or emergency AC power sources.	4.a) Perform tests to demonstrate the capability of the SIS to automatically actuate in response to an SIAS generated by low pressurizer pressure and by high containment pressure. Perform these tests with power supplied from the normal (onsite), standby (offsite), and emergency (diesel generator) AC power sources.	4.a) With power supplied from normal, standby, and emergency AC power sources: low pressurizer pressure or high containment pressure generate an SIAS; an SIAS starts the SI pumps, opens the SI discharge isolation valves, and sends an open signal to the SIT isolation valves.
	b) Perform tests to confirm the SITs and associated valves will respond to an SIAS and will discharge SIT water volume to the RCS.	b) SIT isolation valves open on SIAS and SIT discharge check valves open when SIT pressure exceeds RCS pressure.
5. The SIS operates in the short term injection mode to provide core cooling following a LOCA or non-LOCA event.	5. Perform SIS functional tests to determine as-built system flow vs. RCS pressure and time to rated flow after an SIAS.	5. The analyses results show that the following acceptance criteria are met for LOCA and non-LOCA events: a) For LOCA, the acceptance criteria of 10CFR50.46(b)

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TABLE 1.6.5-1 (Continued)

SAFETY INJECTION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Test, Analyses	Acceptance Criteria
	Inspect as-built drawings and calculate the volume of unborated water in each SI line prior to an actuation, the IRWST volume, SIT volume, SIT inner diameter, and SIT nozzle elevation above the DVI nozzles. From the results of the SIT tests in ITA 4.b), calculate the SIT discharge line K-factor. Compare as-built data from these tests and calculations with the data used in analyses of LOCAs and non-LOCA events which require SIS operation. Confirm the analyses results.	b) For non-LOCA transients, the acceptance criteria of Section 15 of NUREG-0800, Rev. 3.5.
6. The SIS operates in the long term post-LOCA injection mode to provide flow circulation through the core.	6. Manually align the SIS for long term cooling using simultaneous DVI and hot leg injection. Perform tests to determine the hot leg/DVI flow split with four SI pumps running. Compare as-built data from these tests and	6. The SIS can be manually aligned for long term cooling using simultaneous DVI and hot leg injection. The acceptance criteria of 10CFR50.46(b) are met.

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TABLE 1.6.5-1 (Continued)

SAFETY INJECTION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Test, Analyses	Acceptance Criteria
	the tests and calculations in ITA 5 with the data used in the analysis of post-LOCA long-term cooling. Confirm the analysis results.	
7. The SI pumps have sufficient NPSH during all postulated operating conditions.	7. Perform tests to measure suction head available to each SI pump. Suction will be taken from the IRWST under maximum flow conditions in the combined suction lines (i.e., containment spray pumps also running when testing the two SI pumps with shared suction lines). Correct the measured suction head for the IRWST minimum level attainable after an SIAS and the maximum IRWST fluid temperature.	7. Minimum pump NPSH available, as determined based on as-built conditions, meets or exceeds the pump designer's NPSH requirements.
8. The SIS includes a minimum flow recirculation path for each SI pump to protect the pump from overheating.	8. The as-built system configuration and installation will be inspected and minimum flow recirculation verified by analyses, or a minimum flow measurement test will be performed	8. Minimum flow recirculation, as determined based on as-built conditions, meets or exceeds the pump designer's requirements.

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TABLE 1.6.5-1 (Continued)

SAFETY INJECTION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Test, Analyses	Acceptance Criteria
9. The SIS provides automatic indication when SIAS is bypassed and when SIS status is inoperable.	9. Perform tests to verify operation of the SIAS bypass and SIS inoperable indications.	9. SIAS bypass and SIS status inoperable are automatically indicated.
10. SIS redundant piping trains and containment penetrations are physically separated such that failure of one train will not cause failure of other trains.	10. Visual inspections will be performed against construction drawings to verify physical separation of SIS piping trains and containment penetrations.	10. Four quadrant separation is provided for the SIS piping trains and containment penetrations.
11. The SIS permits periodic inspection of important components such as injection nozzles, piping, pumps, and valves.	11. Perform visual inspection of accessibility for periodic inspections of the SIS injection nozzles, piping, pumps, and valves.	11. Access is provided for inspection of SIS injection nozzles, piping, pumps, and valves.
12. The SIS permits functional testing of the full operational sequence that brings the system into operation.	12. Demonstrate that the SIS full operational sequence can be tested by performing tests to show that a low pressurizer pressure condition or high containment pressure condition gen-	12. The initiation signals generate an SIAS, an SIAS actuates the SI pumps and associated valves, and the SIS delivers flow through either the SI lines to the RCS or the test lines to the IRWST.

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TABLE 1.6.5-1 (Continued)

SAFETY INJECTION SYSTEM
Insp ctions, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Test, Analyses	Acceptance Criteria
	an SIAS, an SIAS actuates the SI pumps and associated valves, and the SIS delivers flow to the RCS via the SI lines or to the IRWST via the test lines. These tests may be combined with ITA 4.a).	

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1.7.1 PLANT PROTECTION SYSTEM

Design Description

The System 80+ Plant Protection System (PPS) is a warning and trip system where initial warning and trip decisions are implemented with software logic installed in programmable digital devices. The system provides the limit logic, matrix logic and initiation circuits for both the reactor trip functions and the engineered safety features functions for the System 80+ Advanced Light Water Reactor design. In the System 80+ design the reactor trip function is implemented by process instrumentation, the PPS and the reactor trip switchgear. Similarly, the engineered safety features actuation is implemented by process instrumentation, the PPS, the Engineered Safety Features-Component Control System, motor starters and actuated devices.

The primary functions of the PPS are to: (1) make the logic decisions related to warning and trip conditions of the individual instrument channels, and (2) make the decisions for reactor trip and initiation of engineered safety features based on coincidence of instrument channel trip conditions.

Reactor Trip Actuation Function

The Reactor Trip actuation function of the System 80+ PPS monitors selected plant conditions and automatically effects rapid reactor shutdown (reactor trip) if monitored conditions exceed safety system setpoints. Initiation of a reactor trip for the following conditions has been demonstrated to meet PPS functional requirements:

Variable Overpower
High Logarithmic Power Level
High Local Power Density
Low Departure from Nucleate Boiling Ratio
High Pressurizer Pressure
Low Pressurizer Pressure
Low Steam Generator Water Level
Low Steam Generator Pressure
High Containment Pressure
High Steam Generator Water Level
Low Reactor Coolant Flow

The PPS includes limit logic for simple process-value to set point comparison; the limit logic is implemented in devices referred to as bistable processors. In addition the PPS includes limit logic for complex calculations of the departure from nucleate boiling ratio and the local power density, which is implemented in devices referred to as Core Protection Calculators.

If the setpoint for a trip condition varies with power, the setpoint change is performed automatically within the PPS. For each trip condition, pre-trip alarms are provided. These can provide the operator with an opportunity to take control actions to avoid the trip limit condition.

As shown in Figures 1.7.1a and 1.7.1b, the PPS includes the following elements: bistable trip, local coincidence logic, reactor trip initiation logic, ESF function initiation logic, and automatic testing of PPS logic. The PPS is divided into four channels, and each channel has a pair of bistable processors, referred to as the Plant Protection Calculator (PPC), and a Core Protection Calculator. The bistable trip processors generate trips based on the measurement channel digitized value exceeding digital setpoint. The trip outputs of the bistable processors and the core protection calculators are sent to the local coincidence logic processors. Each local coincidence logic processor receives four trip signals, one from its associated bistable processor or CPC in the channel and one from each of the equivalent bistable processors or CPCs located in the other three channels. The coincidence processors evaluate the local coincidence logic based on the state of the four like trip signals and their respective bypasses. A coincidence of two-out-of-four like trip signals is required to generate a reactor trip or ESF initiation signal. The fourth channel is provided as a spare and allows bypassing of one channel while maintaining a two-out-of-three system.

The coincidence signals are used in the generation of the Reactor Trip Switchgear System (RTSS) or Engineered Safety Features-Component Control System (ESF-CCS) initiation. Upon coincidence of two signals indicating one of the trip conditions, the PPS initiates actuation of the Reactor Trip Switchgear (RTSG). The reactor trip switchgear breakers interrupt power to the Control Element Drive Mechanism (CEDM) coils, allowing all CEA's to drop into the core by gravity. The RTSG can be tripped manually from the Main Control Panel (MCP) and Remote Shutdown Panel (RSP) independent of the PPS bistable and coincidence processors.

Engineered Safety Features Actuation Function

The Engineered Safety Features actuation function of the System 80+ PPS monitors selected plant conditions and automatically initiates engineered safety features actuation signals (ESFAS) to the Engineered Safety Features-Component Control Systems (ESF-CCS) when monitored conditions exceed safety system setpoints.

The ESF-CCS actuates the plant's ESF System components (for example pumps, valves, etc.). The System 80+ PPS initiates the following ESF actuation signals:

Safety Injection Actuation Signal
Containment Isolation Actuation Signal
Containment Spray Actuation Signal
Main Steam Isolation Signal
Emergency Feedwater Actuation Signal-1

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Emergency Feedwater Actuation Signal-2

Conditions which can initiate one or more of the above ESF actuations are:

Low Pressurizer Pressure
Low Steam Generator Water Level
Low Steam Generator Pressure
High Containment Pressure
High Steam Generator Water Level
Low Reactor Coolant Flow
High High Containment Pressure

Using the same method as described above for the reactor trip conditions, these ESF actuation conditions are monitored by the PPS bistable processors and trip outputs are sent to the local coincidence processors for detection of two or more tripped conditions. The functional logic used in the PPS to generate each of the ESF initiation signals is shown in Figures 1.7.1c, 1.7.1d, 1.7.1e and 1.7.1f.

Manual initiation of the ESF actuations independent of the bistable and coincidence processors can be performed at either the Main Control Panel (MCP) or the Remote Shutdown Panel (RSP). The PPS interface at the MCP provides for manual initiation of all ESF actuation signals. The PPS interface at the RSP provides for manual initiation of the Main Steam Isolation Signal. The ESF-CCS interfaces provide for initiation of all ESF functions on a train or component basis at either the MCP or RSP.

PPS Divisional Separation and Isolation

The PPS is a four division system which is designed to provide reliable single failure proof capability to automatically or manually initiate a reactor scram while maintaining protection against unnecessary scrams resulting from single failures in the PPS. All functions of the PPS and the components of the system are safety-related. The PPS and the electrical equipment of the system are also classified as Safety Class 3, Seismic Category I and as IEEE electrical category Class 1E.

PPS components and equipment are separated or segregated from process control system circuits and functions such as to minimize control and protection system interactions. Any necessary interlocks from the PPS to control systems are through isolation devices.

The PPS remains single-failure proof even when one entire division of channel sensors is bypassed and/or when one of the four automatic PPS trip bistable processors or CPCs is out-of-service. All equipment within the PPS is designed to fail into a trip initiating state or other safe state on loss of power or input signals or disconnection of portions of the system. The system also includes trip bypasses and isolated outputs for display, annunciation or performance monitoring. PPS interfaces to the Power Control System, Discrete Indication and Alarm System, Data Processing System, PPC Operator Modules and the Mainte~~ ~~and Test Panel are electrically isolated so that no malfunction of the associated equipment can functionally disable any portion of the PPS. The PPS related equipment is divided into four redundant divisions of sensor (instrument) channels, trip logics and trip actuators, and manual scram controls and scram logic circuitry. The automatic and manual scram initiation logic systems are independent of each other and use diverse methods and equipment to initiate a reactor scram. Once a reactor trip has been initiated, the breakers in the reactor trip switchgear latch open, assuring that the intended fast insertion of all control rods into the reactor core cannot be compromised by any action of the normal power control system. After all of the trip conditions have been cleared, deliberate operator action is required to manually reclose the trip breakers.

Figure 1.7.1b shows the PPS divisional separation aspects and the signal flow from the process instrumentation to the individual channels for initiation of protection system functions. The PPS includes calculators, logic, and other equipment necessary to monitor selected plant parameters and plant conditions and to effect reliable and rapid reactor shutdown (reactor trip) and actuate appropriate ESF System components if monitored conditions approach specified safety system settings. Four measurement channels with electrical and physical separation are provided for each parameter used in the direct generation of trip signals, with the exception of Control Element Assembly (CEA) position which is a two channel measurement. Trip bistable settings associated with initiation of reactor trip are selected to protect the core fuel design limits and the Reactor Coolant System (RCS) pressure boundary for Anticipated Operational Occurrences, and also to provide assistance in mitigating the consequences of accidents.

Basic System Parameters are:

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

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PPS Interfaces and Testing

As shown in Figure 1.7.1a, the PPS interfaces with the following:

- Class 1E safety process instrumentation, including:

  Reed Switch Position Transmission (RSPT) (for CEA position).

  Auxiliary Process Cabinet (APC) (for Pressurizer Pressure, RCS Hot Leg Temperature, RCS Cold Leg Temperature, RCP Speed Pulses, and Ex-Core Neutron Flux Power).

- Manual Actuation Signals for Reactor Trip at the Main Control Panel and the Remote Shutdown Panel.

- Manual Actuation Signals for ESF Systems at the Main Control Panel and the Remote Shutdown Panel.

- Reactor Trip Switch Gear (RTSG).

- Engineered Safety Feature Component Control System (ESF-CCS).

- The Interface and Test Processor portion of the PPS provides optical cable data link interfaces to the following:

  Power Control System
  Discrete Indication and Alarm System
  Data Processing System
  Plant Protection Calculator Operator's Module in the MCP
  Plant Protection Calculator Operator's Module in the RSP

PPS interfaces for operator interaction, alarm annunciation and testing (manual and automatic) are shown in Figure 1.7.1g.

The local and main control room PPS operator's module (one per channel) provides for entering trip channel bypasses, operating bypasses, and variable setpoint resets. These modules also provide indication of status of bypasses, operating bypasses, bistable trip and pre-trip. The local operator module provides the man-machine interface during manual testing of bistable trip functions not tested automatically.

The Interface and Testing Processor (ITP), one per channel, communicates with the bistable trip processors, coincidence processors, operator's modules, ESF-CCS, RTSS and ITP's in the other three channels to monitor, test and control the operational state of the PPS. It also provides selected PPS channel status and test results information to the Data Processing System (DPS), and Discrete Indication and Alarm

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System (DIAS).

Inspections, Tests, Analyses, and Acceptan Criteria

Table 1.7.1-1 provides a definition of the visual inspections, tests and/or analyses, together with associated acceptance criteria for the PPS.

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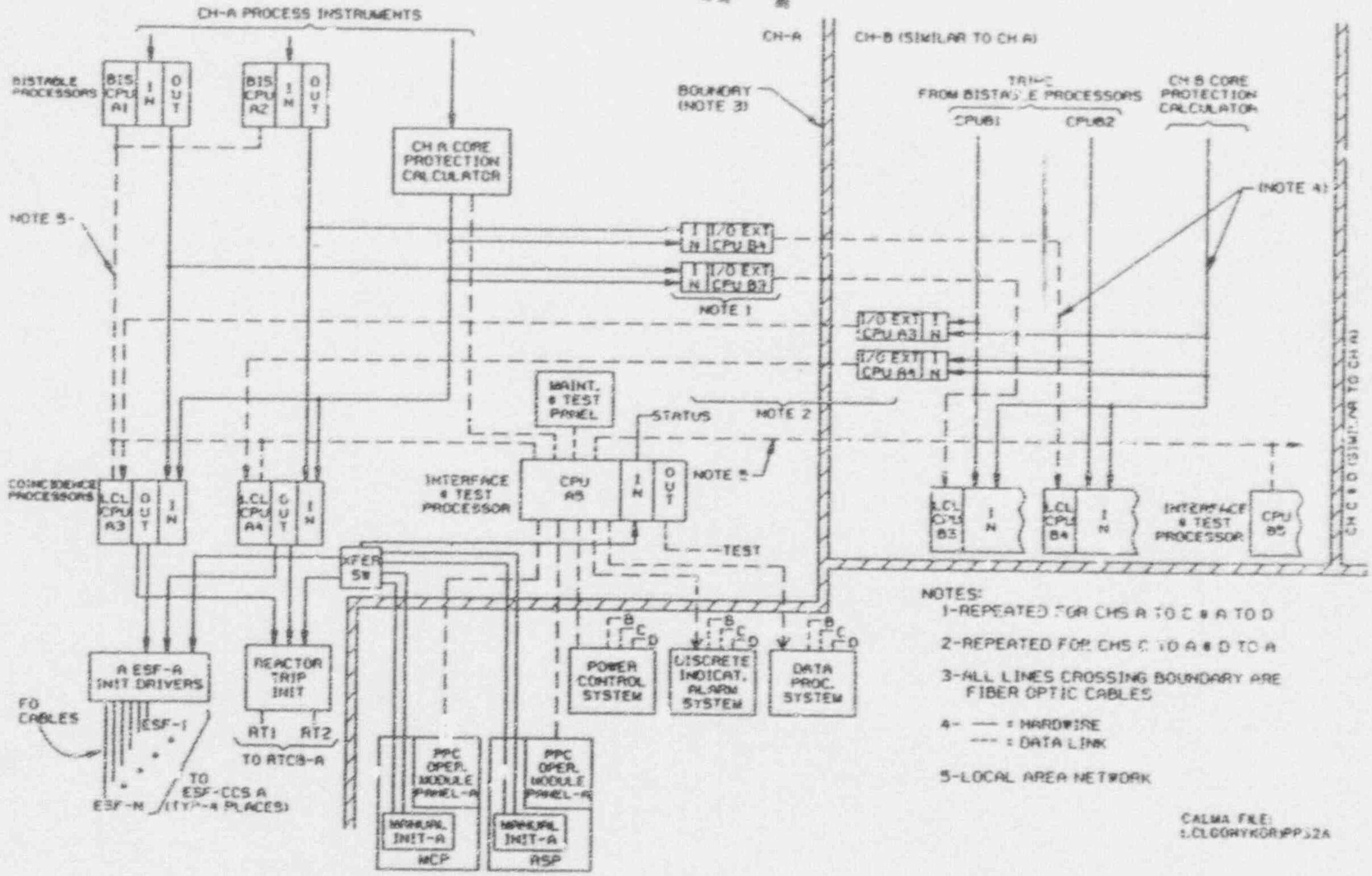

FIGURE 1.7.1a
PLANT PROTECTION SYSTEM INTERFACES

SYSTEM 80+™

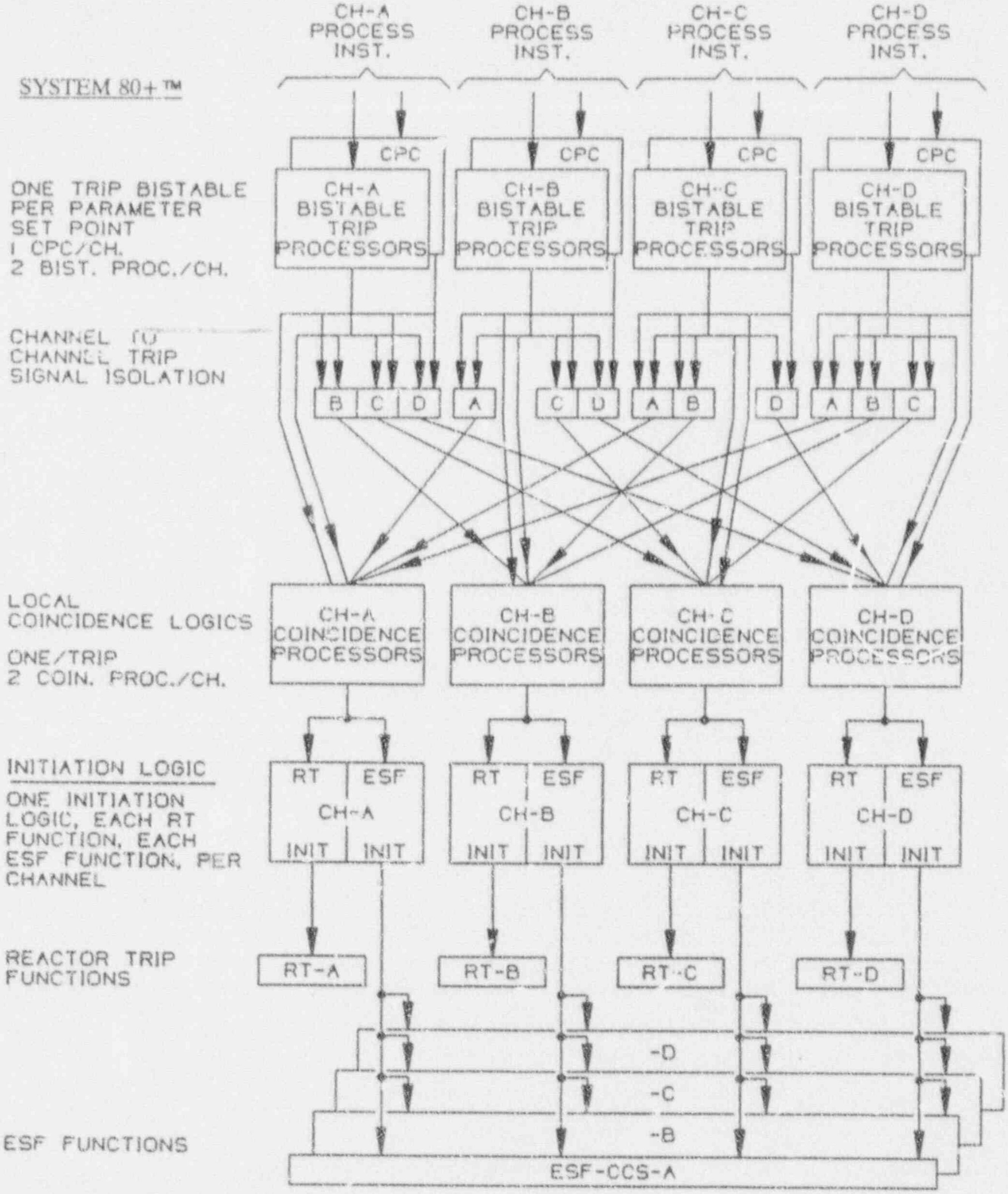

FIGURE 1.7.1b
PPS BASIC BLOCK DIAGRAM

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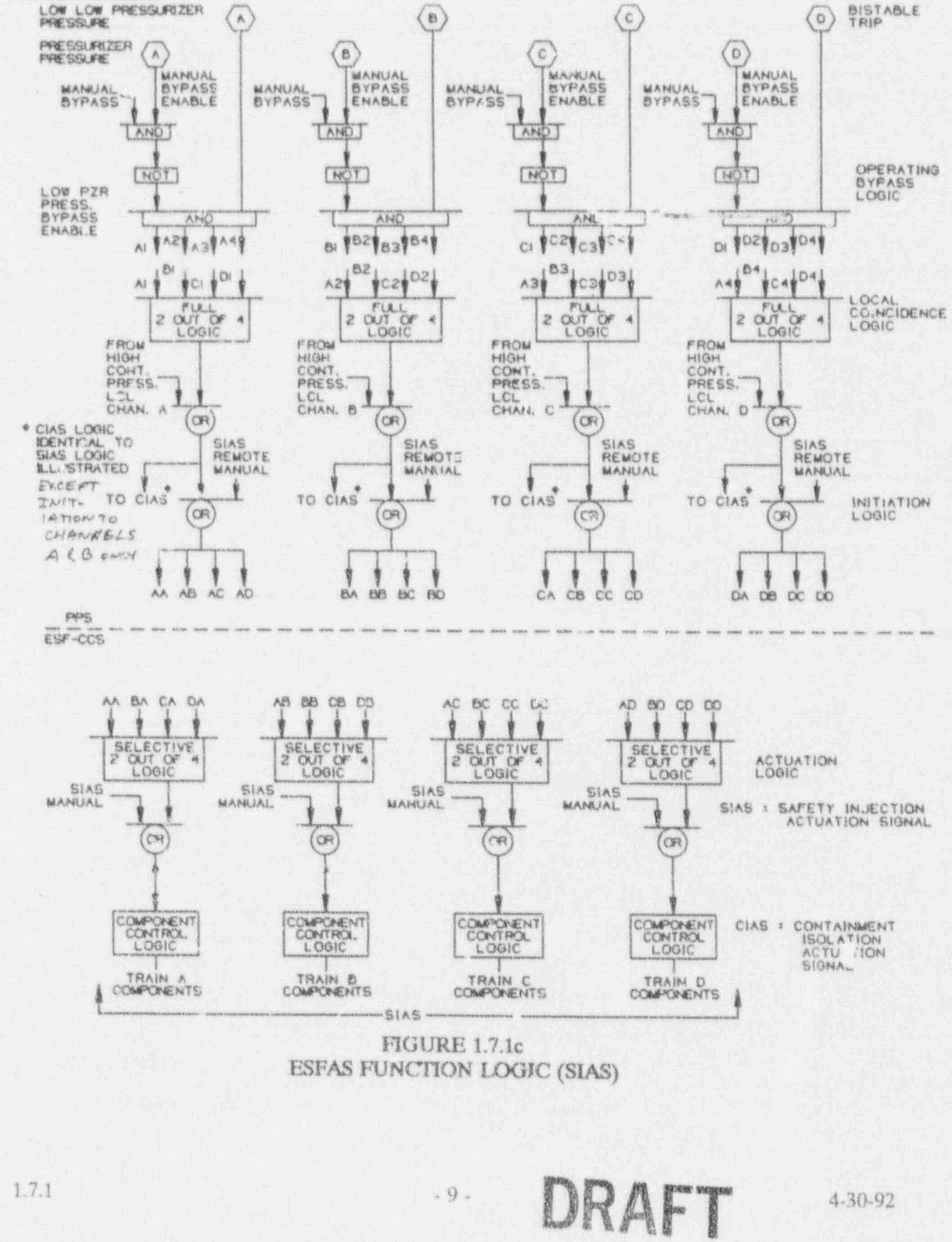

FIGURE 1.7.1c
ESFAS FUNCTION LOGIC (SIAS)

SYSTEM 80+™

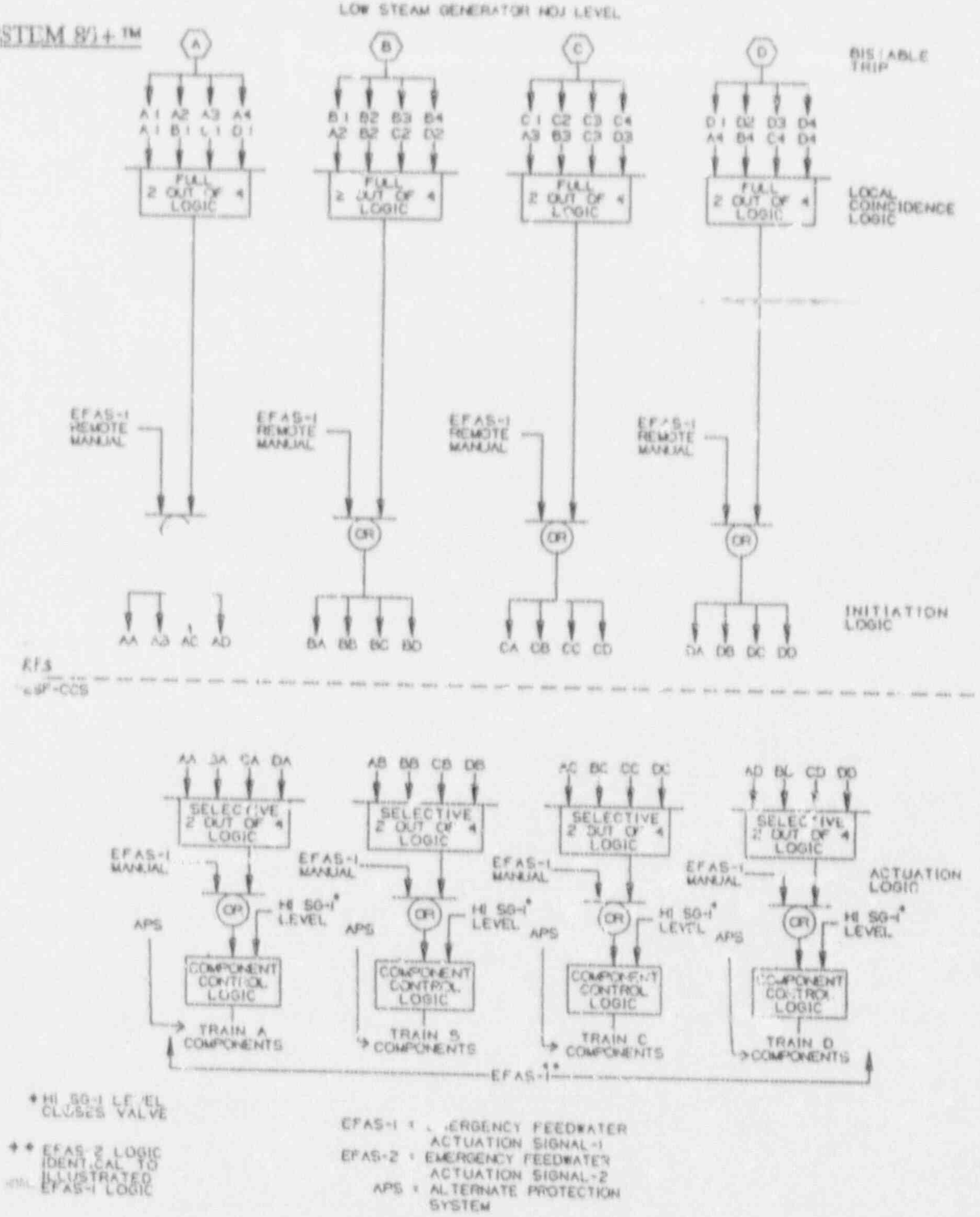

FIGURE 1.7.1d
ESFAS FUNCTIONAL LOGIC (EFAS1, EFAS2)

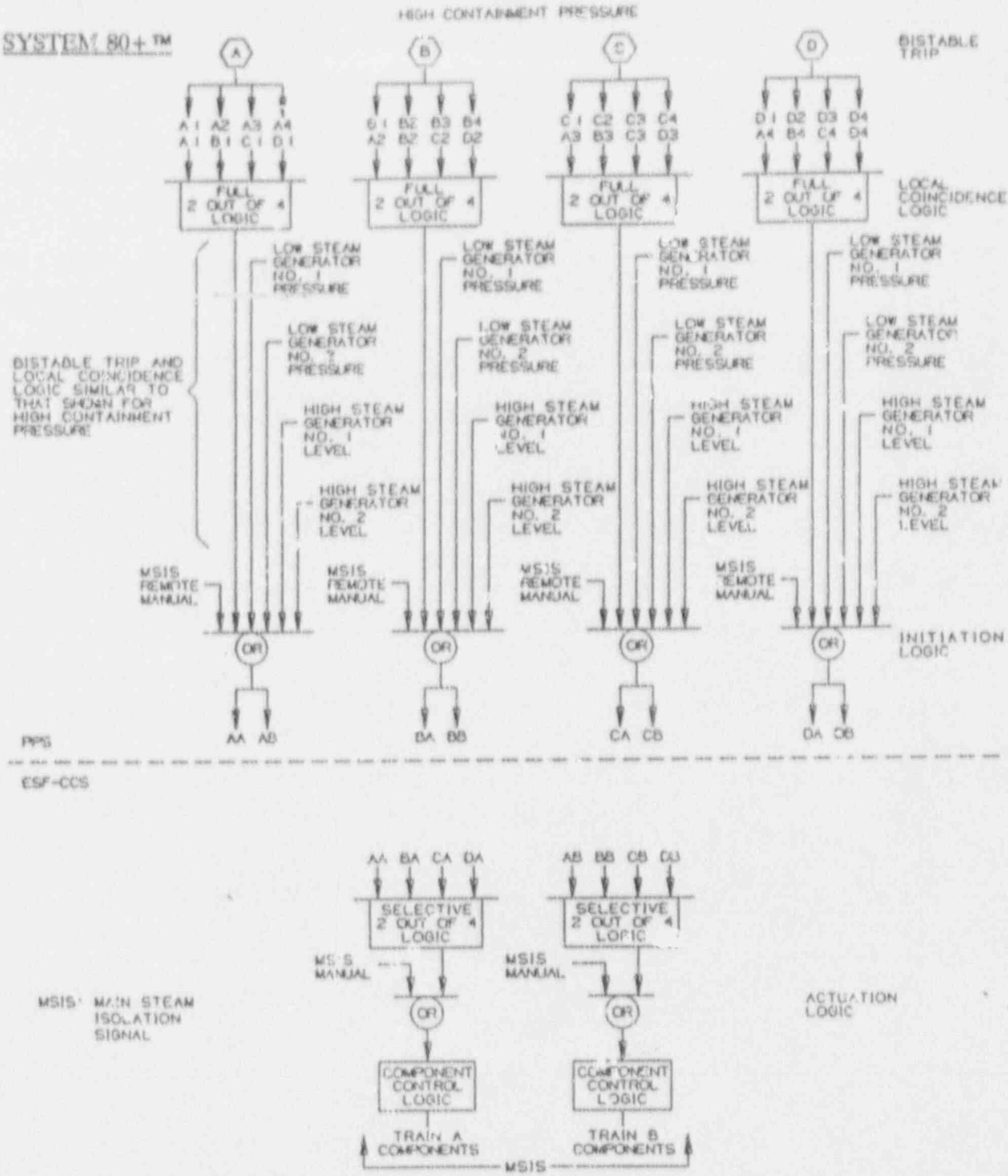

FIGURE 1.7.1e
ESFAS FUNCTIONAL LOGIC (MSIS)

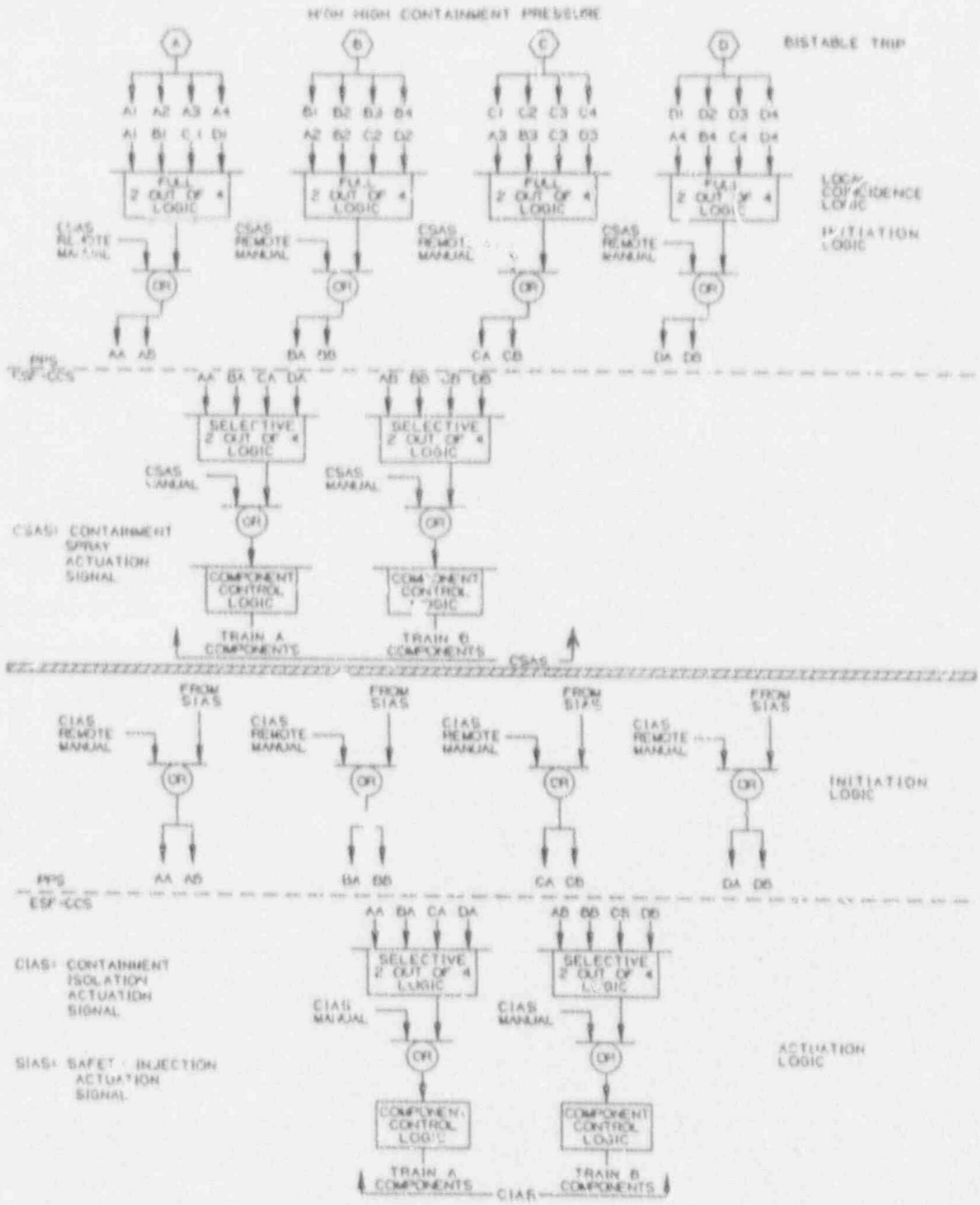

FIGURE 1.7.1f
EFSAS FUNCTIONAL LOGIC (CSAS, CIAS)

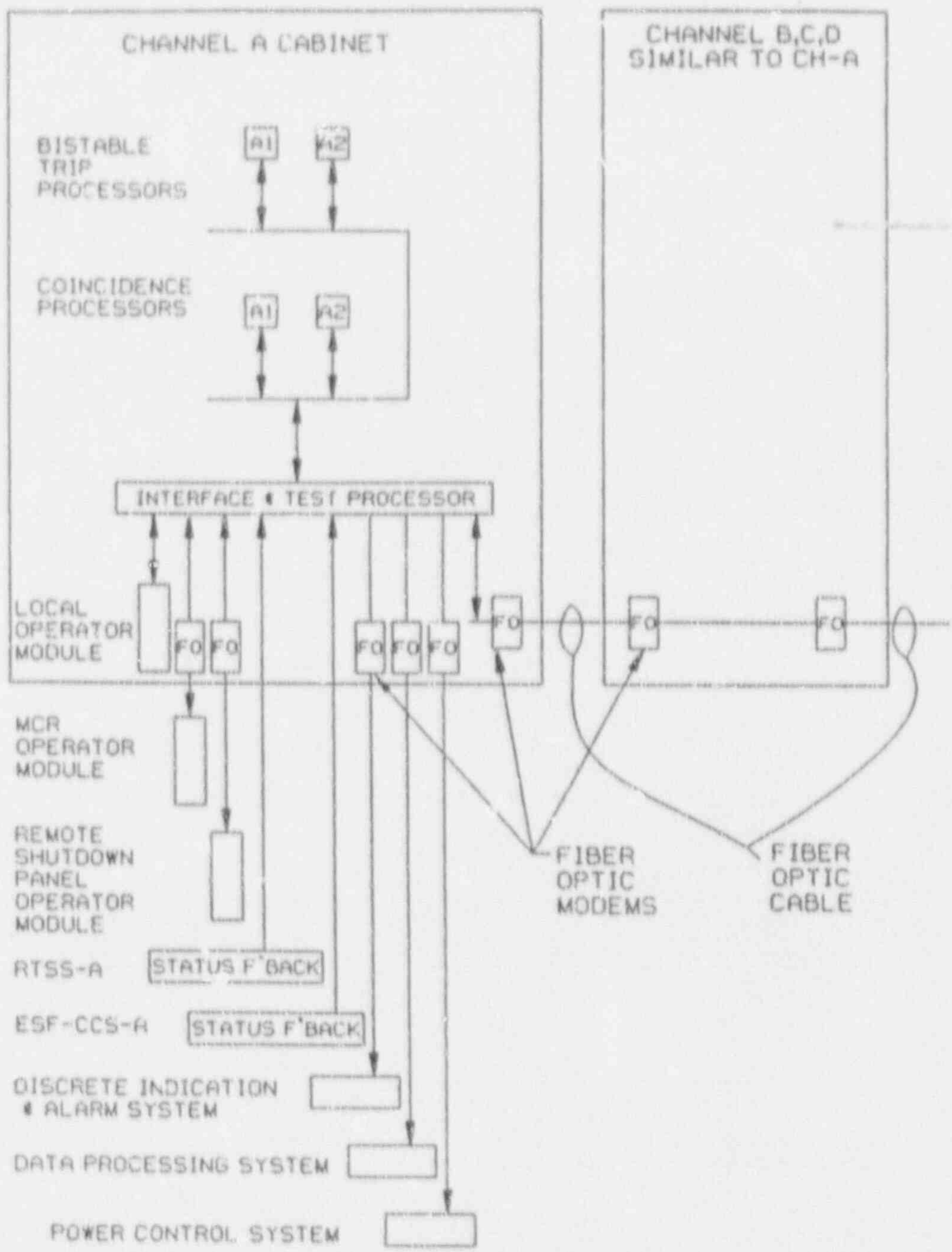

FIGURE 1.7.1g
PPS FUNCTION INTERFACE AND TESTING DIAGRAM

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TABLE 1.7.1-1

PLANT PROTECTION SYSTEM
Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. PPS safety-related software, which is utilized in effecting individual sensor channel trip decisions and trip system coincidence trip decisions, has been developed and verified, the firmware implemented and validated and then integrated with hardware; all according to a formal documented plan.	1. See Generic Software Development verification activities (ITA).	1. See Generic Software Development Acceptance Criteria (AC).
2. Critical parameter trip setpoints are based upon values used in analyses of abnormal operational occurrences. Documented instrument setpoint methodology has been used to account for uncertainties (such as instrument inaccuracies and drift) in order to establish PPS related setpoints.	2. See Generic Setpoint Methodology verification activities (ITA).	2. See Generic Setpoint Methodology Acceptance Criteria (AC).
3. PPS equipment is designed to be protected from effects of noise, such as electromagnetic interference (EMI), and has adequate surge withstand capability (SWC).	3. See Generic EMI/SWC Qualification verification activities (ITA).	3. See Generic EMI/SWC Qualification Acceptance Criteria (AC).

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TABLE 1.7.1-1 (Continued)

PLANT PROTECTION SYSTEM
Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. PPS quipment is qualified for seismic loads and appropriate environment for locations where installed.	4. See Generic Equipment Qualification verification activities (ITA).	4. See Generic Equipment Qualification Acceptance Criteria (AC).
5. PPS components and equipment are kept separate from equipment associated with process control systems.	5. Visual field inspections and analyses of relationship of installed PPS equipment and of installed equipment of interfacing process control systems (and/or tests of interfaces) to confirm appropriate isolation methods used to satisfy separation and segregation requirements.	5. PPS equipment installation acceptable if inspections, analyses and/or tests confirm that any failure in process control systems can not prevent PPS safety functions.
6. Fail-safe failure modes result upon loss of power or disconnection of components.	6. Field tests to confirm that trip conditions and/or bypass inhibits result upon loss of power or disconnection of components.	6. Acceptable if trip conditions and/or bypass inhibits result upon loss of power or disconnection of portions of the PPS.
7. Provisions exist to limit access to trip setpoints, calibration controls and test points.	7. Visual field inspections of the installed equipment will be used to confirm the existence of appropriate administrative controls.	7. The PPS hardware/firmware will be considered acceptable if appropriate methods exist to enforce administrative control for access to sensitive areas.

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TABLE 1.7.1-1 (Continued)

PLANT PROTECTION SYSTEM
Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. The four redundant divisions of PPS equipment are independent from each other except in the area of the required coincidence of trip logic decisions and are both electrically and physically separated from each other.	8. Inspections of fabrication and installation records and construction drawings or visual field inspections of the installed PPS equipment will be used to confirm the quadruple redundancy of the PPS and the electrical and physical separation between divisions.	8. Installed PPS equipment will be determined to conform to the documented description of the design as depicted in Figures 1.7.1b, c, d, e and f.
9. It is possible to conduct verifications of PPS operations, both on-line and off-line, by means of a) individual instrument channel functional tests, b) trip system functional tests and c) total system functional tests.	9. Preoperational tests will be conducted to confirm that system testing such as trip bistable tests, channel functional tests, channel calibrations, coincident logic tests, reactor trip initiation logic tests, manual trip test, and engineered safety feature initiation and actuation logic tests can be performed. These tests will involve simulation of PPS testing modes of operation. Interlocks associated with the reactor mode switch positions, and with other operational and maintenance bypasses or test switches will be tested and annunciation, display and logging functions will be confirmed.	9. The installed reactor protection system configuration, controls, power sources and installations of interfacing systems supports the PPS logic system functional testing and the operability verification of design as follows: a. Installed PPS hardware/firmware initiates trip conditions in all four PPS automatic trip systems upon coincidence of trip conditions in two or more instrument channels associated with the same trip variable(s).

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TABLE 1.7.1-1 (Continued)

PLANT PROTECTION SYSTEM
Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		b. Installed system initiates trip (both under voltage and shunt trip) upon coincidence of trip conditions in two or more of the four PPS automatic trip systems.
		c. Installed system initiates trip conditions two (of four) if manual trip switches are operated.
		d. Installed system initiates appropriate ESFAS actuation signal upon coincidence of appropriate trip conditions in two or more of the four PPS automatic trip systems.
		e. Installed system initiates appropriate ESFAS actuation signal if two of the four associated ESF manual initiation switches are operated.
		f. Trip system and ESF initiation (automatic and manual) trip conditions seal-in and protective actuation signals are maintained.

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TABLE 1.7.1-1 (Continued)

PLANT PROTECTION SYSTEM
Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		g. Installed system provides isolated status and control signals to data logging, display and annunciator systems.
		h. Installed system demonstrates operational interlocks (i.e., trip inhibits or permissives) required for different conditions of reactor operation.
10. The PPS design provides prompt protection against the onset and consequences of events or conditions that threaten the integrity of the fuel barrier.	10. Preoperational tests will be conducted to measure the PPS and supporting systems response times to: (1) monitor the variation of the selected processes; (2) detect when trip setpoints have been exceeded; and, (3) execute the subsequent protection actions when coincidence of trip conditions exists.	10. The PPS hardware/firmware response to initiate reactor scram and Engineered Safety Feature actuation will be considered acceptable if such response is demonstrated to be sufficient to assure that the specified acceptable fuel design limits are not exceeded.

1.9.1 SPENT FUEL STORAGE

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Design Description

The spent fuel storage racks are designed to support and protect spent fuel assemblies and assure a geometrically safe configuration with respect to criticality. The spent fuel storage rack arrangement is shown in Figure 1.9.1-1. The spent fuel storage rack arrangement is shown in Figure 1.9.1-2. There are sufficient spent fuel storage racks to provide for the licensed storage capacity.

The spent fuel storage racks are designed to maintain a neutron multiplication factor less than Keff=.95 for normal loadings including seismic events and impact due to the drop of a fuel assembly plus its handling tool.

The spent fuel storage racks are designed to meet the stress acceptance criteria of the ASME B&PV Code, Section III, Subsection NF, Class 3.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 1.9.1-1 specifies the inspections, tests, analyses and associated acceptance criteria for spent fuel storage.

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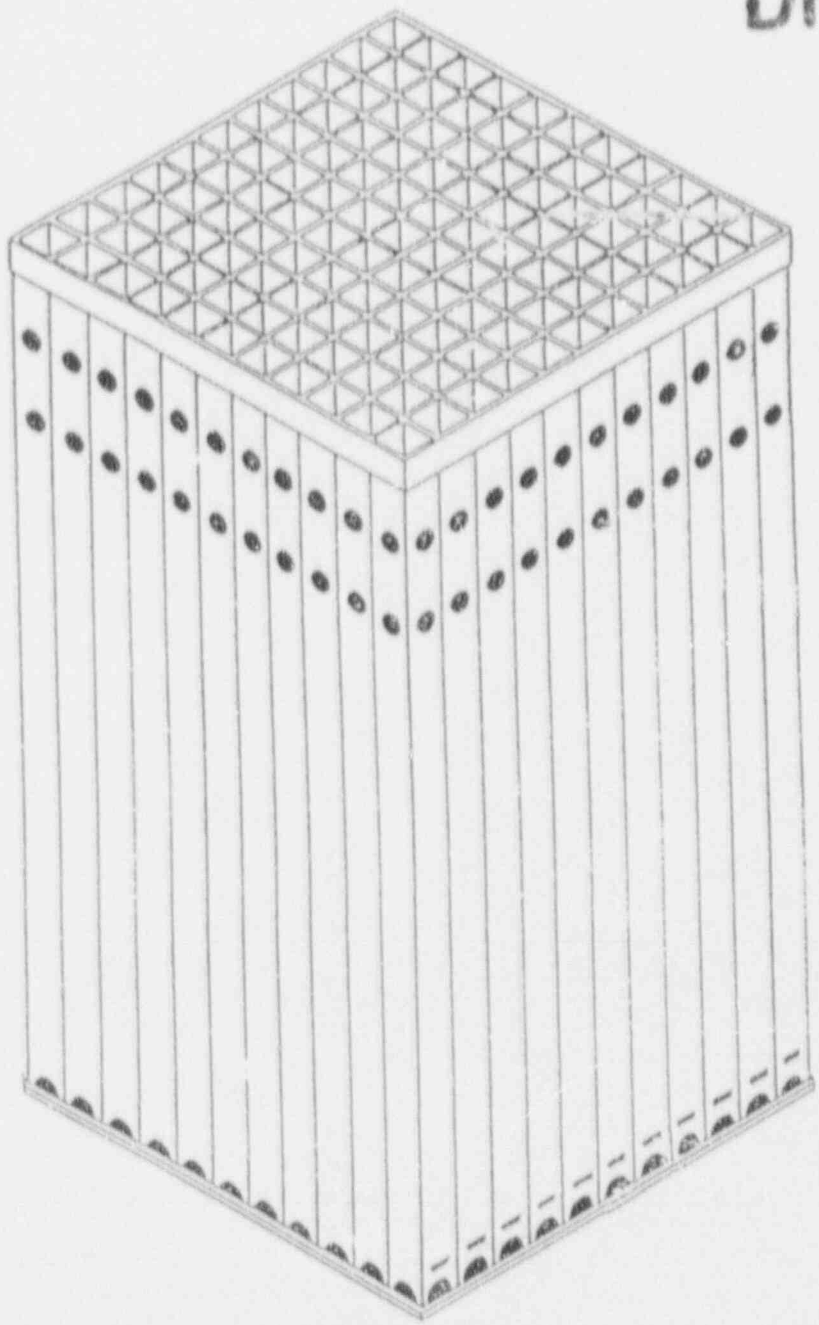

FIGURE 1.9.1-1
TYPICAL SPENT FUEL STORAGE RACK

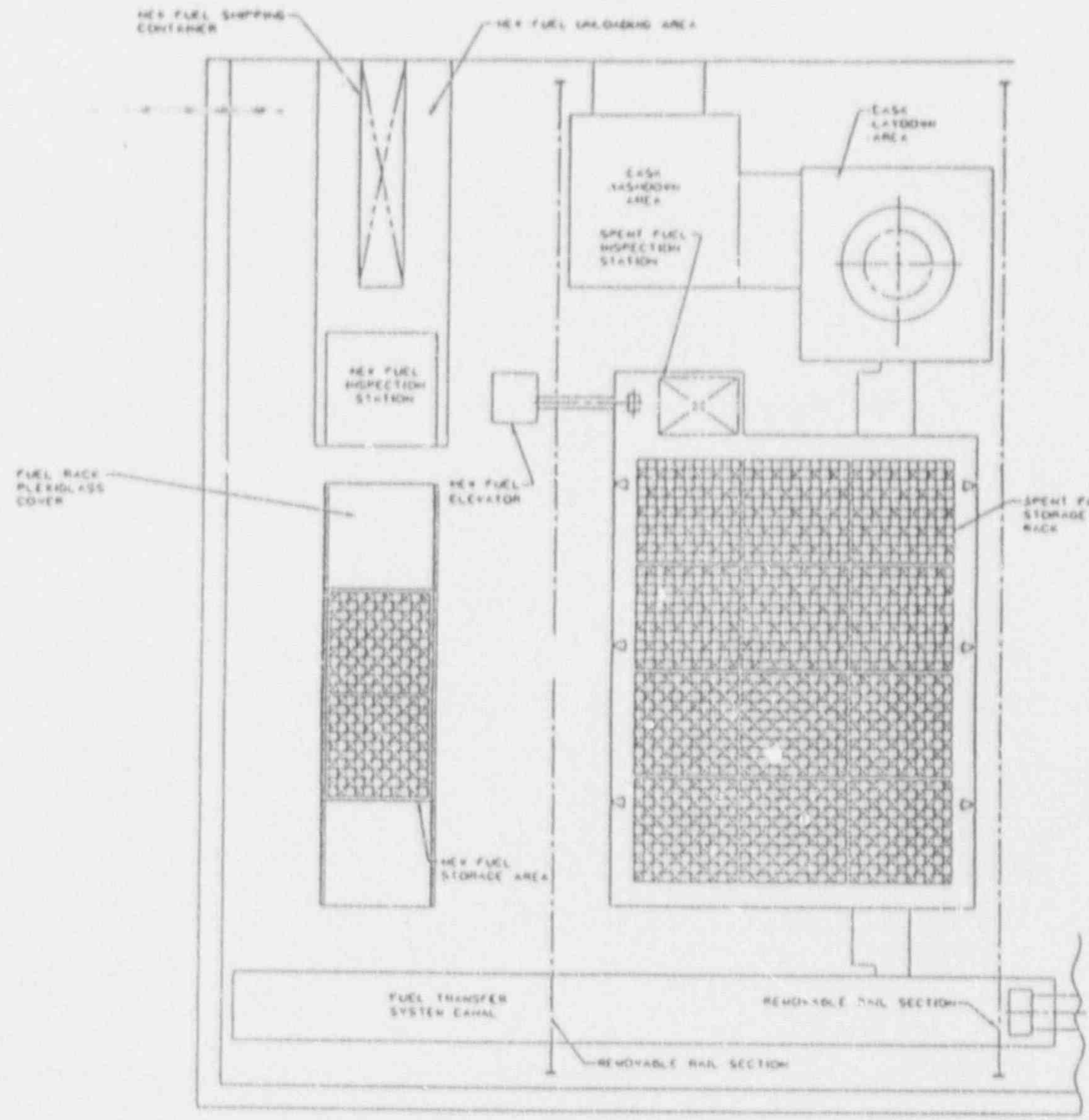

FIGURE 1.9.1-2
TYPICAL SPENT FUEL POOL ARRANGEMENT

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TABLE 1.9.1-1

SPENT FUEL STORAGE
Inspection, Tests, Analysis and Acceptance Criteria

Certified Design Commitment	Inspection, Test, Analysis	Acceptance Criteria
1. The rack arrangement provides storage locations for the licensed spent fuel storage capacity.	1. Visual inspection of the rack verifies rack size and capacity.	1. Visual inspection data report provides verification.
2. The rack is constructed to ASME Code NF.	2. Examine certification data reports.	2. Certification data reports show compliance with ASME Code, Subsection NF.
3. Stress limits are met for SSE loads and drop accidents.	3. Examine the design report.	3. Design report provides actual stress and shows results within allowable limits.
4. Keff less than .95.	4. Examine the criticality analysis and measure the pitch between cells and separation between modules.	4. Criticality evaluation shows Keff < .95. Pitch and separation within drawing limits.

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1.9.2.2 COMPONENT COOLING WATER SYSTEM

Design Description

The Component Cooling Water System (CCWS) is a closed loop cooling water system that is designed to remove heat from the plant's safety related and non-safety related components and heat exchangers as required during normal operation, shutdown cooling, refueling, and design basis accident conditions. The system, in conjunction with the Station Service Water System (SSWS) and the Ultimate Heat Sink (UHS), is capable of removing heat from the essential components and heat exchangers to ensure a safe reactor shutdown and cooling following postulated accidents.

The Component Cooling Water System is an intermediate cooling water system between the Reactor Coolant System (RCS) and the Station Service Water System. The CCWS provides protection against station service water leakage into the Reactor Coolant System. The Component Cooling Water System also provides a barrier to the release of radiological contamination into the environment via the Ultimate Heat Sink.

The CCWS has two 100% capacity divisions (see Figure 1.9.2.2). Each division is connected to its corresponding SSWS division through the component cooling water heat exchangers. Each division has 100% heat dissipation capacity to obtain safe cold shutdown.

Each division of the CCWS includes two component cooling water heat exchangers, a component cooling water surge tank, two component cooling water pumps, piping, valves, controls, and instrumentation. There are no cross connections between the two divisions. A single failure of any component in the CCWS will not impair the ability of the CCWS to meet its functional requirements.

The temperature of the component cooling water leaving each component cooling water heat exchanger is regulated by a component cooling water bypass control valve.

The component cooling water pumps have the capability to supply the plant components and heat exchangers during normal unit operation, during unit cooldown, during refueling, and during emergency situations as required. Inherent system logic is provided to ensure that flow requirements are met and that a minimum flow path is provided as necessary.

The component cooling water surge tanks ensure that required NPSH is provided for the component cooling water pumps. Each tank allows for expansion and contraction of fluid in the system due to temperature changes and provides a means to monitor fluid leakage into and out of the system. Fluid losses are accommodated by the surge tank volumes. System venting and filling are accomplished by the surge tanks. The tanks are also provided with an adequately sized overflow line to protect against overpressurization. In order to maintain surge tank volume in the event of a break in a system cooling loop composed of non-nuclear safety class piping, level indications and controls are provided to isolate these portions of the system in the event of abnormally low surge tank level.

System water chemistry is controlled for the prevention of long term corrosion. The capability is provided to sample the water, and if required, the pH can be adjusted by the addition of chemicals. Organic fouling and inorganic buildups are controlled by proper water treatment. Radiation monitors and system sampling are provided to detect radioactive contamination so that the contaminated water can be processed as liquid waste.

Makeup water to the CCWS is normally supplied by the Demineralized Water Makeup System (DWMS). If the DWMS is unavailable, such as during an accident a safety related backup makeup line of Seismic Category I construction is provided from the Station Service Water System. A removable spool piece is placed on this line to prevent the inadvertent addition of station service water.

Instrumentation and controls are provided to adequately monitor and control the CCWS. All non-safety related instrumentation and controls are designed such that any failure will not cause degradation of any essential equipment function. The CCWS instrumentation facilitates automatic operation, remote control, and continuous indication of the system parameters locally and in the control room. Control room process indications and alarms are provided to enable the operator to evaluate the CCWS performance and to detect malfunctions.

Each division of the CCWS consists of essential and non-essential cooling loops. The essential cooling loop piping and components are designed in accordance with Safety Class 3 requirements. Containment isolation valves and containment penetration piping are designed in accordance with Class 2 requirements. All pneumatic valves fail to pre-determined safe positions upon the loss of instrument air. All Major CCWS pumps, heat exchangers and surge tanks are designed as a minimum to meet ASME III requirements.

The essential portions of the CCWS are designed as Seismic Category I and are contained in Seismic Category I structures. All essential CCWS components are protected from floods, tornado missile damage, internal missiles, pipe breaks and whip, jet impingement, and interaction with other non-seismic systems in the vicinity. Also, failure of the non-essential portions within the CCWS itself does not cause flow degradation to safety related components.

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Each division of the CCWS receives power from its associated Class 1E Auxiliary Power System. In the event of a loss of offsite power, the system receives power from the emergency diesel generators.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 1.9.2.2 provides the inspections, tests, and/or analyses and associated acceptance criteria.

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TABLE 1.9.2.2

COMPONENT COOLING WATER SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The simplified system configuration is shown in Figure 1.9.2.2.	1. Inspections of installation records together with plant walkdowns will be conducted.	1. The system configuration is in accordance with Figure 1.9.2.2.
2. Pneumatic control valves fail to their failed safe positions upon the loss of instrument air.	2. System testing will be conducted to simulate loss of instrument air conditions to verify the response of valves having an instrument air supply.	2. The affected valves respond to a loss of instrument air as designed.
3. Cooling loops composed of non-nuclear safety class component cooling water piping are isolated on an abnormally low surge tank level.	3. System tests will be conducted to simulate an abnormally low surge tank condition to verify the response of valves used to isolate these portions of the system.	3. The affected valves isolate the cooling loops composed of non-nuclear safety class piping in response to an abnormally low surge tank level.
4. The CCWS is capable of accomplishing its nuclear safety functions with Class 1E power from the plant's normal and emergency power sources.	4. System tests will be conducted after installation to confirm that the electrical power supply configurations are in compliance with design.	4. The CCWS is powered with Class 1E power from the plant's normal and emergency power sources.

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TABLE 1.9.2.2 (Continued)

COMPONENT COOLING WATER SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Containment isolation valves and penetration piping meet the specified ASME Code Class 2 requirements.	5. Review of plant records are made to verify compliance.	5. Containment isolation and in-containment piping meets ASME Code Class 2 requirements.
6. The essential portions of the CCWS are designed to Seismic Category I requirements.	6. Evaluation of design characteristics and construction records will be performed to evaluate conformance to design requirements.	6. The essential components of the CCWS meet Seismic Category I requirements.
7. Essential CCWS components are protected from malfunctions caused by floods, tornados, internal missiles, pipe breaks and whip, jet impingement, and interactions with non-seismic systems.	7. Evaluation of the CCWS components will be performed against design requirements.	7. The essential components are protected from the identified hazards.

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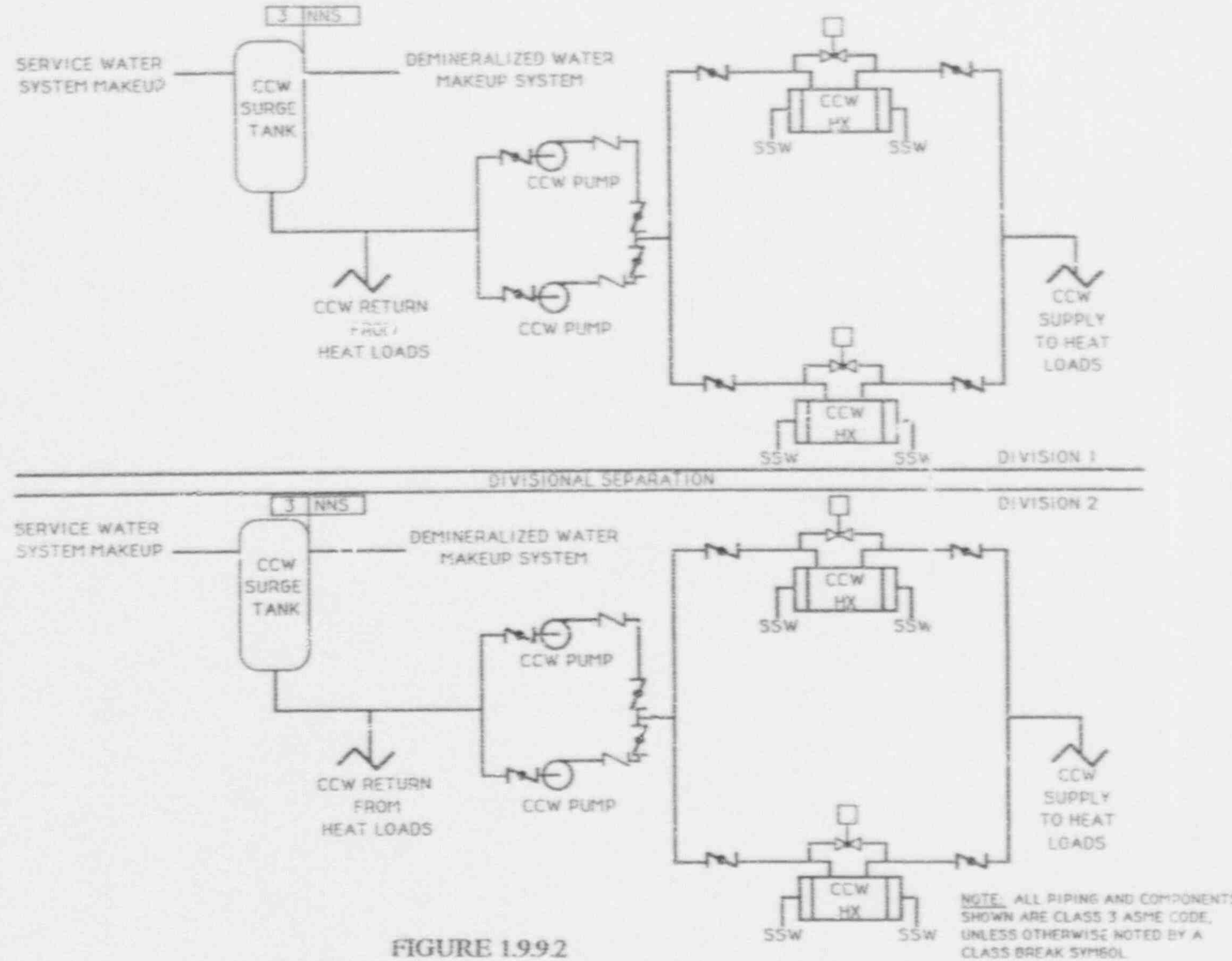

FIGURE 1.9.9.2
COMPONENT COOLING WATER SYSTEM

1.9.6 COMPRESSED AIR SYSTEMS

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Design Description

The Compressed Air Systems are non-safety related systems consisting of the Instrument Air System (IAS), the Station Air System (SAS), and the Breathing Air System (BAS). The Instrument Air System supplies compressed air to air operated instrumentation, controls, and valves. The Station Air System supplies compressed air for air operated tools, miscellaneous equipment, and various maintenance purposes. The Breathing Air System supplies compressed air to various locations in the plant, as required, for breathing protection against airborne contamination for personnel performing certain maintenance and cleaning operations.

The Compressed Air System is not required to achieve a safe reactor shutdown or to mitigate the consequences of an accident. Loss of instrument air due to a failure of the instrument air system during an accident, loss of offsite power, or station blackout (i.e. complete loss of all AC power) will cause all of the pneumatically operated safety-related components to fail to their predetermined safe position. Therefore, failure of the IAS will not prevent any safety-related component or system from performing its intended safety functions.

Each of the four instrument air supply trains is composed of an air intake filter/silencer, an air compressor with intercooler(s), an air receiver, a dryer/filter train, and associated piping and valves as shown in Figure 1.9.6-1. The IAS is capable of supplying instrument air quality compressed air.

The SAS is composed of two parallel, 100% capacity trains of equipment. Each station air supply train consists of an intake filter/silencer, a compressor, an air receiver, a dryer/filter, and associated piping and valves as shown in Figure 1.9.6-2.

The BAS is composed of two parallel, 100% capacity trains of equipment as shown in Figure 1.9.6-3. Each breathing air supply train consists of an intake filter/silencer, a breathing air compressor, an air receiver, a breathing air purifier, and associated piping and valves.

The Compressed Air Systems are Class NNS (Non-Nuclear Safety) with the exception of the containment isolation valves and associated piping which are Safety Class 2 (ASME Code Class 2) and Seismic Category I.

The normal power source for the Compressed Air Systems is the non-class 1E AC Power Source. The nuclear safety-related electric valve actuators on the Compressed Air System containment isolation valves receive power from the Class 1E Alternate AC Source Standby Power Supply to accommodate a loss of offsite power.

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Inspections, Tests, Analyses, and Acceptance Criteria

Table 1.9.6 provides the inspections, tests, and/or analyses and their with associated acceptance criteria.

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TABLE 1.9.6

COMPRESSED AIR SYSTEMS
Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The simplified system configuration for the IAS is as shown in Figure 1.9.6-1; for the SAS in Figure 1.9.6-2; for the BAS in Figure 1.9.6-3.	1. Inspections of installation records together with plant walkdowns will be conducted to confirm that the installed equipment is in compliance with the design configurations.	1. The system configurations are in accordance with Figures 1.9.6-1, 1.9.6-2, and 1.9.6-3.
2. The IAS can be powered from the Non-Class 1E Alternate AC Source Standby Power Supply.	2. System tests will be conducted after installation to confirm that the electric power supply configurations are in compliance with the design.	2. The installed equipment can be powered from the Non-Class 1E Alternate AC Source Standby Power Supply.
3. Containment isolation valves and associated piping meet ASME Code Class 2 requirements.	3. Review plant records to verify compliance.	3. The containment isolation values and in-containment piping associated with the Compressed Gas System conforms to ASME Code Class 2 requirements.

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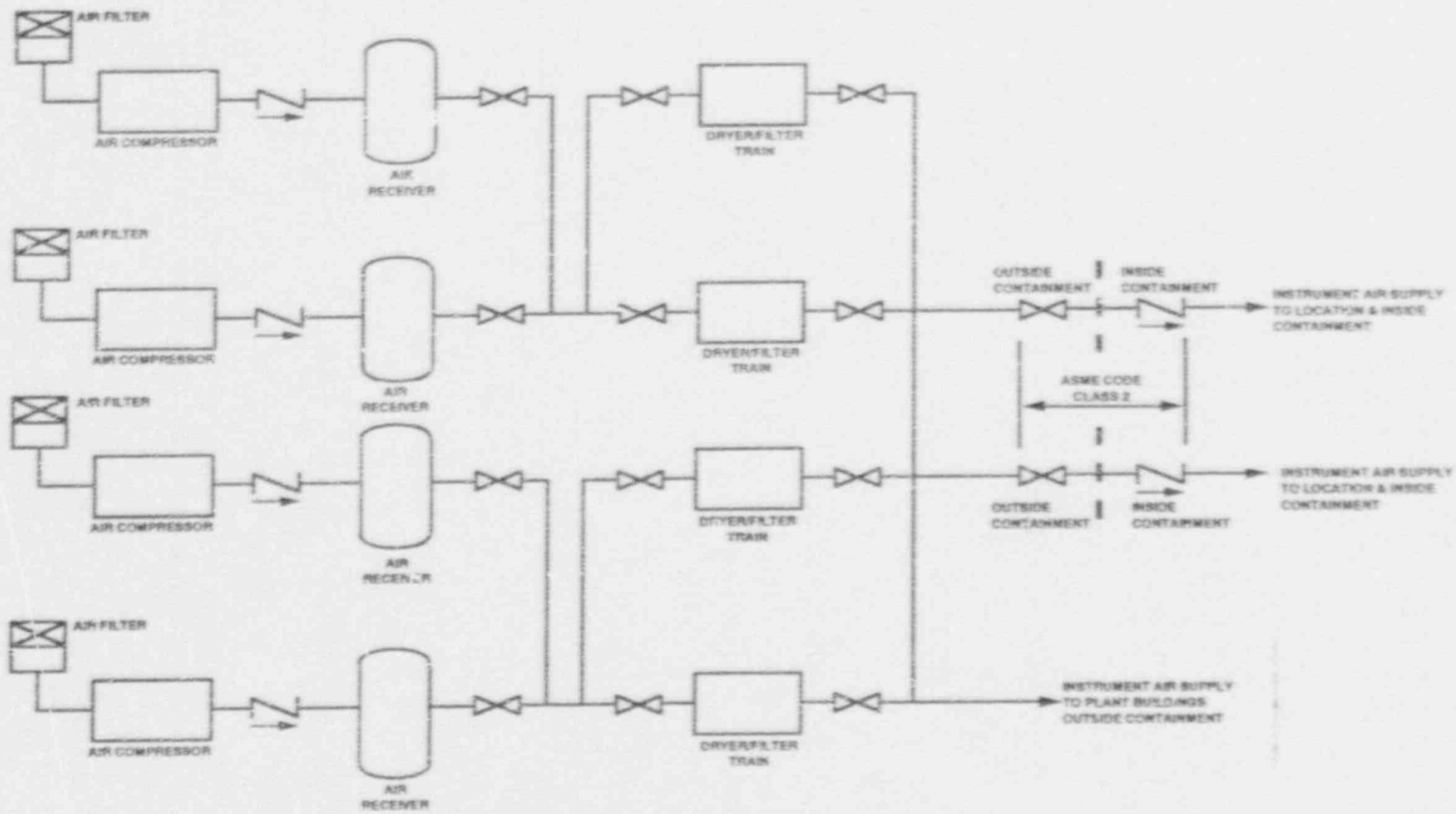

FIGURE 1.9.6-1
INSTRUMENT AIR SYSTEM

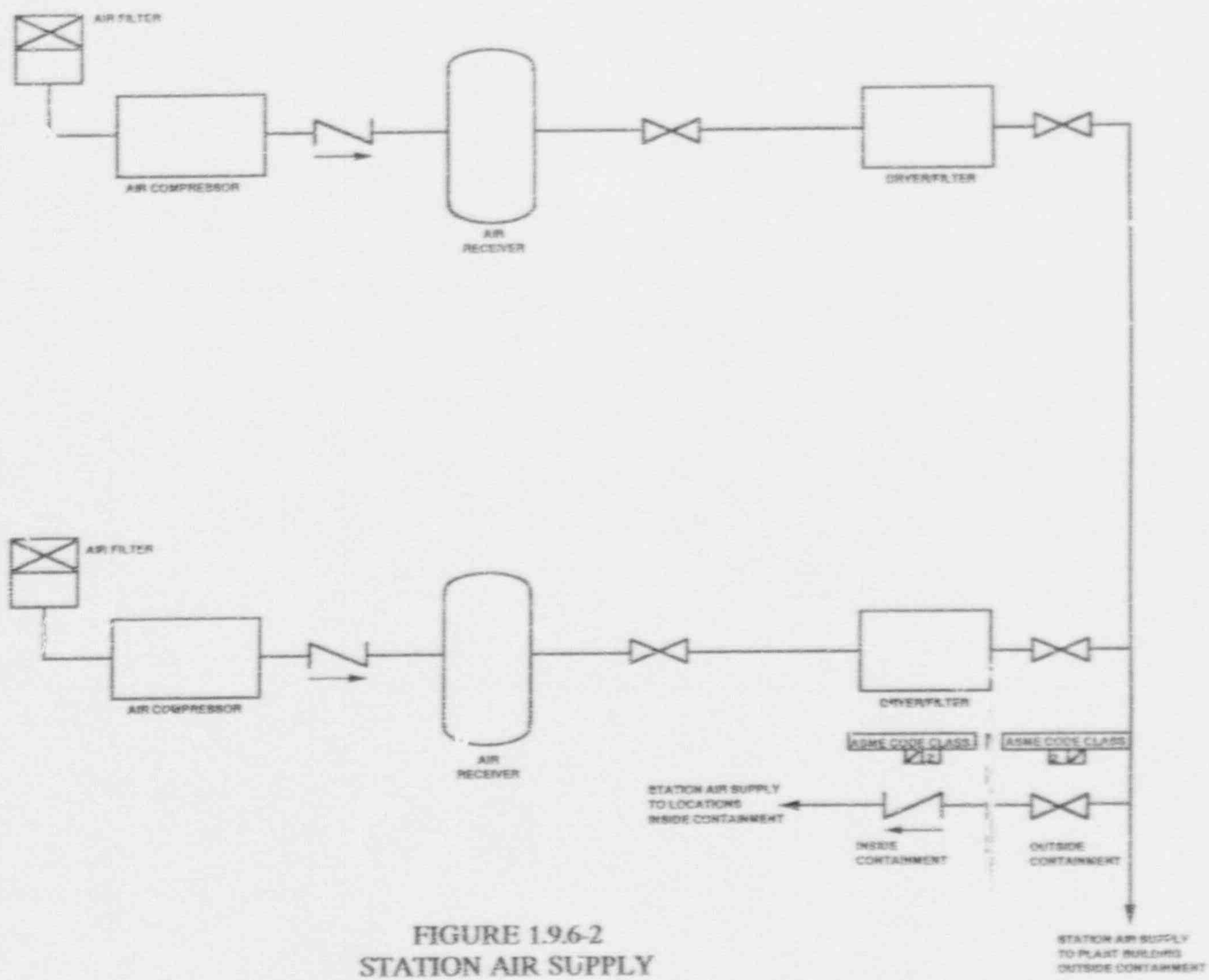

FIGURE 1.9.6-2
STATION AIR SUPPLY

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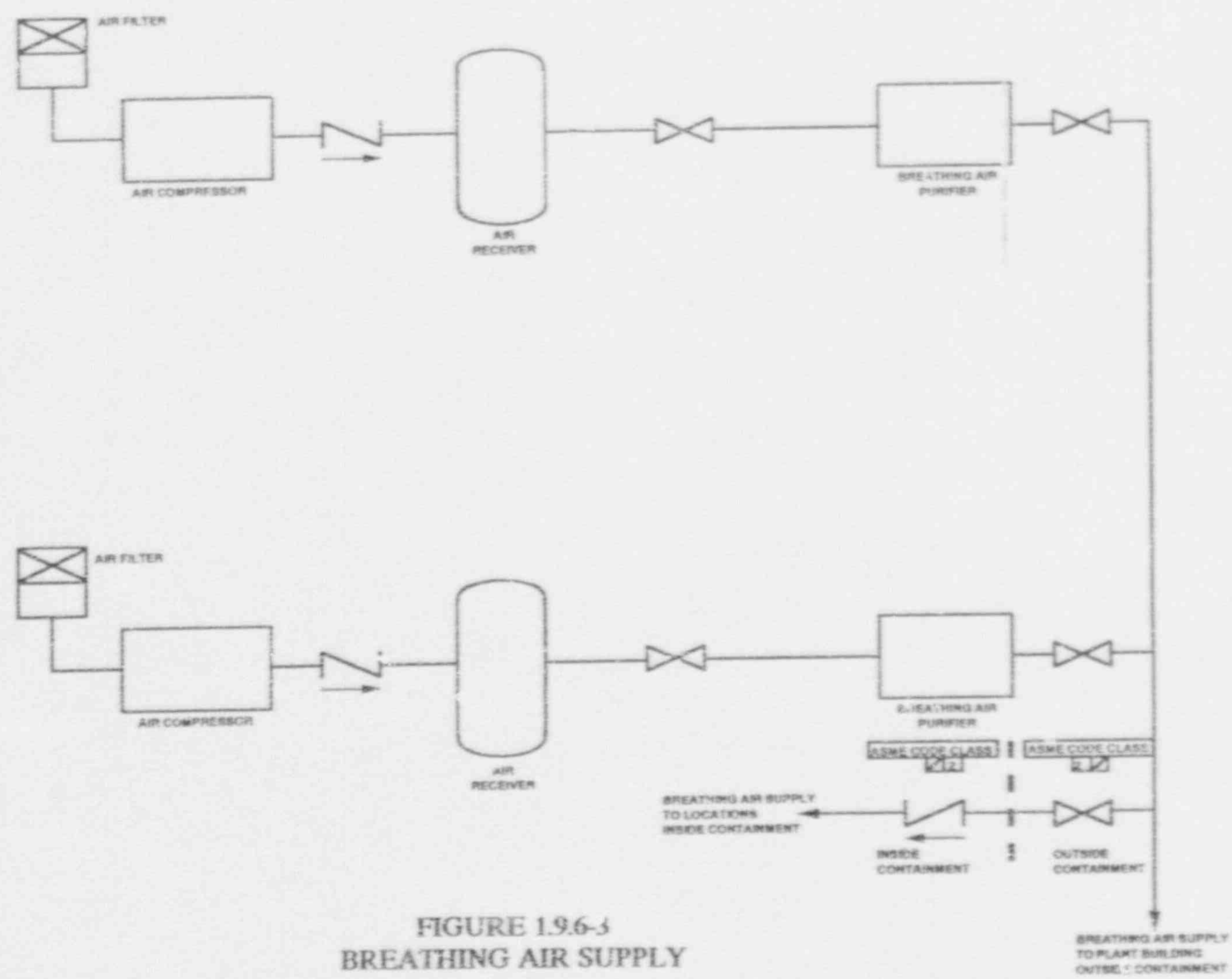

FIGURE 1.9.6-3
BREATHING AIR SUPPLY

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1.9.22.9 STATION SERVICE WATER SYSTEM (SSWS) PUMP STRUCTURE

Design Description

The SSWS Pump Structure interfaces with the station service water system pumps and its design is a site specific requirement of the applicant. The structure includes the SSWS pump intake structure and foundations and is located in the same vital protection area as the main plant and outside the corridor designated as the potential turbine missile path.

The pump structure design meets Seismic Category I requirements. The design provides physical barriers that maintain divisional separation of SSWS components and will withstand the effects of:

a. natural phenomenon, including a safe shutdown earthquake, floods, tornadoes, and hurricanes

b. externally and internally generated missiles

c. fire hazards

The design includes a safety grade screen system located before the SSWS pump inlets. The design provides the capability for periodic cleaning and means to limit ingestion of biofouling organic materials and debris, consistent with the fouling limits of the piping and Component Cooling Water System (CCWS) designs. Pump inlet blockage limits are accommodated in the design.

The SSWS pump well is designed to prevent the formation of air vortices for the complete range of operating water levels in the pump well.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 1.9.22.9 provides a definition of the inspections, tests, and/or analyses and associated acceptance criteria.

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TABLE 1.9.22.9

STATION SERVICE WATER SYSTEM (SSWS) PUMP STRUCTURE
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Test, Analysis	Acceptance Criteria
1. The SSWS pump structure is designed to Seismic Category 1 requirements.	1. The design is evaluated for conformance to Seismic Category I requirements.	1. Seismic Category I requirements are met for the design.
2. The design provides physical barriers to maintain divisional separation of SSWS components.	2. Inspection of the physical layout will be made to evaluate the capability for divisional separation.	2. Divisional separation is provided between the SSWS.
3. The design provides the capability to withstand: 1) Natural phenomena a. SSE b. Floods c. Tornadoes d. Hurricanes 2) Externally and internally generated missiles. 3) Fire Hazards	3. An analysis of the design characteristics is made to verify the capability to withstand the specified conditions.	3. The design is shown to be capable to withstand: 1) Natural phenomena 2) Missiles 3) Fire Hazards

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TABLE 1.9.22.9 (Continued)

STATION SERVICE WATER SYSTEM (SSWS) PUMP STRUCTURE
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Test, Analysis	Acceptance Criteria
4. The SSWS pump structure is located in the same vital protection areas as the main plant.	4. The site layout is inspected to verify this requirement.	4. The SSWS pump structure is within the same vital protected area as the plant.
5. The SSWS pump structure is located outside the turbine missile path.	5. The site layout is inspected to verify this requirement.	5. The SSWS is located outside the projected paths for turbine generated missiles.
6. The SSWS pump structure provides a safety grade screen system prior to the pump inlets.	6. An inspection of the design is made to verify the inclusion of a safety grade screen system located prior to the pump inlets.	6. The design includes a safety grade screen system in a position before the pump inlets.
7. The SSWS pump well is designed to prevent air vortices over the complete range of operational water levels in the pump well.	7. An analysis of the design if made to verify that air vortices will not occur for all operational water levels in the pump well.	7. The design prevents air vortices in the pump well for all operational states.

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1.11.1 LIQUID WASTE MANAGEMENT SYSTEM

Design Description

The Liquid Waste Management System (LWMS) provides the capability to collect, segregate, store, process, sample, and monitor radioactive liquid waste. The LWMS is a non-nuclear safety (NNS) system containing no safety class components except for containment isolation valves and penetrations which are designed to Safety Class 2 requirements.

The liquid waste is segregated into the following categories:

a. Equipment drain waste or clean waste -- degassed reactor grade radioactive liquid waste

b. Floor drain waste or dirty waste -- non-reactor grade radioactive liquid waste

c. Detergent waste -- laundry and hot showers

d. Chemical waste -- non-detergent liquid waste (e.g., decontamination fluids)

Each category of waste is processed by an independent subsystem.

The equipment and floor drain waste subsystems are designed with the provision for filtration, decontamination by demineralizers, batch sampling, and recirculation capability for further processing.

The floor drain waste subsystem is designed with the additional capability for oil/crud removal, flocculent addition to collection tanks, and pH adjustment.

The chemical waste subsystem is designed with the capability for pH adjustment through chemical addition, filtration, batch sampling, and recirculation to floor drain waste subsystem for further processing.

The detergent waste subsystem is designed with the capability for filtration, decontamination by demineralizers, batch sampling, and recirculation to floor drain subsystem for further processing.

This system is designed with collection and storage capacity to process the maximum expected liquid waste volumes. These liquid waste volumes are calculated, based on anticipated peak daily inputs, utilizing vendor, plant specific, and industry data Each large volume subsystem is provided with one or more parallel collection tanks and

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waste monitor or sample tanks. Each small volume subsystem has one collection tank.

The system is designed to process radioactive liquid waste so that the concentration of the liquid effluents at the potable water source is within limits specified by regulatory directives. Criteria are met by the provision of design processing capabilities and adequate dilution flow at the plant discharge.

This system is designed so that releases of radioactive materials to the environment will be controlled and monitored in accordance with 10 CFR 50, Appendix A (General Design Criteria 60, 61, and 64). The system is designed so that release of processed liquid waste will require an operator action. This system is designed with the capability to batch sample and monitor processed liquid waste prior to release to the environment. The radiation monitor, located upstream of the plant discharge, is designed to terminate the release if a pre-set limit is exceeded.

The LWMS is housed in a Radwaste Facility building.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 1.11.1 provides the inspections, tests and/or analyses and their associated acceptance criteria.

TABLE 1.11.1

LIQUID WASTE MANAGEMENT SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The LWMS is designed with sufficient collection and storage capacity.	1. Vendor, plant specific, and industry information will be reviewed to verify: a) Provision for at least one collection, waste monitor, and sample tank per subsystem. b) Sizing of collection, waste monitor and sample tanks. c) Provision for emergency storage capacity by the Steam Generator Drain Tank. d) Provision of level indication for collection, waste monitor, and sample tanks.	1. The LWMS provides storage capacity to accommodate the maximum expected radioactive liquid waste input during normal and anticipated occurrences.

TABLE 1.11.1 (Continued)

LIQUID WASTE MANAGEMENT SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. The LWMS is designed with sufficient processing capabilities to ensure the concentration of the liquid effluents at the potable water source is within design limits.	2. Vendor records, installation information, and performance test results will be reviewed to verify that assumptions made during the analysis are conservative with respect to: a) Process decontamination capability provided. b) Dilution flow available at plant discharge. c) Recirculation capability provided.	2. The LWMS designs conforms to 10 CFR 20, Appendix B, Table II, Column 2 and 10 CFR 50, Appendix I limits.
3. The LWMS is designed to provide for a controlled monitored release. This will be ensured through the capability to sample and monitor each batch prior to release to the environment.	3. Inspection of installation records together with plant walkdowns will be conducted to confirm that batch sampling and radiation monitoring capabilities are provided in the system design upstream of the plant discharge.	3. The LWMS design includes provision for controlled monitored release in accordance with 10 CFR 50, Appendix A (General Design Criteria 60, 61, and 64).

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TABLE 1.11.1 (Continued)

LIQUID WASTE MANAGEMENT SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		a) Sampling capability is provided for each collection tank prior to processing and for each waste monitor tank and sample tank prior to batch release. b) Radiation monitoring is provided upstream of plant discharge.
4. Interfacing systems are designed with the capability to segregate radioactive liquid waste prior to collection and processing in the LWMS.	4. Inspection of installation records together with walk-downs will be conducted to confirm that the interfacing system design configurations provide for segregation of the radioactive liquid waste streams.	4. Radioactive liquid waste streams are segregated prior to collection and processing by independent subsystems in the LWMS.

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1.11.2 GASEOUS WASTE MANAGEMENT SYSTEM

Design Description

The Gaseous Waste Management System (GWMS) provides the capability to collect, store, process, sample, and monitor radioactive gaseous waste. The GWMS is a non-nuclear safety (NNS) system containing no safety class components except for containment isolation valves and penetrations which shall be designed Safety Class 2.

The GWMS is a gas delay system. The GWMS is designed to operate continuously, as well as periodically, at flow rates established by systems feeding the GWMS, such as the Chemical and Volume Control System.

This system is designed to include conditioning equipment, such as a cooler-condenser for humidity control, charcoal guard bed to protect the charcoal adsorbers from excessive moisture or contamination, and charcoal adsorbers for delay of noble gases.

This system is designed to include dual hydrogen analyzers, which monitor the concentration of hydrogen and oxygen in the GWMS, and nitrogen purge capability to maintain the concentration of hydrogen and oxygen less than 4% in accordance with 10 CFR 50, Appendix A (General Design Criterion 3).

The system is designed to process radioactive gaseous waste so that the concentration of the gaseous effluents at the exclusion area boundary is within limits specified by regulatory directives. Criteria are met by the provision of sufficient processing capability and adequate dispersion of the effluent released from the unit vent.

This system is designed to ensure releases of radioactive materials to the environment can be controlled and monitored in accordance with 10 CFR 50, Appendix A (General Design Criteria 60, 61, and 64). This system is designed with the capability to continuously monitor processed gaseous waste prior to release to the environment. The radiation monitor, is designed to automatically isolate the GWMS discharge if a pre-set limit is exceeded. This system is designed so that leakage rates of processing equipment ensure releases of radioactive gases from the GWMS to the environment is controlled.

The GWMS is located in the Radwaste Facility.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 1.11.2 provides the inspections, tests and/or analyses and their associated acceptance criteria.

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TABLE 1.11.2

GASEOUS WASTE MANAGEMENT SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The GWMS is designed with processing capabilities to ensure the concentration of the gaseous effluents at the exclusion area boundary is within design limits.	1. Vendor records, site specific information, industry data, and preoperational tests will be reviewed to verify that the bases of the design analysis are conservative with respect to:	1. Conformance to 10 CFR 20, Appendix B, Table II, Column 1 and 10 CFR 50, Appendix I limits.
	a) Delay time for each isotope calculated based on:	Delay time for xenon and krypton are at least 60 days and 3 days, respectively.
	i) Carrier gas flow rate	
	ii) Mass of charcoal in absorber.	
	iii) Absorbtivity of charcoal for each isotope.	
	b) Dispersion of effluents at plant unit vent.	

TABLE 1.11.2 (Continued)

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GASEOUS WASTE MANAGEMENT SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. The GWMS is designed to limit a buildup of hydrogen and oxygen to explosive limits.	2. Vendor records, pre-operational tests, and inspections of installation records together with plant walkdowns will be reviewed.	2. The GWMS conforms to 10 CFR 50, Appendix A (General Design Criterion 3) requirements for hydrogen control. Hydrogen and oxygen concentrations will be maintained less than 4%.
3. The GWMS has provisions for controlled monitored releases.	3. Inspection of installation records together with plant walkdowns will be conducted to confirm radiation monitoring capabilities are provided. Inspection of vendor records and leak tests will be conducted to confirm installed equipment is leak-tight.	3. The GWMS conforms to 10 CFR 50, Appendix A (General Design Criteria 60, 61, and 64). Radiation monitoring is provided upstream of plant unit vent. Leakage rates from individual components within a zone are within design limits.