ABB

ASEA BROWN BOVERI

April 29, 1994
LD-94-032

Docket No. 52-002

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

Subject: Response to Comments on Design Certification Material

Reference: NRC Letter, T. Boyce (NRC) to C. Brinkman (ABB-CE), dated March 24, 1994

Dear Sirs:

Enclosed are responses to NRC comments on ABB-CE's Design Certification Material transmitted by the referenced letter. The responses reflect comment resolutions reached in our meeting with the NRC staff on April 19 and 20, 1994.

Should you have questions on the enclosed material, please contact me or Mr. George Hess at (203) 285-5218 or Mr. John Rec at (203) 285-2861.

Very truly yours,

COMBUSTION ENGINEERING, INC.

G. D. Hess for

C. B. Brinkman
Director
Nuclear Systems Licensing

GDH/sab
Enclosure: As Stated

cc: T. Boyce (NRC)
T. Wambach (NRC)
M. Franovich (NRC)
P. Lang (DOE)
J. Trotter (EPRI)

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CE 80+ ITAAC Independent Review Comments

ITAAC No. GENERAL

No.	Comments	Cat.	Resolution
1 (2)	CESSAR Fig. 1.7-1, Table 4 should include Note 18 as one of the references because this note describes safety class 4.	1	Agree - Change CESSAR DC Figure 1.7.1, Table 4 to add Note 18. See Markup
2 (4)	The CDM figures do not show all valves designated as "active" in CESSAR Table 3.9-15. Criteria for selecting active valves for inclusion in figures should be stated and applied consistently on all figures.	1	Disagree. See attached response.
3 (5)	CESSAR (Chapter 11) and CDM (2.9.4) use the words "control room" to refer to the MCR or the Radwaste Building control room. Consistent terminology such as "Main Control Room" or "Radwaste Building Control Room" as appropriate should be used.		See markups
4 (6)	Verification of independence between Class 1E channels is not consistently covered in all system CDMs. For example, CDM 2.6.3 and 2.9.4 explicitly state this requirement in the Design Description and require its verification in an ITAAC. The other CDMs do not.	1	Class 1E channel requirements should be deleted from 2.9.4 since this requirement is only applied to I&C Systems

Response to General Comment 2(4)

The active MOVs are shown in the respective DCM figures except as follows:

1) The equipment-specific service water system strainer backwash valves are not shown;
2) Containment isolation valves not shown in a DCM system figure are covered by DCM Section 2.4.5, Containment Isolation System;
3) Valves in the emergency feedwater pump turbine continuous steam drain lines are not shown, because this is a level of detail not included in the DCM; and
4) Valves in the hydrogen recombiner inlet and outlet lines are not shown because the number of valves will be determined by the detailed design and a DCM commitment would overly constrain the design.

Comment 1 (2)

ABBREVIATIONS

SYSTEM ABBREVIATIONS

CCWS — COMPONENT COOLING WATER
CVCS — CHEMICAL AND VOLUME CONTROL
EFS — EMERGENCY FEEDWATER
FWCS — FEEDWATER CONTROL
GWMS — GASEOUS WASTE MANAGEMENT
IWSS — INCONTAINMENT WATER STORAGE
MLWMS — MISC LIQUID WASTE MANAGEMENT
MSS — MAIN STEAM
PCPS — POOL COOLING AND PURIFICATION
PLCS — PRESSURIZER LEVEL CONTROL
PPCS — PRESSURIZER PRESSURE CONTROL
PPS — PLANT PROTECTION
RCGVS — REACTOR COOLANT GAS VENT
RCS — REACTOR COOLANT
RRS — REACTOR REGULATING
SBCS — STEAM BYPASS CONTROL
SCS — SHUTDOWN COOLING
SIS — SAFETY INJECTION
SS — SAMPLING
SWMS — SOLID WASTE MANAGEMENT
APS — ALTERNATE PROTECTION
DPS — DATA PROCESSING
SDS — SAFETY DEPRESSURIZATION
DIAS — DISCRETE INDICATION AND ALARM
COLSS — CORE OPERATING LIMIT SUPERVISOR
CEDMCS — CONTROL ELEMENT DRIVE MECHANISM CONTROL
ALMS — ACOUSTIC LEAK MONITORING
LPMS — LOOSE PARTS MONITORING

SIGNAL ABBREVIATIONS

SIAS — SAFETY INJECTION ACTUATION
CIAS — CONTAINMENT ISOLATION ACTUATION
CSAS — CONTAINMENT SPRAY ACTUATION
EFAS — EMERGENCY FEEDWATER ACTUATION
MSIS — MAIN STEAM ISOLATION

EQUIPMENT ABBREVIATIONS

BABT — BORIC ACID BATCHING TANK (CVCS)
BAC — BORIC ACID CONCENTRATOR (CVCS)
BACIX — BORIC ACID CONDENSATE ION EXCHANGER (CVCS)
BAMP — BORIC ACID MAKEUP PUMP (CVCS)
BAST — BORIC ACID STORAGE TANK (CVCS)
EDT — EQUIPMENT DRAIN TANK (CVCS)
GS — GAS STRIPPER (CVCS)
HT — HOLDUP TANK (CVCS)
HP — HOLDUP PUMP (CVCS)
IRWST — INCONTAINMENT REFUELING WATER STORAGE TANK (SIS)
PHIX — PRE-HOLDUP ION EXCHANGER (CVCS)
PZR — PRESSURIZER (RCS)
RCP — REACTOR COOLANT PUMP (RCS)
RDP — REACTOR DRAIN PUMP (CVCS)
RDT — REACTOR DRAIN TANK (CVCS)
RMWT — REACTOR MAKEUP WATER TANK (CVCS)
VCT — VOLUME CONTROL TANK (CVCS)

DRAINS

D — LOCAL DRAIN
DXXX — DRAIN TO
IDH — ION EXCHANGER DRAIN HEADER (CVCS)
MSH — MAKEUP SUPPLY HEADER (CVCS)
RDH — RECYCLE DRAIN HEADER (CVCS)
RSSH — RESIN SLUICE SUPPLY HEADER (CVCS)

MISCELLANEOUS

A/E — ARCHITECT ENGINEER
A/S — AIR SUPPLY
ATMOS — ATMOSPHERE
BOP — BALANCE OF PLANT
CBO — CONTROLLED BLEEDOFF
LS — LOCAL SAMPLE
MISC — MISCELLANEOUS
N_2 — NITROGEN
H_2 — HYDROGEN
RC — REACTOR COOLANT
V — VENT TO ATMOSPHERE

FIG 1.7.1

CHANNEL NUMBERS (INSTRUMENTS)

SUFFIX (S) (NOTE 9)
FUNCTION (TABLE 1)
PXXX-Y (PI)
CHANNEL NO.
1YY-RCS
1XYY-MSS
2YY-CVCS
3YY-SIS
4YY-PCPS
5YY-SS
6YY-WMS — MLW / GW / SW
8YY-CCS

TABLE 1 (NOTE 1)

INSTRUMENTATION IDENTIFICATION LETTERS		
	FIRST LETTER	SUCCEEDING LETTERS
	MEASURED OR INITIATING VARIABLE	READOUT OR FUNCTION
A	ANALYSIS	ALARMING
B	BORON	
C	CONDUCTIVITY (ELEC)	CONTROL
D	DENSITY OR SPECIFIC GRAVITY	DIFFERENTIAL
E	VOLTAGE (EMF)	ELEMENT
F	FLOW RATE	
G	GAGING (DIMENSIONAL)	GLASS
H	HAND INITIATED (MANUALLY)	
I	CURRENT (ELEC)	INDICATION (NOTE 17)
J	POWER	
K	TIME OR TIME SCHEDULE	
L	LEVEL	LIGHT
M	MOISTURE OR HUMIDITY	
N	VIBRATION	SENSING
O	ROTATION	ORIFICE (RESTRICTION)
P	PRESSURE OR VACUUM	
Q	QUANTITY OR EVENT	INTEGRATE OR TOTALIZER
R	RADIOACTIVITY	RECORDING (NOTE 17)
S	SPEED OR FREQUENCY	SWITCH
T	TEMPERATURE	TRANSMITTING
U	ACOUSTIC	
V	ACCELEROMETER	VALVE
W	WEIGHT	WELL
X	REFRACTOMETER	TEST POINT
Y		CONVERTOR
Z	POSITION	

TABLE 4 (NOTES 8, 14 & 18)

AA-BB-C-DD-E
SAFETY CLASS (1,2,3,4)
SIZE (NOMINAL PIPE) *
MATERIAL CODE
DESIGN TEMPERATURE CODE (°F)
DESIGN PRESSURE CODE (PSIG)

MATERIAL CODE

A-AUSTENITIC STAINLESS STEEL
B-CARBON STEEL
C-INCONEL 690

* TUBING O.D. INDICATED BY $\boxed{DD}$

PRESSURE AND TEMPERATURE CODE		
	1ST LETTER	2ND LETTER
A	0	ATMOSPHERIC
B	1	00
C	2	[illegible]
D	3	10
E	4	15
F	5	20
G	6	25
H	7	30
I	8	35
J	9	40
K	10	45
L	11	50
M	12	55
N	14	60
O	15	65
P	16	70
Q	17	75
R	18	80
S	19	85
T	20	90
U	21	95
V	22	
W	23	
X	24	
Y	25	
Z	27	
a	28	
b	29	
d	30	

INSTRUM

Comment 3(5) (23 sheets)

considered expended for one of these reasons will also be nearing limits for the others as well. Normally, the first bed in the process flow will become expended. When it does, it will be isolated in preparation for transferring the contents to the SWMS, and a new bed will be added at the end of the process flow path. In this way, each bed will first be the final step of a series of processes and will advance to the first position over its life. The number of beds in a particular series will be expected to change with circumstances and is left for the operators to determine.

Each pair of Waste Monitor Tanks will also alternate as the receiver of the process stream. The one that is filling will have the fluid driven mixing started above the low level permissive so that when the tank is full, a representative sample will be immediately available. The details of the effluent release are provided in Section 11.2.6.

11.2.3 SAFETY EVALUATION

The LWMS has no safe shutdown or accident mitigation function. Accidental releases will not exceed the limits of 10 CFR 20. Accidental releases due to a major component failure or LWMS leak, will be contained in the Radwaste Building.

11.2.4 INSPECTION AND TESTING REQUIREMENTS

A program of testing requirements appropriate to assure that the LWMS is operating as intended is developed prior to fuel loading. Emphasis is placed on verifying remote function, and instrumentation important to the design objectives. Testing of the waste process streams for the most effective and economical process is required periodically during normal operation.

11.2.5 INSTRUMENTATION REQUIREMENTS

Instrumentation and indication important to the design basis of the LWMS are as follows:

A. Level Indicators

All Waste Collection and Waste Monitor Tanks are equipped with continuous level indicators. In addition, redundant means of detecting high level are provided along with non-redundant low level indicators. High level is alarmed both locally and in the radwaste Building control room. Levels in the area sumps and tanks which feed the LWMS Collection Tanks are also indicated in the Radwaste Building control room.

Local instrumentation and alarms will be provided in the Condensate Polishing Control Room, e.g., neutralization tank level and high alarm, with a common system trouble alarm in the Main Control Room.

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Local instrumentation and alarms will be provided in the Condensate Polishing Control Room, e.g., neutralization tank level and high alarm, with a common system trouble alarm in the Main Control Room.

B. Radioactive Liquid Effluent Monitor

Prior to release, waste liquid is held in a monitor tank from which a representative sample is taken. Inlet valves on tanks being prepared for release are closed, providing for a batch release. However, all releases are made through an effluent monitor. The effluent monitor set point is adjusted so that it will only alarm on unexpected high activity (relative to batch release sample information). The alarm also automatically terminates the release.

The setpoint for the liquid waste discharge radiation monitor is determined by the COL Applicant and provided in the Offsite Dose Calculation Manual to ensure compliance with 10 CFR 20, Appendix B of Sections 20.1001 through 20.2402 effluent concentrations. The radioactive liquid effluent monitor is located downstream of the last possible point of input of radioactive liquid waste.

(use initial caps) High activity is alarmed in the Radwaste Building control room and in the main control room via the Data Processing System (DPS) and the Discrete Indication and ALARM System (DIAS). X

The COL Applicant will provide the operational setpoint for the termination of the liquid waste management system discharge to the environment in the plant-specific offsite dose calculation manual (ODCM). This setpoint ensures that the ratio of instantaneous concentrations of radionuclides in the liquid effluent in unrestricted areas to the corresponding values given in 10 CFR 20, Appendix B of Section 20.1001-20.2402, Table 2, Column 2 summed over the radionuclides in the liquid effluent does not exceed 10.

Prior to release, the regenerant waste water is held in one of the neutralization tanks from which a representative sample is taken. Releases from the neutralization tanks will be batch releases. All releases are made through a process radiation monitor. Upon detection of a radiation signal above the radiation monitor setpoint, the release is terminated automatically. The contents of the neutralization tank would then be sampled by the operator and the flow manually diverted to the Floor Drain Tank in the low level waste subsystem of the LWMS for further processing.

C. Differential Pressure

Both Filter and Media Bed Process Vessels are equipped with differential pressure measurement instrumentation to monitor the loading of the filter or bed media. Differential pressure indication is provided in the Radwaste Building control room.

D. Flow

Each Process Pump is equipped with flow measurement to assist the operators in regulating the process within the appropriate operating range. Flow rate information, in conjunction with differential pressure information, is also important for the operator to assess filter media condition. Flow indication is provided in the Radwaste control room.

X

Building

E. Area Radiation

Area radiation monitors are discussed in Section 11.5.1.2.5. Area monitors will have local visual and audible alarms.

11.2.6 ESTIMATED LIQUID RELEASES

The estimated quantity of radioactivity released in liquid effluents during normal operation, including operational occurrences, is shown in Table 11.2-1.

The methodology of NUREG-0017 (Reference 1) is used in determining liquid radioactive releases. The sources, estimated volumes, and activity levels of LWMS waste input streams as well as other NUREG-0017 model input parameters and assumptions are summarized in Tables 11.2-2 and 11.2-3. A simplified liquid pathway release assessment process model is provided in Figure 11.2-2.

The GWMS is designed to preclude the buildup of an explosive mixture of hydrogen and oxygen in accordance with the Standard Review Plan, Section 11.3. The charcoal vessels, condenser cooler, piping, analyzer pressure boundary and valves within the GWMS will be designed to withstand a hydrogen explosion (i.e., twenty times normal operating pressure) in accordance with ANSI Standard 55.4. One hydrogen and one oxygen gas analyzer is utilized to monitor H_2 and O_2 gas concentrations in the GWMS. Alarms are provided locally in the Nuclear Annex and in the Main Control Room to alarm on high oxygen concentration.

(use initial capitalize) TB

11.3.1.2 Codes and Standards

The GWMS is designed in accordance with the guidance of Regulatory Guide 1.143 from applicable regulatory positions (C.2, C.4, C.5 and C.6). These include:

A. The GWMS is designed and tested in accordance with regulatory position C.2 of Regulatory Guide 1.143.

   1. The GWMS is designed and tested to the codes and standards listed in Table 1 supplemented by regulatory positions 2.1.2 and 4.

   2. Materials used for pressure retaining portions of the GWMS are designed in accordance with requirements specified in Section II of the ASME Boiler and Pressure Vessel Code. Materials used in the GWMS are compatible with the chemical, physical, and radioactive environment during normal and anticipated operating conditions. Malleable, wrought, or cast iron and plastics are not used in the GWMS.

      The GWMS is designed to preclude the buildup of an explosive mixture of hydrogen and oxygen. Gas analyzers are provided to monitor the concentration of hydrogen and oxygen in the GWMS. Alarms are provided locally in the Nuclear Annex and in the main control room to high alarm on 1% oxygen concentration.

      use Initial caps

   3. The Nuclear Annex houses the charcoal adsorber beds, which delay the release of radioactive gaseous waste from GWMS. The foundations and walls of structures housing the GWMS are designed to meet the requirements specified in regulatory position C.5. The Nuclear Annex is designed as a seismic Category I building and is designed to withstand a plant Safe Shutdown Earthquake (SSE).

B. The GWMS is designed and tested in accordance with regulatory position C.4 of the Regulatory Guide 1.143.

   1. The GWMS is housed in the Nuclear Annex. The GWMS is designed to control leakage. In addition, sufficient space is provided to facilitate access, operation, inspection, testing, and maintenance to maintain personnel exposures ALARA in accordance with Regulatory Guide 8.8 guidelines.

   2. A quality assurance (QA) program will be applied with the provisions as specified in regulatory position C.6 of Regulatory Guide 1.143.

The charcoal guard bed is provided upstream of the charcoal process adsorber beds. The guard bed is normally not in service but provides additional capacity in the event of excess moisture. Nitrogen purge is available to dry charcoal beds in the event of excessive moisture contamination. A guard bed and six charcoal absorbers containing a total of 15,300 pounds of charcoal are employed for xenon and krypton delay. The tanks are located in a shielded vault.

The GWMS operates at pressures slightly above atmospheric thus limiting the potential for oxygen in-leakage. Leakage from the GWMS is further limited through the use of welded connections wherever not restricted due to maintenance requirements. All control valves are provided with bellows seals to minimize leakage through the valve topworks.

The total charcoal mass requirements for the adsorber beds is determined using the following equation:

$$M = 8.98E4\ FT_i/K_i$$

Where:
- M = Charcoal mass, lbs
- F = Flow rate of carrier gas, SCFM
- T_i = Average delay time for the i-th isotope, days
- K_i = Dynamic adsorption coefficient for the i-th isotope, cc/gm
- 8.98E4 = Conversion factor (lbs-cc-min)/(gm-ft^3-day)

For the GWMS these parameters are as follows:

- M = 15,300 lbs.
- F = 1.0 SCFM
- K(Kr) = 18 cc/gm at 104°F
- K(Xe) = 178 cc/gm at 104°F
- T(Kr) = 3 days
- T(Xe) = 30 days

Where the potential for explosive mixtures of hydrogen and oxygen exists, the GWMS is designed to maintain system integrity by first, preventing the formation or buildup of explosive mixtures and secondly, monitoring and purging any concentrations above 1% oxygen in the atmosphere.

Gas analyzers are used to detect the formation of gas mixtures. The system is designed to alarm both locally and in the main control room for remedial action. The gas analyzers take continuous samples from the GWMS and in addition, from sources to the system, i.e., the gas stripper, volume control tank, equipment drain tank, and reactor drain tank.

use Initial Caps

Earthquake is not in the design basis for the System 80+™. Although the likelihood of the OBE is greater than the SSE, the loads associated with the SSE are higher and govern the design of the plant. The structural design of the Radwaste Building meets Regulatory Guide 1.143 requirements.

D. The Quality Assurance (QA) program for the design, installation, procurement, and fabrication of SWMS components complies with regulatory position C.6 of Regulatory Guide 1.143. Table 3.2-1 of the CESSAR-DC identifies seismic category, quality and safety class for each of the respective components in the SWMS in accordance with a constructors QA program in compliance with federal regulations. The COL Applicant will develop a construction and operations QA program.

11.4.1.3 Features

The following features assist in meeting the design criteria.

A. The system has provisions to accommodate leased equipment which may provide the most economical choice at particular times or for particular waste.

B. Normal system operations are remotely controlled from a the ~~centralized control panel within~~ the Radwaste Building control room, which permits operators to most effectively coordinate activities.

C. Active and replaceable components have crane or monorail hoist access to facilitate removal and repair.

11.4.2 SYSTEM DESCRIPTION

11.4.2.1 General Description

Primary functions of the SWMS include providing means by which spent resin, filters, etc. from the LWMS and primary letdown systems are processed to ensure economical packaging within regulatory guidelines, as well as handling dry, low activity wastes for shipment to a licensed burial facility.

The spent resin transfer system is designed to transfer expended radioactive demineralizer and ion exchanger resins from their vessels to the spent resin tank. The system also provides holdup of the resin and transfer of the resin to the solidification system. The major components of this system are spent resin tanks, spent resin surge tanks, spent resin transfer pumps, and filters. The spent resin transfer system is shown on Figure 11.4-1 (Sheet 1 and 2).

The spent resin tanks provide settling capacity for radioactive bead resins transferred from various demineralizers. Capability is provided for solidification of dewatered resins or sluicing to containers approved for shipping and disposal of dewatered ion exchange resins. Also, connections are provided for use of vendor supplied services such as rapid dewatering or waste drying systems when it is determined that the use of these methods represents a savings over the permanently installed alternatives.

A shielded onsite storage area is provided to allow for interim storage of higher activity packaged wastes. The facility is sized such that it is capable of storing the maximum number of full shipping containers generated in any one year period containing the greatest expected waste generation. The process and storage areas include a dedicated overhead crane with direct access to adjacent truck bays with sufficient overhead clearance to facilitate direct trailer loading of waste packages. Crane operation may be performed remotely with the aid of crane-mounted video cameras or locally to provide additional flexibility.

Building space is also provided to sort miscellaneous contaminated dry solids from uncontaminated solids for appropriate and cost effective packaging and disposal. Miscellaneous solid waste consisting of contaminated or potentially contaminated rags, paper, clothing, glass, and other small items is received by the Solid Radwaste System when it arrives at the low-level handling and packaging area. Although waste forms are segregated and bagged at generation points throughout the plant, this area provides space where the waste is further segregated (e.g., compactible versus non-compactible, radioactive versus non-radioactive) on sorting tables. When a sufficient quantity of contaminated waste has been accumulated, the compactor is operated. Radioactivity of filled containers is monitored so that proper handling, storage, and disposal are assured. Filled containers may be stored in the low-level package storage area until shipped.

11.4.2.2 Components Description

Design parameters for the equipment in the SWMS are provided in Table 11.4-1. Component arrangement is shown on the system flow diagrams provided in Figure 11.4-1 (Sheet 1 and 2).

11.4.2.2.1 Spent Resin Tank

Three stainless steel spent resin tanks with conical bottoms hold resins from radioactive or potentially radioactive plant demineralizers. Non-clogging screens prevent the flow of resins out of the tank through the spent resin tank pump suction lines and the service air injection and vent lines. Instrumentation which monitors resin and water levels in the tank and resin water content is read from the remote panel located in the Radwaste Building control room.

The Spent Resin Tanks provide several functions in the resin transfer and disposal process. They provide a source of water which is used to flush the demineralizer resin beds. They perform a phase separator function in accumulating resin while providing water for the flushing process. Separation is accomplished by an underdrain system within the tank that retains resin but allows the spent resin waste transfer pump to draw water for continued resin flushing.

Resin slurry is removed via a drain nozzle located in the tank bottom and the resin waste forwarding pump. The resin slurries can be transferred to the dewatered waste processing area located in the radwaste building.

Normally, the tank is vented to the room exhaust duct which is handled by the Radwaste Building Ventilation system. During resin transfers, the vent line is closed to allow tank pressurization by either water or air. A relief valve on each tank prevents overpressurization. Resin transfers may be terminated from the Radwaste control room or the dewatered waste processing area using an emergency cutoff to actuate valve closure in the resin transfer line and service air supply to the spent resin tank. x

Building

11.4.2.2.2 Spent Resin Surge Tanks

The two Spent Resin Surge Tanks are stainless steel tanks with dished heads, one Spent Resin Surge Tank services the Low Activity Spent Resin Tanks. The other Spent Resin Surge Tank services the High Activity Spent Resin Tank.

The Spent Resin Surge Tank is required to provide a surge space in the otherwise closed loop resin transfer system. It also serves the purpose of keeping the demineralizers and resin hold tanks full of water during resin transfer operations. It is arranged at the highest elevation in the system above the demineralizers and spent resin tanks. It accommodates surges in system water inventory that occur during operation thus minimizing the amount of makeup water required by the system and the amount of radioactive liquid that must be processed by the liquid waste system.

11.4.2.2.3 Spent Resin Transfer Pumps

Each Spent Resin Tank is provided with a Spent Resin Transfer Pump. The three Spent Resin Transfer Pumps are stainless steel pumps.

The Spent Resin Transfer Pump takes suction from its respective spent resin tank and is used during the resin transfer from the various ion exchangers, demineralizers and storage vessels to the spent resin tank. The piping arrangement allows the discharge of the Spent Resin Transfer Pump to be directed to the demineralizer in Tables 11.4-2 and 11.4-3, respectively. Table 11.4-4 lists the estimated burial volume and activity estimates for the various solid waste types that will be shipped for disposal from the System 80+. Radionuclide specific activities for each waste type are provided in Table 11.4-5.

11.4.4 SAFETY EVALUATION

The SWMS has no safe shutdown or accident mitigation function. Finally, accidental releases from this system, will not exceed the limits of 10 CFR 20, Sections 20.1001-20.2402 of Appendix B, Table 2, Column 2. Accidental releases due to a major component failure or SWMS leak will be contained in the Radwaste Building.

11.4.5 INSPECTION AND TESTING REQUIREMENTS

A Process Control Program appropriate to assure that the SWMS is operating as intended is developed prior to fuel loading. Procedures for each phase of system operation including resin transfer and batching help ensure that design objectives are met. Emphasis is placed on verifying instrumentation and remote functions important to these design objectives.

11.4.6 INSTRUMENTATION REQUIREMENTS

Instrumentation and indications important to the Design Basis of the SWMS are as follows:

A. Level Indicators

High level indication will be provided to prevent overflow of tanks during fill and resin transfer/sluice operations. These indications will be read in the ~~r~~Radwaste Building control room. Also, video observation of all fill processes is included. X

Densitometers are provided on the spent resin storage tanks and used to verify correct resin-to-water ratio when a batch of bead resin is to be solidified.

B. Flow and Pressure Indicators

Pump discharge flow and suction metering as well as pump discharge pressure indication will be provided to properly control the bed transfer process.

C. Radiation Monitoring

Area radiation monitors will be provided as discussed in Section 11.5.

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The process and effluent radiological monitoring and sampling systems are used to measure, record, and control releases of radioactive materials in plant process systems and effluent streams. The monitoring and sampling systems consist of permanently installed sampling and monitoring equipment designed to indicate routine operational radiation releases, equipment or component failure, system malfunction or misoperation, or potential radiological hazards to plant personnel or to the general public. The area radiation monitoring system which is also described in this section, supplements the area radiation survey provisions of Chapter 12 to ensure proper personnel radiation protection.

Collectively, the monitoring systems are referred to as the Radiation Monitoring System (RMS). For some systems, the RMS is also used for radiological sampling purposes, while for other systems, other sampling equipment is utilized. These systems include both nuclear safety-related and non-safety-related equipment which interface with both nuclear safety-related and non-safety-related process, control, and information systems consistent with the guidelines of Regulatory Guide 1.97.

11.5.1 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEMS

Continuous monitoring equipment is located in selected airborne, gaseous, and liquid process and effluent streams to detect activity generated during normal operations, including anticipated transients, and during and after postulated accidents. The area radiation monitoring system provides radiation level indication and alarms for selected areas of the plant to alert plant ~~personnel and control room~~ operators, of increasing or abnormally high radiation levels.

and other personnel

The Radiation Monitoring System assists plant operators in evaluating and controlling the radiological consequences of a potential equipment failure, system malfunction, or system misoperation. Tables 11.5-1, 11.5-2, and 11.5-3 list the gaseous and liquid process and effluent, and airborne radiation monitors. Table 11.5-4 lists the area radiation monitors for the plant, while Table 11.5-5 lists special purpose area monitors which are also used for special monitoring functions.

11.5.1.1 Design Basis

The Radiation Monitoring System is designed to perform the following basic functions:

A. Provide early warning to station personnel of equipment, component, or system malfunction or misoperation, or potential radiological hazards within the station consistent with 10 CFR 20 and 10 CFR 50 Appendix I.

B. Provide continuous monitoring of radioactive liquid and airborne releases consistent with the requirements of 10 CFR 20, 10 CFR 50 GDCs 60, 63, 64, and Regulatory Guide 1.21.

C. Provide monitoring of liquid and airborne activity in selected locations and effluent paths for postulated accidents in accordance with the requirements of 10 CFR 50, NUREG-0737, and Regulatory Guides 1.45, 1.97, and 8.12.

D. The System 80+ design of the Process and Effluent Radiological Monitoring and Sampling Systems provides instrumentation to measure, record, and readout in the control room as well as control releases of radioactive materials in plant process systems and effluent streams. This system is designed to provide for continuous sampling and monitoring of radioactive iodine and particulate, as well as the capability to take grab samples in gaseous process and/or effluent streams in all potential accident release points.

   A particulate/iodine fixed filter cartridge is provided for all plant ventilation systems, with the exception of the nuclear annex and radwaste building ventilation system, which have their own particulate and iodine monitoring systems. Except for the turbine building exhaust, containment purge, the main condenser evacuation system, the Nuclear Island ventilation systems, and the Gaseous Waste Management System, exhausts discharge through the unit vent. Provisions for taking grab samples are provided as specified in Table 11.5-6. Additional discussion regarding sampling capabilities for gaseous process and effluent streams is addressed in Section 11.5.2.2.

   A fixed iodine absorption filter and detector assembly, as well as a moving filter and detector assembly are provided for the unit vent monitor as discussed in Section 11.5.1.2.3.1.B. The ventilation systems are provided with a fixed iodine absorption filter and detector assembly only, with the exception of the Nuclear Annex and Radwaste Building ventilation systems which are provided with its own particulate and iodine detection systems discussed in Section 11.5.1.2.4.B and E.

   The capability for taking grab samples from the unit vent and ventilation system exhausts are provided, as specified in Table 11.5-6 at the respective radiation monitor locations. These grab samples are taken for analysis, at a

The ranges and sensitivities of the monitors are based upon the maximum and minimum expected concentrations for normal plant operation, including anticipated transients, and postulated accidents in accordance with 10 CFR 20 limits and regulatory guidance. The range and sensitivity values listed in Tables 11.5-1 through 11.5-5 represent design values which meet or exceed the regulatory guidance and are generally commercially available.

11.5.1.2 System Description

11.5.1.2.1 Monitor Design and Configuration

Process and effluent, and airborne radiation monitors typically consist of components such as a microprocessor, one or more detectors, a shielded detection chamber, a sample pump, flow instrumentation, and associated tubing and cabling. Three basic types of process and effluent monitoring configurations are used: off-line, on-line, and in-line.

In the off-line type system, a representative sample is taken from the fluid and routed through a filter or chamber for detection of activity. Off-line type systems allow optimized design of detector geometry and shielding, and for detection chambers to be located away from areas of high background radiation. Most off-line monitors are designed as factory-built package skids; however, some applications have a microprocessor which is mounted remotely from the detector/shield assembly. While most off-line monitors utilize a sample pump to draw a sample, some off-line monitors use system pressure to create sample flow.

The on-line system configuration has the detector located next to or on the pipe, duct, or tubing carrying the process or effluent fluid. This type system has the advantage of being simpler (fewer active components) and having a quicker response time. These type monitors generally have less shielding than off-line monitors and have remote mounted microprocessors.

In-line type systems are very similar to the on-line type systems except that the detector is located in the actual process stream. The detector is generally mounted inside a well which allows the detector to be surrounded by the process or effluent fluid.

Each process and effluent, and airborne monitor is located in an easily accessible area and is provided with sufficient shielding to ensure that the required sensitivity is achieved at the design background radiation level for the area. A checksource is used to periodically check the operability of each detector. The checksource is automatically activated by the monitor microprocessor at pre-set intervals. In addition, manual actuation of the checksource is available in the Main control room and at the microprocessor location. Instrumentation and sensors are

(Should read, "Main Control Room")

they discharge through a common plant vent. Sections 11.5.1.2.3.1 and 11.5.1.2.4 provide for monitor descriptions for the monitors listed below. These monitors are designed to meet high-quality commercial grade standards.

- Containment Atmosphere Monitor
- Containment High Purge Exhaust Monitor
- Containment Low Purge Exhaust Monitor
- Reactor Building Annulus Monitor
- Reactor Building Subsphere Ventilation Monitor
- Nuclear Annex Building Ventilation Monitors
- Main Condenser Evacuation System Monitor
- Fuel Building Ventilation Monitor
- Radwaste Building Ventilation Monitor

The post-accident radiation monitors consistent with Regulatory Guide 1.97 are the high range containment monitors, primary coolant monitors, main steam line monitors, unit vent monitor, unit vent post-accident monitor, and selected area radiation monitors which cover areas where access may be required to service equipment important to safety. The post-accident area radiation monitor locations are selected based on the results of post-accident shielding analysis and design information on equipment location and access requirements. (See post-accident dose assessment in Chapter 12.)

11.5.1.2.2 Main Control Room Interface

Primary indication of radiation levels and alarms is handled through the DIAS and DPS systems including both post-accident and non-post-accident monitors. ~~Control Room~~ Display in the main Control Room of post-accident radiation monitoring parameters is in compliance with the requirements of Regulatory Guide 1.97 as described in Chapter 7.

Via the DPS and DIAS systems, ~~control room~~ plant operators can obtain detailed information on monitor readings, alarm setpoints, and operating status. A digital communications network is used to interface these systems with each monitor microprocessor. Operators can access information on monitor configuration and historical trends, and diagnose problems from operation status alarms. A failure in any individual microprocessor does not affect the operation of any other microprocessor nor does it fail the communications network.

Dedicated operator control modules are also available to change microprocessor database items, initiate certain monitor control functions, and change monitor alarm setpoints. These control functions include starting or stopping sample pumps, manual checksource actuation, monitor purge initiation, and moving filter paper advance. Alert alarm setpoints are set at a level determined by operating personnel to allow the observation of differential changes in activity levels. High alarm setpoints to indicate and correlate primary-to-secondary leakage. A sample tap is provided to allow the collection of periodic grab samples.

11.5.1.2.3.2 Liquid Process and Effluent Monitors

An itemized description of each Liquid Process and Effluent Monitor follows. Also, a list of each monitor and associated parameters are given in Table 11.5-2.

A. Component Cooling Water System Monitors

Each division of the Component Cooling Water System is provided with an off-line type radiation monitor. Samples are withdrawn from the system downstream of the Component Cooling Water pumps and continuously monitored by a gamma scintillation detector mounted in a shielded liquid sampler. After passing through the monitor, the sample is returned to the Component Cooling Water System.

Activity detected above background is indicative of a leak into the Component Cooling Water System from the Reactor Coolant System or one of the other systems containing radioactive fluids which reject heat to the Component Cooling Water System.

| B. Liquid Waste Management System Discharge Monitor

The liquid waste effluent discharge pipe is monitored downstream of the last possible point of radioactive liquid waste addition, by means of an off-line, shielded liquid sampler using a gamma scintillation detector system. Effluents being monitored include discharge from the Waste Monitor Tanks, Detergent Sample Tanks, and Chemical Sample Tank. In the event that radioactivity in excess of a preset limit is detected in the waste liquid discharge flow, the Liquid Waste Management System Discharge Monitor will actuate an alarm in the Radwaste Building control room, and the Control Room, and terminate the discharge. The radiation setpoint for the Liquid Waste Discharge Monitor is determined prior to each batch release based on expected concentrations (by sampling) and discharge flow rate.

C. Steam Generator Blowdown Sample Monitor

This off-line monitor samples the steam generator blowdown for radioactivity which would be indicative of primary-to-secondary leakage. Samples from each of the steam generators are continuously monitored individually by a detector mounted in a shielded liquid sampler. After being monitored, the sample passes back to the steam generator blowdown system.

D. Reactor Coolant Gross Activity Monitor

This monitor is located in the Process Sampling System. Gross activity in the reactor coolant is continuously monitored by a gamma scintillation detector in a lead shielded sampler assembly. The sampler assembly is located in an accessible area of the Nuclear Annex. To permit decay of activity not indicative of fuel clad failure, such as N-16, a delay is incorporated in the sample transport from the reactor coolant system to the detector.

Large variations in activity levels are possible depending on the amount of fission products leaked into the reactor coolant system. Abnormal conditions of high activity or loss of sample flow are alarmed in the main control room. The setpoint for high activity is adjustable over the full range of the instrument. The high activity setpoint is adjusted to alarm a significant change in reactor coolant activity so that laboratory sample analysis can be performed and appropriate action taken.

E. Turbine Building Drains Monitor

This is an off-line monitor with a shielded liquid sampler and gamma scintillation detection system to continuously monitor the effluent from the Turbine Building Drains System. Detection of high activity automatically terminates releases from the system and initiates alarms to plant operators.

F. Station Service Water System Monitor

This monitor is also an off-line type monitor which continuously monitors the Station Service Water System for gross gamma activity. Samples are withdrawn from the service water side system downstream of the Component Cooling Water System Heat Exchangers. Two sample lines (one from each division) combine before the sample enters the shielded sampler.

G. Steam Generator Drain Tank Discharge Monitor

This monitor uses an off-line shielded liquid sampler and a gamma scintillation detection system to continuously monitor the effluent from the Steam Generator Drain Tank System. Detection of high activity automatically terminates releases from the system and initiates alarms to plant operators.

H. Containment Cooler Condensate Tank Monitor

This monitor uses an off-line shielded liquid sampler and a gamma scintillation detection system to continuously monitor the effluent from the Containment Cooler Condensate Tank system. Detection of high activity automatically terminates releases from the system and initiates alarms to plant operators.

I. Neutralization Tank Discharge Monitor

This is an offline monitor with a shielded liquid sampler and gamma scintillation detection system to monitor the releases from the neutralization tank. In the event that activity above a preset limit is detected, the discharge from this tank would be automatically terminated and an alarm would be actuated in the control room. The setpoint for this monitor is determined prior to release based on expected concentrations based on sampling and discharge flow rate.

J. Steam Generator Blowdown System Discharge Monitor

This is an offline monitor with a shielded liquid sampler and gamma scintillation detection system to monitor the releases from the flash tank. In the event that activity above a preset limit is detected, the discharge from this tank would be automatically terminated and an alarm would be actuated in the control room. The setpoint for this monitor is determined prior to release based on expected concentrations based on sampling and discharge flow rate. However, typically the steam generator blowdown liquid will be recycled and not released to the environment.

11.5.1.2.4 Airborne Radiation Monitors

Airborne radiation monitoring equipment is provided in selected areas and ventilation systems to provide plant operating personnel with continuous information concerning the airborne radioactivity levels throughout the plant. An itemized description of each airborne radiation monitor follows. Also, a list of each monitor and associated parameters are given in Table 11.5-3. Monitor locations are indicated on the applicable air flow diagrams in Section 9.4.

A. Containment Atmosphere Monitor

The containment atmosphere monitor draws a sample from the containment atmosphere recirculation system or the containment filtration system to monitor airborne radioactivity levels in the containment structure. This monitor is similar in design to the unit vent monitor described in Section 11.5.1.2.3.1 with particulate, iodine, and gaseous detection channels. After monitoring, the sample flow is returned to the containment atmosphere.

G. Control Room Air Intake Monitors

Each of the two main Control Room air intakes is continuously monitored for airborne radioactivity by means of off-line shielded gaseous radiation monitors. In order to provide redundancy, there are four safety class 3 monitors (two for each intake). The capability for auto selection and closure of the intake with the highest radiation level is provided. A particulate/iodine fixed filter cartridge is included in the inlet sample tubing to each monitor for collecting periodic grab samples.

H. Reactor Building Annulus Monitor

Annulus air is continuously monitored by this gas monitor to indicate radioactivity resulting from equipment failure or leakage. When an entry into the annulus is required this monitor can give station personnel information on airborne activity. Sample tubing is routed to give a representative sample of annulus air, particularly including areas where station personnel are likely to perform maintenance or surveillance activities. A particulate/iodine fixed filter cartridge is included in the inlet sample tubing to this monitor for collecting periodic grab samples. Under post-accident conditions, this monitor can be used as a supplement to Regulatory Guide 1.97 monitors to measure activity from expected containment leakage or from an unexpected breach in containment.

I. Reactor Building Subsphere Ventilation Monitor

Each division is continuously monitored by an off-line monitor. These monitors continuously sample the exhaust from both divisions of the Reactor Building Subsphere Ventilation System. Sample points are upstream of the exhaust filters and downstream of the last entry point to the exhaust subsystem. Detection of activity is indicative of equipment failure or leakage in the subsphere areas. A particulate/iodine fixed filter cartridge is included in the sample inlet for grab sample collection.

J. Portable Airborne Monitor

This monitor includes detector channels for particulate, iodine, and gaseous activity. The samplers, detectors, auxiliary equipment, and associated electronics are assembled on a mobile cart. This monitor can be moved to areas where work or surveillance activities are at an unusual risk of airborne exposure. Design and operation of this monitor allows for the transfer of the particulate sample filters and iodine sample cartridges to the counting room for further sample analysis. The Portable Airborne

Monitor meets the equipment requirements stated in Section III.D.3.3 of NUREG-0737. This includes requirements on sample media, purging, and calibration.

K. Emergency Operations Facility (EOF) Ventilation Monitor

While it is in use during an emergency, the EOF is continuously monitored in the same manner as the TSC described above. While it is in use during an emergency, air entering the EOF is continuously monitored by a shielded off-line gaseous activity detector and returned to the ventilation duct downstream of the intake. If the gaseous activity exceeds a preset limit, an alarm is actuated in the EOF.

11.5.1.2.5 Area Monitors

The Area Radiation Monitoring System monitors the radiation levels in selected areas throughout the plant. Most area monitors are designed to provide normal operation indication of unusual radiological events in order to warn operators and station personnel. Some area monitors are designed for post-accident indication for areas where access for maintenance to equipment important to safety may be necessary. These post-accident monitors are designed to the standards required by Regulatory Guide 1.97. Area radiation monitors will have local visual and audible alarms. High noise areas may have additional visual indication provided if needed to insure prompt recognition by nearby personnel of high radiation conditions. One exception would be the Control Room Area Monitor which will use the existing Control Room indications in order not to create a nuisance or distraction due to a spurious alarm. A list of area radiation monitors and their ranges is presented in Table 11.5-4. Area monitor locations are provided in Table 12.3-5.

11.5.1.2.6 Special Purpose Area Monitors

Listed below are area monitoring systems which are used for process monitoring functions or other special monitoring applications.

A. Main Steam Line Area Monitors

These monitors are located upstream of the safety relief valves on each pair of main steam lines. Detectors are mounted within close proximity of the process lines to detect radioactivity due to a steam generator tube rupture. This monitoring system meets Regulatory Guide 1.97 requirements, including Category 2 environmental qualification and applicable range requirements.

B. Purification Filter Area Monitors

A monitor is located in the immediate vicinity of each reactor coolant purification filter. The readings from the purification filter area monitors are trended to indicate suspended solids concentrations in the RCS. In addition, these monitors can be used to indicate when the filters should be replaced to prevent the filters from becoming too radioactive for normal disposal. These also provide general area dose rate information for ALARA planning.

C. Primary Coolant Loop Monitors

The primary coolant monitors consist of two physically independent and electrically separate high range area monitoring channels to monitor Reactor Coolant System radiation levels. A high range ion chamber detector is located next to each of the Reactor Coolant System hot legs to provide a seismically and environmentally qualified indication of a breach of fuel cladding following a loss of coolant accident. Main Control Room indication and alarms are provided in compliance with Regulatory Guide 1.97 requirements for post-accident monitoring as described in Chapter 7.

D. High Range Containment Area Monitors

The High Range Containment Area Monitors consist of two physically independent and electrically separated ion chambers located inside the reactor containment away from the influence of the Reactor Coolant System to measure high range gamma radiation. This monitor gives operators a seismically and environmentally qualified indication of containment airborne activity. The design and qualification of these monitors meet the requirements of Regulatory Guide 1.97 for Category I instruments. Dose rate readings are correlated to determine airborne concentrations based on expected accident source terms and the time after an accident. Main Control Room indication and alarms are provided in compliance with Regulatory Guide 1.97 requirements for post-accident monitoring as described in Chapter 7. The High Range Containment Area Monitors will meet all of the requirements of Section II.F.1-3 of NUREG-0737. This includes Table II.F.1-3 requirements for range, detector response, redundancy, separation, in situ calibration, and environmental/design qualification.

E. N-16 Steam Line Radiation Monitors

N-16 radiation monitors are located outside of containment downstream of the main steam isolation valves. One monitor per generator will be mounted within close proximity of the steam lines to detect radioactivity resulting from a steam generator tube (SGTR) rupture. Alarms located in the Main Control Room to alert the operator when these monitors detect specified primary to secondary leakage.

11.5.1.3 Calibration and Maintenance

Commercially available equipment with industry proven technology is incorporated into the design of the Radiation Monitoring System. Monitoring equipment is factory tested and calibrated with provisions made for periodic field calibrations to verify

TABLE 11.5-4

(Sheet 1 of 2)

AREA RADIATION MONITORS[a]

Monitor	Typical Range (mR/hr)	Power Source	Seismic Category[d]
Reactor Containment Entrance	0.1 - 1E+7	non-1E	None
Refueling Bridge Crane	0.1 - 1E+4	non-1E	II
In-core Instrumentation Equipment	0.1 - 1E+4	non-1E	None
Decontamination Area	0.1 - 1E+4	non-1E	None
Sample Room	0.1 - 1E+4	non-1E	None
Control Room	0.1 - 1E+4	non-1E	None
Primary Chemistry Laboratory	0.1 - 1E+4	non-1E	None
New Fuel Storage Area (2)	0.1 - 1E+4	non-1E	None
Spent Fuel Pool Bridge	0.1 - 1E+4	non-1E	II
Fuel Building Area	0.1 - 1E+7	non-1E	None
Nuclear Annex [b] (normal operation)	0.1 - 1E+4	non-1E	None
Nuclear Annex [c] (post-accident)	100 - 1E+7	non-1E	None
Solid Waste Drum Storage and Handling Area	0.1 - 1E+4	non-1E	None
Radwaste Building Loading Bay	0.1 - 1E+4	non-1E	None
Hot Machine Shop	0.1 - 1E+4	non-1E	None
Hot Instrument Shop	0.1 - 1E+4	non-1E	None
Radwaste Building Areas [b]	0.1 - 1E+4	non-1E	None
Reactor Building Subsphere [b] (normal operation)	0.1 - 1E+4	non-1E	None
Reactor Building Subsphere [c] (post-accident)	100 - 1E+7	non-1E	None

COMMENT 3 (5) AND 4 (6)

2.9.4 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEM

Design Description

The Process and Effluent Radiological Monitoring and Sampling System (PERMSS) provides components to monitor liquid and gaseous effluents prior to release to unrestricted areas, and to monitor for inplant radioactivity.

Components of the PERMSS are located in the nuclear island structures, the radwaste building, the turbine building, and the station service water pump structure.

The PERMSS has components that provide radiological monitoring of gaseous and liquid processing systems and their effluents, airborne radioactivity, radiation areas, and specified plant equipment.[1]

The system provides radiological monitoring during plant operation and post-accident conditions. The two high range containment area radiation monitors provide indication of the radiation levels in Containment throughout the course of a design basis accident.

The PERMSS is non-safety-related with the exception of the following each of which is safety-related, Seismic Category I, and Class 1E:

a. MAIN control room (MCR) intake radiation monitor (2/intake),
b. high range containment area radiation monitor (2),
c. containment atmosphere radiation monitor (particulate channel only),
d. primary coolant loop radiation monitors (2).

Independence is provided between Class 1E Divisions, and between Class 1E Divisions and non-Class 1E equipment, in the PERMSS. ~~Independence is also provided between Class 1E channels and between Class 1E channels and non-class 1E equipment in the PERMSS.~~ 3(5)

The MCR ~~control room~~ intake radiation monitors shall have the capability for auto selection and closure of the most contaminated intake.

[1] The radiation monitors that monitor gaseous and liquid processing systems and their effluents and the response of these systems to detection of radiation are addressed in the individual systems which they support.

Displays of the PERMSS safety-related instrumentation (the ~~control room~~ MCR air intake radiation monitors, the reactor coolant radiation monitors, the high range containment area monitors and the containment atmosphere particulate monitors) exist in the ~~main control room~~ (MCR) or can be retrieved there.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.9.4-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Process and Effluent Radiological Monitoring and Sampling System.

PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The PERMSS has components that provide radiological monitoring of gaseous and liquid processing systems and their effluents, airborne radioactivity, radiation areas, and specified plant equipment.	1. Inspection of the PERMSS components will be performed.	1. The PERMSS provides the components specified in Table 2.4.9-2.
2. Displays and alarms of the PERMSS safety-related instrumentation (the ~~control room~~ MCR air intake radiation monitors, the reactor coolant radiation monitors, the high range containment area monitors and the containment atmosphere particulate monitors) exist in the MCR or can be retrieved there.	2. Inspection for the existence or retreivability in the MCR of instrumentation displays and alarms will be performed.	2. Displays and alarms of the PERMSS safety-related instrumentation (the ~~control room~~ MCR air intake radiation monitors, the reactor coolant radiation monitors, the high range containment area monitors and the containment atmosphere particulate monitors) exist in the MCR or can be retrieved there.
3. The ~~control room~~ MCR intake radiation monitors shall have the capability for auto selection and closure of the most contaminated intake.	3. Testing of each monitor will be conducted using manual controls and simulated automatic initiation signals.	3. Each ~~control room~~ MCR intake monitor is activated upon receipt of test signals and the associated control room intake is closed automatically.
4. Operation of each safety-related PERMSS division can be manually activated from the ~~control room~~ MCR or automatically.	4. Testing of each division (including each channel of the safety-related portion of the area radiation monitoring system) will be conducted using manual controls and simulated automatic initiation signals.	4. Each division is activated upon receipt of test signal.

TABLE 2.9.4-1 (Continued)

PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Each safety-related area radiation monitor channel monitors the radiation level in its assigned area, and indicates its respective ~~Main Control Room (~~MCR~~)~~ alarm and local audible and visual alarm (if provided) when the radiation level exceeds a preset level.	5. Testing of each channel of the safety-related area radiation monitors will be conducted using simulated input signals.	5. MCR and local alarms are initiated when the simulated radiation level exceeds a preset limit.
6. The following PERMSS safety-related instrumentation shall be provided: a. MCR ~~control room~~ intake radiation monitor (2/intake), b. high range containment area radiation monitor (2), c. containment atmosphere radiation monitor (particulate channel only), d. primary coolant loop radiation monitors (2).	6. Inspection of the as-built system will be conducted.	6. The as-built PERMSS conforms with the design description.
7. The PERMSS safety-related instrumentation (the MCR ~~control room~~ intake radiation monitors, high range containment area radiation monitors, containment atmosphere radiation monitor (particulate channel), and the primary coolant loop radiation monitors) are classified Seismic Category I.	7. Seismic analyses of the as-built PERMSS safety-related instrumentation will be performed.	7. An analysis report exists which concludes that the PERMSS safety-related instrumentation (the ~~control room~~ MCR intake radiation monitors, high range containment area radiation monitors, containment atmosphere radiation monitor (particulate channel), and the primary coolant loop radiation monitors) are classified Seismic Category I.

TABLE 2.9.4-1 (Continued)

PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria	
8.~~a)~~ Independence is provided between Class 1E Divisions, and between Class 1E Divisions and non-Class 1E equipment, in the PERMSS.	8.~~a)~~ Inspection of the as-installed Class 1E Divisions of the PERMSS will be performed.	8.~~a)~~ Physical separation exists between Class 1E Divisions in the PERMSS. Physical separation exists between Class 1E Divisions and non-Class 1E equipment in the PERMSS.	3(5)
8.b) Independence is also provided between Class 1E Channels and between Class 1E Channels and non-Class 1E equipment in the PERMSS. a			

TABLE 2.9.4-2

GASEOUS PROCESS AND EFFLUENT MONITORS

Gaseous waste management system waste gas discharge (1)

Unit Vent - Normal (1)

Unit Vent - Post Accident (1)

Containment high purge exhaust (1)

Containment low purge exhaust (1)

Main Condenser Evacuation System (1)

LIQUID PROCESS AND EFFLUENT MONITORS

Component cooling water system (1 /division)

Liquid waste management system liquid waste discharge (1)

Steam generator blowdown (1)

Reactor coolant gross activity (1)

Turbine floor building drains (1)

Station service water system (1/Division)

Containment cooler condensate tank (1)

Condensate Cleanup System Neutralization Tank Discharge (1)

AIRBORNE RADIATION MONITORS

Containment atmosphere (1)

Radwaste building ventilation exhaust (1)

Fuel building ventilation exhaust (1)

Ventilation systems multisampler (2 monitors - 1/division in Nuclear Annex) (1 in Radwaste Building)

Nuclear annex building ventilation (2 monitors - 1/division)

Main control room air intake (2/intake)

Reactor building annulus exhaust (2 monitors - 1/division)

Reactor building subsphere ventilation exhaust (2 monitors - 1/division)

Portable airborne

Emergency operations facility ventilation (1)

TABLE 2.9.4-2 (Continued)

AREA RADIATION MONITORS

Reactor Containment entrance

Refueling bridge crane

In-core instrumentation equipment

Decontamination area

Sample room

Main control room

Primary chemistry laboratory

New fuel storage area

Spent fuel pool bridge

Fuel building area

Nuclear annex building (normal operation)

Nuclear annex building (post accident)

Reactor building subsphere (normal operation)

Reactor building subsphere (post accident)

Solid waste drum storage and handling area

Radwaste building loading bay

Hot machine shop

Hot instrument shop

Radwaste building areas

Technical support center

SPECIAL PURPOSE AREA RADIATION MONITORS

Main steam lines area (2 monitors - 1/loop)

Purification filters (1/filter)

Primary coolant loops (2 monitors - 1/loop)

High range containment area monitors (2)

Main steam lines (primary-to-secondary leakage) (2 monitors - 1/SG)

CE 80+ ITAAC Independent Review Comments

ITAAC No. 1.3 (Figure Legend)

No.	Comments	Cat.	Resolution
1	Symbol is listed as a diaphragm operator. However, symbol is used throughout the CDM to generically show all types of pneumatic operators, including pneumatic diaphragm operators. Suggest delete the diaphragm symbol from CDM section 1.3.	1	Agree - Delete symbol

FIGURE LEGEND (continued)

Valves

Gate Valve 

Globe Valve 

Check Valve 

Butterfly Valve 

Ball Valve 

Relief Valve 

Three Way Valve 

Post Indicator Valve 

Valve Type Not Specified 

Valve Operators

Operator Of Unspecified Type 

Fluid Powered Operator 

Motor Operator 

Solenoid Operator 

Diaphragm Operator 

Hydraulic Operator 

Pneumatic Operator 

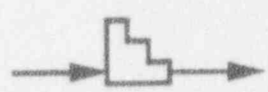
Position Indications For
Hydraulic And Pneumatic Operators

-Fails As Is FAI

-Fails Closed FC

-Fails Open FO

Mechanical Equipment

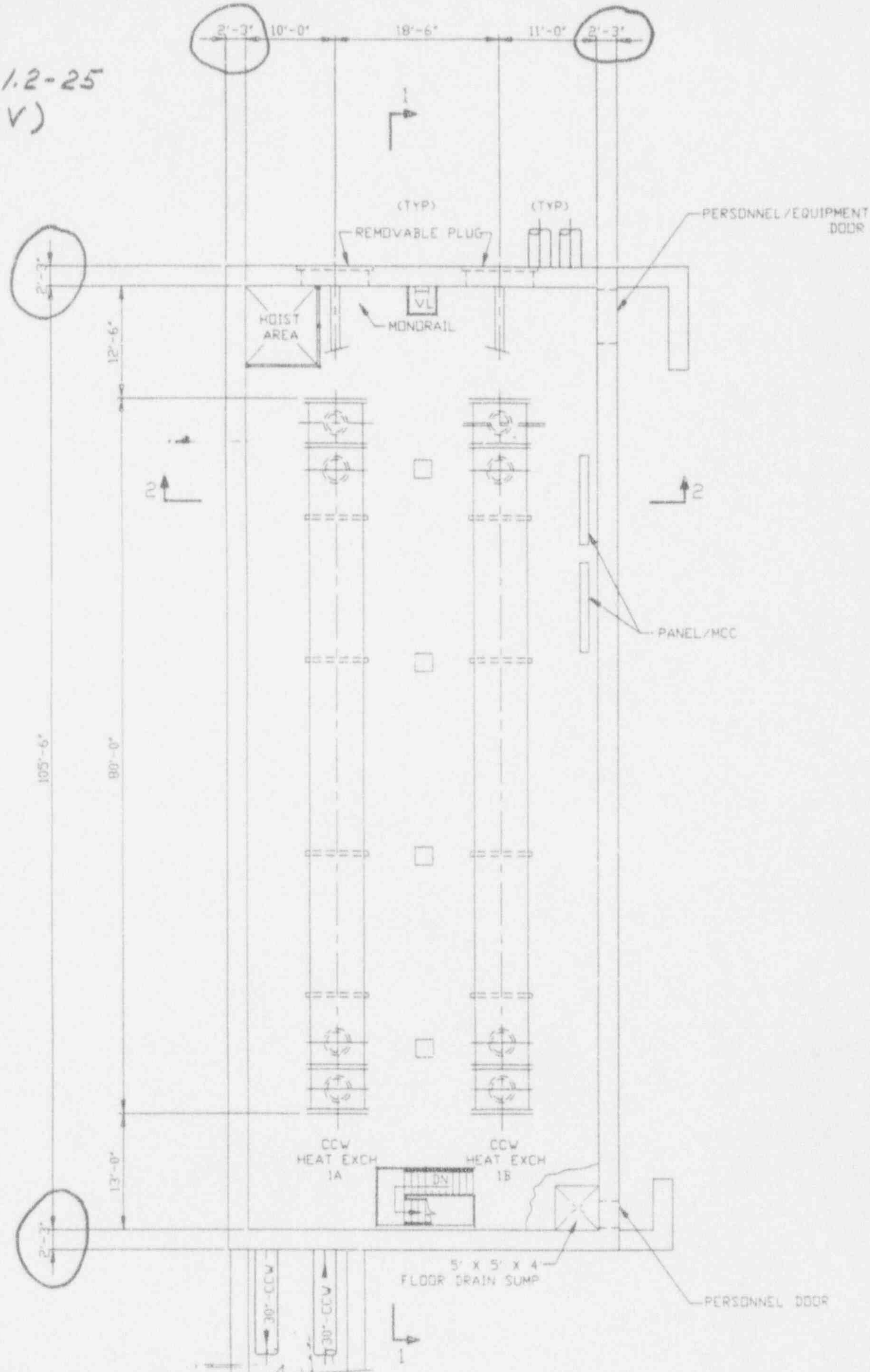
Positive Displacement Pump 

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.1.3 (CCW Heat Exchanger Structures)

No.	Comments	Cat.	Resolution
1	The Design Criteria (CESSAR Appendix 3.8B, Section 7.2.4, Page 3.8B-36) indicates that the North-South (long direction) walls are 4 foot walls while the Drawing (Fig. 1.2-25) shows that their thicknesses are 2 ft..	1	Agree - CESSAR-DC Appendix 3.8B, Section 7.2.4 and Figure 1.2-25 have been revised in Amd. V to show the correct wall thickness of 2'-3". Attached are a revised Figure 1.2-25 and a marked revision page for Appendix 3.8B, Section 7.2.4. No ITAAC revision is required.

Figure 1.2-25
(Amd. V)



PLAN - GRADE

7.2.3 LOADS AND LOAD COMBINATIONS

The Component Cooling Water Heat Exchanger Structure is evaluated for the loads and load combinations specified in Sections 3.8A.5.1 and 3.8A.5.2, respectively, for Seismic Category I concrete structures.

The major loadings affecting the design of the structure are dead loads (i.e., self weight and equipment weight from the CCW heat exchangers), temperature, static and dynamic lateral soil and ground water pressures, wind loads, earthquake loads, and tornado loads.

The critical load combinations are equations 5.2.2.1(a), 5.2.2.1(d), and 5.2.2.2(a) of Section 3.8A.5.2, i.e.,

$$U = 1.4D + 1.7L$$

$$U = 0.75\,(1.4D + 1.7F + 1.7L + 1.7H + 1.7T_o + 1.7R_o)$$

$$U = D + F + L + H + T_o + R_o + E'$$

7.2.4 ANALYSES AND RESULTS

The reinforced concrete members of Seismic Category I structures are designed to the criteria specified in ACI 349 and NRC Regulatory Guide 1.142, except as modified by Appendix 3.8A (see 3.8A.6.2). In general, symmetrical reinforcing steel (i.e., the same area and configuration on opposite faces of members), is provided except in local areas. Concrete joints shall be detailed in accordance with the criteria specified in ACI 318, Chapter 21 (see Section 3.8A.6.2.1.1.1 and Section 6.0 of this appendix).

Foundation Mat:

The primary reinforcing for the four-foot thick foundation mat consists of a rectangular grid of #9 at 10 inches ~~each way~~ each face, [i.e., 1.20 in²/ft] in the long direction and #11 at 6 inches each face, [i.e., 3.12 in²/ft] in the short direction.
No transverse shear reinforcing is required.

East and West Walls (Short Direction):

The primary reinforcing for these ~~two-foot~~ 2'-3" thick walls consists of a rectangular grid of #11 at 6 inches ~~each way~~ each face, [i.e., 3.12 in²/ft] vertically and #11 at 10 inches each face, [i.e., 1.87 in²/ft] horizontally.
No transverse shear reinforcing is required.

North and South Walls (Long Direction):

The primary reinforcing for these ~~four-foot~~ 2'-3" thick walls consists of a rectangular grid of #11 at 6 inches vertically each face and #11 at 10 inches horizontally, [i.e., 3.12 in²/ft and 1.87 in²/ft, respectively].

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.1.6 (Reactor Vessel Internals (RVI))

No.	Comments	Cat.	Resolution
1	See attached pages for comments. (Pgs 1-2 of Design Description and Pg 1 of ITAAC)	1	Agree. Comments incorporated as shown on marked revision pages to ITAAC. No CESSAR-DC changes are required.

2.1.6 REACTOR VESSEL INTERNALS

Design Description

The Reactor Vessel Internals consist of a Core Support Barrel (CSB) Assembly and an Upper Guide Structure (UGS) Assembly.

The Basic Configurations of the CSB and the UGS are as shown on Figures 2.1.6-1 and 2.1.6-2, respectively. The Reactor Vessel Internals are safety-related.

Dimensions of the core support barrel and the upper guide structure assembly are listed in Table 2.1.6-1.

The ~~Core Support Barrel (~~CSB~~)~~ assembly is suspended from the reactor vessel flange. The CSB assembly provides support and location positioning for the fuel assembly lower end fittings. The CSB assembly contains structural elements that provide an instrumentation guide path from the lower vessel, and hydraulic flow paths through the vessel from the inlet nozzles to the upper end of the fuel assemblies.

The core barrel assembly contains a grid structure which supports the core and provides flow distribution from the lower plenum region to the bottom of the fuel assemblies. The core shroud is part of the CSB assembly and provides an envelope to direct the primary coolant flow through the core. Instrument nozzles in the grid structure provide a guide path for in-core instruments from the reactor vessel lower head to the fuel assemblies.

The ~~Upper Guide Structure (~~UGS~~)~~ assembly is supported by the CSB upper flange and extends into the CSB assembly to engage the top of the fuel assemblies. The UGS assembly provides an insertion path for the control element assemblies (CEA). The UGS assembly contains structural elements which provide both a guide path and lateral support for the upper portion of the control element assemblies and extension shafts in the reactor vessel upper plenum region. The UGS assembly also provides guide paths for heated junction thermocouple (HJTC) assemblies.

The CSB and UGS assemblies are designed and constructed in accordance with ASME Code Section III Subsection NG requirements and are classified Seismic Category I. The reactor vessel internals maintain their integrity during normal operation, transients, and during SSE and design basis accident conditions not eliminated by leak-before-break evaluations. The material of construction for the CSB and UGS components is austenitic stainless steel with the exception of the Holdown Ring, which is made of martensitic stainless steel. Cobalt base material, if used, is used only for hardsurfacing of wear parts.

The Reactor Vessel Internals withstand the effects of flow induced vibration caused by the operation of the reactor coolant pumps.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.1.6-~~1~~2 specifies the inspections, tests, analyses and associated acceptance criteria for the Reactor Vessel Internals.

TABLE 2.1.6-2

REACTOR VESSEL INTERNALS
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the Reactor Vessel Internals is as shown on Figures 2.1.6-1 and 2.1.6-2.	1. Inspection of the as-built Reactor Vessel Internals will be conducted.	1. For the components and equipment shown on Figures 2.1.6-1 and 2.1.6-2, the as-built Reactor Vessel Internals conform with the Basic Configuration.
2. The Core Support Barrel and Upper Guide Structure are designed and constructed in accordance with ASME Code Section III Subsection NG requirements and are qualified Seismic Category I.	2. Inspection will be performed of the ASME Code Section III required Owner's Review of the ASME Design Report. ~~Document~~	2. The completed ASME Code Section III required Owner's Review of the ASME Design Report ~~Document~~ exists.
3. The Reactor Vessel Internals withstand the effects of flow induced vibration caused by operation of the reactor coolant pumps.	3.a) Testing will be performed to subject the Reactor Vessel Internals to flow induced vibration. Pre- and post-test visual inspection will be performed on the Reactor Vessel Internals.	3.a) Testing and inspection results demonstrate that the Reactor Vessel Internals retain their integrity.
	3.b) A vibration type test will be conducted on the prototype reactor vessel internals.	3.b) A vibration type test report exists and concludes that the prototype reactor vessel internals retain their integrity and have no loose parts as a result of the test.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.1.7 (In-Core Instrument (ICI) Guide Tube)

No.	Comments	Cat.	Resolution
1	CDM Table 2.1.7-1 should be supplemented with appropriate ITAAC (similar to 2.3.1-1, item 10b for Class 1) verifications to confirm ASME Section III Class 1 items shown on Figure 2.1.7-1 are designed and constructed properly.	1	Agree. Comments incorporated as shown on marked revision pages to ITAAC 2.1.7 to be consistent with ITAAC 2.3.1. No revisions to CESSAR-DC are required.
2	See attached mark-up pages. (Pg. 2 of ITAAC)	3	Agree. Comments incorporated as shown on marked revision pages to ITAAC 2.1.7 to be consistent with ITAAC 2.3.1. No revisions to CESSAR-DC are required.

TABLE 2.1.7-1

IN-CORE INSTRUMENT GUIDE TUBE SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration for the ICI Guide Tube System is as shown on Figure 2.1.7-1.	1. Inspection of the as-built ICI Guide Tube System configuration will be conducted.	1. For the components and equipment shown on Figure 2.1.7-1, the as-built ICI Guide Tube System conforms with the Basic Configuration.
2.a) The ICI guide tubes and seal housings retain their pressure boundary integrity under internal pressures that will be experienced during service.	2.a) A pressure test will be conducted on those portions of the ICI Guide Tube System required to be pressure tested by the ASME Code Section III.	2.a) The results of the pressure test of ASME Code Section III components of the ICI guide tubes and seal housings conform with the pressure testing acceptance criteria in ASME Code Section III, ~~Subsection ND~~.
2.b) Components shown as ASME Code Class 1 on Figure 2.1.7-1 are designed and constructed in accordance with ASME Code Class 1 requirements.	2.b) Inspection of the ASME design reports will be conducted.	2.b) The ASME Code Section III design reports exist for the ICI Guide Tube System Class 1 components.

Comment 2.1.7 – 1,2

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.3.1 (Reactor Coolant System)

No.	Comments	Cat.	Resolution
1 (3)	The requirement that the RCP rotating inertia be such that adequate core cooling is maintained when electrical power to the RCP is disconnected should be stated. Slowing the pump flow coastdown does not necessarily assure adequate core cooling unless certain minimum flowrate (and pressure) is met.	1	ABB-CE disagrees. The Design Description and Design Commitment 7.c) are intended only to state a physical attribute of a component. ITAAC 7.c) confirms that the RCP rotating assembly has at least the minimum inertia assumed in Tier 2 safety analyses. The analyses in Tier 2 demonstrate that adequate core cooling is maintained when power is removed from the RCP. Post-core testing is performed (CESSAR DC Section 14.2.12.2.3) to verify adequate flow coastdown. **NRC Staff concurs.**

Note: Markups reflect the change to reactor coolant level instrumentation requested by ABB-CE.

ABB-CE-initiated change (Not associated with NRC Independent Review comments)

2.3.1 REACTOR COOLANT SYSTEM

Design Description

The Reactor Coolant System (RCS) removes heat generated in the reactor core and transfers the heat to the steam generators. The reactor coolant system forms part of the pressure and fission product boundary between the reactor coolant and the Containment atmosphere.

The Basic Configuration of the RCS is as shown on Figures 2.3.1-1 through 2.3.1-4. The pressure retaining components of the RCS and the RCS instrumentation shown on the figures, except as noted on the figures, are safety related.

The RCS is located in the Containment and has a reactor vessel (RV), two vertical, U-tube steam generators (SGs), four vertical, shaft sealed reactor coolant pumps (RCPs), one pressurizer (PZR), four pressurizer safety valves, piping, heaters, controls, instrumentation and valves.

The reactor vessel has a vessel assembly and a removable closure head assembly. The vessel assembly has a shell, lower head, and vessel flange forgings, welded together. The closure head assembly has a dome and head flange forgings, welded together. Forged reactor coolant inlet and outlet nozzles are welded to a shell section. Nozzles for control element drive mechanisms and instrumentation are welded to the closure head assembly, and nozzles for instrumentation are welded to the lower head forging.

RCP seal injection flow is provided by the Chemical and Volume Control System (CVCS). The RCPs have anti-reverse rotation devices.

The RCPs circulate reactor coolant water in loops through the RV to the SGs and back to the RV. The PZR provides a surge volume for the reactor coolant and pressurizes the RCS.

RCS instrumentation has core exit thermocouples (CETs) in the in-core instrumentation (ICI) detector assemblies, heated junction thermocouples (HJTCs) in the HJTC probe assemblies, and differential pressure-based level detectors between the shutdown cooling system (SCS) suction lines and two safety injection system (SIS) direct vessel injection (DVI) lines, and differential pressure-based level detectors between the SCS suction lines and the reactor coolant gas vent subsystem (RCGVS) in the safety depressurization system (SDS). Instrumentation is also provided to measure reactor coolant level across the vertical span of the reactor vessel outlet nozzles.

The pressurizer safety valves provide overpressure protection for reactor coolant pressure boundary components in the RCS. Low temperature overpressure protection for the RCS is provided by the shutdown cooling system (SCS).

instrumentation to measure reactor coolant level across the vertical span of the reactor vessel outlet nozzles

Controls exist in the MCR to start and stop the RCPs, open and close those power operated valves shown on Figures 2.3.1-1 through 2.3.1-4, and energize or de-energize the pressurizer heaters.

Two pressurizer backup heater banks are powered from different Class 1E Divisions. The other pressurizer heaters, the reactor coolant pump motors, and power-operated valves shown on Figure 2.3.1-1 are powered from non-Class 1E sources. Instrumentation shown on Figures 2.3.1-1 through 2.3.1-4 is powered from its respective Class 1E Division except as follows: the ~~narrow range heated junction thermocouple instruments shown on Figure 2.3.1-3~~, the refueling water level instruments between the SCS suction lines and safety injection system lines and the refueling water level instruments between the SCS suction lines and the SDS on Figure 2.3.1-1 are powered from non-Class 1E sources. Independence is provided between Class 1E Divisions, and between Class 1E Divisions and non-Class 1E equipment, in the RCS.

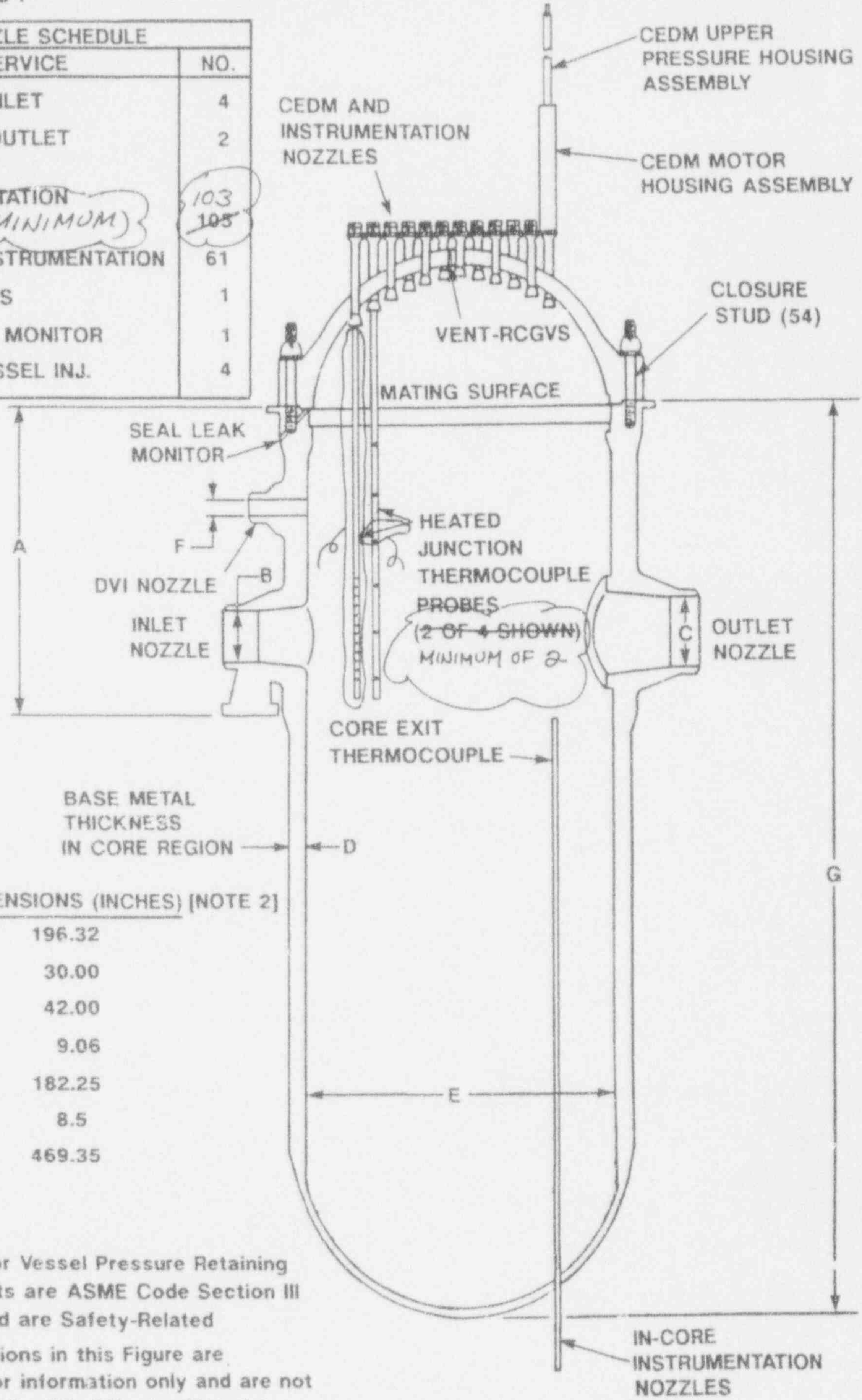
Valves with response positions indicated on Figure 2.3.1-1 change position to that indicated on the figure upon loss of motive power.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.3.1-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Reactor Coolant System.

SYSTEM 80+™

NOZZLE SCHEDULE	
SERVICE	NO.
COOLANT INLET	4
COOLANT OUTLET	2
CEDM AND INSTRUMENTATION NOZZLES (MINIMUM)	~~105~~ 103
IN-CORE INSTRUMENTATION	61
VENT-RCGVS	1
SEAL LEAK MONITOR	1
DIRECT VESSEL INJ.	4



LETTER	DIMENSIONS (INCHES) [NOTE 2]
A	196.32
B	30.00
C	42.00
D	9.06
E	182.25
F	8.5
G	469.35

NOTES:

1. The Reactor Vessel Pressure Retaining Components are ASME Code Section III Class 1 and are Safety-Related
2. The dimensions in this Figure are provided for information only and are not part of the Certified Design Material.

FIGURE 2.3.1-3
REACTOR COOLANT SYSTEM
(REACTOR VESSEL)

TABLE 2.3.2-1

SHUTDOWN COOLING SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the SCS is as shown on Figure 2.3.2-1.	1. Inspection of the as-built SCS configuration will be conducted.	1. For the components and equipment shown on Figure 2.3.2-1, the as-built SCS conforms with the Basic Configuration.
2.a) Each SCS Division has the heat removal capacity to cool the reactor coolant from SCS entry conditions to cold shutdown conditions.	2.a) Testing and analysis of the SCS to measure pump head and the shutdown cooling flow at the combined discharge of the SCS heat exchanger and heat exchanger bypass line will be performed. Testing, inspection, and analyses will be performed to determine the heat removal capability of the SCS heat exchanger.	2.a) Flow through the SCS heat exchanger and heat exchanger bypass line can be adjusted while maintaining a flow of no less than 5000 gpm per Division. Each SCS pump provides at least 400 feet of head at a flow rate no less than 5000 gpm. The heat removal capability of one SCS Division, as measured by the product of the service heat transfer coefficient and the effective heat transfer area of the SCS heat exchanger is no less than 1.38 x 10^6 BTU/hr -deg F
2.b) Each SCS Division has the heat removal capacity to cool the IRWST after design bases events or feed and bleed operation using the SIS and SDS.	2.b) Testing and analyses of the SCS to measure pump head and flow at the combined discharge of the SCS heat exchanger, with suction and return lines aligned to the IRWST, will be performed.	2.b) Each SCS pump develops at least 400 feet of head at a flow rate no less than 5000 gpm.

2.3.2
Comment No. 2

TABLE 2.12.2-2
REMOTE SHUTDOWN ROOM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the RSR is as shown on Figure 2.12.2-1.	1. Inspection of the as-built RSR configuration will be conducted.	1. For the components and equipment shown on Figure 2.12.2-1, the as-built RSR conforms with the Basic Configuration.
2. The RSR makes available the annunciators, displays and controls identified in Table 2.12.2-1.	2. Human Factors Engineering (HFE) availability verification inspection of the as-built RSR will be performed.	2.a) The as-built RSR makes available the annunciators, displays, and controls necessary to achieve and maintain prompt hot shutdown of the reactor.
		2.b) The as-built RSR provides capability for RSR operators to perform RSR tasks to achieve subsequent cold shutdown of the plant.
3. The RSR provides suitable workspace and environment for use by RSR operators.	3. HFE suitability inspection against verification criteria will be performed.	3. The RSR workspace and environment are determined to be suitable for use by RSR operators.
4. The RSR permits execution of RSR tasks performed by RSR operators to shutdown the plant and maintain safe shutdown conditions.	4. Testing and analysis against the validation criteria using a facility that physically represents the RSR configuration and dynamically represents the operating characteristics of the System 80+ design will be performed.	4. The test and analysis results demonstrate validation of RSR task execution by RSR operators to achieve and maintain safe shutdown conditions.
X. Controls exist in the RSR to stop the RCPs, trip the reactor, control the EFW steam-driven pump turbine speed, and to start and stop those other pumps, open and close those valves, and energize or de-energize those pressurizer heaters listed in Table 2.12.2-1.	X. Testing will be performed using the controls in the RSR.	X. Controls in the RSR operate to stop the RCPs, trip the reactor, control the EFW steam-driven pump turbine speed, and to start and stop those other pumps, open and close those valves, and energize or de-energize those pressurizer heaters listed in Table 2.12.2-1.

Resolves 2.3.2 SCS Comment No. 3 (SRXB)

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.3.2 (Shutdown Cooling System)

No.	Comments	Cat.	Resolution
1	CESSAR Figure 6.3.2-1A shows that SIAS or CSAS starts the SCS pumps. However, CDM Fig. 2.3.201 shows only the CSAS signal.	1	ABB-CE disagrees. Both an SIAS and a CSAS are provided to start the shutdown cooling pump when it is aligned for containment spray. However, only the pump start on CSAS, not on SIAS, is credited in the Chapter 6 and 15 safety and containment analyses. Therefore, only the pump start on CSAS is verified in Tier 1. **NRC Staff concurs.**
2	The heat removal capability of 1.38×10^6 Btu/hr for the SCS heat exchanger stated in the acceptance criteria for ITAAC 2a.. appears very low for the expected delta-T. The method of arriving at this number stated in the acceptance criteria is also incorrect unless the unit is changed to Btu/(hr).(deg F).	1	ABB-CE agrees. Acceptance criteria 2.a) will be revised to 1.38×10^6 Btu/hr-deg F, as shown in the attached markup. **NRC Staff concurs.**

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.3.2 (Shutdown Cooling System)

No.	Comments	Cat.	Resolution
3	General comment on all systems: There is no requirement for testing operability of equipment that are operable from the Remote Shutdown Panel (RSP) similar to testing from the MCR (for example, ITAAC 9b.). An ITAAC similar to 9b. should be including for testing from RSP either in each applicable system or in 2.12.2 Remote Shutdown Room.	1	Operability of equipment controls at the Remote Shutdown Panel will be included in ITAAC 2.12.2 Remote Shutdown Room, as shown in the attached markup. **NRC Staff concurs.**

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.3.3 (RCS Component Supports)

No.	Comments	Cat.	Resolution
1	CDM Figure 2.3.3-1 provides no details on the base plate slots as described in the design description. Revise ITAAC item 3 to verify the basic configuration conforms with the "design description" which includes both text and figures.	1	Agree. ITAAC Figure 2.3.3-1 is changed as shown on the marked revision page to be consistent with the design description and CESSAR-DC. No change to CESSAR-DC is required.

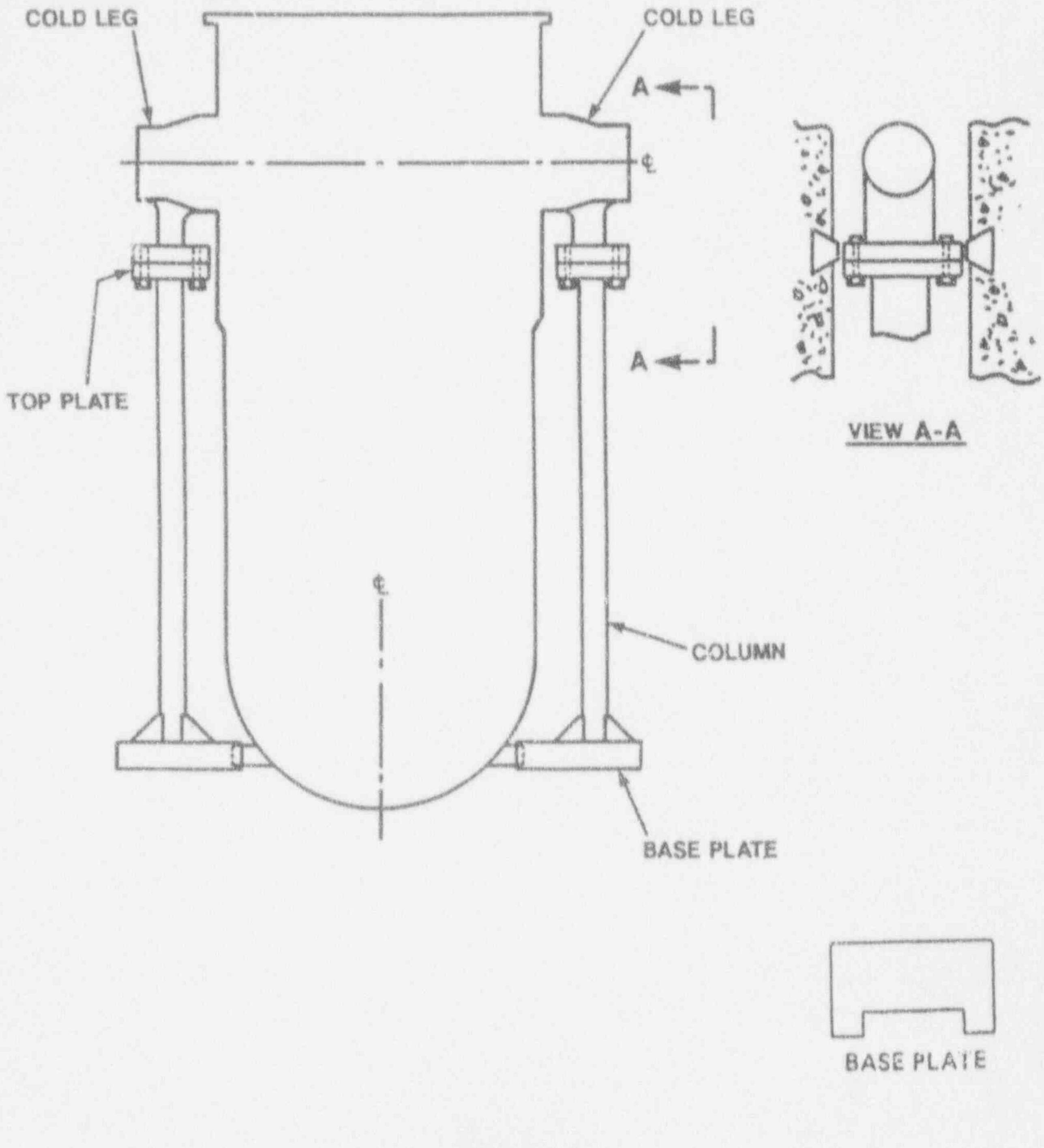
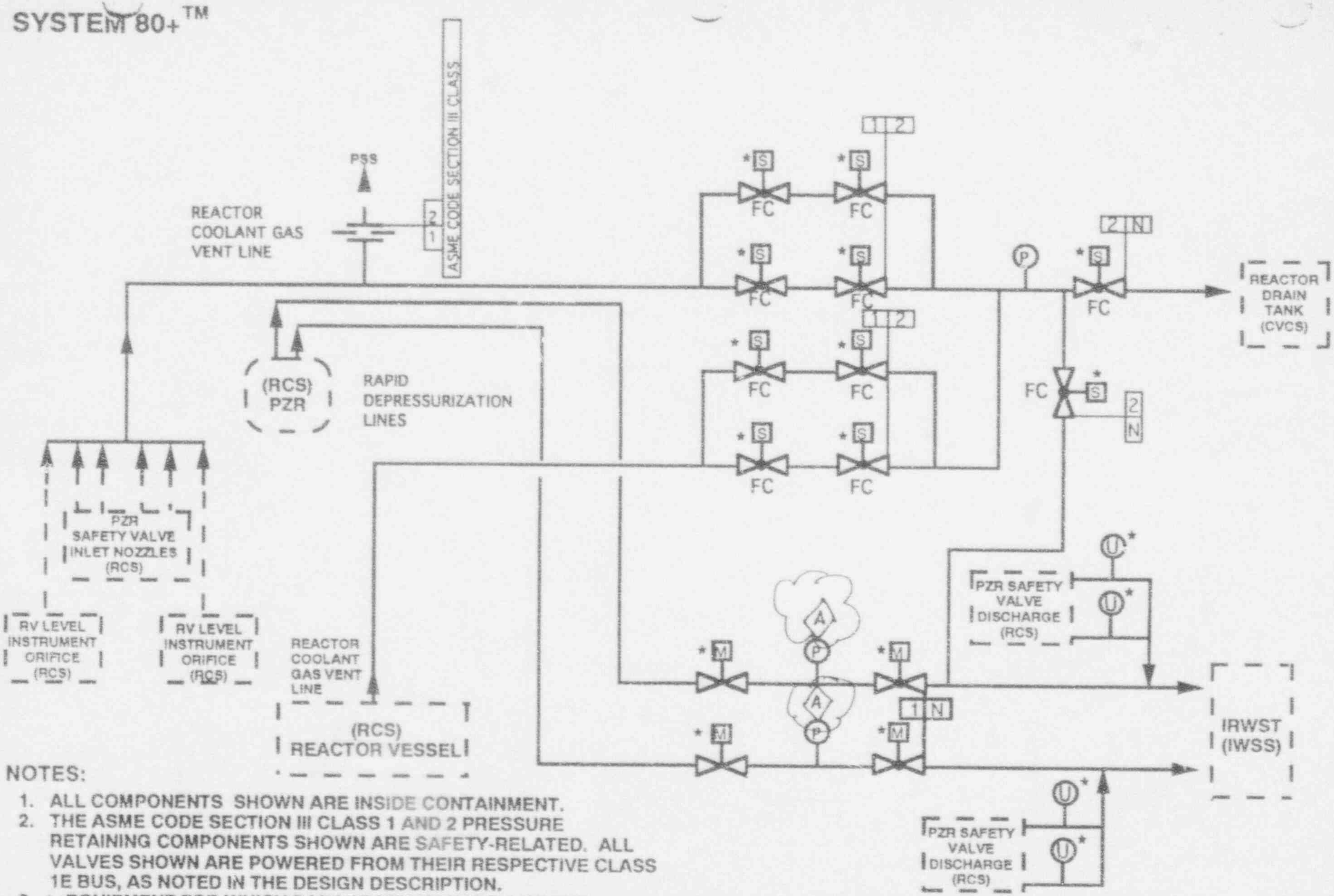


FIGURE 2.3.3-1
REACTOR VESSEL SUPPORTS

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.4.1 (Safety Depressurization System)

No.	Comments	Cat.	Resolution
1	Design Description and ITAAC 5c. state that alarms shown on Figure 2.4.1-1 are provided in the MCR. But the figure does not show any alarms.	1	ABB-CE agrees. Figure 2.4.1-1 will be revised to add alarms on the RDS line pressure instruments, as shown in the attached markup. **NRC Staff concurs.**
2	The temperature indicators on the PZR safety valve discharge lines should be shown on Fig. 2.4.1-1.	1	ABB-CE disagrees. The temperature instruments on the PZR safety valve discharge lines perform no safety-related function, nor are they required for post-accident monitoring by Reg. Guide 1.97. **NRC Staff concurs.**
3	The acceptance criteria in ITAAC 2 is only applicable to the pressurizer vent portion of the RCGVS according to Chapter 6.7.1.2.1 of CESSAR. A different criteria for the RVUH vent portion is given in the CESSAR. This should be reconciled.	1	ABB-CE disagrees. Per previous agreement with the Reactor Systems Branch, the RCS depressurization rate using the RCGVS RVUH vent path is not required in ITAAC. The success of the natural circulation cooldown analysis is not sensitive to the RVUH depressurization rate, and is therefore not as significant as the depressurization rate using the RCGVS PZR vent. Pre-operational testing and analyses will be performed for both vent paths as described in CESSAR DC Sections 6.7.4.1 and 14.2.12.1.39 to verify the RCS depressurization rates stated in Section 6.7.1.2.1 and assumed in the NCC analysis in Appendix 5D. **NRC Staff concurs.**



NOTES:

1. ALL COMPONENTS SHOWN ARE INSIDE CONTAINMENT.
2. THE ASME CODE SECTION III CLASS 1 AND 2 PRESSURE RETAINING COMPONENTS SHOWN ARE SAFETY-RELATED. ALL VALVES SHOWN ARE POWERED FROM THEIR RESPECTIVE CLASS 1E BUS, AS NOTED IN THE DESIGN DESCRIPTION.
3. * :EQUIPMENT FOR WHICH PARAGRAPH NUMBER 3 OF THE "VERIFICATION FOR BASIC CONFIGURATION FOR SYSTEMS" SECTION OF THE GENERAL PROVISIONS (SECTION 1.2) APPLIES.

2.4.1 Comment No. 1

FIGURE 2.4.1-1
SAFETY DEPRESSURIZATION SYSTEM

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.4.2 (Annulus Ventilation System)

No.	Comments	Cat.	Resolution
1	In CESSAR section 6.2.3.3, page 6.2-42, paragraph 4. absorber should be adsorber.	3	Agree. See markup of CESSAR-DC page 6.2-42.
2	The grilla on the CESSAR Figure 6.2.3-1 is not connected to the system duct.	3	Agree. See revision to the annulus ventilation figure.
3	From Acceptance Criteria 3, a 110 sec time requirement is identified. What is the basis and where is it discussed in the CESSAR.	1	Agree. See markup of CESSAR-DC page 14.2-202.

The system has no containment penetrations.

The system is 100% redundant, although ducting inside the annulus is shared.

The system has complete electrical separation between the two redundant trains. Each train is powered by its respective Class 1E Emergency Diesel Generator.

The Annulus Ventilation System is an engineered safety feature and is credited in analyzing the consequences of design basis accidents. No credit has been taken for the carbon ~~absorbers~~ adsorbers in analyzing the consequences of a design basis accidents.

6.2.3.4 Inspection and Testing Requirements

Test and inspections will be performed to assure and demonstrate the capability of components and the system to perform the assigned function in accordance with design criteria. Bypass leak paths will be tested by local leak rate tests as defined in Appendix J of 10 CFR 50.

6.2.3.4.1 Manufacturer Testing

The manufacturer will be required to verify by appropriate tests the following:

A. High Efficiency Filters:

Testing in compliance with Regulatory Guide 1.52. HEPA filters will be tested for efficiency, initially at the factory and at the USNRC Quality Assurance Station in accordance with MIL-STD-282.

B. Fan:

Certified head and flow characteristics.

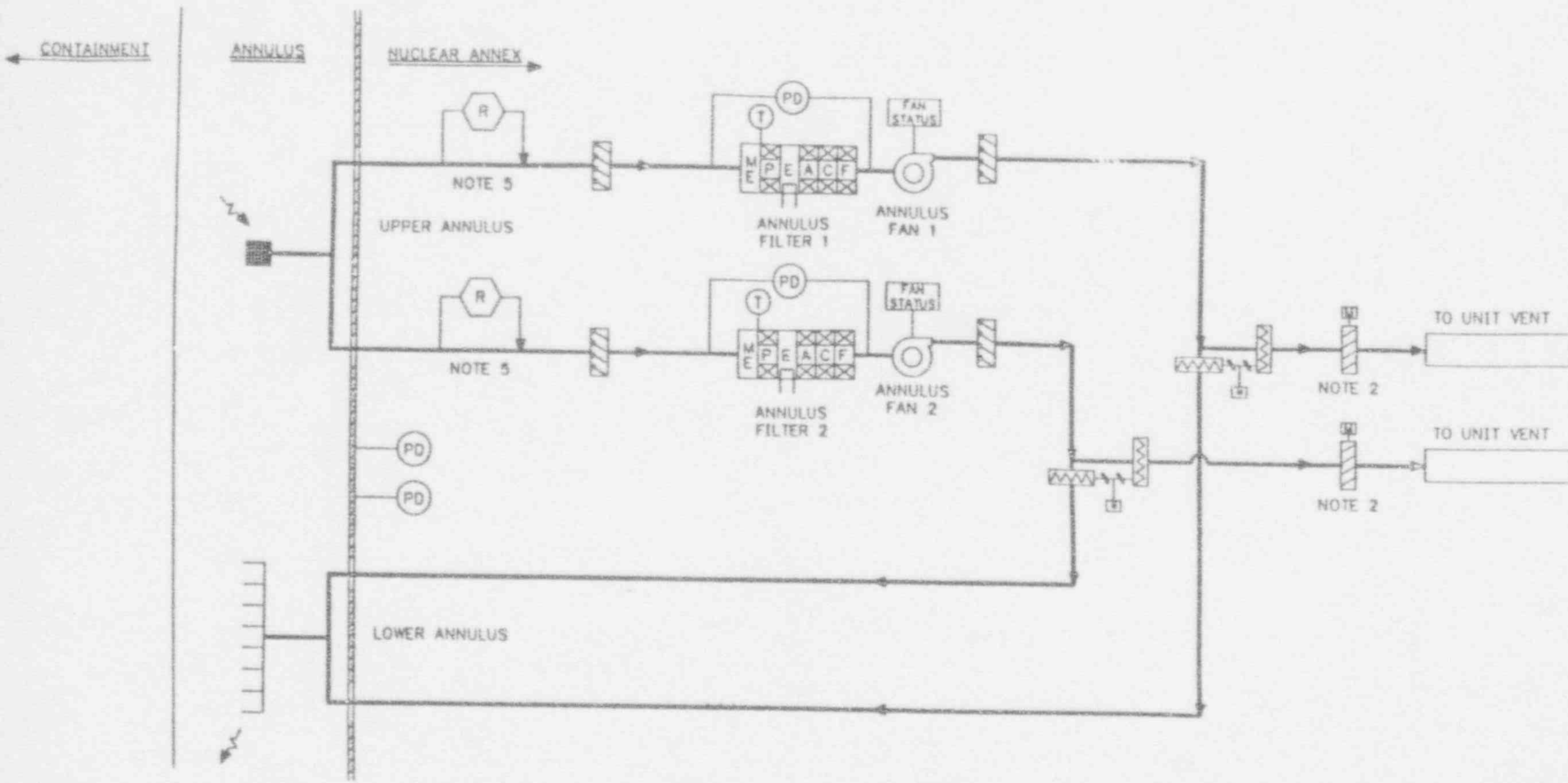
6.2.3.4.2 System Testing and Inspection

Operational testing will be performed prior to initial startup to demonstrate proper functioning of the system. Testing will include the following:

A. Leak tightness of components and system to be in accordance with ASME N510.

B. System functional test (flow, vacuum pressure)

C. HEPA filter efficiency test.



CONTAINMENT
ANNULUS
NUCLEAR ANNEX
R
NOTE 5
UPPER ANNULUS
T
PD
M
E
P
E
A
C
F
ANNULUS
FILTER 1
FAN
STATUS
ANNULUS
FAN 1
ANNULUS
FILTER 2
ANNULUS
FAN 2
PD
PD
LOWER ANNULUS
NOTE 2
TO UNIT VENT
NOTE 2
TO UNIT VENT

14.2.12.1.109 Annulus Ventilation System Test

1.0 OBJECTIVE

1.1 To demonstrate the capability of the Annulus Ventilation System to produce and maintain a negative pressure in the annulus following a LOCA and to minimize the release of radioisotopes following a LOCA by recirculating a large volume of filtered annulus air relative to the volume discharged for negative pressure maintenance.

2.0 PREREQUISITES

2.1 Construction activities on the containment wall and shield wall are complete with all penetrations sealed in place.

2.2 Construction activities on the Annulus Ventilation System have been completed.

2.3 Annulus Ventilation System instrumentation has been calibrated.

2.4 Support systems required for operation of the Annulus Ventilation System are complete and operational.

2.5 Test instrumentation is available and calibrated.

3.0 TEST METHOD

3.1 Verify all control logic, including response to ESFAS.

3.2 Verify the proper operation, failure mode, stroking speed, and position indication of control valves and dampers.

3.3 Demonstrate that the Annulus Ventilation System will achieve a negative pressure in the Annulus greater than or equal to 0.25 inches ~~of -0.5 in.~~ water ~~guage~~ gauge within 110 seconds of actuation.

3.4 Verify the proper operation of all protective devices, controls, interlocks, instrumentation, and alarms.

3.5 Verify design air flow for normal and emergency operation.

3.6 Perform filter and carbon adsorber efficiency test.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.4.3 (Combustible Gas Control)

No.	Comments	Cat.	Resolution
1	ITAAC item #5 requires connection of the hydrogen recombiner units to the Containment Hydrogen Recombiner System; however, the CDM does not contain this. Section 6.2.5.2.1 of the CESSAR DC gives a descreption of the connection. This needs to be added to the design description.	2	Disagree. Covered by basic configuration. The ITAAC figure shows connection. **NRC Staff concurs.**
2	ITAAC item #7 states that, "forty hydrogen igniters to be powered by one Division of Class 1E power sources..." CESSAR DC sections 6.2.5.1.2 (b) and 5.2.5.2.2.3 requires thirty four hydrogen igniters per Division of power. Amendment U, Dec. 31, 1993 updated the change; however the ITAAC was not.	1	Agree. Already incorporated in ITAAC.

CE 80+ ITAAC Independent Review Comments

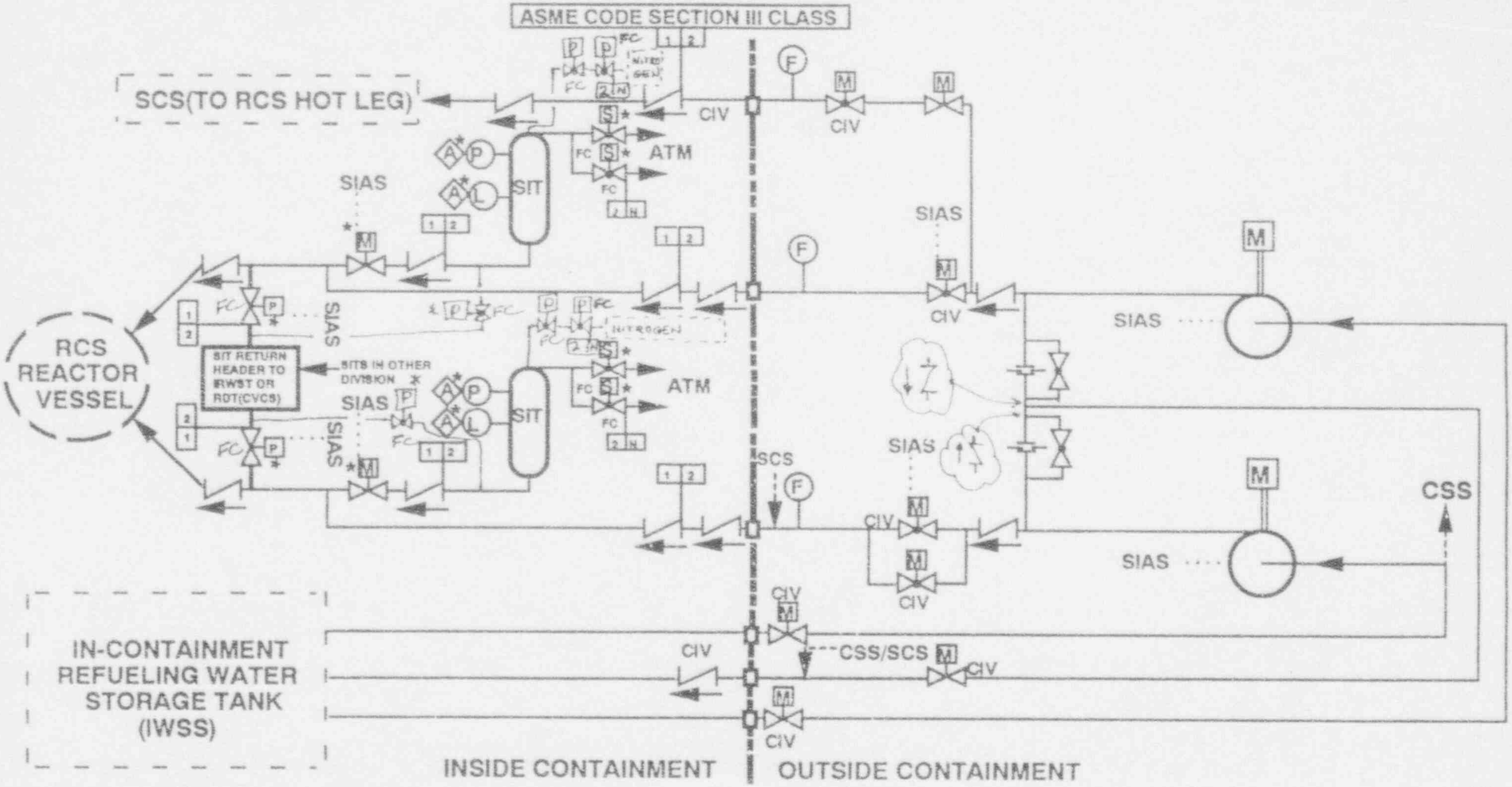
ITAAC No. 2.4.4 (Safety Injection System)

No.	Comments	Cat.	Resolution
1 (2)	The safety-related portion of the nitrogen cover gas piping and valves and the fill line to the SIT should be shown on Fig. 2.4.4-1.	1	ABB-CE agrees. The safety-related portion of the nitrogen cover gas connections to the SITs will be added to Figure 2.4.4-1, as shown on the attached markup. **NRC Staff concurs.**
2 (3)	ITAAC 2 should include verification of the minimum volume in the SIT used as input parameter for CESSAR Chapter 6 analysis. Also, the SIT low level alarm setpoint should be verified to assure that the required min. volume is available in the tank.	1	ABB-CE disagrees. The Chapter 6 and 15 analyses assume SIT water volumes based on the Tech Spec limits. Water level and alarm setpoints are operational considerations and not appropriate for ITAAC. **NRC Staff concurs.**
3 (4)	The SI pump differential pressures in ITAAC 2 acceptance criteria could not be found in CESSAR 6.3. The required flows at the DVI nozzle pressures used in the DBA analysis (for example, selected data points from CESSAR Table 6.3.3.3-1 or Table 6.3.2-5) should be specified as acceptance criteria. Specifying pump differential pressures without stating the limits on as-built system hydraulic resistance is incomplete.	1	ABB-CE disagrees. The SI pump differential pressures specified in ITAAC 2 correspond to the RCS pressures for zero injection flow in CESSAR DC Tables 6.3.2-5 and 6.3.3.3-1. With no injection flow (pump at minimum recirculation flow), there is no hydraulic resistance in the injection line. The elevation difference between the water source (IRWST) surface and the DVI nozzles is neglected.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.4.4 (Safety Injection System)

No.	Comments	Cat.	Resolution
3 (4)	(continued)		(continued) With no injection line flow or hydraulic resistance, the pump differential pressure is essentially equal to the RCS pressure minus the IRWST pressure (containment atmospheric). The runout flow of 980 to 1232 gpm does account for line resistance, and matches the analyses assumptions stated in CESSAR DC Tables 6.3.2-5 and 6.3.3.3-1. **NRC Staff concurs.**
4 (5)	Active valves such as, SIT fill and drain valves, miniflow line check valves, and SIT fill line containment isolation valves are not shown on Fig. 2.4.4-1.	1	ABB-CE agrees, with modification. The active SIT fill and drain valves and the miniflow line check valves will be added to Figure 2.4.4-1. The SIT fill line containment isolation valves are shown in ITAAC 2.4.5, Containment Isolation System, Table 2.4.5-2, item 29.



NOTES:
1. SAFETY-RELATED ELECTRICAL COMPONENTS AND EQUIPMENT SHOWN ON THIS FIGURE ARE CLASS 1E. ALARMS ARE NOT SAFETY-RELATED AND NOT CLASS 1E.
2. * EQUIPMENT FOR WHICH PARAGRAPH NUMBER 3 OF THE "VERIFICATION FOR BASIC CONFIGURATION FOR SYSTEMS" SECTION OF THE GENERAL PROVISIONS (SECTION 1.2) APPLIES.
3. THE ASME CODE SECTION III CLASS 1 AND 2 PRESSURE RETAINING COMPONENTS SHOWN ARE SAFETY-RELATED

FIGURE 2.4.4-1
SAFETY INJECTION SYSTEM
(ONE OF TWO DIVISIONS)

12-31-93

2.4.4 Comment Nos. 2+5

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.4.5 (Containment Isolation System (CIS)) Page 1 of 4

No.	Comments	Cat.	Resolution
1 (4)	Table 2.4.5-2 Items 13, 14, 15, 16 listed the check valves with "Note 4" which signifies a code class break. P&ID Figure 6.3.2-1C does not show such code class break. Suggest to delete Note 4.	1	Agree. Note 4 needs to reflect current CESSAR-DC consideration that these check valves are not containment isolation valves. Note 4 can just say: "Check valve is not a containment isolation valve." See markup of the ITAAC
2 (5)	Table 2.4.5-2 Items 17 & 18 are Arrangement 11 which showed the inside containment isolation valves as ASME Section III Code Class 2, whereas CESSAR Fig. 6.3.2-1C showed the same valves as Code Class 1 valves. Please resolve discrepancy.	1	Agree. This comment will be resolved by modifying CDM Figure 2.4.5-1 configuration 11. The class break (1/2) will be moved from the left side of the inside-containment remotely operated valve to the right side of the same valve, the 1/2 class break will be consistent with the P&ID. See markup of the ITAAC
3 (6)	Table 2.4.5-2 Items 19 & 20 corresponding to Figure 2.4.5-1 Arrangement 2: the inside containment check valve is shown as ASME Section III Code Class 2 whereas CESSAR Figure 6.3.2-1C is showing Code Class 1. Please resolve this discrepancy.	1	Agree. Comment is that for hot leg injection penetration, the configuration on CDM Figure 2.4.5-1 doesn't show the class break from 2 to 1 (towards containment). The valves in question are Class 1 valves. Therefore a new configuration will be added to Figure 2.4.5-1. See markup of the ITAAC

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.4.5 (Containment Isolation System (CIS))

No.	Comments	Cat.	Resolution
4 (8)	Table 2.4.5-2 Item 52 corresponds to CESSAR Table 6.2.4-1 Item 57, which is identified as a part of CESSAR Figure 9.3.2-1. This item cannot be located on the abovc figure.	1	Agree. Comment states that holdup volume penetration valves do not appear on CESSAR-DC Figure 9.3.2-1. Apparently, they do not. One valve (not numbered) appears cn Figure 9.3.2-2, however, it is not a complete P&ID. The reference to figure in CESSAR-DC Table 6.2.4-1 for this penetration will be deleted. See markup of Table 6.2.4-1.
5 (9)	CESSAR Table 6.2.4-1 Item 53 is shown as "INTENTIONALLY BLANK", whereas on Figure 9.3.2-1 its the Containment Penetration between valves SS-235 and SS-236.	1	Agree. Actually, penetration #53 was added to CESSAR-DC Figure 9.3.2-1, but never included in CESSAR-DC Table 6.2.4-1. See markup of Table 6.2.4-1.
6 (10)	Table 2.4.5-2 Items 53 to 58 inclusive corresponding to CESSAR Tale 6.2.4-1 Items 58 to 63 inclusive need a drawing/figure reference, currently none is available. Furthermore, valves SS-220 to SS-227 inclusive can be found on CESSAR Figure 9.3.2-1 but are not associated with any containment penetrations. Please resolve this confusion.	1	Disagree. ABB-CE does not intend to add flow diagrams for the SC system at this time. Currently, items 58, 59, 60, 61, 62, 63 show a "-", indicating there is no figure. Second part of this comment involves a mistake by the commenter. The referenced valves are SC valves, not SS valves, as the comment indicates. CESSAR-DC Table 6.2.4-1 is correct as is. **NRC Staff concurs.**

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.4.5 (Containment Isolation System (CIS))

No.	Comments	Cat.	Resolution
7 (11)	See attached pages 17 to 19 for typos	1	Agree. See markups.
8 (12)	Table 2.4.5-2: A) Item 90, this is Item 36 of CESSAR Table 6.2.4-1, and Containment Penetration 99 of Figure 6.8-3. B) Item 91, this is Item 38 of CESSAR Table 6.2.4-1, and Containment Penetration 98 of Figure 9.3.4-1 sh. 2 of 4. Please resolve these discrepancies.	1	A. Agree. On Cessar-DC Figure 6.8-3, penetration number "99" is incorrect. This will be changed to "36" to match Table 6.2.4-1. B. Agree. On CESSAR-DC Figure 9.3.4-1.2 penetration number "98" is incorrect. This will be changed to "38" to match Table 6.2.4-1. See markups.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.4.5 (Containment Isolation System (CIS))

No.	Comments	Cat.	Resolution
9 (13)	Table 2.4.5-2 item numbers corresponded to the Containment Penetration Numbers similar to CESSAR Table 6.2.4-1 up to and included Item 35. Item numbers greater than 35 are no longer in agreement. The deviation of higher numbers must be resolved so that the correspondence is reestablished for all the CDM item numbers to penetration numbers.	1	Disagree. The numbering correspondence cannot be achieved unless the "intentionally blank" entries are deleted from Table 6.2.4-1. Doing so will necessitate figure amendments. It was never intended that there be a one to one correlation between Table 2.4.5-2 and CESSAR-DC Table 6.2.4-1. The airlocks, equipment hatch, and fuel transfer tube flange are addressed by the Nuclear Island Structures ITAAC (CDM 2.1.1); however, they are addressed directly in CESSAR-DC Table 6.2.4-1 in the traditional manner. Some modifications to CDM Table 2.4.5-2 will be performed to ensure better numbering correspondence. Total consistency, however, cannot be achieved unless significant amendments to figures and tables are performed. A title will be added to Table 2.4.5-2. See markup.
10 (14)	A statement of purpose for Table 2.4.5-2 in the text portion of the CDM similar to that of Figure 2.4.5-1 and Table 2.4.5-1 is needed.	1	Disagree. Table 2.4.5-2 is included as acceptance criteria for ITAAC in Table 2.4.5-1. By agreement, Table 2.4.5-2 would not be referred to in the design description. **NRC Staff concurs.**

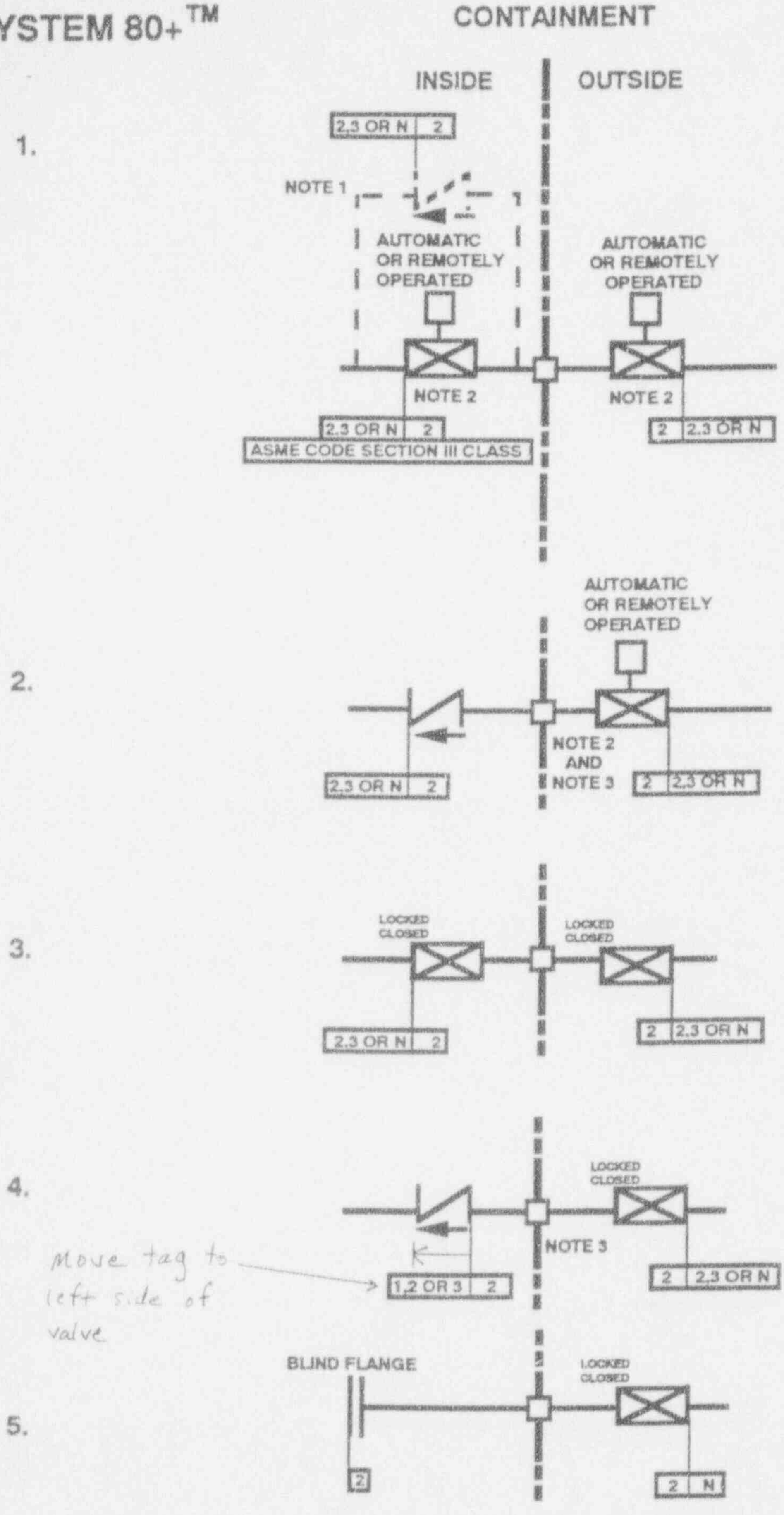


FIGURE 2.4.5-1 (PAGE 1 OF 4)
CONTAINMENT ISOLATION VALVE CONFIGURATION

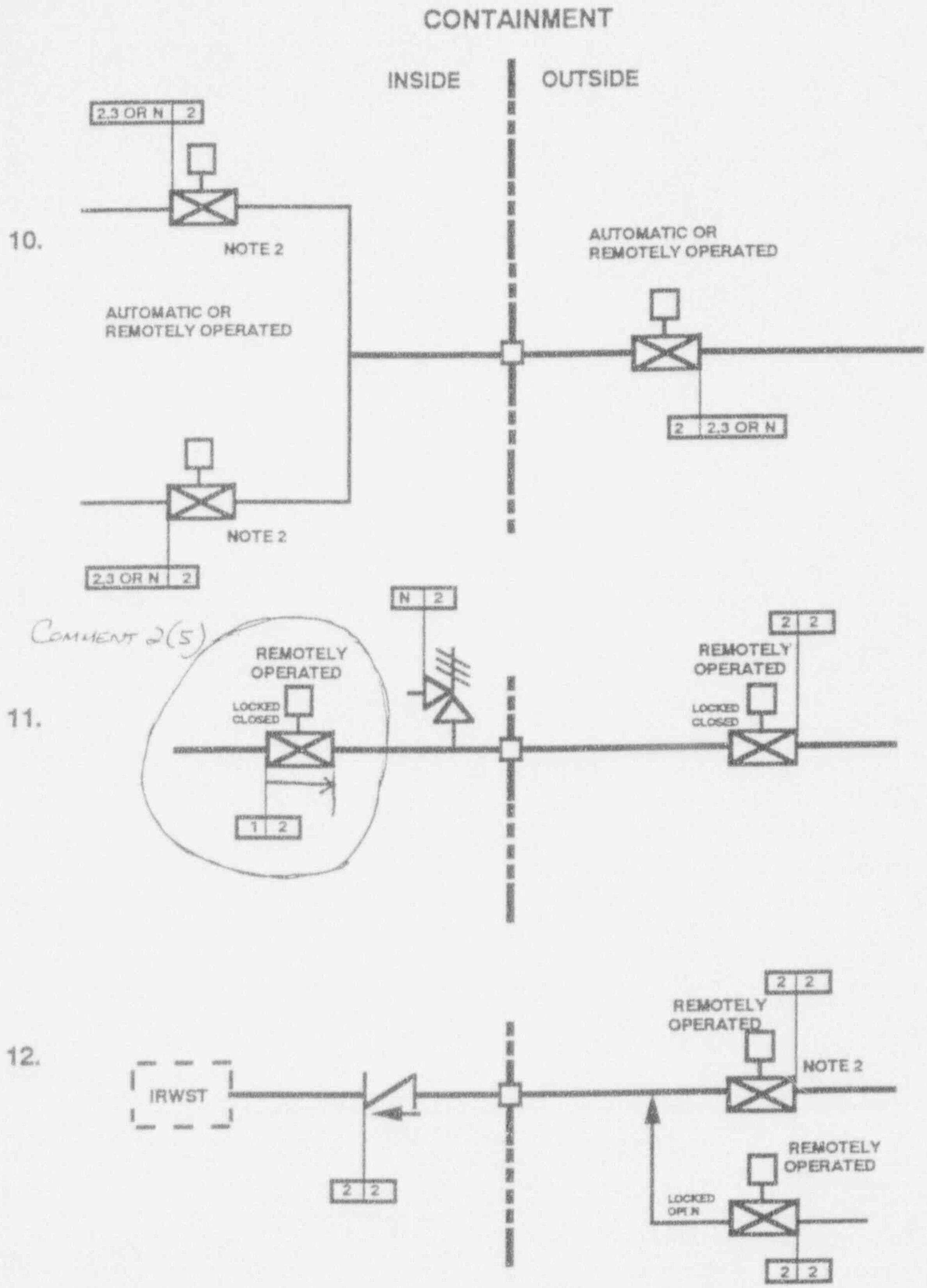
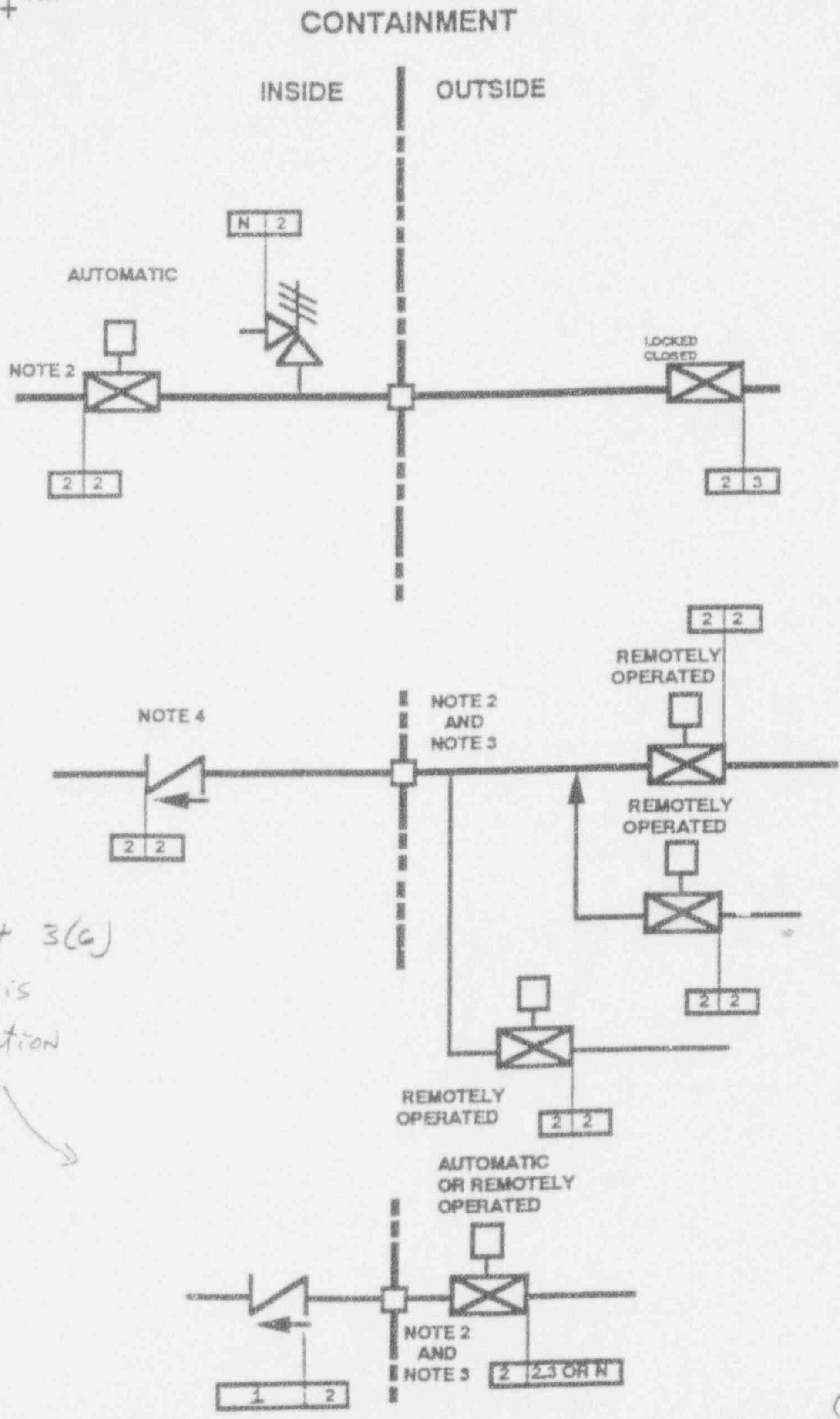


FIGURE 2.4.5-1 (PAGE 3 OF 4)
CONTAINMENT ISOLATION VALVE CONFIGURATION

13. 14. 15.



Comment 3(6)
Add this
Configuration

Comment 1(4)

NOTES:
1. LIQUID RELIEF VALVE CAN BE INCLUDED IN CONFIGURATION
2. VALVE CAN BE OPEN OR CLOSED IN NORMAL POSITION
3. FLOW ELEMENT/ROOT VALVES OMITTED FOR CLARITY, WHERE APPLICABLE.
4. CHECK VALVE IS NOT A CONTAINMENT ISOLATION VALVE

moved note

FIGURE 2.4.5-1 (PAGE 4 OF 4)
CONTAINMENT ISOLATION VALVE CONFIGURATION

TABLE 2.4.5-1

CONTAINMENT ISOLATION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the Containment isolation valves for piping which penetrates Containment is as shown on Figure 2.4.5-1; each Containment isolation valve arrangement is as shown in one of the configurations on the figure.	1. Inspection of the as-built CIS configuration will be conducted.	1. For the components and equipment shown on Figure 2.4.5-1 and specified in Table 2.4.5-2, the as-built CIS conforms with the specified Basic Configuration shown on Figure 2.4.5-1.
2. The ASME Code Section III valves shown on Figure 2.4.5-1 retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A pressure test will be performed on those components of the CIS required to be pressure tested by ASME Code Section III.	2. The results of the pressure test of ASME Code Section III components of the CIS specified in Table 2.4.5-2 conform with the pressure testing acceptance criteria in ASME Code Section III.
3.a) Electrically-powered Containment isolation valves are Class 1E. These Class 1E loads are powered from their respective Class 1E Divisions.	3.a) Testing will be performed on the Contain-ment isolation valves by providing a test signal in only one Class 1E Division at a time.	3.a) Within the CIS, a test signal exists only at the equipment powered from the Class 1E Division under test.
3.b) The Containment equipment hatch trolley receives Class 1E power.	3.b) Inspection of the as-built Containment equipment hatch trolley will be performed.	3.b) The Containment equipment hatch trolley receives Class 1E power.
3.c) Independence is provided between Class 1E Divisions and between Class 1E Divisions and non-Class 1E equipment in the CIS.	3.c) Inspection of the as-installed Class 1E Divisions in the CIS will be performed.	3.c) Physical separation exists between Class 1E Divisions in the CIS. Separation exists between Class 1E Divisions and non-Class 1E equipment in the CIS.

TABLE 2.4.5-1 (Continued)

CONTAINMENT ISOLATION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Redundant Containment isolation valves which require electrical power are powered from different Class 1E Divisions.	4. Testing will be performed on the Containment isolation valves by providing a test signal in only one Class 1E Division at a time.	4. Within the CIS, a test signal exists only at the equipment powered from the Class 1E Division under test.
5.a) Displays of CIS valve positions for remotely operated and automatic Containment isolation valves exist in the MCR or can be retrieved there.	5.a) Inspection for the existence or retrievability in the MCR of displays of Containment isolation valve positions will be performed.	5.a) Displays of CIS valve positions for remotely operated and automatic Containment isolation valves exist in the MCR or can be retrieved there.
5.b) Controls exist in the MCR to open and close CIS power operated valves.	5.b) Testing will be performed using the Containment isolation valve controls in the MCR.	5.b) Controls in the MCR operate to open and close power operated Containment isolation valves.
6.a) Only those valves required to close automatically for Containment isolation are closed by a CIAS.	6.a) Testing of the isolation function will be performed using a signal simulating CIAS.	6.a) Containment isolation valves respond to a signal simulating CIAS as specified in Table 2.4.5-2.
6.b) Containment isolation valves that receive a CIAS close within the time allocated to the function performed.	6.b) Testing of the closure times of automatically actuated Containment isolation valves will be performed using a signal that simulates a CIAS.	6.b) Containment isolation valves close upon receipt of a signal that simulates a CIAS in less than or equal to the time specified in Table 2.4.5-2, if specified.
6.c) Containment isolation valves that receive a CIAS, upon closure, do not reopen as a direct result of reset of the CIAS.	6.c) Following closure of Containment isolation valves on a signal that simulates a CIAS, tests will be performed to verify that the valves do not reopen when a signal that simulates the CIAS reset is applied.	6.c) Containment isolation valves, once closed by a signal that simulates a CIAS, do not reopen as a direct result of a signal that simulates resetting the CIAS.

TABLE 2.4.5-1 (Continued)

CONTAINMENT ISOLATION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. Pneumatic Containment isolation valves close upon loss of motive or control power to the valve.	7. Testing will be performed on each pneumatic Containment isolation valve to simulate a loss of motive power and a loss of control power.	7. Pneumatic Containment isolation valves close.
8. Motor-operated valves (MOVs) that receive a CIAS will close under differential pressure or fluid flow conditions, and under temperature conditions.	8. Testing to close MOVs that receive a CIAS will be conducted under preoperational differential pressure or fluid flow conditions, and under temperature conditions.	8. Each MOV that receives a CIAS closes.
9. Containment isolation check valves having an active safety function will close under system pressure, fluid flow conditions, or temperature conditions.	9. Testing of Containment isolation check valves will be conducted under system preoperational pressure, fluid flow conditions, or temperature conditions.	9. Each Containment isolation check valve specified in Table 2.4.5-2 closes.
10.a) Containment isolation valves required to close against containment atmosphere are designed to close against at least containment design pressure.	10.a) Inspection and analysis will be performed on Containment isolation valves required to close against containment atmosphere.	10.a) Reports exist which conclude that containment isolation valves required to close against containment atmosphere are designed to close against at least containment design pressure.
10.b) Containment isolation valves and piping between CIVs are designed for pressures at least equal to the containment design pressure.	10.b) Inspection and analysis of containment isolation valves and piping between CIVS will be performed.	10.b) Reports exist which conclude that containment isolation valves and piping between CIVs are designed for pressures at least equal to the containment design pressure.

TABL .4.5-2

CONTAINMENT PENETRATIONS

Repeat title for all Table 2.4.5-2 pages

Item No.	Service	(Note 1) Valve Arrangement	(Note 2) Closes On CIAS (Yes, No)	(Note 3) Maximum Valve Closure Time on CIAS
1	Main Steam Line #1 from Steam Generator #1	9	No	
	Remotely Operated			–
	Safety Valve			–
	Safety Valve			–
	Safety Valve			–
	Safety Valve			–
	Safety Valve			–
	Remotely Operated			–
	Remotely Operated			–
	Remotely Operated			–
	Manual Valve			–
	Manual Valve			–
2	Main Steam Line #2 from Steam Generator #1	9	No	
	Remotely Operated			–
	Safety Valve			–
	Safety Valve			–
	Safety Valve			–
	Safety Valve			–
	Safety Valve			–
	Remotely Operated			–
	Remotely Operated			–
	Remotely Operated			–
	Manual Valve			–
3	Main Steam Line #1 from Steam Generator #2	9	No	
	Remotely Operated			–
	Safety Valve			–
	Safety Valve			–
	Safety Valve			–
	Safety Valve			–
	Safety Valve			–
	Remotely Operated			–
	Remotely Operated			–
	Remotely Operated			–
	Manual Valve			–

TABLE 2.4.5-2 (Continued)

CONTAINMENT PENETRATIONS

Item No.	Service	(Note 1) Valve Arrangement	(Note 2) Closes On CIAS (Yes, No)	(Note 3) Maximum Valve Closure Time on CIAS
16	Safety Injection Pump #1 Discharge	14	No	
	Remotely Operated			–
	Remotely Operated			–
	Check Valve (Note 4)			–
	Remotely Operated			–
17	SCS Pump #2 Suction	11	No	
	Remotely Operated			–
	Relief Valve			–
	Remotely Operated			–
18	SCS Pump #1 Suction	11	No	
	Remotely Operated	COMMENT 3(c)		–
	Relief Valve			–
	Remotely Operated			–
19	Hot Leg Injection Loop #2	~~2~~ 15	No	
	Remotely Operated			–
	Check Valve			–
20	Hot Leg Injection Loop #1	~~2~~ 15	No	
	Remotely Operated			–
	Check Valve			–
21	Containment Spray Pump #2 Discharge	2	No	
	Remotely Operated			–
	Check Valve			–
22	Containment Spray Pump #1 Discharge	2	No	
	Remotely Operated			–
	Check Valve			–

TABLE 2.4.5-2 (Continued)

CONTAINMENT PENETRATIONS

Item No.	Service	(Note 1) Valve Arrangement	(Note 2) Closes On CIAS (Yes, No)	(Note 3) Maximum Valve Closure Time on CIAS
54	Steam Generator #1 Hot Leg Sample	1	No	
	Remotely Operated			–
	Remotely Operated			–
55	Steam Generator #1 Downcomer Sample	1	No	
	Remotely Operated			–
	Remotely Operated			–
56	Steam Generator #2 Cold Leg Sample	1	No	
	Remotely Operated			–
	Remotely Operated			–
57	Steam Generator #2 Hot Leg Sample	1	No	
	Remotely Operated			–
	Remotely Operated			–
58	Steam Generator #2 Downcomer Sample	1	No	
	Remotely Operated			–
	Remotely Operated			–
59	High Volume Containment Purge System Supply #1	1	~~No~~ yes	
	Remotely Operated			60 sec
	Remotely Operated			60 sec
60	High Volume Containment Purge System Supply #2	1	Yes	
	Remotely Operated			60 sec
	Remotely Operated			60 sec
61	High Volume Containment Purge System Exhaust #1	1	Yes	
	Remotely Operated			60 sec
	Remotely Operated			60 sec

TABLE 2.4.5-2 (Continued)
CONTAINMENT PENETRATIONS

Item No.	Service	(Note 1) Valve Arrangement	(Note 2) Closes On CIAS (Yes, No)	(Note 3) Maximum Valve Closure Time on CIAS
86	Division 1 Hydrogen Recombiner Discharge to Containment Remotely Operated Check Valve	2	Yes	 60 sec –
87	Division 2 Hydrogen Recombiner Discharge to Containment Remotely Operated Check Valve	2	Yes	 60 sec –
88	Steam Generator Wet Layup Recirculation Return to Steam Generator #1 Manual Valve Check Valve	4	No	 – –
89	Steam Generator Wet Layup Recirculation Return to Steam Generator #2 Manual Valve Check Valve	4	No	 – –
90	SI IRWST Boron Recovery Supply to CVCS Remotely Operated Remotely Operated	1	Yes	 60 sec 60 sec
91	CVCS IRWST Boron Recovery Return Remotely Operated Check Valve	2	Yes	 60 sec –

NOTES:

1. Valve arrangements are in accordance with the Containment isolation valve configurations shown on Figure 2.4.5-1.
2. Paragraph Number 3 of the General Provisions (Section 1.2) applies to Containment isolation valves which receive a CIAS.
3. A dash (–) denotes NOT APPLICABLE
4. Not a containment isolation valve. ~~shown only to establish ASME Code Section III class break location.~~

TABLE 6.2.4-1 (Cont'd)

(Sheet 8 of 15)

CONTAINMENT ISOLATION VALVE AND ACTUATOR DATA

Comment 5(9)

Item No.	Service	Valve No.	Figure No.	(Note 18) Valve Type	Location Relative to Containment	Flow Direction Relative to Containment	(Note 1) Valve Arrangement (GDC)	(Note 5) Valve Position Normal	Fail Safe	Shutdown	Accident	(Note 4) Actuator Type	(Note 2) Actuation Signal	(Note 3) Type	Vent and Drain for Type A Test	(Note 6) Type C Test	Justification for Not Testing	(Note 17) Essential/ Nonessential
50	Refueling Pool Cleanup Suction Line	PC-258 PC-257	9.1-3	Packless Packless	Outside Inside	Out	21 (56)	LC LC	-	O/C O/C	LC LC	HW HW	-	M M	Yes	Yes		Nonessential
51	Refueling Pool Cleanup Return Header	PC-291 PC-292	9.1-3	Packless Packless	Outside Inside	In	21 (56)	LC LC	-	O/C O/C	LC LC	HW HW	-	M M	Yes	Yes		Nonessential
52	Fuel Transfer Tube Quick Closure Hatch			DBLE Seal Bl. Flange	Inside	None	22 (N/A)	C		C	C			M	No	No	Note 10	Nonessential
53	SIT Sample Line	SS-236 SS-235	9.3.2-1	Globe* Globe*	Outside Inside	Out	20 (55)	C C	C C	C C	C C	S S	CIAS CIAS	A,R,M A,R,M	Yes	Yes	—	Nonessential
54	Pressurizer Liquid Sample Line	SS-201 SS-204	9.3.2-1	Globe* Globe*	Outside Inside	Out	20 (55)	C C	C C	C C	C C	S S	CIAS CIAS	A,R,M A,R,M	Yes	Yes		Nonessential
55	Pressurizer Steam Space Sample Line	SS-202 SS-205	9.3.2-1	Globe* Globe*	Outside Inside	Out	20 (55)	C C	C C	C C	C C	S S	CIAS CIAS	A,R,M A,R,M	Yes	Yes		Nonessential
56	Hot Leg Sample Line	SS-200 SS-203	9.3.2-1	Globe* Globe*	Outside Inside	Out	20 (55)	C C	C C	C C	C C	S S	CIAS CIAS	A,R,M A,R,M	Yes	Yes		Nonessential
57	Holdup Volume Tank Sample Line	SS-208 SS-210 SS-211	~~9.3.2-1~~	Globe* Globe* Globe*	Outside Inside Inside	Out	24 (56)	C C C	C C C	C C C	C C C	S S S	CIAS CIAS CIAS	A,R,M A,R,M A,R,M	Yes	Yes		Nonessential
58	Steam Generator #1 Cold Leg Sample	SC-219 SC-204		Globe* Globe*	Outside Inside	Out	27 (57)	O O	C C	C C	C C	S S	MSIS/EFAS/AFAS MSIS/EFAS/AFAS	A,R,M A,R,M	No	No	Note 7	Nonessential
59	Steam Generator #1 Hot Leg Sample	SC-228 SC-211		Globe* Globe*	Outside Inside	Out	27 (57)	O O	C C	C C	C C	S S	MSIS/EFAS/AFAS MSIS/EFAS/AFAS	A,R,M A,R,M	No	No	Note 7	Nonessential
60	Steam Generator #1 Downcomer Sample	SC-221 SC-220		Globe* Globe*	Outside Inside	Out	27 (57)	O O	C C	C C	C C	S S	MSIS/EFAS/AFAS MSIS/EFAS/AFAS	A,R,M A,R,M	No	No	Note 7	Nonessential

* Maximum valve closure time on CIAS is 60 sec.

Comment 4(8)

TABLE 6.2.4-1 (Cont'd)

(Sheet 12 of 15)

CONTAINMENT ISOLATION VALVE AND ACTUATOR DATA

Item No.	Service	Valve No.	Figure No.	(Note 18) Valve Type	Location Relative to Containment	Flow Direction Relative to Containment	(Note 1) Valve Arrangement (GDC)	(Note 5) Valve Position Normal	Fail Safe	Shutdown	Accident	(Note 4) Actuator Type	(Note 2) Actuation Signal	(Note 3) Type	Vent and Drain for Type A Test	(Note 6) Type C Test	Justification for Not Testing	(Note 17) Essential/ Nonessential
90	Personnel Airlock #2 Equalization Line	-	-	Check Check	Outside Inside	None	(N/A)	C C		C C	C C				Yes	Yes		N/A
91	Containment Sump Pump Discharge Line	-		Gate* Gate*	Outside Inside	Out	14 (56)	O O	AI AI	O O	C C	P P	CIAS/HRAS CIAS/HRAS	A,R,M A,R,M	Yes	Yes		Nonessential
92	Containment Ventilation Units' Condensate Drain Header	- - -	-	Gate* Gate* Check	Outside Inside Inside	Out	16 (56)	O O C	AI AI -	O O C	C C C	E E	CIAS/HRAS CIAS/HRAS	A,R,M A,R,M	Yes	Yes		Nonessential
93	Reactor Drain Tank Gas Space to GWMS	-	-	Globe* Globe*	Outside Inside	In/Out	25 (56)	O O	AI AI	O O	C C	E E	CIAS CIAS	A,R,M A,R,M	Yes	Yes		Nonessential
94	Decontamination Line	-	-	Globe Globe	Outside Inside	In	21 (56)	LC LC		LC LC	LC LC	HW HW		M M	Yes	Yes		Nonessential
95	Division 1 Hydrogen Recombiner Suction from Containment	- -	6.2.5-1	Globe* Globe*	Outside Inside	Out	23 (56)	C C	AI AI	C C	O/C O/C	E E	CIAS CIAS	A,R,M A,R,M	Yes	Yes		Essential
96	Division 2 Hydrogen Recombiner Suction from Containment	- -	6.2.5-1	Globe* Globe*	Outside Inside	Out	23 (56)	C C	AI AI	C C	O/C O/C	E E	CIAS CIAS	A,R,M A,R,M	Yes	Yes	-	Essential
97	Division 1 Hydrogen Recombiner Discharge to Containment	- -	~~6.2.6-1~~ 6.2.5-1	Globe* Check	Outside Inside	In	4 (56)	C C	AI -	C C	O/C O/C	E -	CIAS CIAS	A,R,M -	Yes	Yes		Essential
98	Division 2 Hydrogen Recombiner Discharge to Containment	- -	~~6.2.6-1~~ 6.2.5-1	Globe* Check	Outside Inside	In	4 (56)	C C	AI -	C C	O/C O/C	E -	CIAS CIAS	A,R,M -	Yes	Yes		Essential

* Maximum valve closure time on CIAS is 60 sec.

TABLE 6.2.4-1 (Cont'd)

(Sheet 13 of 15)

CONTAINMENT ISOLATION VALVE AND ACTUATOR DATA

Item No.	Service	Valve No.	Figure No.	(Note 18) Valve Type	Location Relative to Containment	Flow Direction Relative to Containment	(Note 1) Valve Arrangement (GDC)	(Note 5) Valve Position Normal	Fail Safe	Shutdown	Accident	(Note 4) Actuator Type	(Note 2) Actuation Signal	(Note 3) Type	Vent and Drain for Type A Test	(Note 6) Type C Test	Justification for Not Testing	(Note 17) Essential/ Nonessential
99	Steam Generator Wet Layup Recirculation Return to Steam Generator #1	- -	-	Gate Check	Outside Inside	In	9 (57)	LC C	- -	O/C O/C	LC C	HW -	- -	M -	No	No	Note 7	Nonessential
100	Steam Generator Wet Layup Recirculation Return to Steam Generator #2	- -	-	Gate Check	Outside Inside	In	9 (57)	LC C	- -	O/C O/C	LC C	HW -	- -	M -	No	No	Note 7	Nonessential

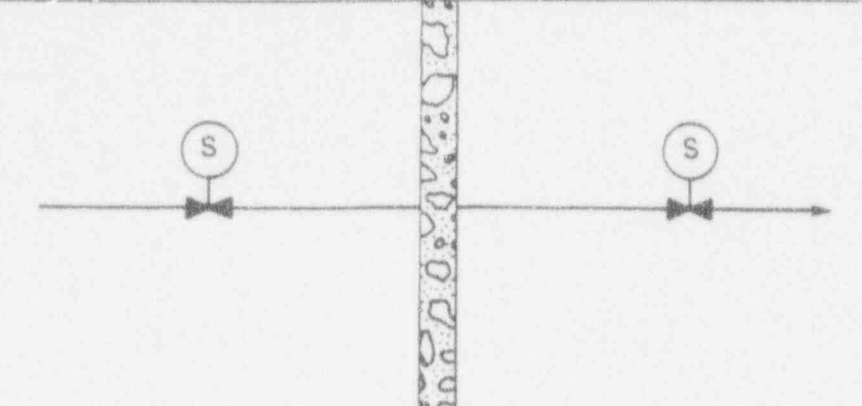
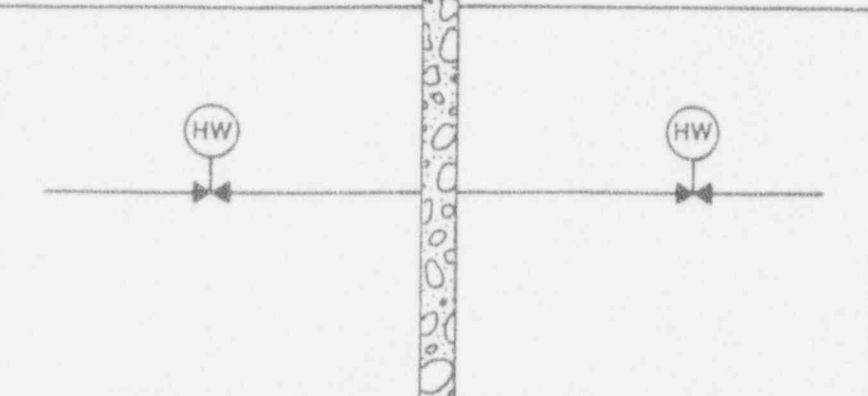
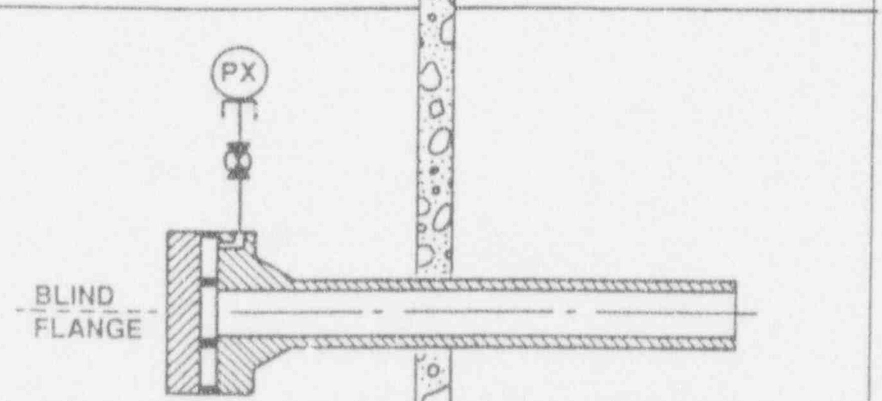
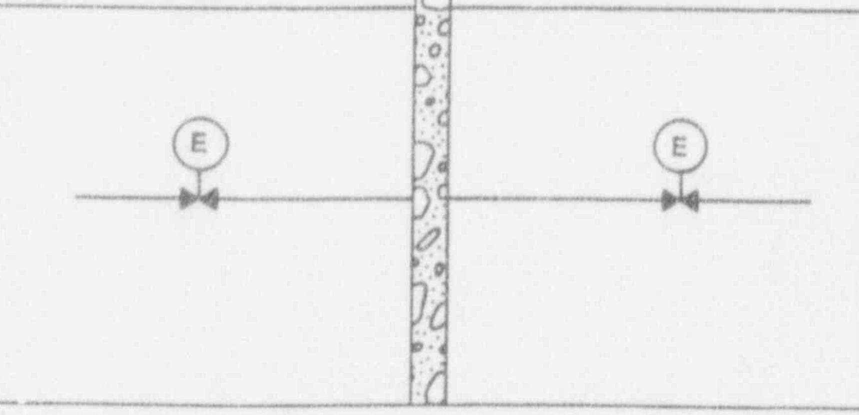
NOTES:

1. Valve arrangements are shown in Figure 6.2.4-1. The COL Applicant is to provide pipe sizes and distances of isolation valve from containment penetration.

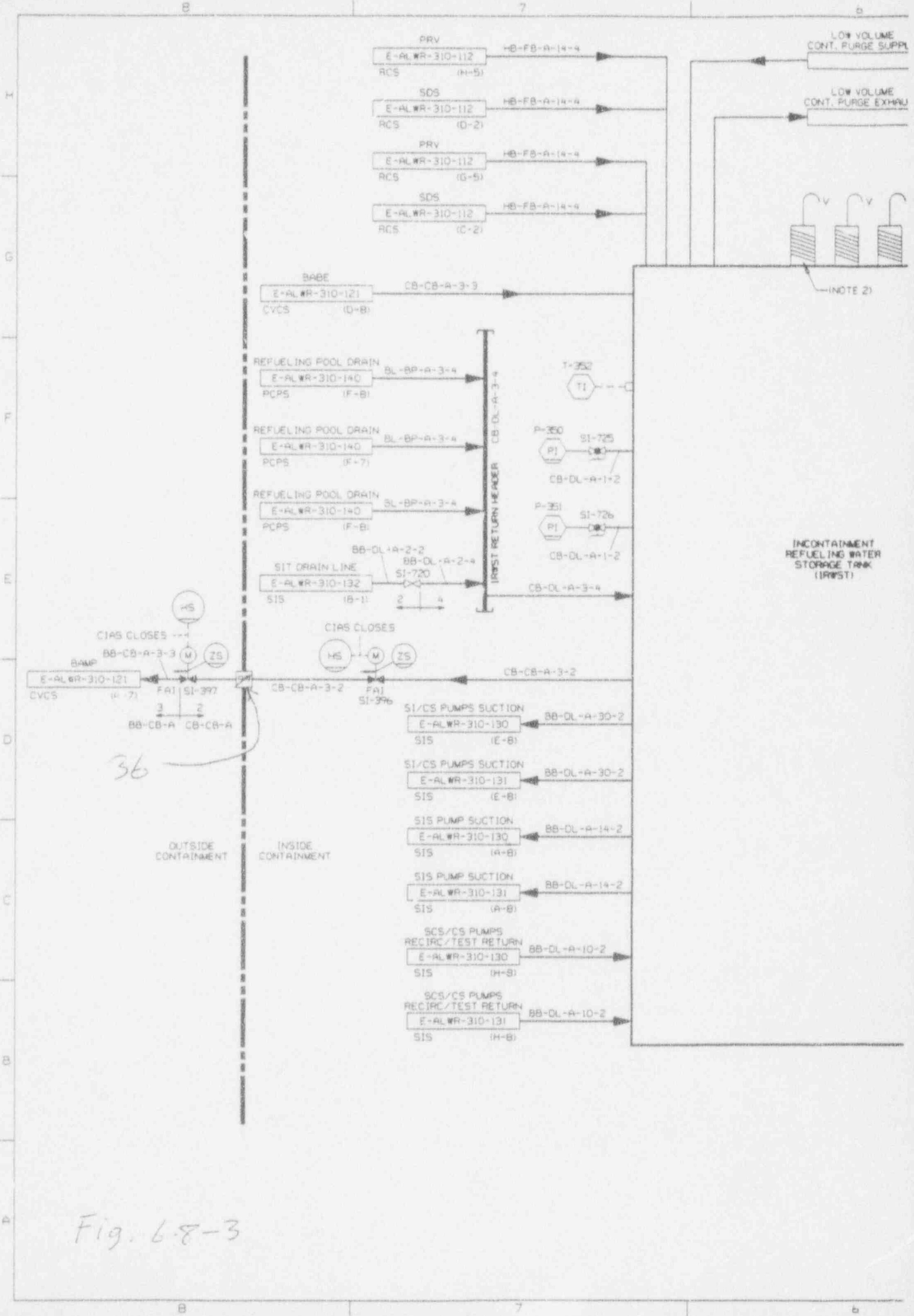
2. Definition of Actuation Signals

CIAS - Containment Isolation Actuation Signal
CSAS - Containment Spray Actuation Signal
MSIS - Main Steam Isolation Signal
EFAS - Emergency Feedwater Actuation Signal
AFAS - Alternate Feedwater Actuation Signal
HRAS - High Radiation Actuation Signal
HHAS - High Humidity Actuation Signal
SIAS - Safety Injection Actuation Signal
CCWLLSTAS - Component Cooling Water Low-Low Surge Tank Actuation Signal

All of these signals are Engineered Safety Features Actuation Signals, except HRAS, HHAS, CCWLLSTAS, and AFAS. Valves receiving ESF signals are classified as ESF valves.

VALVE ARRANGEMENT NO. (GDC)	INSIDE CONTAINMENT	OUTSIDE CONTAINMENT	ITEM NO.
20 (55)			53 54, 55, 56
21 (56)			50, 51, 81, 94
22 (N/A)			52, 86
23 (56)			36, 85, 95, 96

Amendment R
July 30, 1993



PRV
E-AL#R-310-112
RCS (H-5)
HB-FB-A-14-4
SDS
E-AL#R-310-112
RCS (D-2)
HB-FB-A-14-4
PRV
E-AL#R-310-112
RCS (G-5)
HB-FB-A-14-4
SDS
E-AL#R-310-112
RCS (C-2)
HB-FB-A-14-4
LOW VOLUME CONT. PURGE SUPPL
LOW VOLUME CONT. PURGE EXHAU
V
V
(NOTE 2)
BABE
E-AL#R-310-121
CVCS (D-8)
CB-CB-A-3-3
REFUELING POOL DRAIN
E-AL#R-310-140
PCPS (F-8)
BL-BP-A-3-4
REFUELING POOL DRAIN
E-AL#R-310-140
PCPS (F-7)
BL-BP-A-3-4
REFUELING POOL DRAIN
E-AL#R-310-140
PCPS (F-8)
BL-BP-A-3-4
IRWST RETURN HEADER
CB-DL-A-3-4
T-352
TI
P-350
PI
SI-725
CB-DL-A-1-2
P-351
PI
SI-726
CB-DL-A-1-2
SIT DRAIN LINE
E-AL#R-310-132
SIS (B-1)
BB-DL-A-2-2
BB-DL-A-2-4
SI-720
2
4
CB-DL-A-3-4
INCONTAINMENT REFUELING WATER STORAGE TANK (IRWST)
HS
CIAS CLOSES
BB-CB-A-3-3
M
ZS
BAMP
E-AL#R-310-121
CVCS (F-7)
FAI
SI-397
3
2
BB-CB-A
CB-CB-A
CB-CB-A-3-2
CIAS CLOSES
HS
M
ZS
FAI
SI-396
CB-CB-A-3-2
36
SI/CS PUMPS SUCTION
E-AL#R-310-130
SIS (E-8)
BB-DL-A-30-2
SI/CS PUMPS SUCTION
E-AL#R-310-131
SIS (E-8)
BB-DL-A-30-2
SIS PUMP SUCTION
E-AL#R-310-130
SIS (A-8)
BB-DL-A-14-2
SIS PUMP SUCTION
E-AL#R-310-131
SIS (A-8)
BB-DL-A-14-2
SCS/CS PUMPS RECIRC/TEST RETURN
E-AL#R-310-130
SIS (H-8)
BB-DL-A-10-2
SCS/CS PUMPS RECIRC/TEST RETURN
E-AL#R-310-131
SIS (H-8)
BB-DL-A-10-2
OUTSIDE CONTAINMENT
INSIDE CONTAINMENT

Fig. 6.8-3

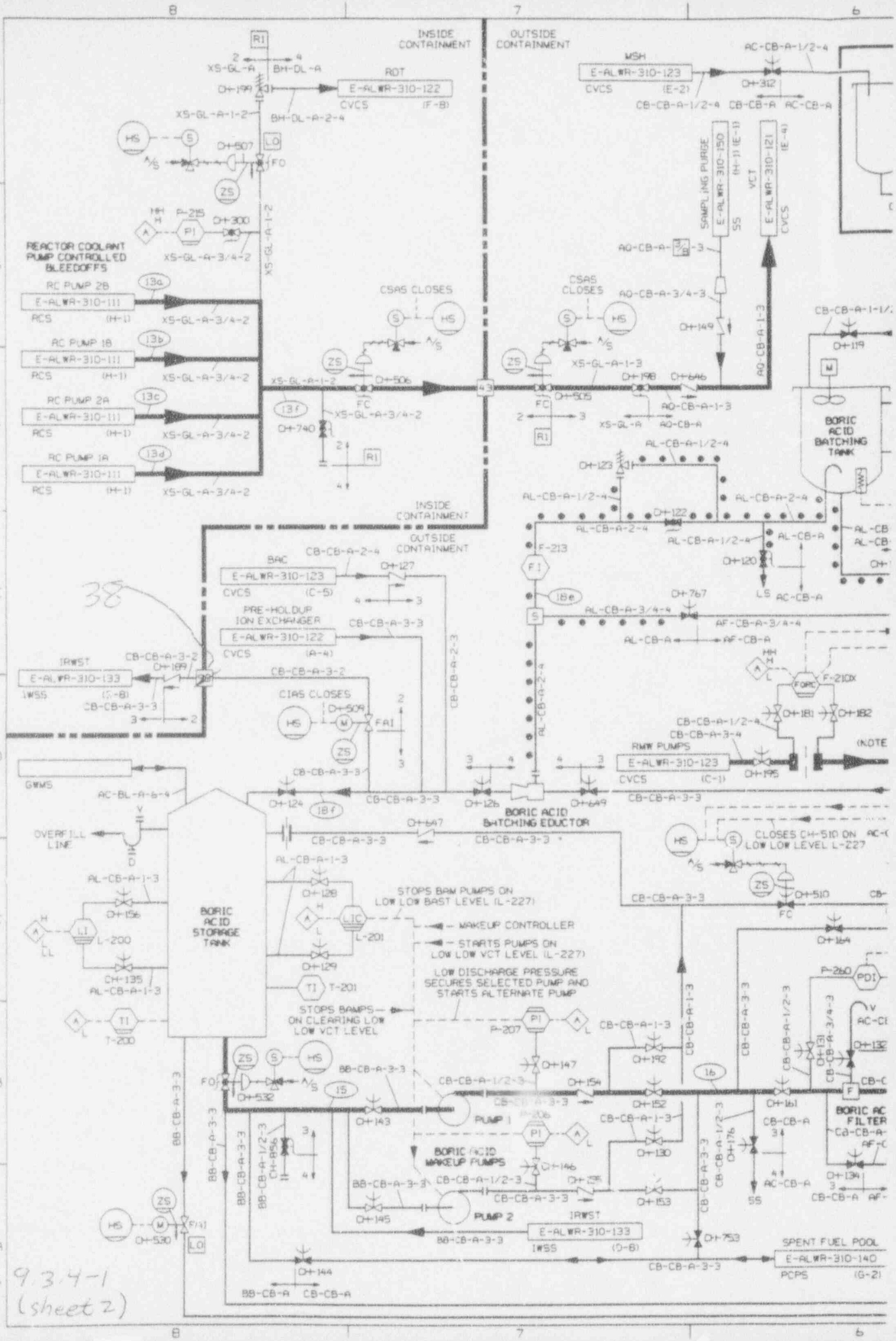


Fig. 9.3.4-1
(sheet 2)

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.4.6 (Containment Spray System (CSS))

No.	Comments	Cat.	Resolution
1	Page 2, 6th paragraph and Figure 2.4.6-1 showed that "The CSS pumps are started upon receipt of a containment spray actuation signal (CSAS),---.", whereas in various CESSAR sections (6.5.1.1, 7.3.1.1.10.2, 7.3.2.2.1) and P&ID Figure 6.3.2-1 showed that "The containment spray pumps starts upon the receipt of a Safety Injection Actuation Signal (SIAS) or a Containment Spray Actuation Signal (CSAS)." Please resolve this discrepancy.	1	ABB-CE disagrees. Both an SIAS and a CSAS are provided to start the containment spray pump. However, only the pump start on CSAS, not on SIAS, is credited in the Chapter 6 and 15 safety and containment analyses. Therefore, only the pump start on CSAS is verified in Tier 1. **NRC Staff concurs.**
2	Page 2 paragraph 7 discussed MOVs with active safety function. Does this statement apply to all MOVs of Figure 2.4.6-1? Please confirm.	1	ABB-CE agrees. All Containment Spray System MOVs shown on Figure 2.4.6-1 perform an active safety function. CESSAR DC Table 3.9-4 will be revised to add containment spray IRWST recirculation line isolation valves SI-686 and SI-696, as shown on the attached markup. **NRC Staff concurs.**

2.4.6 CSS Comment No. 2

SI-686	CSS 1 IRWST Recirculation Line Isolation	Gate	2	Motor
SI-696	CSS 2 " " " "	"	"	"

TABLE 3.9-4 (Cont'd)

(Sheet 5 of 7)

SEISMIC I ACTIVE VALVES

VALVE NO.	SYSTEM NAME (safety function)	VALVE TYPE	ASME SECTION III CODE CLASS	ACTUATOR TYPE
SI 614	Safety Injection Sys. (Operate)	Check	1	None
SI 422,423	Shutdown Cooling Sys. (Operate)	Relief	2	None
SI 305	IRWST Isolation	Gate	2	Motor
SI 304	IRWST Isolation	Gate	2	Motor
SI 309	IRWST Isolation	Gate	2	Motor
SI 308	IRWST Isolation	Gate	2	Motor
SI 300,301	CS/SCS IRWST Recirculation Isolation	Gate	2	Motor
SI 302,303	SI IRWST Recirculation Isolation	Gate	2	Motor
SI 695	CS Header 2 Block Valve	Gate	2	Motor
SI 687	CS Header 1 Block Valve	Gate	2	Motor
SI 657	CSS 1 IRWST Recirculation Line Flow Control	Globe	2	Motor
SI 658	CSS 2 IRWST Recirculation Line Flow Control	Globe	2	Motor
SI 314	SCS 1 IRWST Recirculation Line Flow Control	Globe	2	Motor
SI 315	SCS 2 IRWST Recirculation Line Flow Control	Globe	2	Motor
SI 688	SCS 1 IRWST Recirculation Line Isolation	Gate	2	Motor
SI 693	SCS 2 IRWST Recirculation Line Isolation	Gate	2	Motor
SI 682	SIT Fill Line Isolation	Relief	2	None
SI 390	Cavity Flooding Sys. (Open)	Gate	2	Motor
SI 391	Cavity Flooding Sys. (Open)	Gate	2	Motor
SI 392	Cavity Flooding Sys. (Open)	Gate	2	Motor
SI 393	Cavity Flooding Sys. (Open)	Gate	2	Motor
SI 394	Cavity Flooding Sys. (Open)	Gate	2	Motor
SI 395	Cavity Flooding Sys. (Open)	Gate	2	Motor
SI 396,397	Safety Injection Sys. (Close)	Gate	2	Motor
CH 189	CVCS Makeup to IRWST Check (Close)	Check	2	None

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.1 (Reactor Protection System)

No.	Comments	Cat.	Resolution
1	Figure 2.5.1-2, "PPS Interconnection" does not clearly specify 2 coincidence/bistable processors per channel. See 2.5.1, Pg 1 (attached)	1	The ITAAC figures and Design Description are intended to specify the Tier 1 design. This has meant limiting the level of detail from that given in CESSAR-DC text and figures. In some cases, detail has been provided in the design description text rather than figures and vice versa. It is intended that the sum of the ITAAC text and figures reflect the Tier 1 description of the design in total. This approach was agreed to with the NRC.
2	Page 1 of design description states that the Interface and Test processor communicates with the bistable trip processors and coincidence processors. Figure 2.5.1-2 indicates that the ITP communicates with the Core protection calculator, control room and remote shutdown panels, ESF-CS, maintenance and test panel and the initiation logics. The design description should be clarified. See Figure 2.5.1-2 (attached).	1	Refer to response 1 of N.2.5.1, above.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.1 (Reactor Protection System)

No.	Comments	Cat.	Resolution
3	Page 2 of CDM states that EMI qualification is applied to equipment with known EMI susceptibility based on operating environment and/or inherent design characteristics. How is "known susceptibility" to be determined? Suggest revising statement to: EMI qualification is applied for equipment based on the operating environment and/or inherent design characteristics. See also CESSAR 7.2.1.2 Design Bases (attached).	1	Agree. EMI qualification is applied for equipment with known EMI susceptibility based on operating environment and/or inherent design characteristics. ABB-CE interprets equipment with known EMI susceptibility to typically include but not be limited to power supplies, programmable logic controllers, digital computers, communications interface equipment and input/output equipment. It is known based on previous experience that certain equipment such as electromechanical switches, contactors, medium and heavy duty relays, terminal blocks, connectors, indicator lamps, ... etc. are not susceptible to radiated or conducted EMI effects. The suggested change will be incorporated in the CDM.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.1 (Reactor Protection System)

No.	Comments	Cat.	Resolution
4	Page 2 states that a site survey will be performed upon completion of system installation. This should be clarified to state whether this is a control room or plant wide survey. A commitment to update/review the EMI map based on equipment or environmental changes would be appropriate to include in the design description (external and internal to the plant).	2	ABB-CE will change the design description and appropriate sections of CESSAR-DC to state "EMI qualification is applied for equipment based on operating environment and/or inherent design characteristics." The ITAAC acceptance criteria references the site survey. The site survey will be performed at locations where susceptible equipment is located and will include the MCR as well as other areas of the plant. The site survey establishes a baseline from which subsequent changes in electromagnetic environment may be determined should EMI problems be encountered. Separate emissions testing is not performed, however, the impact of one system on another is accounted for by the site survey.
5	Deleted		

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.1 (Reactor Protection System)

No.	Comments	Cat.	Resolution
6	Page 3, Item c references a graded approach to software development based on relative importance to safety. The details of this need to be amplified in the CESSAR (i.e. guiding industry standards, applicability to RPS).	2	No Change Required. The graded approach to software development is fully described in the "Nuplex 80+ Software Program Manual." This manual is referenced in CESSAR-DC and has been approved for use by the USNRC. The FSER stipulates that any later changes to the specified sections of this manual be reviewed and approved by the USNRC. Software has historically been graded in two classes; safety and non-safety. With NUREG 0696, the USNRC initiated V&V for Safety Parameter Display software. This non-safety software became distinguished from other non-safety software by its level of scrutiny during the design process. The level of scrutiny was not intended to be as rigorous as for safety software, such as for much simpler core protection software. Thus, this defacto third grade of software became evident in the design process. This grade of software has never been formally defined in software standards. The use of a graded approach was developed by the A/E and ABB-CE during the USDOE HWRF program.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.1 (Reactor Protection System)

No.	Comments	Cat.	Resolution
7	Page 3. Commercial dedication (software/hardware) acceptance criteria may be interpreted as less rigorous than that specified for PPS. See Page 2, Items a and b. Clarification is needed as to whether the reference to commercial grade software is limited only to the software required for system development (programming language, operation system)? The reference to PPS software also needs clarification (see attached).	2	No Char ge Required. Software to be commercially dedicate · must be supplied in accordance with "Requirei .ents for the Supply of Commercial Digital Computer Hardware and software to be used in Nuplex 80+ Safety Systems." This document is referenced in CESSAR-DC. The dedication process must be in accordance with Section 3.10 of the "Nuplex 80+ Software Program Manual." This manual has been reviewed and approved by the USNRC. Commercially dedicated software meets the design, test, installation and maintenance process which is described in the ITAAC design description. There is no absolute restriction on the use of commercial software in safety systems provided it successfully completes the dedication program. The PPS will utilize both installed commercial software and commercial software for off-line system development.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.1 (Reactor Protection System)

No.	Comments	Cat.	Resolution
8	Page 3. Setpoint methodology Item b references design basis events, instrument accuracy and drift. Design description (and CESSAR) should be augmented to include a commitment for measurement and test equipment accuracy (confirmation that setpoint assumptions reflect actual plant surveillance M&TE practices).	2	No Change Required. CESSAR-DC, Section 7.2.2.3.2.J, Equipment Design Criteria, Capability For Test and Calibration, defines that the RPS design complies with IEEE Std. 338-1977, "Periodic Testing of Nuclear Generating Station Class 1E Power and Protection Systems". This standard imposes requirements for surveillance practices including verification of test instrument accuracies. It is ABB-CE's position that plant surveillance M&TE practices are accommodated in the setpoints assumptions and addressed as a Tier 2 item.
9	Page 6. First sentence add abbreviation to "reactor trip switchgear" (RTSG) or (RTSS) to be consistent with Figure 2.5.1-1, and 2.5.1-2. See CESSAR 7.2.1.1, 2nd paragraph (attached).	3	Agree. The term Reactor Trip Switchgear System (RTSS) will be used and consistency will be verified between the design description and CESSAR-DC.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.1 (Reactor Protection System)

No.	Comments	Cat.	Resolution
10	Figure 2.5.1-2. The figure indicates that communication between the CPC and the ITP and ITP to coincidence processors is in only one direction. Is this correct? Core protection calculator output is unclear (drafting error). The coincidence processors (Figure 2.5.1-2) indication two output paths. What is the distinction between the two? (it appears that the initiation logic block should be split to represent ESF logic in the figure and be consistent with Figure 2.5.1-3. In Figure 2.5.1-1 the RTSG is shown feeding the CEDMCS. Figure 2.5.1-2 states CEDMS. The design description references the CEDM system (see Page 5, last paragraph)? System notations are not consistent. (see attached).	1	Agree. Communications shown as one way between the ITP and the CPC and coincidence processors on Figure 2.5.1-2 will be revised to show two way communication. The figure will also be revised to correct the drafting error to clarify the CPC output. On this figure, one output path of the Coincidence Processor block represents the RPS trip initiation signal; the other output path represents an ESFAS initiation signal. Figure 2.5.1-2 will be revised to split the Initiation Logics block to represent ESF logic consistently with Figure 2.5.1-3. The term Control Element Drive Mechanism Control System (CEDMCS) will be used and consistency verified between the ITAAC document and CESSAR-DC.
11	CESSAR 7.1.2.2 Page 7.1-2 lists Reactor Trip Switchgear System (RTSS) the CDM lists Reactor Trip Switchgear (RTSG). See 2.5.1, Page 6, first paragraph. (attached-see 9 above). These should be made consistent.	1	Agree. The term Reactor Trip Switchgear System (RTSS) will be used and consistency will be verified between the design description and CESSAR-DC.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.1 (Reactor Protection System)

No.	Comments	Cat.	Resolution
12	CESSAR 7.1.2.15. List conformance to RG 1.11. This should be Safety Guide 11.	3	Regulatory Guide 1.11 is the correct reference. No change required.
13	CESSAR 7.1.2.15, Safety Guide 11. The containment pressure transmitters and instrument lines located outside of containment are considered part of containment. This is an exception to the safety guide and should be justified. Suggest Section "B" of the safety guide. Instrument lines are field run and designed. An ITAAC verification should be considered (see attached).	1	Containment pressure is monitored by pressure transmitters that are located outside of containment. Penetrations for containment monitoring are qualified diaphragms and transmitter sensing lines. Diaphragms and sensing lines are rated beyond maximum containment design pressure. See Item 77 on Table 19.12.2.2.4-1 attached. CESSAR-DC Section 7.1.2.15 will be revised for clarification.
14	CESSAR 7.1.2.22, Conformance to RG 1.62. This section paraphrases the RG but does not specify manual initiation at the "system level" as discussed in the RG. (see attached) This aspect should be addressed.	1	Agree. CESSAR-DC Section 7.1.2.22 will be revised to define that manual actuation is provided at the system level.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.1 (Reactor Protection System)

No.	Comments	Cat.	Resolution
15	CESSAR 7.2.1.1, states that the forth channel is a spare while maintaining a two out of three system. Is the CE80+ to be licensed as a 3 channel plant? Will a channel be allowed to be in bypass indefinitely? Is this supported by analysis? FMEA? Do the TS surveillance AOTS and surveillance intervals reflect the above? Please clarify. Also see 7.2.1.1.7 and TS 3.3.2 attached.	1	The RPS design is a four channel, 2-out-of-4 system with capability to bypass one inoperable channel. An inoperable channel is allowed to be bypassed as defined in the Technical Specification. The channel bypass capabilities are supported by PRA analysis. The FMEA was performed for a three channel system. Technical Specification surveillance requirements and intervals support the above.
16	Typos - See attached.	3	Agree. Typographical error will be corrected.
17	CESSAR 7.2.1.1.8, Page 7.2-27 states that alternate bistable trips are available should the initial trip function fail (functional diversity). This is accomplished by assuring that backup trips are not processed through the same PPS processor or bistable processor. There should be an ITAAC entry for this design feature. See 7.2.1.18 attached.	2	Agree. The ITAAC will be revised to include an entry for this design feature.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.1 (Reactor Protection System)

No.	Comments	Cat.	Resolution
18	TS surveillance requirements need to be revised to specify the allowed outage time for surveillance (channel functional test or calibration) TS Table 3.3.2-1.	2	Disagree. Surveillance testing is covered by bypass allowances for channels. That is, the surveillance testing is done with the channel in bypass.
19	CESSAR 7.2.1.1.9.2, Page 7.2-31. Automatic bistable testing states that the test task removes the test signal before the initiation circuit timer runs out. It is also stated that should the test input signal not be removed by the automatic test the timing logic built into the bistable trip logic will remove it. The action of the test circuit is not clear. Is the bistable timing logic designed to run out before the initiation circuit timer? Can the initiation circuit respond to a test signal should the test input signal not be removed? Is spurious actuation possible? Can the bistable be locked out during testing such that the bistable cannot respond to a valid input signal? The CESSAR needs to be clarified accordingly.	2	No Change Required. CESSAR-DC, Section 7.2.1.1.9 addresses automatic bistable testing. The bistable timing logic is designed to run out before the initiation circuit timer. The timer settings will be such that the potential for spurious actuation is minimized. The bistable is designed such that a valid trip condition cannot be blocked by testing. Detailed design of this test feature and its timer relationships are dependent on the characteristics of the as-procured equipment.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.1 (Reactor Protection System)

No.	Comments	Cat.	Resolution
20	CESSAR 7.2.1.1.9.8, Item B, references on-line spectral analysis for measuring analog sensor response time. What is the justification and bases for this test methodology? (see attached). This reference should be considered for inclusion in the CESSAR.	2	Change CESSAR. ANSI/ISA-S67.06-1984, Response Time Testing of Nuclear Safety-Related Instrument Channels in Nuclear Power Plants addresses response time testing methods. Appendix A, Noise Analysis Techniques, Section A.1, Power Spectral Density defines the test methodology. CESSAR-DC describes one method which may be used for response time testing of certain parameters. The standard will be included in CESSAR-DC as a reference.
21	Revise the design description as-marked to be consistent with Technical Specification description of trip setpoints.	1	Agree. The design description will be revised as marked.

An environmental qualification program assures the PPS equipment is able to perform its intended safety function for the time needed to be functional, under its design environmental conditions. The environmental conditions, bounded by applicable design basis events, are: temperature, pressure, humidity, chemical effects, radiation, aging, seismic events, submergence, power supply voltage & frequency variations, electromagnetic compatibility and synergistic effects which may have a significant effect on equipment performance. The environmental qualification of PPS equipment is achieved via tests, analyses or a combination of analyses and tests.

Comments
2.5.1.3
2.5.1.4

EMI qualification is applied for equipment ~~with known EMI susceptibility~~ based on operating environment and/or inherent design characteristics.

The PPS is qualified according to an established plan for Electromagnetic Compatibility (EMC).

The qualification plan requires the equipment to function properly when subjected to the expected operational electrical surges, electromagnetic interference (EMI), electrostatic discharge (ESD), and radio frequency interference (RFI).

The equipment to be tested will be configured for intended service conditions.

A site survey is performed upon completion of system installation to characterize the installed EMI environment.

PPS software is designed, tested, installed and maintained using a process which:

a. Defines the organization, responsibilities, and software quality assurance activities for the software engineering life cycle that provides for:

- establishment of plans and methodologies
- specification of functional, system and software requirements and standards, identification of safety critical requirements
- design and development of software
- software module, unit and system testing practices
- installation and checkout practices
- reporting and correction of software defects during operation

b. Specifies requirements for:

- software management, documentation requirements, standards, review requirements, and procedures for problem reporting and corrective action
- software configuration management, historical records of software, and control of software changes

Qualification is applied for equipment based on operating environment and/or inherent design characteristics

The reactor protective system sensor response times, reactor trip delay times, and analysis setpoints used in Chapter 15 are representative of the manner in which the RPS and associated instrumentation will operate. These quantities are used in the transient analysis documented in Chapter 15. Note that the reactor trip delay times shown in Chapter 15 do not include the sensor response times. Actual RPS equipment uncertainties, response times and reactor trip delay times are obtained from calculations and tests performed on the RPS and associated instrumentation. The verified system uncertainties are factored into all RPS settings and/or setpoints to assure that the system adequately performs its intended function when the errors and uncertainties combine in an adverse manner.

J. All system components are qualified for environmental and seismic conditions in accordance with IEEE Standard 323-1983, and IEEE Standard 344-1987. Compliance is addressed in Sections 3.10 and 3.11, respectively. In addition, the system is capable of performing its intended function under the most degraded conditions of the energy supply, as addressed in Section 8.3.

COMMENTS
2.5.1.3
2.5.1.4

K. System components ~~with known susceptibility~~ are qualified according to an established plan for electromagnetic compatibility (EMC) that requires the equipment to function properly when subjected to electrical surges, electromagnetic interference (EMI), electrostatic discharge (ESD) and radio frequency interference (RFI). EMI qualification is performed in accordance with applicable requirements of MIL-STD-461C, 1986 (Sections RS03, RS02, CS01, CS02 and CS06), "Electromagnetic Emission and Susceptibility Requirements for the Control of Electromagnetic Interference." Radiated and conducted EMI envelopes are established for qualification. A site-specific EMI survey is then performed to ensure that system exposure to EMI is within qualification envelope limits.

L. The RPS is considered a vital system. Vital instrumentation cabinet doors are locked and equipped with "door open" alarms. Refer to Chapter 13, Appendix 13A for additional details.

7.2.1.3 System Drawings

The RPS MCBDs, signal logics, block diagrams, and test circuit block diagrams are shown in Figures 7.2-1 through 7.2-30.

7.2.2 ANALYSIS

7.2.2.1 Introduction

The RPS is designed to provide the following protective functions:

Comments 2.5.1.9 2.5.1.11

The reactor trip switchgear system (RTSS) can be tripped manually from the Main Control Room or the Remote Shutdown Room. The manual reactor trip uses hardwired circuits which are independent of the PPS bistable and coincidence processors. Once a reactor trip has been initiated, the breakers in the ~~reactor trip switchgear~~ RTSS latch open.

Upon coincidence of two like signals indicating a condition for generating an ESFAS, the ESF initiation logic transmits the respective initiation signal to the ESF-CCS.

The PPS interfaces in the Main Control Room allow for manual activation of each of the ESF initiating signals input to the ESF-CCS. The PPS interfaces in the Remote Shutdown Room allow for manual activation of the initiating signals for Main Steam Isolation. Manual activation of these initiating signals is independent of the PPS bistable and coincidence processors.

The PPS operator's modules at the Main Control Room, the Remote Shutdown Room and at the maintenance and test panel allow operators to enter trip channel bypasses, operating bypasses, and variable setpoint resets. These modules provide indication of bypass status and bistable trip and pre-trip status.

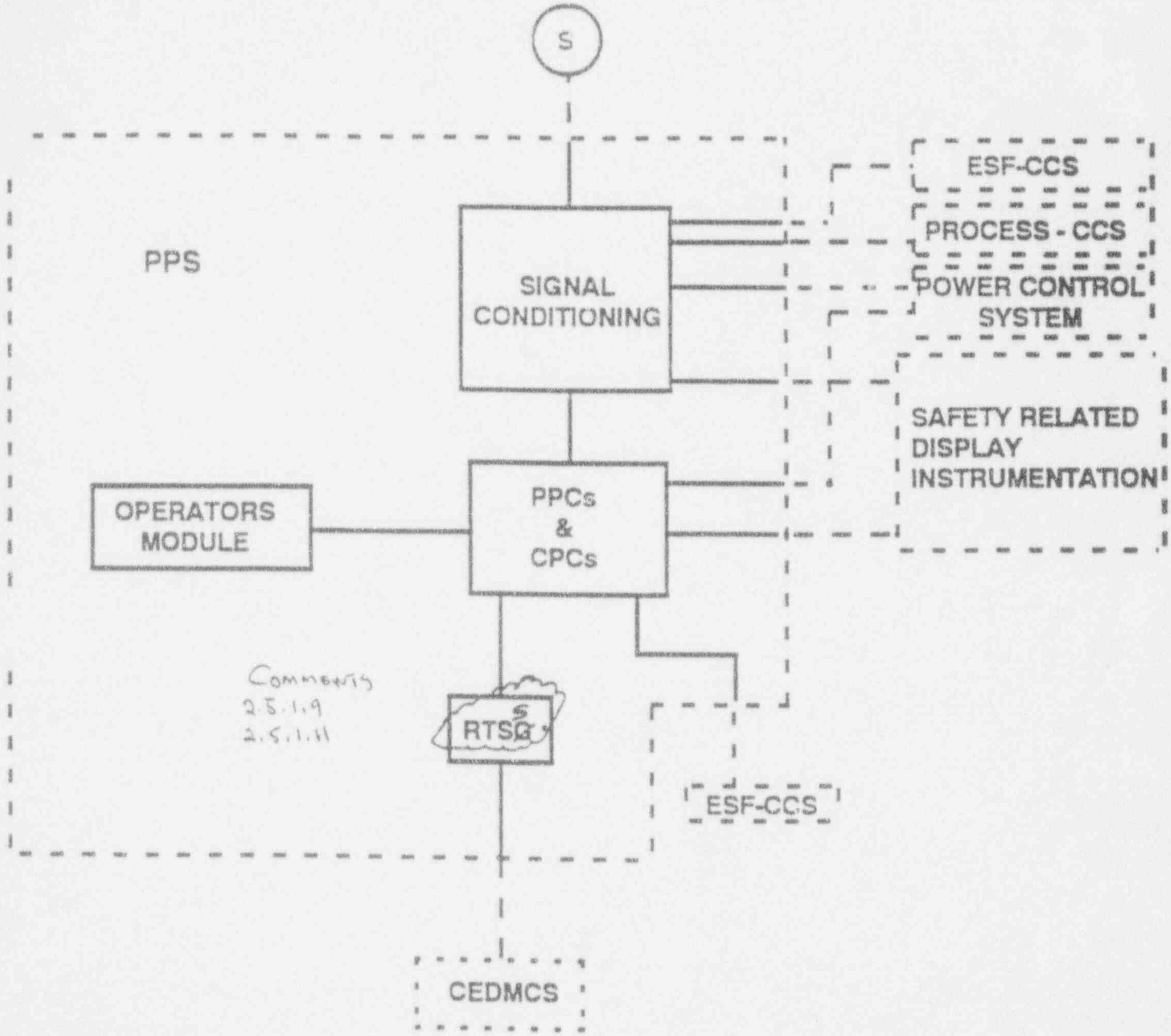
Manual control capability for the PPS is transferred from the Main Control Room to the Remote Shutdown Room upon actuation of the Master Transfer Switches via signals from the ESF-CCS for all control functions except reactor trip. The manual reactor trip switches are active in both locations at all times. Provision for transferring PPS control capability back to the Main Control Room is provided at the maintenance and test panel.

Loss of power to, or disconnection of a reactor trip path component in a PPC or CPC will cause a trip initiating state to be detected in a downstream component in that channel.

Periodic testing to verify operability of the PPS can be performed with the reactor at power or when shutdown without interfering with the protective function of the system. Overlap in individual tests assures that all functions are tested from sensor input through to the actuation of a reactor trip circuit breaker and to the generation of protection function initiation signals provided to the ESF-CCS.

The ITP monitors the on-line continuous automatic PPC and CPC hardware testing and performs on-line periodic automatic software logic functional testing of PPS logic.

Where automatic testing is implemented in the PPS, it does not degrade the capability of the PPS to perform its protective function. Indication of the automatic test system status and test results are provided to the operator via the Interface and Test Processor interface to the DIAS and DPS.



NOTES:

1. PPS EQUIPMENT SHOWN ON THE FIGURE IS CLASS 1E.
2. PPS EQUIPMENT IS POWERED FROM CLASS 1E SUPPLIES.
3. EACH PPS CHANNEL (4 IN NUMBER) IS POWERED FROM A SEPARATE CLASS 1E BUS.

FIGURE 2.5.1-1
PPS CONFIGURATION

COMMENTS
2.5.1.10
2.5.1.9 & 11

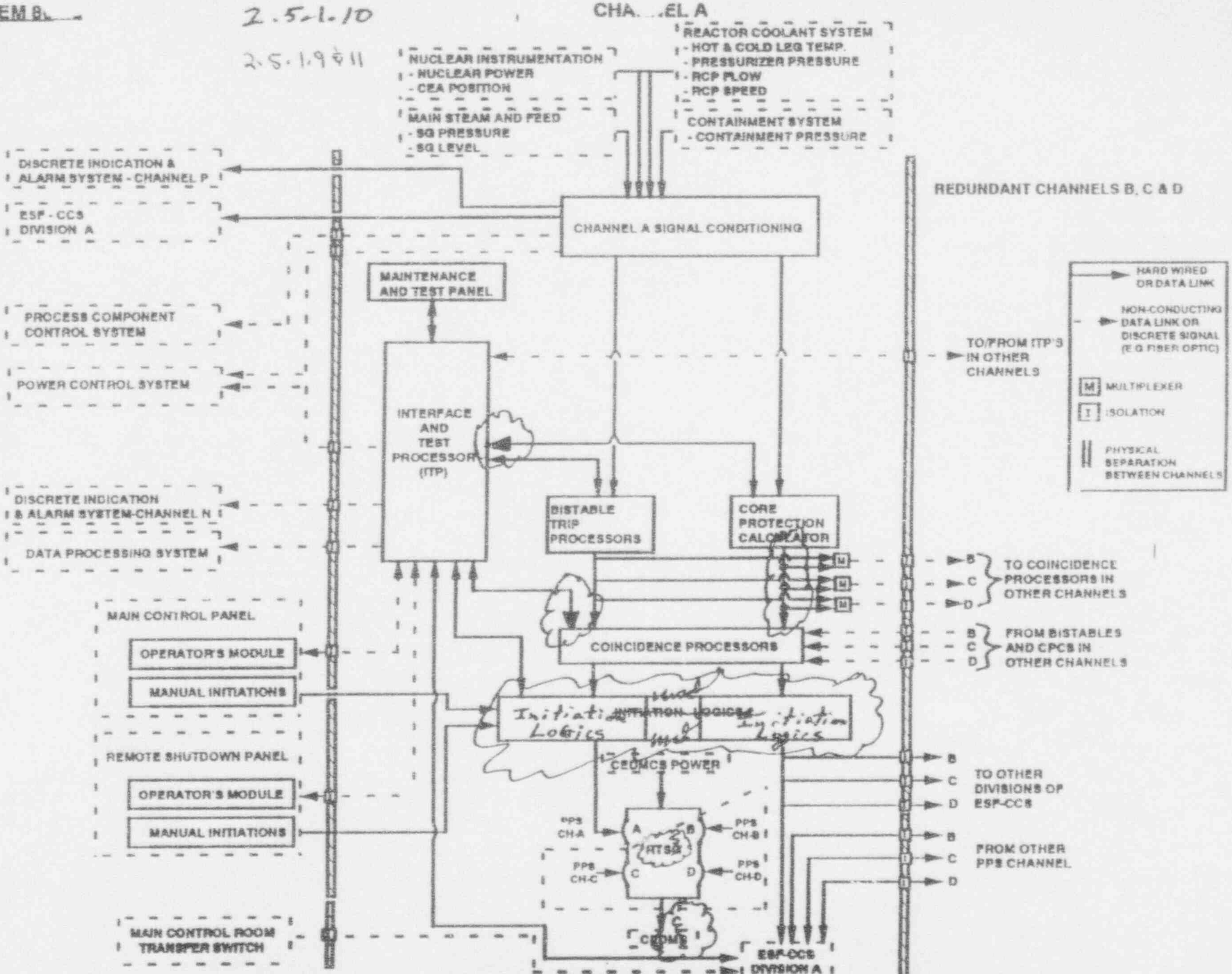


FIGURE 2.5.1-2
PLANT PROTECTION SYSTEM INTERCONNECTIONS

credible failures on the non-1E side of the isolation device will affect the PPS side and that independence of the PPS is not jeopardized.

7.1.2.11 Conformance to IEEE 387-1984

Conformance to IEEE 387-1984 , "IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations," as criteria in the design of these systems is discussed in Sections 8.3.1, and 9.5.4 through 9.5.8.

7.1.2.12 Conformance to IEEE 450-1980

Conformance to IEEE 450-1980, "IEEE Recommended Practice for Large Lead Storage Batteries for Generating Stations and Substations," as criteria in the design of these systems is discussed in Chapter 8.

7.1.2.13 Conformance to IEEE 603-1980, as Augmented by Regulatory Guide 1.153

The safety systems such as PPS, ESF-CCS and RTSS conform to the requirements of IEEE 603-1980, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations," as augmented by Regulatory Guide 1.153, "Criteria for Power, Instrumentation, and Control Portion of Safety Systems." For descriptions of conformances, refer to Sections 7.1.2.2, 7.1.2.3, 7.1.2.5, 7.1.2.7, 7.1.2.9 and 7.1.2.10.

7.1.2.14 Comparison of Design with Regulatory Guide 1.6

See Chapter 8.

7.1.2.15 Conformance to Regulatory Guide 1.11

Comments 2.5.1.12 2.5.1.13

Guidelines for instrument lines which penetrate primary reactor containment, and which are part of the reactor coolant pressure boundary or are connected directly to the containment atmosphere do not apply, since there are no lines which fall into this category. Containment pressure is monitored by four redundant pressure transmitters ~~located~~ outside of containment which monitor containment atmosphere. The lines both inside and outside containment are kept as short as possible. These lines and the transmitter diaphragm are considered an extension of the containment building. No other instrument lines penetrate reactor containment.

and are seismically qualified and designed for higher pressure than containment design pressure.

CONTAINMENT PENETRATION SUMMARY

Item	Service	Flow Dir.	Isolation Inside Containment	Isolation Outside Containment	Expected Isolation Status for SA	Calc. Type	Failure Probability
70	Steam Generator #1 Combined Blowdown	Out	1 check valve in parallel with 1 MOV (N.O.)	1 MOV (N.O.)	Assume open during DBE. Receives CIAS to close.	Calculation Type 22 for SGTR	P = 2.52E-04 for SGTR P = 0 for non-SGTR
71	Steam Generator #2 Combined Blowdown	Out	1 check valve in parallel with 1 MOV (N.O.)	1 MOV (N.O.)	Assume open during DBE. Receives CIAS to close.	Calculation Type 22 for SGTR	P = 2.52E-04 for SGTR P = 0 for non-SGTR
72	Fire Protection Water Supply to Containment	In	1 check valve	1 MOV (N.C.)	Assume closed during DBE. Receives CIAS to close.	Calculation Type 23	P = 9.84E-14
73	NCWS Supply to Containment Ventilation Units and CEDM Units	In	1 check valve	1 MOV (N.O.)	Valves assumed open during DBE. Closed loop inside CTMT.	Calculation Type 3	P = 1.30E-10
74	NCWS Return from Containment Ventilation Units	Out	1 check valve in parallel with 1 MOV (N.O.)	1 MOV (N.O.)	Assumed open during DBE. Closed loop inside CTMT.	Calculation Type 25	P = 6.56E-09
75	Containment Radiation Monitor (Inlet)	Out	1 MOV (N.O.)	1 MOV (N.O.)	Assume open during DBE. Receives CIAS to close. Direct connection to atmosphere.	Calculation Type 19	P = 6.66E-09
76	Containment Radiation Monitor (Outlet)	In	1 MOV (N.O.)	1 MOV (N.O.)	Assume open during DBE. Receives CIAS to close. Direct connection to atmosphere.	Calculation Type 19	P = 6.66E-09
77	Containment Pressure Sensing Line	-	-	-	Closed system piping (instrument tubing) not postulated to fail - seismically qualified and designed for higher pressure than CTMT design pressure.	Calculation Type 24	P = 0
78	ILRT Pressure Sensing Line	None	1 manual valve (N.C.)	1 manual valve (N.C.)	Assume valves closed during DBE. Direct connection to atmosphere; normally closed.	Calculation Type 15	P = 2.64E-16

COMMENT
2.5.1.13

remote operator's modules located in the control room. In addition, the status of each bypass is provided to the plant Data Processing System.

7.1.2.21.3 **ESF Components Inoperable**

The bypassed and/or inoperable condition of ESF components is monitored by the ESF-CCS, as described in Section 7.3. ESF-CCS status outputs are provided to the Data Processing System (DPS) which processes logic to indicate at the system level, the bypassing, inoperability or deliberate inducing of inoperability of an ESF system. The DPS also provides status information at the component level. The operator has the ability to activate each ESF system level bypass indicator manually in the control room. Inoperable indication is shown on the DPS CRTs, Integrated Process Status Overview (IPSO) panel and Discrete Indication and Alarm System (DIAS) alarm tiles as further described in Sections 7.7.1.4 and 7.7.1.5.

7.1.2.22 **Conformance to Regulatory Guide 1.62**

Manual initiation of the RPS is described in Sections 7.2.1.1.1.11 and 7.2.2.3.2. Manual initiation of the ESFAS is described in Section 7.3.2.3.2. Conformance to Regulatory Guide 1.62, "Manual Initiation of Protective Actions," is as follows:

A. Each of the above systems can be manually actuated.

Comment 2.5.1.14

B. Manual initiation of a protective action is provided at the system level and causes the same actions to be performed by the protection system as would be performed if the protection system had been initiated by automatic action.

C. Manual switches are located in the control room, ESF-CCS and at the RTSS for use by the operators. Some ESF functions also have manual actuation at the Remote Shutdown Panel.

D. The amount of equipment common to the manual and automatic initiation paths is kept to a minimum, usually just the actuation devices. No single credible failure in the manual, automatic, or common portions of the protective system will prevent initiation of a protective action by manual or automatic means.

E. Manual initiation requires a minimum of equipment consistent with the needs of A, B, C, and D above.

F. Once initiated, manual protective action will go to completion.

The design is based upon the use of Programmable Logic Controller (PLC) type equipment in each safety channel. All protective channel process loop inputs, protective channel trip functions, and the 2/4 Logic Matrix functions will be processed within the PLC's in that safety channel.

The reactor trip signal deenergizes the Control Element Drive Mechanism (CEDM) coils, allowing all CEAs to drop into the core.

PPS interfaces (RPS and ESFAS) for functions, such as operator interaction, alarm annunciation and testing (manual and automatic), are shown on Figure 7.2-2.

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 main control room (MCR) panels provide means to manually initiate engineered safeguards.

The Remote Shutdown Panel (RSP) provides selected functions needed for safe shutdown and cooldown, as described in Section 7.4.

Each PPS channel cabinet contains a manual transfer switch that enables the RSP or MCR for PPS channel functions that are common to both.

Comment

gateway

2.5.1.16 The Interface and Testing Processor (ITP), one per channel, ~~consists~~ of a data bus and three functional blocks: i.e., two ~~gateways~~ blocks and one test/bypass block, as shown in Figure 7.2-17. Gateway #1 interfaces to: the PPS Operators Module at the RSP; the Data Processing System, to provide selected PPS and CEAC channel status and test results information; and the CEAC, to retrieve status information. Gateway #2 interfaces to: the PPS Operators Module at the MCR; the Discrete Indication and Alarm System, to provide selected PPS and TLC channel status and test results information; the TLC, to retrieve status information; and the Power Control System, to retrieve status information. The test and bypass processor performs automatic on-line and manual testing of the PPC, processes the bypass logic and interfaces to the ITP's in other PPS channels via the data bus interfaces to the bistable processors and coincidence processors. A data bus bridge interfaces to the ESF-CCS.

Amendment Q
June 30, 1993

COMMENT
2.5.1.17

System protective functions are distributed between bistable processors to provide functional diversity.

Initiation Logic
Reactor Trip Switchgear
Interface and Test Processor —
Operator's Modules
Switches for Manual Activation of Reactor Trip Signals
Switches for Manual Activation of ESF Initiating Signals

Figure 2.5.1-2 shows the plant systems in which process instrumentation is implemented for generation of the sensor signal input to the PPS. Limit logic for process-value to setpoint comparison is implemented in bistable processors in each channel. The bistable processors generate trip signals based on the channel digitized value exceeding a digital setpoint. The PPS maintenance and test panels provide the capability for trip limit setpoint changes. Limit logic for calculated departure from nucleate boiling ratio and high linear heat generation rate are implemented in each channel in a section of the PPS referred to as the Core Protection Calculator (CPC).

The trip output signals of the bistable processors and the CPC in each channel are sent to the local coincidence logic processors in all four PPS channels. Therefore, for each trip condition, the local coincidence logic processor in each channel receives four trip signals, one from its associated bistable processors or CPC from within the channel, and one from the equivalent bistable processors or CPC located in each of the other three redundant 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 any two like trip signals is required to generate a reactor trip or ESF initiation signal.

Operating bypasses are implemented in the PPS to provide for the bypass of trip functions which are plant mode specific. These bypasses are manually activated. The PPS automatically removes an operating bypass if the plant approaches conditions for which the associated trip function is designed to provide protection. Bistable trip channel bypasses allow one channel of the bistable inputs to the coincidence processors to be bypassed for each trip function. This converts the local coincidence logic to two-out-of-three coincidence for each trip function for which a bistable trip channel bypass is initiated. For each trip function, the PPS allows only one bistable trip channel to be bypassed at a time.

Upon coincidence of two like signals indicating one of the conditions for reactor trip, the PPS logic initiates actuation of a channel of the reactor trip switchgear. As shown on Figure 2.5.1-2, actuation of a selective two single channels of the reactor trip switchgear is required to cause a reactor trip. The reactor trip switchgear breakers interrupt power to the Control Element Drive Mechanism (CEDM) coils, allowing all Control Element Assemblies to drop into the core by gravity.

B. Bistable Trip Channel Bypass Testing

A description of testing bistable trip channel bypasses is included as part of the local coincidence logic testing described in Section 7.2.1.1.9.4.

7.2.1.1.9.8 Response Time Tests

Response time testing of the complete Reactor Protective System, is accomplished by the combined use of portable field installed test equipment and test features provided as part of the PPS test function.

Measurement Channel Response Time Tests, which include portions of the system (such as cables and sensors) may be conducted on a system basis or an overlapping subsystem basis.

Methods which ~~are~~ may used to conduct these tests include:

A. Perturbation and monitoring of plant parameters - either during operation or while shutdown. This method is applicable to RTDs (monitored following a plant trip), reactor coolant pump speed sensors (monitored following turn-off of pump), and CEA position reed switches (monitored during CEA motion).

Comment 2.5.1.20

B. On-line power spectral density analysis. This method would be applicable to analog sensors as defined in ANSI/ISA-S67.06-1984, "Response Time Testing of Nuclear Safety-Related Instrument Channels in Nuclear Power Plants"

C. Off-line injection of step or ramp changes for RPS inputs. This method would be applicable to sensors (via special pressure test rigs, hot oil baths or hot sand boxes) or electronics and logic (via special electrical test boxes).

D. The test function in the course of its normal testing implicitly verifies that the response time of the PPS is less than a known upper limit. The upper limit is bounded by the bistable logic processor execution time (fixed) plus the coincidence processor execution time (fixed) plus the worst case skew time due to the asychronous operation of the processor. An independent timer monitors the fixed execution time and provides overrun status. The test function reads this status and will annunciate a failure.

E. Operation and monitoring of actuated devices. This method would be applicable to the CEDMs, including their control logic and switchgear.

ANSI/ISA-S67.06-1984
Approved August 29, 1986

Comment
2.5.1.20

American National Standard

Response Time Testing of Nuclear Safety-Related Instrument Channels in Nuclear Power Plants

Instrument Society of America

APPENDIX A

Noise Analysis Techniques

A.1 Power Spectral Density

Power spectral density analysis involves the determination of signal power per unit frequency as a function of frequency. For sensor response evaluation, the noise signal analyzed is the sensor output that results from normal process fluctuations. If the process fluctuations have a constant power spectrum (white noise over the nominal sensor bandwidth), then the power spectral density of the sensor output signal is proportional to the square of the frequency response gain of the sensor. Consequently, the sensor response characteristics can be evaluated by fitting a transfer function to the measured power spectral density if the white noise assumption is valid. This empirically determined transfer function then may be used to predict the response of the sensor to any input of interest.

If the white noise assumption is not valid, then the above procedure cannot be used. However, changes in the sensor response characteristic may alter a measured power spectral density.

A.2 Autoregressive Analysis

Autoregressive analysis involves fitting a simple formula to the measured data. The formula has the form

$$\gamma_k = \sum_{i=1}^{i=N} a_i \gamma_{k-i}$$

where:

γ_k = sample k of the output
N = order of the fit
a_i = an autoregressive coefficient

The fit provides estimates of the a_i (usually obtained by least squares fitting techniques). Once the a_i are known, the autoregressive model may be used to evaluate sensor response characteristics. As with the power spectral density approach, the results are quantitative only if the process fluctuations have white noise characteristics.

A.3 Zero-Crossing

The rate at which a sensor output crosses its average value in response to a specific fluctuating input decreases as the sensor time constant increases. Consequently, a device that monitors the crossing rate can be used to detect changes in sensor time constant and/or changes in input fluctuations. Masking of any effects due to changes in sensor time constant by exactly compensating changes in input fluctuations is implausible. Therefore, measuring the crossing rate will detect changes in the sensor time constant if the sensitivity of the crossing rate to changes in the time constant is large enough. For temperature sensors (where the response is governed by heat diffusion), the sensitivity is unity (an x percent increase in time constant causes an x percent decrease in crossing rate). The usual practice is to remove the average value of the signal and measure the rate of crossing of the zero value in the remaining signal. Consequently, the method is often called the zero-crossing technique.

Steam Generator Pressure - Low
Containment Pressure - High
Reactor Coolant Flow - Low
Departure from Nucleate Boiling Ratio - Low
Linear Heat Generation Rate - High

Setpoints for initiation of a reactor trip are installed for each monitored condition to provide for initiation of a reactor trip prior to exceeding reactor fuel thermal limits and the Reactor Coolant System pressure boundary limits for anticipated operational occurrences. If a monitored condition ~~exceeds~~ reaches its setpoint, the PPS automatically actuates the reactor trip switchgear.

Engineered Safety Features Initiation Function

Process instrumentation, the PPCs, the ESF-CCS, motor starters and other actuated devices function to initiate the engineered safety feature systems. The process instrumentation provides sensor data input to the PPCs, which monitor the following plant conditions to initiate the engineered safety features systems.

Pressurizer Pressure - Low
Steam Generator Water Level - Low or High
Steam Generator Pressure - Low
Containment Pressure - High

COMMENT 2.5.1.21

If a monitored condition ~~exceeds~~ reaches its setpoint, the PPCs automatically generate one or more of the following Engineered Safety Feature Actuation Signals (ESFAS).

Safety Injection Actuation Signal
Containment Isolation Signal
Containment Spray Actuation Signal
Main Steam Isolation Signal
Emergency Feedwater Actuation Signals

These initiating signals are provided to the ESF-CCS, which responds by actuating the engineered safety feature systems.

Elements Of The PPS

The PPS is divided into four redundant channels. The following elements, depicted in Figures 2.5.1-2 and 2.5.1-3, are included in each channel of the PPS.

Process Instrumentation
Signal Conditioning Equipment
Limit Logic (PPC Bistables and CPCs)
Local Coincidence Logic

Initiation Logic
Reactor Trip Switchgear
Interface and Test Processor
Operator's Modules
Switches for Manual Activation of Reactor Trip Signals
Switches for Manual Activation of ESF Initiating Signals

COMMENT 2.5.1.21

reaching

Figure 2.5.1-2 shows the plant systems in which process instrumentation is implemented for generation of the sensor signal input to the PPS. Limit logic for process-value to setpoint comparison is implemented in bistable processors in each channel. The bistable processors generate trip signals based on the channel digitized value ~~exceeding~~ a digital setpoint. The PPS maintenance and test panels provide the capability for trip limit setpoint changes. Limit logic for calculated departure from nucleate boiling ratio and high linear heat generation rate are implemented in each channel in a section of the PPS referred to as the Core Protection Calculator (CPC).

The trip output signals of the bistable processors and the CPC in each channel are sent to the local coincidence logic processors in all four PPS channels. Therefore, for each trip condition, the local coincidence logic processor in each channel receives four trip signals, one from its associated bistable processors or CPC from within the channel, and one from the equivalent bistable processors or CPC located in each of the other three redundant 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 any two like trip signals is required to generate a reactor trip or ESF initiation signal.

Operating bypasses are implemented in the PPS to provide for the bypass of trip functions which are plant mode specific. These bypasses are manually activated. The PPS automatically removes an operating bypass if the plant approaches conditions for which the associated trip function is designed to provide protection. Bistable trip channel bypasses allow one channel of the bistable inputs to the coincidence processors to be bypassed for each trip function. This converts the local coincidence logic to two-out-of-three coincidence for each trip function for which a bistable trip channel bypass is initiated. For each trip function, the PPS allows only one bistable trip channel to be bypassed at a time.

Upon coincidence of two like signals indicating one of the conditions for reactor trip, the PPS logic initiates actuation of a channel of the reactor trip switchgear. As shown on Figure 2.5.1-2, actuation of a selective two single channels of the reactor trip switchgear is required to cause a reactor trip. The reactor trip switchgear breakers interrupt power to the Control Element Drive Mechanism (CEDM) coils, allowing all Control Element Assemblies to drop into the core by gravity.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.2 (Engineered Safety Features-Component Control System

No.	Comments	Cat.	Resolution
1	Page 1 of the DCM states that for components of the ESF-CCS EMI qualification is applied for equipment with known EMI susceptibility based on operating environment and/or inherent design characteristics. Describe how "known susceptibility" is to be determined? Revise statement to "EMI qualification is applied for equipment based on operating environment, inherent design characteristics and anticipated operating occurrences." See question 3 PPS and CESSAR 7.3.1.2 page 7.3-33 fifth paragraph.	1	Agree. EMI qualification is applied for equipment with known EMI susceptibility based on operating environment and/or inherent design characteristics. Equipment with known EMI susceptibility includes but is not limited to power supplies, programmable logic controllers, digital computers, communications interface equipment and input/output equipment. It is known based on previous experience that certain equipment such as electromechanical switches, contactors, medium and heavy duty relays, terminal blocks, connectors, indicator lamps, ... etc. are not susceptible to radiated or conducted EMI effects. The suggested change will be incorporated in the CDM.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.2 (Engineered Safety Features-Component Control System

No.	Comments	Cat.	Resolution
2	The DCM states that a site survey will be performed upon system installation to characterize the installed EMI environment. Define the scope of the survey. A commitment to update the survey based on plant modification and/or environmental (external and internal to the plant). Describe what actions are to be taken should the site survey indicate that the EMI equipment qualification is inadequate.	2	ABB-CE will change the design description and appropriate sections of CESSAR-DC to state "EMI qualification is applied for equipment based on operating environment and/or inherent design characteristics." The ITAAC acceptance criteria references the site survey. The site survey will be performed at locations where susceptible equipment is located which will include the control room as well as other areas of the plant. The site survey establishes a baseline from which subsequent changes in electromagnetic environment may be determined should EMI problems be encountered. Separate emissions testing is not performed, however, the impact of one system on another is accounted for by the site survey.
3	The DCM does not list the ESF-CCS initiation signals from the PPS.	1	ESF initiation signals from the PPS are intentionally excluded from this ITAAC. These signals are listed in the ITAAC for the PPS N.2.5.1 but are not required to be a design commitment since they may change over the life of the plant. This approach has been agreed to in the resolution to previous ITAAC comment 2.5.2-9B.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.2 (Engineered Safety Features-Component Control System

No.	Comments	Cat.	Resolution
4	Page 3 CDM. The ESF-CCS control capability for non-ESF systems does not list control complex ventilation system. This system is listed in the ITAAC.	1	Agree. The Control Complex Ventilation System will be added to Page 3 of the CDM.
5	Page 3 CDM the ESF is stated as providing control capability for listed safety related systems. In addition the ITAAC references status indication capability (see ITAAC item 8), add this to the CDM.	1	Agree. The Item 8 Design Commitment will be revised to include status indication capability.
6	The CDM material indicates that the ESF-CCS integrates ESF initiation and diesel generator load sequencing. Is this consolidation addressed within the TS? For example: If the load sequencer is inoperative is the DG inoperable and the ESF-CCS? If the ESF-CCS be declared inoperable will the load sequencer and DSG be inoperable as well?	1	ESF initiation and diesel load sequencing are integrated functions performed by the ESF-CCS as described in the CDM. The Technical Specification is structured to address operability of functions. The declaration of inoperability is addressed on a functional level in the Technical Specification. For the examples given, operability of supported system would depend on the function that had failed.
7	Deleted	1	

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.2 (Engineered Safety Features-Component Control System

No.	Comments	Cat.	Resolution
8	Page 4 DCM references diverse manual actuation switches as an alternate means for manual actuation of ESF components. The DCM should also reference that High level manual control is also available in all four divisions to be consistent with ITAAC item 12.	1	No Change Required. System level manual actuation of ESFAS functions is addressed in ITAAC 2.5.1, Item 11 for the Plant Protection System. CESSAR-DC also addresses system level manual actuation in Sections 7.1.2.22 and 7.3.2.2. ABB-CE believes that system level manual actuation of ESFAS functions is adequately addressed. The diverse manual switches defined on DCM page 4 are independent from the system level manual actuation.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.2 (Engineered Safety Features-Component Control System

No.	Comments	Cat.	Resolution
9	Page 5 DCM references a graded approach to software development based on relative importance to safety. Where is this approach reflected in a standard (GDC 1?) Where would a graded approach to software development be applicable to RPS? See also ITAAC item 20	2	No Change Required. The graded approach to software development is fully described in the "Nuplex 80+ Software Program Manual." This manual is referenced in CESSAR-DC and has been approved for use by the USNRC. The FSER stipulates that any later changes to the specified sections of this manual be reviewed and approved by the USNRC. Software has historically been graded in two classes; safety and non-safety. With NUREG 0696, the USNRC initiated V&V for Safety Parameter Display software. This non-safety software became distinguished from other non-safety software by its level of scrutiny during the design process. The level of scrutiny was not intended to be as rigorous as for safety software, such as for much simpler core protection software. Thus, this defacto third grade of software became evident in the design process. This grade of software has never been formally defined in software standards. The use of a graded approach was developed by the A/E and ABB-CE during the USDOE HWRF program.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.2 (Engineered Safety Features-Component Control System

No.	Comments	Cat.	Resolution
10	Page 5 DCM Commercial dedication (software/hardware) dedication process appears less rigorous than that specified for the ESF-CCS specifically. Is the reference to commercial grade software intended to be limited to software required for system development (programming language, operating system)? The reference to commercial dedication of ESF-CCS software is unclear. See also ITAAC item 22.	2	Software to be commercially dedicated must be supplied in accordance with "Requirements for the Supply of Commercial Digital Computer Hardware and Software to be used in Nuplex 80+ Safety Systems." This document is referenced in CESSAR-DC. The dedication process must be in accordance with Section 3.10 of the "Nuplex 80+ Software Program Manual." This manual has been reviewed and approved by the USNRC. Commercially dedicated software meets the design, test, installation and maintenance process which is described in the ITAAC design description. There is no absolute restriction on the use of commercial software in safety systems provided it successfully completes the dedication program. The ESF-CCS will utilize both installed commercial software and commercial software for off-line system development.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.2 (Engineered Safety Features-Component Control System

No.	Comments	Cat.	Resolution
11	Page 6 CDM Item b should add process measurement accuracies, environmental effects, response time and test equipment accuracy. Additionally a new item (G) should be added for test equipment accuracy and calibration (control of MTE) and that the "allowable tolerance" or "lease alone zone" components are defined (uncertainty terms). See also ITAAC Item 19.	2	No Change Required. CESSAR-DC, Section 7.3.2.3.2.J, Equipment Design Criteria, Capability for Test and Calibration, defines that the the ESF-CCS design complies with IEEE Std 338-1977, "Periodic Testing of Nuclear Generating Station Class 1E Power and Protection Systems." This standard imposes requirements for surveillance practices including verification of test instrument accuracies. It is ABB-CE's position that plant surveillance MT&E practices are accommodated in setpoint assumptions and are addressed as a Tier 2 item.
12	Figure 2.5.2-1 and 2.5.2-2 list outputs ESF-CCS as ESF components only. The DCM material referenced "non-ESF" and additional safety related components. See page 3 of the CDM. Revise the figure to reflect the CDM description.	1	The ITAAC figures and Design Description are intended specify the Tier 1 design. This has meant limiting the level of detail from that given in CESSAR-DC text and figures. In some cases, detail has been provided in the design description text and vice versa. It is intended that the sum of the ITAAC text and figures reflect the Tier 1 description of the design in total. This approach was agreed to with the NRC.

CE 80+ ITAAC independent Review Comments

ITAAC No. 2.5.2 (Engineered Safety Features-Component Control System

No.	Comments	Cat.	Resolution
13	ITAAC item 6 The acceptance criteria 6.b states that ESF initiation signals that satisfy the selective 2 out of 4 logic result in actuation signals to related system components. This should read PPS initiation signals ... result in ESF actuation signals ... to reflect the CDM description.	1	Agree. ITAAC Item 6 will be revised.
14	ITAAC item 7, Acceptance criteria 7b. Same as 13 above. Additionally, references the control complex ventilation system as actuated by the ESF-CCS. This system is not listed in the CDM material. See page CDM page 3 and question 4 above.	1	Agree. ITAAC Item 7 and the CDM will be revised.
15	ITAAC 10b The emergency feedwater actuation signal is listed for steam generator 1 and steam generator 2. This is inconsistent with the ESF initiation signals listed elsewhere in the ITAAC/DCM (lists emergency feedwater actuation) but consistent with the TS functional units and the CESSAR.	1	Agree. The DCM will be revised to reflect Steam Generator 1 and 2.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.2 (Engineered Safety Features-Component Control System

No.	Comments	Cat.	Resolution
16	ITAAC 14.a The master transfer switches and the ESF-CCS maintenance test panel (transfer) appear to be located in the same fire zone. Is this acceptable per App "R" requirements? See APP. "R" III-G.	2	Agree. The Master Transfer Switches are located within the Main Control Room fire boundary. The Maintenance and Test Panels are located in plant equipment room fire boundaries. This is to meet Appendix R fire requirements. ITAAC Item 14.a will be revised for clarity.
17	Section 7.3.1.1 states that the ESF functions are assigned such that the effect of a single group failure to selected ESF functions in a given division. This functional diversity approach is not discussed in the CDM material.	1	The NRC was requested to clarify the comment.
18	The "General Design Criteria" in the CESSAR Section 7.3.2.3, page 7.3-38 is inconsistent with the standard review plan criteria. Specifically, the SRP lists GDC-19 and 29 but is not listed in the CESSAR. GDCs 1, 16, 34, 35, 37, 38, 40, 41, 43, 44 and 46 are listed in the CESSAR but not applicable per the SRP. Revise the CESSAR or discuss the basis for the discrepancy.	3	Criteria 19 - Control Room is addressed in CESSAR-DC Section 3.1.15. Criteria 29 - Reactor Coolant Makeup is discussed in Section 3.1.29. Suitable references to other sections are provided therein. ABB CE elects to retain the remainder as they are met by the design.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.2 (Engineered Safety Features-Component Control System

No.	Comments	Cat.	Resolution
19	Typographical error, Figure 7.3-13b, should read "contactor"	3	Agree. The typographical error will be corrected.

2.5.2 ENGINEERED SAFETY FEATURES - COMPONENT CONTROL SYSTEM

Design Description

The Engineered Safety Features-Component Control System (ESF-CCS) is a safety-related instrumentation and control system which provides automatic actuation of Engineered Safety Features (ESF) systems upon receipt of ESF initiation signals from the Plant Protection System (PPS). The ESF-CCS also provides the capability for manual actuation of ESF systems, manual control of ESF system components and manual control of other safety-related systems and components identified below.

The ESF-CCS is located in the nuclear island structures.

The Basic Configuration of the ESF-CCS is as shown on Figure 2.5.2-1.

The ESF-CCS is classified Seismic Category I.

The ESF-CCS equipment is classified Class 1E.

An environmental qualification program assures the ESF-CCS equipment is able to perform its intended safety function for the time needed to be functional, under its design environmental conditions. The environmental conditions, bounded by applicable design basis events, are: temperature, pressure, humidity, chemical effects, radiation, aging, seismic events, submergence, power supply voltage & frequency variations, electromagnetic compatibility and synergistic effects which may have a significant effect on equipment performance. The environmental qualification of ESF-CCS equipment is achieved via tests, analyses or a combination of analyses and tests.

COMMENT

2.5.2.1

2.5.2.2

EMI qualification is applied for equipment ~~with known EMI susceptibility~~ based on operating environment and/or inherent design characteristics.

The ESF-CCS is qualified according to an established plan for Electromagnetic Compatibility (EMC).

The qualification plan requires the equipment to function properly when subjected to the expected operational electrical surges or electromagnetic interference (EMI), electrostatic discharge (ESD), and radio frequency interference (RFI).

The equipment to be tested will be configured for intended service conditions.

7.3.1.2 Design Bases

The design bases of the ESF Systems are discussed in Chapter 6. The ESFAS is designed to provide initiating signals for ESF components which require automatic actuation following the design bases events shown on Table 7.3-2.

The systems are designed in compliance with the applicable criteria of the NRC, "General Design Criteria for Nuclear Power Plants," Appendix A, 10 CFR 50. System testing conforms to the requirements of IEEE Std. 338-1977, "Standard Criteria for Periodic Testing of Nuclear Power Generating Station Protection Systems," and the intent of Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions."

Specific design criteria for the ESFAS are detailed in IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," Section 3. The following is a discussion of the specific items in IEEE Std. 279-1971 and their implementation.

The generating station conditions requiring actuation of the ESFAS are listed on Table 7.3-2, which also shows which system will actuate for each event. The monitored variables required for ESF System protective action are listed on Table 7.3-3, which also shows which signals are generated by the variable. The number and location of the sensors required to monitor the variables are listed in Table 7.3-4. The normal operating ranges, actuation setpoints, the nominal full power value, and the margin between the last two are listed on Table 7.3-5. The ranges of the ESFAS variables are listed on Table 7.3-6.

COMMENT 2.5.2.1 2.5.2.2

The ESFAS is designed with consideration given to unusual events which could degrade system performance. System components are qualified for the environmental conditions discussed in Section 3.11 and the seismic conditions discussed in Section 3.10. System components ~~with known susceptibility to~~ electromagnetic interference (EMI) are qualified by methods defined in Section 7.2.1.2.K. A single failure within the system will not prevent proper protective action at the system level. The single failure criterion is discussed in Section 7.3.2.3.2.

The ESFAS minimum response times are specified in the Technical Specifications. The accuracies of the ESFAS measurement channels are given as ALLOWED VARIATION in the Technical Specifications. The total ranges of ESFAS variables are provided in Table 7.3-6.

The ESF-CCS provides control capability and, upon receipt of initiation signals from the PPS, automatically generates actuation signals to the following ESF systems within allocated response times:

safety injection system,
containment isolation system,
containment spray system,
main steam isolation, and
emergency feedwater system.

Once initiation signals are received from the PPS, the ESF-CCS actuation logic signals remain following removal of the initiation signal.

The ESF-CCS provides control capability and, upon receipt of initiation signals from the PPS, automatically generates actuation signals to the following non-ESF systems:

annulus ventilation system,
component cooling water system,
onsite power system, ~~and~~
Comment diesel generators, and
2.5.2.4 control complex ventilation system

The ESF-CCS provides control capability for the following safety-related systems:

shutdown cooling system,
safety depressurization system,
atmospheric dump system,
station service water system,
heating, ventilating and air conditioning systems, and
hydrogen mitigation devices.

Upon receipt of ESF initiation signals for safety injection, containment spray or emergency feedwater, the ESF-CCS initiates an automatic start of the diesel generators and automatic load sequencing of ESF loads.

Upon detecting loss of power to Class 1E Division buses through protective devices, the ESF-CCS automatically initiates startup of the diesel generators, shedding of electrical load, transfer of Class 1E bus connections to the diesel generator, and sequencing of the reloading of safety-related loads to the Class 1E bus. In performing load sequencing, normally used safety related plant loads are loaded first in a predetermined sequence unless an ESF actuation signal is generated. Upon ESF actuation, the normal load sequence is interrupted and priority is given to loading the actuated ESF systems and associated safety-related systems. The sequence for loading the normally used safety related plant loads is then resumed.

TABLE 2.5.2-1 (Continued)

ENGINEERED SAFETY FEATURES COMPONENT CONTROL SYSTEM
Inspections, Tests, Analyses and Acceptance Criteria

Comment 2.5.2.5

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. (Continued)	7.b) Tests will be performed using signals simulating ESF initiation to the ESF-CCS.	7.b) ESF initiation signals which satisfy the selective 2 out of 4 criteria results in actuation signals for related system components for the following systems: annulus ventilation system, component cooling water system, onsite power system, diesel generators, and control complex ventilation system.
8. The ESF-CCS provides control and display capability for the following safety-related systems: shutdown cooling system, safety depressurization system, atmospheric dump system, station service water system, heating, ventilating and air conditioning systems, and hydrogen mitigation devices.	8. Tests will be performed on the as-built ESF-CCS control and display interface equipment.	8. The control and display interface equipment provide component status and control capability for the following systems: shutdown cooling system, safety depressurization system, atmospheric dump system, station service water system, heating, ventilating and air conditioning systems, and hydrogen mitigation devices.

TABLE 2.5.2-1 (Continued)

ENGINEERED SAFETY FEATURES COMPONENT CONTROL SYSTEM
Inspections, Tests, Analyses and Acceptance Criteria

Comment 2.5.2.13

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. (Continued) Once initiation signals are received from the PPS, the ESF-CCS actuation logic signals remain following removal of the initiation signal.	6.b) Tests will be performed using signals simulating ~~ESF~~ PPS initiation to the ESF-CCS.	6.b) ~~ESF~~ PPS initiation signals which satisfy the selective 2 out of 4 criteria result in ESF actuation signals for related system components for the following systems: safety injection system, containment isolation system, containment spray system main steam isolation, and emergency feedwater system.
	6.c) Tests will be performed using signals simulating ~~ESF~~ PPS initiation to the ESF-CCS.	6.c) Measured response times are less than or equal to the response time values required for each ESF actuation signal.
	6.d) Testing will be performed using signals simulating ~~ESF~~ PPS initiation to the ESF-CCS.	6.d) Once initiated, ESF-CCS actuation logic signals remain following removal of the initiation signal.
7. The ESF-CCS provides control capability and, upon receipt of initiation signals from the PPS, automatically generates actuation signals to the following non-ESF systems: annulus ventilation system, component cooling water system, onsite power system, diesel generators, and control complex ventilation system.	7.a) Tests will be performed on the as-built ESF-CCS control and display interface equipment.	7.a) The control and display interface equipment provide control capability for the following systems: annulus ventilation system, component cooling water system, onsite power system, diesel generators, and control complex ventilation system.

TABLE 2.5.2-1 (Continued)

COMMENT 2.5.2.14

ENGINEERED SAFETY FEATURES COMPONENT CONTROL SYSTEM
Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. (Continued)	7.b) Tests will be performed using signals simulating ~~ESF~~ PPS initiation to the ESF-CCS.	7.b) ~~ESF~~ PPS initiation signals which satisfy the selective 2 out of 4 criteria results in ESF actuation signals for related system components for the following systems: annulus ventilation system, component cooling water system, onsite power system, diesel generators, and control complex ventilation system.
8. The ESF-CCS provides control capability for the following safety-related systems: shutdown cooling system, safety depressurization system, atmospheric dump system, station service water system, heating, ventilating and air conditioning systems, and hydrogen mitigation devices.	8. Tests will be performed on the as-built ESF-CCS control and display interface equipment.	8. The control and display interface equipment provide component status and control capability for the following systems: shutdown cooling system, safety depressurization system, atmospheric dump system, station service water system, heating, ventilating and air conditioning systems, and hydrogen mitigation devices.

TABLE 2.5.2-1 (Continued)

ENGINEERED SAFETY FEATURES COMPONENT CONTROL SYSTEM
Inspections, Tests, Analyses and Acceptance Criteria

Comment 2.5.2.14

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10.b) Upon ESF actuation, the normal load sequence is interrupted and priority is given to loading the actuated ESF systems and associated safety-related systems.	10.b) A test will be performed using a simulated loss of power to the Class 1E buses and simulated ~~ESF~~ PPS initiation signals input to the ESF-CCS during the reloading sequence for each of the following ESF initiation signals: safety injection actuation signal, containment spray actuation signal, emergency feedwater actuation signal to steam generator 1, and emergency feedwater actuation signal to steam generator 2.	10.b) Upon receipt of the ~~ESF~~ PPS initiation signal, the ESF-CCS automatically interrupts the loading sequence to load the equipment associated with the ESF equipment associated with the ESF initiation signal and then resumes the reloading sequence.
10.c) Loss of power in an ESF-CCS Division results in ESF-CCS outputs assuming fail-safe output operation.	10.c) Testing will be performed simulating loss of power in the ESF-CCS Division.	10.c) Loss of power in an ESF-CCS Division results in ESF-CCS outputs assuming fail-safe output operation.
10.d) Protective devices are designed to detect loss of power if a setpoint is exceeded.	10.d) Inspection of the as-built protective devices will be performed.	10.d) Protective devices are installed to detect loss of power, if a setpoint is exceeded.

The ESF-CCS provides control capability and, upon receipt of initiation signals from the PPS, automatically generates actuation signals to the following ESF systems within allocated response times:

COMMENT 2.5.2.15

- safety injection system,
- containment isolation system,
- containment spray system,
- main steam isolation, and
- emergency feedwater system.

steam generator 1 and steam generator 2

Once initiation signals are received from the PPS, the ESF-CCS actuation logic signals remain following removal of the initiation signal.

The ESF-CCS provides control capability and, upon receipt of initiation signals from the PPS, automatically generates actuation signals to the following non-ESF systems:

- annulus ventilation system,
- component cooling water system,
- onsite power system, and
- diesel generators.

The ESF-CCS provides control capability for the following safety-related systems:

- shutdown cooling system,
- safety depressurization system,
- atmospheric dump system,
- station service water system,
- heating, ventilating and air conditioning systems, and
- hydrogen mitigation devices.

Upon receipt of ESF initiation signals for safety injection, containment spray or emergency feedwater, the ESF-CCS initiates an automatic start of the diesel generators and automatic load sequencing of ESF loads.

Upon detecting loss of power to Class 1E Division buses through protective devices, the ESF-CCS automatically initiates startup of the diesel generators, shedding of electrical load, transfer of Class 1E bus connections to the diesel generator, and sequencing of the reloading of safety-related loads to the Class 1E bus. In performing load sequencing, normally used safety related plant loads are loaded first in a predetermined sequence unless an ESF actuation signal is generated. Upon ESF actuation, the normal load sequence is interrupted and priority is given to loading the actuated ESF systems and associated safety-related systems. The sequence for loading the normally used safety related plant loads is then resumed.

TABLE 2.5.2-1 (Continued)

ENGINEERED SAFETY FEATURES COMPONENT CONTROL SYSTEM
Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. (Continued) Once initiation signals are received from the PPS, the ESF-CCS actuation logic signals remain following removal of the initiation signal.	6.b) Tests will be performed using signals simulating ESF initiation to the ESF-CCS.	6.b) ESF initiation signals which satisfy the selective 2 out of 4 criteria result in actuation signals for related system components for the following systems: safety injection system, containment isolation system, containment spray system main steam isolation, and emergency feedwater system. COMMENT 2.5.2.15 [steam generator 1 and Steam generator 2
	6.c) Tests will be performed using signals simulating ESF initiation to the ESF-CCS.	6.c) Measured response times are less than or equal to the response time values required for each ESF actuation signal.
	6.d) Testing will be performed using signals simulating ESF initiation to the ESF-CCS.	6.d) Once initiated, ESF-CCS actuation logic signals remain following removal of the initiation signal.
7. The ESF-CCS provides control capability and, upon receipt of initiation signals from the PPS, automatically generates actuation signals to the following non-ESF systems: annulus ventilation system, component cooling water system, onsite power system, diesel generators, and control complex ventilation system.	7.a) Tests will be performed on the as-built ESF-CCS control and display interface equipment.	7.a) The control and display interface equipment provide control capability for the following systems: annulus ventilation system, component cooling water system, onsite power system, diesel generators, and control complex ventilation system.

TABLE 2.5.2-1 (Continued)

ENGINEERED SAFETY FEATURES COMPONENT CONTROL SYSTEM
Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. The operator interface devices of the ESF-CCS in the MCR provide for automatic and manual control of ESF systems and components.	12. Addressed in 6.a), 7.a) and 8.	12. Addressed in 6.a), 7.a) and 8.
13. In the remote shutdown room, operator interface devices provide for manual control of ESF system components needed to achieve hot standby.	13. Tests will be performed on the as-built ESF-CCS control and display interface devices in the remote shutdown room following a transfer of control capability to the remote shutdown room. COMMENT 2.5.2.15 steam generator 1 and steam generator 2	13. Control capability is provided at the ESF-CCS control and display interface devices in the remote shutdown room for the following systems: safety injection system, emergency feedwater system, component cooling water system, onsite power system, diesel generators, shutdown cooling system, safety depressurization system, atmospheric dump system, station service water system, and heating, ventilating and air conditioning systems.

ENGINEERED SAFETY FEATURES COMPONENT CONTROL SYSTEM
Inspections, Tests, Analyses and Acceptance Criteria

Comment 2.5.2.16

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
14.a) Actuation of master transfer switches at either exit ~~of~~ to the MCR transfers control capability from the ESF-CCS control and display interface devices depicted in the MCR to those in the remote shutdown room. Indication of transfer status is provided in the MCR.	14.a) Tests will be performed using master transfer switches at each exit ~~of~~ to the MCR and each of the ESF-CCS control and display interface devices in the MCR and the remote shutdown room.	14.a) Upon actuation of the master transfer switches at either MCR exit: 1) control actions at the ESF-CCS control and display interface devices do not cause the ESF-CCS to generate the associated control signals, and 2) control actions at the ESF-CCS control and display interface devices in the remote shutdown room cause the ESF-CCS to generate the associated control signals. 3) indication of transfer status is provided in the MCR.
14.b) Each ESF-CCS division's maintenance and test panel provides capability to transfer control from the MCR to the remote shutdown panel for its respective ESF-CCS division and to transfer control back to the MCR for its respective ESF-CCS division.	14.b) Testing will be performed using each ESF-CCS division's maintenance and test panel and control and display interface devices in the MCR and the remote shutdown room.	14.b) Upon actuation of the master transfer switching function from each ESF-CCS division's maintenance and test panel: 1) control actions at the ESF-CCS control and display interface devices in the MCR for that ESF-CCS division do not cause the ESF-CCS to generate the associated control signals, and

The ESF-CCS provides control capability and, upon receipt of initiation signals from the PPS, automatically generates actuation signals to the following ESF systems within allocated response times:

safety injection system,
containment isolation system,
containment spray system,
main steam isolation, and
emergency feedwater system.

Once initiation signals are received from the PPS, the ESF-CCS actuation logic signals remain following removal of the initiation signal.

Comment 2.5.2.17 Insert A

The ESF-CCS provides control capability and, upon receipt of initiation signals from the PPS, automatically generates actuation signals to the following non-ESF systems:

annulus ventilation system,
component cooling water system,
onsite power system, and
diesel generators.

The ESF-CCS provides control capability for the following safety-related systems:

shutdown cooling system,
safety depressurization system,
atmospheric dump system,
station service water system,
heating, ventilating and air conditioning systems, and
hydrogen mitigation devices.

Upon receipt of ESF initiation signals for safety injection, containment spray or emergency feedwater, the ESF-CCS initiates an automatic start of the diesel generators and automatic load sequencing of ESF loads.

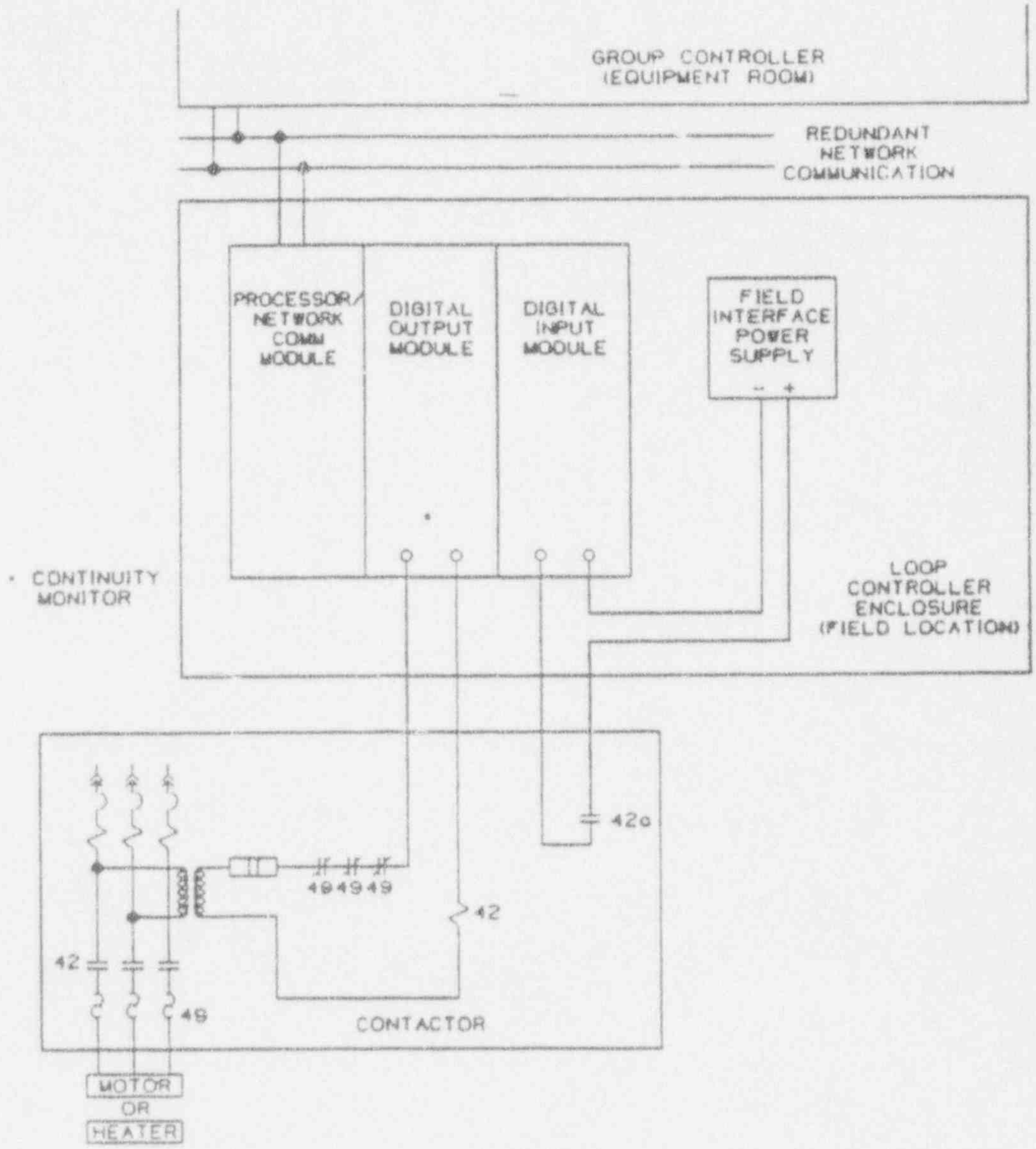
Upon detecting loss of power to Class 1E Division buses through protective devices, the ESF-CCS automatically initiates startup of the diesel generators, shedding of electrical load, transfer of Class 1E bus connections to the diesel generator, and sequencing of the reloading of safety-related loads to the Class 1E bus. In performing load sequencing, normally used safety related plant loads are loaded first in a predetermined sequence unless an ESF actuation signal is generated. Upon ESF actuation, the normal load sequence is interrupted and priority is given to loading the actuated ESF systems and associated safety-related systems. The sequence for loading the normally used safety related plant loads is then resumed.

Comment 2.5.2.17

Insert A

ESF functions are assigned to individual group control segments within each ESF-CCS division. This functional assignment approach limits the effect of a single group failure to selected ESF functions in a given division.

Additional segmentation of functional assignment is applied within each ESF-CCS group control segment. This practice limits the effect of a single multiplexer or module failure to selected ESF functions in the division. ESF system interfaces are also confined within group control segments to minimize reliance on the Intradivision Communication Network for ESF operability.



COMMENT
2.5.2.19

Amendment N
April 1, 1993

SYSTEM 80+™	TYPICAL ELECTRICAL INTERFACE FOR A CONTRACTOR OPERATED COMPONENT	Figure 7.3-13b

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.3 (Discrete Indication and Alarm System

No.	Comments	Cat.	Resolution
1	ITAAC item 10 DIAS/DPS software development process is not described in the CDM. The software methodology listed is described in the PPS and ESFAS ITAACs for safety related software. Is the DIAS and DPS software to be developed with the above methodology or is the above intended for the PAMI system.	1	Agree with added clarification. DIAS/DPS software is developed in accordance with the "Nuplex 80+ Software Program Manual" which includes each of the elements of ITAAC Item 10. It is not intended that the Software Program Manual apply to only PAMI software. CESSAR-DC Section 7.7.1.7.4 invokes the Software Program Manual for DPS software. A similar reference will be added to CESSAR-DC for DIAS software.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.3 (Discrete Indication and Alarm System

No.	Comments	Cat.	Resolution
2	The above ITAAC references a graded approach to software development. See previous comments regarding PPS and ESFAS ITAACs.	2	No Change Required. The graded approach to software development is fully described in the "Nuplex 80+ Software Program Manual." This manual is referenced in CESSAR-DC and has been approved for use by the USNRC. The FSER stipulates that any later changes to the specified sections of this manual be reviewed and approved by the USNRC. Software has historically been graded in two classes; safety and non-safety. With NUREG 0696, the USNRC initiated V&V for Safety Parameter Display software. This non-safety software became distinguished from other non-safety software by its level of scrutiny during the design process. The level of scrutiny was not intended to be as rigorous as for safety software, such as for much simpler core protection software. Thus, this defacto third grade of software became evident in the design process. This grade of software has never been formally defined in software standards. The use of a graded approach was developed by the A/E and ABB-CE during the USDOE HWRF program.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.3 (Discrete Indication and Alarm System

No.	Comments	Cat.	Resolution
3	DCM states that EMI qualification applies to equipment with known EMI susceptibility based on operating environment and/or inherent design characteristics. The DCM states that a site survey will be performed after installation to characterize the EMI environment. The site survey requirement is not listed in the ITAAC. Will emissions testing also be performed? To what standard?	1	Agree. EMI qualification is applied for equipment with known EMI susceptibility based on operating environment and/or inherent design characteristics. ABB-CE interprets equipment with known EMI susceptibility to typically include but not be limited to power supplies, programmable logic controllers, digital computers, communications interface equipment and input/output equipment. It is known based on previous experience that certain equipment such as electromechanical switches, contactors, medium and heavy duty relays, terminal blocks, connectors, indicator lamps, ... etc. are not susceptible to radiated or conducted EMI effects. The suggested change will be incorporated in the CDM.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.3 (Discrete Indication and Alarm System

No.	Comments	Cat.	Resolution
4	ITAAC 18 describes commercial dedication of software. Is this applicable to DIAS and DPS only or just post accident monitoring (PAMI)? If applicable to PAMI it should be stated that commercial dedication is applicable to system development (programming language, operating system) only.	1	The requirements for the commercial dedication of software apply to both DIAS and DPS. This dedication program is described in Section 3.10 of the "Nuplex 80+ Software Program Manual." PAMI software is classified as Important to Safety and must meet the more rigorous supplier requirements defined in "Requirements for Supply of Commercial Digital Computer Hardware and Software Components to be used in Nuplex 80+ Safety Systems." There is no absolute restriction on the use of commercial software in safety systems provided it successfully completes the dedication program.
5	The DCM and ITAAC state that the DIAS displays and processors are designed for room ambient temperatures and humidity environmental conditions. The CESSAR 7.5.2.5, page 7.5-18 states that the temperature and humidity qualification exceeds the most severe equipment environment by a design margin. Clarify the CESSAR to indicate that listed temperature and humidity qualification applies to the PAMI system and not the DIAS or DPS portions of post accident monitoring.	1	Agree. The statement that "Temperature and Humidity qualification exceeds the most severe equipment environment by a design margin" is accurate with regard to DIAS and DPS equipment locations. CESSAR will be revised to indicate these environmental conditions apply to PAMI.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.3 (Discrete Indication and Alarm System

No.	Comments	Cat.	Resolution
6	Typographical error - CESSAR, page 7.5-18 first paragraph repeats "channel up to the ...". See page 7.5-17 last paragraph.	3	Agree. CESSAR-DC will be corrected.
7	The CDM material indicates that communication provided to the DIAS-P are diverse from the communication software used in the plant protection system (PPS) and the engineered safety features-component control system. From Figures 2.5.3.1 and 2.5.3.2 and the DCM material it is unclear how diverse communication is implemented between the protection system and DIAS-P and PAMI and PAMI to DIAS-P, DIAS channel N, and DPS. Additionally, communication link depicted for protection system SC-A and B to the DIAS channel P displays should be shown hardwired. See Figure 7.5-1 CESSAR. Clarification needed.	1	Disagree. The approach in the design description and ITAAC was agreed to with the Staff. This approach provides the desired flexibility in the design. NRC Concurs

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.3 (Discrete Indication and Alarm System

No.	Comments	Cat.	Resolution
8	The DCM states that on a loss of electrical power the DIAS display will result in a blank screen, inactive running indicator or bad data symbol. Provide a description of the diagnostics and system alarms (error detection and diagnostics) available to inform the operator/maintenance of annunciator failures or problems beyond that discussed in the CDM for power supply failure. (see unrecognized loss of annunciators IN 93-47)	1	Disagree. ABB-CE believes that the current level of detail in the ITAAC is sufficient. The DIAS and DPS are independent systems and information is available through either system. NRC Concurs
9	Are the display rates for the DIAS adequate to provide information to the operator in a timely manner based on expected operational occurrences (transients). This is not discussed in the CDM.	1	The DIAS is designed to provide a two second update of dynamic data under worst-case loading. This is sufficient to follow plant transients without introducing a data readability problem. This is not considered Tier 1 level of detail. NRC Concurs

7.7.1.4.5 DIAS Quality Classification

The DIAS performs no direct plant safety function since it strictly monitors and displays data. However, due to its importance, DIAS is designed in accordance with a quality program to assure product quality commensurate with the intended use of the equipment.

Although the DIAS is a non-safety system, it receives both Class 1E and non-Class 1E input signals. All DIAS input/output equipment is qualified to not generate faults that would degrade the channel A, B, C and D Class 1E inputs. As shown in Figures 7.5-1 and 7.7-17, the Class 1E signal interface equipment uses qualified fiber-optic network interfaces to maintain isolation.

Comment 2.5.3.1

Insert 7.7.1.4.6

7.7.1.5 <u>Integrated Process Status Overview (IPSO)</u>

The IPSO provides a single location to allow quick assessment of key information indicative of critical plant power production and safety functions. The IPSO displays information that both the operators and supervisory personnel require for quickly assessing overall plant status. It indicates existence of Priority 1 alarms, deviations from control setpoints, key parameter values, and system operational status and non-operational availability in a schematic representation. The IPSO is implemented as a big board mimic display located above the Master Control Console (MCC) in the Nuplex 80+ control room (see Chapter 18) and as a top level DPS CRT display page.

The plant systems represented on the IPSO will be the major heat transport path systems and systems that are required to support the major heat transport process, either power or safety related. These systems include those that require availability monitoring per Regulatory Guide 1.47.

The following systems have dynamic representations on IPSO:

CCW - Component Cooling Water
CD - Condensate
CI - Containment Isolation
CS - Containment Spray
CW - Circulating Water
EF - Emergency Feedwater
FW - Feedwater
IA - Instrument Air
SDC - Shutdown Cooling
RCS - Reactor Coolant
SI - Safety Injection
SW - Service Water
TB - Turbine Bypass

Comment Q.5.3.1

Insert 7.7.1.4.6

7.7.1.4.6 DIAS Verification and Validation Requirements

The DIAS is subject to a thorough and systematic verification and validation program to assure that the system is correctly implemented and satisfies all its functional requirements. The program is implemented in accordance with Reference 3.

Electrical isolation devices are provided at DIAS-N and DPS interfaces to the PPS, ESF-CCS, PCS/P-CCS and at interfaces to display devices in the MCR and remote shutdown room.

Electrical isolation is provided between the DIAS-P display devices and protection system signal conditioning equipment, as shown on Figure 2.5.3-2.

DIAS uses redundant networks for communications. The networks utilize isolation technology (e.g., fiber optics) to ensure electrical independence of the redundant safety channels and electrical independence of the MCR and the RSR. The DIAS communications network provide communication paths to allow display of information from safety-related I&C systems. Data communications is on a cyclical basis, independent of plant transients.

A loss of electrical power to DIAS or DPS equipment will result in a blank screen, inactive running indicator, or bad data symbol.

Comment 2.5.3.3

EMI qualification is applied for equipment ~~with known EMI susceptibility~~ based on operating environment and/or inherent design characteristics.

The DIAS/DPS is qualified according to an established plan for Electromagnetic Compatibility (EMC).

The qualification plan requires the equipment to function properly when subjected to the expected operational electrical surges, electromagnetic interference (EMI), electrostatic discharge (ESD), and radio frequency interference (RFI).

The equipment to be tested will be configured for intended service conditions.

A site survey is performed upon completion of system installation to characterize the installed EMI environment.

The use of commercial grade computer hardware and software items in the DIAS/DPS is accomplished through a process that has:

- requirements for supplier design control, configuration management, problem reporting and change control;
- review of product performance;
- receipt acceptance of the commercial grade item;
- final acceptance, based on equipment qualification and software validation in the integrated system.

DIAS/DPS software is designed, tested, installed and maintained using a process which:

Alarms are categorized into 3 priorities (Priority 1, Priority 2, and Priority 3) to help establish a hierarchy for responding to abnormal conditions. All Priority 1 and Priority 2 alarms and Priority 3 alarms which degrade to Priority 2 and 1 conditions are processed by DIAS.

Refer to Chapter 18 for more information regarding the alarm logic algorithms, prioritization, mode dependencies, first out alarms, other dynamic features and HFE design aspects.

Individual DIAS segments are designed such that a failure of one segment's processor or a communications link will not affect any segment's alarms. Additional failures may result in the degradation of one (or more) segment's displays. If this occurs, alarms and alarm discrepancies are still provided independently by the DPS CRT displays and printer logs.

Since message displays are driven by the DIAS segments as previously described, their failure modes are the same as described above for the alarms. If they should fail, descriptive alarm information will be available independently via the DPS CRT displays and printer logs.

The failure of an individual alarm indication and message display has no adverse impact on that segment's CPU. The CPU still functions to generate alarms that are transmitted to the DPS for display and acknowledgement.

7.7.1.4.4 DIAS Environmental Qualification

Class 1E instrument channels are seismically and environmentally qualified up to and including the channel isolation device (fiber optic modems) such that the instrument channel is not degraded.

The DIAS displays and central processing units are non-Class 1E designed to meet control room and electronic equipment room ambient temperature, pressure and humidity requirements. Each cabinet containing DIAS computer equipment is provided with a temperature switch and associated alarm in the main control room, to alert the operator if the temperature within the cabinet reaches the upper limit specified for the environment in that location.

All DIAS displays and CPUs are seismically qualified for physical and functional integrity to enhance control room information availability.

Comment 2.5.3.3 DIAS equipment ~~with known susceptibility~~ is qualified according to an established plan for electromagnetic compatibility (EMC) that requires the equipment to function properly when subjected to electrical surges, electromagnetic interference (EMI), electrostatic discharge (ESD) and radio frequency interference (RFI). Qualification is applied for equipment based on operating environment and/or inherent design characteristics.

Comment 2.5.3.5

[For DPS and DIAS display equipment including equipment used for PAMI display.

channel up to the channel isolation device. Class 1E signals are isolated either prior to transmission to or within qualified I/O sections of the DIAS and DPS. The DIAS displays and processing units are non-Class 1E, but are considered important to safety; therefore, they are seismically qualified to enhance channel availability. The DPS also displays all Category 1 variables, though it is designed as a non-safety system with no functional seismic qualification.

Temperature and humidity qualification exceeds the most severe equipment environment by a design margin. DIAS and DPS cabinet temperature alarms are provided to alert the operator if the cabinet temperature exceeds the limit specified for that location.

Category 2: Available displays for Category 2 variables are provided by the DIAS channel N and/or the DPS. All Category 2 variables are available on the DPS. Qualification for Category 2 variables is the same as for Category 1, except that there is no specific seismic qualification of the display devices.

Category 3: No specific qualification requirements apply for sensors or displays. These variables are presented by DIAS channel N and/or the DPS, as appropriate for each variable.

A more detailed discussion of the environmental qualification is provided in Section 3.11.

7.5.2.5.2 Redundancy

Category 1: Redundancy with respect to Category 1 variables is provided for both the instrument channels supplying the signal and for the displays in the control room. Instrument channels are electrically independent and physically separated from each other and from non-safety equipment by qualified isolation devices. Credited redundancy for the display of Category 1 variables is 2provided by the channel P and channel N DIAS displays. These displays are electrically independent and physically separated. To minimize technical specification limitations for conditions when a DIAS channel is out of service, each Category 1 variable is also presented on the DPS. The DPS is physically separated and independent from both DIAS channels. Channel availability is further discussed in Section 7.5.2.5.4.

The following design criteria were used in providing the CEA position indication function:

A. Position readouts of all CEAs may be obtained.

B. Continuous position indication of all CEAs is provided.

C. A means is provided to alert the operator of CEA deviations within a group.

D. A permanent record may be made of the position of any or all CEAs.

E. The "full-in" and "full-out" indications are provided for each CEA.

F. Redundant and diverse means of monitoring and indicating CEA position are provided.

7.5.2.5 Analysis of Post-Accident Monitoring Instrumentation

The Post-Accident Monitoring Instrumentation (PAMI) that is identified in Table 7.5-3 is provided for remote monitoring of post-accident conditions. Post-accident conditions are defined as those conditions which exist during and following an accident.

The extensive instrumentation identified in Table 7.5-3 provides the plant operator with long-term monitoring and surveillance capabilities of post-accident conditions within the primary containment. Table 7.5-3 identifies Category 1, 2, and 3 variables from Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

PAMI shall function with precision and reliability to display the appropriate monitored variables. Each instrument's performance characteristics, response time and accuracy are compatible with the design goal of providing the operator with reliable information.

The guidance of Regulatory Guide 1.97 is applicable to the design of the PAMI and are applied to the design of this instrumentation by appropriate category for each variable as follows.

7.5.2.5.1 Equipment Qualification

Comment 7.5.3.6

Category 1: Available displays for Category 1 variables are the DPS and DIAS channels P and N. Class 1E qualification includes the entire instrument ~~channel up to the~~

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.4 (PCS/P-CCS)

No.	Comments	Cat.	Resolution
1	PCS/P-CCS System provides various functions which include the display of safety-related variables, the required interfacing logic circuitry for safety related control circuits, the actuation of alternate reactor trips and turbine trips, etc. but since it is functionally classified as non-safety related and the presentation is so nebulous, the paramount questions seems to be why it is included in the design certification material? Any subsequent questions are dependent on the response to this question. However, if it is decided to allow it to remain in certified material, then the writeup in Section 2.5.4 in regard to the actual bounds and limitations of this system should be made much more clearer.	1	No action required per discussion with USNRC.
2	The level of detail provided in Figure 2.5.4-1 does not seem to warrant the satisfactory completion of ITAAC #1 with any degree of confidence since the bounds and internals of the system are so vaguely defined. (See General comment on basic configuration.)	1	No action required per discussion with USNRC.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.4 (PCS/P-CCS)

No.	Comments	Cat.	Resolution
3	All control interfaces should be more clearly defined since the SAR does not provide any specific information as to what they consist of generically but addresses this particular point in the vague writeups/sketches of the affected subsystems which comprise the PCS/P-CCS System. Define these interfaces with enough detail so as to determine whether their failure has any impact on the affected safety related systems.	1	Agree with added clarification. Control functions are defined in CESSAR-DC Section 7.7. Control interfaces are represented in this SAR material. This CESSAR-DC section will be revised to generically define the interfaces in sufficient detail to define their affect on safety related systems.
4	ITAAC No.4 should be performed actually by injecting test signals into the affected circuits instead of verification of design documentation.	1	Disagree. ITAAC No. 4 addresses the independence and diversity of alternate reactor trip, turbine trip, and emergency feedwater actuation. The diversity and independence of these functions from the protection system actuation circuits can be adequately demonstrated by verification of design documentation. NRC CONCURS
5	ITAAC No. 5 does not appear to be actually simulating the function of this system in regard to signal validation and also determining the particular sensor that is either bypassed/failed.	1	No action required per discussion with USNRC.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.5.4 (PCS/P-CCS)

No.	Comments	Cat.	Resolution
6	ITAAC No. 6 should also verify that the MCR displays can not be reset utilizing the master transfer switches at the MCR exits.	1	No action required per discussion with USNRC.
7	ITAAC Nos. 9.a) and 9.b) try to verify that isolation devices are installed between this system and the affected systems with which it interfaces. However to accomplish this task the number of inputs from each subsystem which require individual isolation devices should be identified either in the certified material or the SAR since the applicable sketches lack the required detail.	1	No action required per discussion with USNRC.

Class 1E/Safety (PPS, ESF-CCS)

- Channelized Redundancy
- Seismic
- Independent Verification
- Independent QA
- Complete Configuration Control
- Single Task CPUs
- No Interrupts
- Deterministic Design
- Off-line Changes Only
- Read-only Data Communications with Isolation

These systems are provided as an integral part of the Nuplex 80+ Advanced Control Complex (ACC). As such, they provide the operating staff the ability to monitor the plant's operating status, change its operating mode and take those actions necessary to maintain the plant within its design basis for all normal modes of plant operation. The ACC systems and equipment described below include the control systems, main control panels and monitoring systems.

Comment 2.5.4.3

Insert B

7.7.1.1 Control Systems

The general description given below permits an understanding of the reactor and important subsystem control methodology.

The design reactivity feedback properties of the NSSS will inherently cause reactor power to match the total NSSS load. The resulting reactor coolant temperature at which this occurs is a controlled parameter and is adjusted by changes in total reactivity as implemented through CEA position changes or through boric acid concentration changes in the primary coolant.

The ability of the NSSS to follow turbine load changes is dependent on the ability of the control systems or operator to adjust reactivity, feedwater flow, bypass steam flow, reactor coolant inventory, and energy content of the pressurizer such that NSSS conditions remain within normal operating limits.

Except as limited by Xenon conditions, the major control systems described below provide the capability to automatically follow design load changes. Additionally, these automatic systems provide the capability to accommodate load rejections of any magnitude or the loss of one of two operating feedwater pumps.

Comment 2.5.t.3

Insert B

Non-1E Category 1 and Category 2 control and monitoring systems which interface with Class 1E Safety systems are designed such that credible failures in the control and monitoring systems will not impact the operation of 1E safety systems.

Interfaces between these systems employ isolation devices to maintain electrical independence between channels. Isolation devices are qualified for design base events including seismic, environmental, electromagnetic interference (EMI) and electrical fault isolation.

Where 1E safety related transmitters and signal conditioning devices provide parameters for control and/or monitoring, signal isolation is applied between the safety systems and the control and monitoring systems. Signal validation is performed within the control systems as described in Section 7.7.1.1.13.

The DIAS and DPS monitoring systems receive data from both Class 1E and non-Class 1E systems via qualified fiber-optic network interfaces to maintain isolation.

Main Control Panels and the Remote Shutdown Panel are designed to support human-machine interface devices for each Class 1E safety channel as well as non-Class 1E control channels. Low energy circuits are used and all panel mounted control and monitoring devices are isolated such that credible electrical faults originating in one channel will not affect other channels. Control panels are designed to maintain structural integrity, such that no control room missile hazard will occur as a result of a seismic event.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.6.1 (AC Electrical Power Distribution System

No.	Comments	Cat.	Resolution
1 (3)	DD and ITAAC 24 need to be revised as shown in the attached mark-up.	1	Agree. DD and DC should add "transformers". See markup of ITAAC 2.6.1. Acceptance Criteria O.K. as is.
2 (5)	DD needs to be revised to show controls and displays provided in the MCR for EPDS. Appropriate ITAAC needs to be provided to verify this.	1	Agree. An appropriate DD and DC/ITA/AC will be added. See markup of ITAAC 2.6.1.
3 (6)	The same list of abbreviations/acronyms are shown in pages viii and ix of CESSAR, Chapter 8. Once of the pages needs to be deleted.	1	CESSAR-DC Effective pages does not list page IX. Therefore clerical removal of this page from the CESSAR-DC set is all that is required. Dale Thatcher notified of this by phone 4/8/94 (MDC).

Note: Markups contain additional comments added at the suggestion of Charles Thomas (NRC). They also include revisions to resolve a question concerning associated circuits, deletion of "channel", and relocation of the Seismic Category I statement.

2.6.1 AC ELECTRICAL POWER DISTRIBUTION SYSTEM

DESIGN DESCRIPTION

(L/Cs)

The AC Electrical Power Distribution System (EPDS) consists of the transmission system, the plant switching stations, the Unit Main Transformer (UMT), two Unit Auxiliary Transformers (UATs), two Reserve Auxiliary Transformers (RATs), a Main Generator (MG), Generator Circuit Breaker (GCB), buses, switchgear, load centers, motor control centers (MCCs), breakers and cabling. The EPDS includes the power, instrumentation and control cables and buses to the distribution system loads, and electrical protection devices (circuit breakers and fuses) for the power, instrumentation and control cables and buses. The portion of the EPDS from the high sides of the UMT and RATs to the distribution system loads constitutes the EPDS Certified Design scope. Interface requirements for the transmission system, plant switching stations, UMT and RATs are specified below under the heading, "Interface Requirements."

Insert A

Two Emergency Diesel Generators (EDGs) provide Class 1E power to the two independent Class 1E Divisions. ~~as described in Section 2.6.2.~~

See page 3 #

A non-safety-related Alternate AC Source (AAC) (i.e., combustion turbine) supplies non-Class 1E power to the EPDS. ~~as described in Section 2.6.5.~~

The Basic Configuration of the Class 1E portion of the EPDS is as shown on Figure 2.6.1-1.

During plant power operation, the MG supplies power through the GCB through the UMT to the transmission system, and to the UATs. When the GCB is open, power is backfed from the transmission system through the UMT to the UATs.

The UATs are sized to supply the design operating requirements of their respective Class 1E buses and non-Class 1E medium voltage non-safety and permanent non-safety buses.

The UMT and UATs are separated from the RATs.

UMT, UATs, and RATs are provided with their own oil pit, drain, fire deluge system, grounding, and lightning protection systems.

The MG and GCB are separated from the RAT power feeders. The MG and GCB instrumentation and control circuits are separated from the RAT's instrumentation and control circuits.

Each RAT is sized to supply the design operating power requirements of at least its respective Class 1E buses and permanent non-safety bus, and one reactor coolant

Insert A:

"The Backup Pressurizer Heaters, Emergency Lighting, RCP Seal Injection Pump, and RCP Seal Injection Pump Room Ventilation Fan are the only electrical loads classified as non-Class 1E which are directly connectable to the Class 1E buses."

pump and its reactor coolant pump support loads. Each RAT has the capability of supplying power directly (i.e., not through any bus supplying non-Class 1E loads) to its respective Class 1E buses.

UAT power feeders, and instrumentation and control circuits are separated from the RAT's power feeders, and instrumentation and control circuits

Power feeders, and instrumentation and control circuits for the UMT and its switching station are separated from power feeders, and instrumentation and control circuits for the RATs and their switching station.

EPDS medium voltage switchgear, low voltage switchgear and their respective transformers, MCCs, and MCC feeder and load circuit breakers are sized to supply their load requirements. EPDS medium voltage switchgear, low voltage switchgear and their respective transformers, and MCCs are rated to withstand fault currents for the time required to clear the fault from its power source.

The GCB, medium voltage switchgear, low voltage switchgear, and MCC feeder and load circuit breakers are rated to interrupt fault currents.

EPDS interrupting devices (circuit breakers and fuses) are coordinated so that the circuit interrupter closest to the fault is designed to open before other devices.

Instrumentation and control power for Class 1E Divisional medium voltage switchgear and low voltage switchgear is supplied from the Class 1E DC Power System in the same Division.

The GCB is equipped with redundant trip ~~coils~~ devices supplied from separate non-Class 1E DC power systems.

EPDS cables and buses are sized to supply their load requirements. EPDS cables and buses are rated to withstand fault currents for the time required to clear the fault from its power source.

For the EPDS, Class 1E power is supplied by two independent Class 1E Divisions. Independence is maintained between Class 1E Divisions, and between Class 1E Divisions and non-Class 1E equipment.

Class 1E medium voltage switchgear, low voltage switchgear, and MCCs are identified according to their Class 1E Division/~~Channel~~. Class 1E medium voltage switchgear, low voltage switchgear, and MCCs are located in Seismic Category I structures and in their respective Division areas.

Displays of EPDS voltage, amperage, frequency, watts, and vars instrumentation exist in the main control room (MCR) or can be retrieved there.
Controls exist in the MCR to operate the EPDS.

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Class 1E EPDS cables and raceways are identified according to their Class 1E Division. Class 1E EPDS cables are routed in Seismic Category I structures and in their respective raceways.

Class 1E equipment is not prevented from performing its safety functions by harmonic distortion waveforms.

The EPDS supplies an operating voltage at the terminals of the Class 1E equipment which is within the equipment's voltage tolerance limits.

Class 1E equipment is protected from degraded voltage conditions.

transformers,

An electrical grounding system is provided for (1) instrumentation, control, and computer systems, (2) electrical equipment (switchgear, motors, distribution panels), and (3) mechanical equipment (fuel and chemical tanks). Lightning protection systems are provided for buildings, ~~and for~~ structures located outside of the buildings. Each grounding system and lightning protection system is separately grounded to the plant ground grid.

and transformers

Class 1E equipment is classified as Seismic Category I.

There are no automatic connections between Class 1E Divisions.

Interface Requirements

The offsite system shall consist of a minimum of two independent offsite transmission circuits from the transmission system.

move page 1 @ #

The offsite transmission circuits shall be sized to supply their load requirements, during all design operating modes, of their respective Class 1E divisions and non-Class 1E loads.

The UMT and RATs shall be connected to independent switching stations. Switching stations and their circuit breakers shall be sized to supply their load requirements and be rated to interrupt fault currents.

Voltage variations of the transmission system shall not cause voltage variations at the loads of more than plus or minus 10% of the loads' nominal voltage rating.

The normal steady-state frequency of the offsite system shall be within plus or minus 2 Hertz of 60 Hertz during recoverable periods of system instability.

The transmission system does not subject the reactor coolant pumps to sustained frequency decays of greater than 3 Hertz per second.

~~Each switchyard shall have two redundant and independent 125V DC power systems to provide 125V DC power for all relaying, controls, and monitoring equipment in the switchyards.~~

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.6.1-1 specifies the inspections, tests, analysis, and associated acceptance criteria for the AC Electrical Power Distribution System.

AC ELECTRICAL POWER DISTRIBUTION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. EPDS medium voltage switchgear, low voltage switchgear and their respective transformers, MCCs, and MCC feeder and load circuit breakers are sized to supply their load requirements.	9.~~a)~~ Analysis for the as-built EPDS to determine load requirements will be performed.	9.~~a)~~ Analysis for the as-built EPDS exists and concludes that the capacities of the Class 1E medium voltage switchgear, low voltage switchgear and their respective transformers, MCCs, and MCC feeder and load circuit breakers, as determined by their nameplate ratings, exceed their analyzed load requirements.
	~~9.b) Testing of the as-built Class 1E medium voltage and low voltage switchgear and MCCs and their respective load circuit breakers will be performed by operating connected Class 1E loads in the ranges of 9% to 10% above and 9% to 10% below design voltage.~~	~~9.b) Connected Class 1E loads operate in the ranges of 9% to 10% above and 9% to 10% below design voltage.~~
10.a) EPDS medium voltage switchgear, low voltage switchgear and their respective transformers, and MCCs are rated to withstand fault currents for the time required to clear the fault from its power source.	10.a) Analysis for the as-built EPDS to determine fault currents will be performed.	10.a) Analysis for the as-built EPDS exists and concludes that the current capacities of the Class 1E medium voltage switchgear, low voltage switchgear and their respective transformers, and MCCs exceed their analyzed fault currents for the time required, as determined by the circuit interrupting device coordination analyses, to clear the fault from its power source.

TABLE 2.6.1-1 (Continued)

AC ELECTRICAL POWER DISTRIBUTION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10.b) The GCB, medium voltage switchgear, low voltage switchgear, and MCC feeder and load circuit breakers are rated to interrupt fault currents.	10.b) Analysis for the as-built EPDS to determine fault currents will be performed.	10.b) Analysis for the as-built EPDS exists and concludes that the analyzed fault currents do not exceed the GCB and ~~Class 1E~~ medium voltage switchgear, low voltage switchgear, and MCC feeder and load circuit breakers interrupt capacities, as determined by their nameplate ratings.
11. EPDS interrupting devices (circuit breakers and fuses) are coordinated so that the circuit interrupter closest to the fault is designed to open before other devices.	11. Analysis for the as-built EPDS to determine circuit interrupting device coordination will be performed.	11. Analysis for the as-built EPDS exists and concludes that the analyzed Class 1E circuit interrupter closest to the analyzed fault will open before other devices.
12. Instrumentation and control power for Class 1E Divisional medium voltage switchgear and low voltage switchgear is supplied from the Class 1E DC power system in the same Division.	12. Testing of the as-built Class 1E medium and low voltage switchgear will be conducted by providing a test signal in only one Class 1E Division at a time.	12. A test signal exists in only the ~~circuit~~ Class 1E Division under test.
13. The GCB is equipped with redundant trip devices which are supplied from separate non-Class 1E DC power systems.	13. Testing of the as-built GCB will be conducted by providing a test signal in only one trip circuit at a time.	13. A test signal exists in only the circuit under test.
14. EPDS cables and buses are sized to supply their load requirements.	14. Analysis for the as-built EPDS cables and buses will be performed.	14. Analysis for the as-built EPDS exists and concludes that Class 1E cables and bus capacities, as determined by cable and bus ratings, exceed their analyzed load requirements.

TABLE 2.6.1-1 (Continued)

AC ELECTRICAL POWER DISTRIBUTION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
15. EPDS cables and buses are rated to withstand fault currents for the time required to clear the fault from its power source.	15. Analysis for the as-built EPDS to determine fault currents will be performed.	15. Analysis for the as-built EPDS exists and concludes that Class 1E cables and buses will withstand the analyzed fault currents for the time required, as determined by the circuit interrupting device coordination analyses, to clear the analyzed faults from their power sources.
16. For the EPDS, Class 1E power is supplied by two independent Class 1E Divisions. Independence is maintained between Class 1E Divisions/Channel, and between Class 1E Divisions and non-Class 1E equipment.	16.a) Testing on the as-built EPDS will be performed by providing a test signal in only one Class 1E Division/Channel at a time.	16.a) A test signal exists in only the Class 1E Division/Channel under test in the EPDS.
	16.b) Inspection of the as-built EPDS Class 1E Divisions/Channels will be conducted.	16.b) In the EPDS, physical separation or electrical isolation exists between Class 1E Divisions. Physical separation or electrical isolation exists between Class 1E Channels. Physical separation or electrical isolation exists between these Class 1E Divisions/Channels and non-Class 1E equipment. ~~Raceways containing Class 1E cables do not contain non-Class 1E cables.~~
17. Class 1E medium voltage switchgear, low voltage switchgear, and MCCs are identified according to their Class 1E Division.	17. Inspection of the as-built EPDS Class 1E medium voltage switchgear, low voltage switchgear, and MCCs will be conducted.	17. As-built Class 1E medium voltage switchgear, low voltage switchgear, and MCCs are identified according to their Class 1E Division.

TABLE 2.6.1-1 (Continued)

AC ELECTRICAL POWER DISTRIBUTION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
18. Class 1E medium voltage switchgear, low voltage switchgear, and MCCs are located in Seismic Category I structures and in their respective Divisional areas.	18. Inspection~~s~~ of the as-built Class 1E medium voltage switchgear, low voltage switchgear, and MCCs will be conducted.	18. As-built Class 1E medium voltage switchgear, low voltage switchgear, and MCCs are located in Seismic Category I structures and in their respective Divisional areas.
19. Class 1E EPDS cables and raceways are identified according to their Class 1E Division.	19. Inspection of the as-built Class 1E EPDS Divisional cables and raceways will be conducted.	19. As-built EPDS cables and raceways are identified according to their Class 1E Division.
20. Class 1E Division~~/Channel~~ cables are routed in Seismic Category I structures and in their respective raceways.	20. Inspection of the as-built EPDS Division~~/Channel~~ cables and raceways will be conducted.	20. As-built Class 1E Division~~/Channel~~ cables are routed in Seismic Category I structures and in their respective Division~~/Channel~~ raceways.
21. Class 1E equipment is not prevented from performing its safety functions by harmonic distortion waveforms.	21. Analysis for the as-built EPDS to determine harmonic distortions will be performed.	21. Analysis for the as-built EPDS exists and concludes that harmonic distortion waveforms do not exceed 5 percent voltage distortion on the Class 1E EPDS.
22. The EPDS supplies an operating voltage at the terminals of the Class 1E equipment which is within the equipment's voltage tolerance limits.	22.a) Analysis for the as-built EPDS to determine voltage drops will be performed. 22.b) Tests of the as-built Class 1E EPDS will be performed by operating connected Class 1E loads at the analyzed minimum voltage.	22.a) Analysis for the as-built EPDS exists and concludes that the analyzed operating voltage supplied at the terminals of the Class 1E equipment is within the equipment's voltage tolerance limits, as determined by their nameplate ratings. 22.b) Connected Class 1E loads operate at the analyzed minimum voltage as determined by the voltage drop analysis.

TABLE 2.6.1-1 (Continued)

AC ELECTRICAL POWER DISTRIBUTION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
23. Class 1E equipment is protected from degraded voltage conditions.	23.a) Analysis for the as-built EPDS to determine the trip conditions for degraded voltage conditions will be performed.	23.a) Analysis for the as-built EPDS exists and concludes that the Class 1E preferred offsite power feeder breakers to the Class 1E medium voltage switchgear will trip before Class 1E loads experience degraded voltage conditions exceeding those voltage conditions for which the Class 1E equipment is qualified.
	23.b) Testing for each as-built Class 1E medium voltage switchgear will be conducted by providing a simulated degraded voltage signal.	23.b) As-built Class 1E feeder breakers from preferred offsite power to the Class 1E medium voltage switchgear trip when a degraded voltage conditions exists.
24. An electrical grounding system is provided for (1) instrumentation, control, and computer systems, (2) electrical equipment (switchgear, ~~motors~~ transformers, distribution panels, and motors), and (3) mechanical equipment (fuel and chemical tanks). Lightning protection systems are provided for major plant structures, transformers and equipment located outside buildings. Each grounding system and lightning protection system is separately grounded to the plant ground grid.	24. Inspection of the plant grounding and lightning protection systems will be performed.	24. The as-built EPDS instrumentation, control, and computer grounding system, electrical equipment and mechanical equipment grounding system, and lightning protection systems provided for buildings and for structures and transformers located outside of the buildings, are separately grounded to the plant ground grid.
25. There are no automatic connections between Class 1E Divisions.	25. Inspection~~s~~ of the as-built Class 1E Divisions will be conducted.	25. There are no automatic connections between Class 1E Divisions.

TABLE 2.6.1-1 (Continued)

ELECTRICAL POWER DISTRIBUTION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
26.a) The EPDS displays identified in the Design Description (Section 2.6.1) exist in the MCR or can be retrieved there.	26.a) Inspection of the MCR will be conducted.	26.a) EPDS displays identified in the Design Description (Section 2.6.1) exist in the MCR or can be retrieved there.
26.b) The EPDS controls identified in the Design Description (Section 2.6.1) exist in the MCR.	26.b) Inspection of the MCR will be conducted.	26.b) EPDS controls identified in the Design Description (Section 2.6.1) exist in the MCR.

Add DC/ITA/AC 26.a and 26.b.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.6.2 (EDG/Mechanical Auxiliaries)

No.	Comments	Cat.	Resolution
1	Design description mark-ups attached. The DD should acknowledge existence of safety-related/Seismic Category I non-ASME EDG auxiliary systems (ANSI Class?) and ensure associated ITAAC verification is performed.	1	Disagree. The comment states that the DD should acknowledge existence of safety-related/Seismic Category I non-ASME EDG auxiliary systems and ensure associated ITAAC verification is performed. The last sentence of second paragraph, first page of the DD, states: "The EDG engine and ASME Code Class 3 portions of its respective support systems are classified Seismic Category I." The term "The EDG Engine" includes both non-ASME and ASME Code Class 3 components. Thus by stating that "The EDG engine," will be classified Seismic Category I, any non-ASME components on the engine will also be classified Seismic Category I. **NRC Staff concurs.**
2	Add a statement to the design description to reflect operation of the fuel oil gravity feed and operation of the fuel oil transfer valve. Add an appropriate ITAAC test verification.	2	Disagree. This is really a support system preoperational test only. NRC Staff concurs.
3 (4)	Add standard ITAAC verification for ASME component pressure boundary integrity for ASME components in auxiliary systems.	1	Agree. An appropriate DD/ITA/AC will be added to Table 2.6.2-1, which will address ASME pressure testing. See markup of ITAAC 2.6.2.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.6.2 (EDG/Mechanical Auxiliaries)

No.	Comments	Cat.	Resolution
4 (5)	SSAR mark-up attached.	3	See CESSAR-DC markup of pages 9.5-51, 56 and 57. a. Note that the Diesel Fuel Storage Structures are not specified Seismic Category I in Section 2.6.2. This duty is performed by Section 2.1.4, "Diesel Fuel Storage Structure." It is in Section 2.1.4, and accompanying ITAAC table that the specification for Seismic Category and associated testing exists. b. Technical Specifications markup. Comment was made that SR 3.8.3.1 should state "45000" instead of "[*]" for gallons of fuel. This is acceptable, provided that the "45000" remain inside brackets "[ ]". The reason for this is that the brackets indicate a value to be determined by detailed design (i.e., when diesel generators are actually final sized and purchased). CESSAR-DC Section 9.5.4.1.1 will be modified to describe the 45000 gallons value as typical (i.e., based on detailed final design). Later sections of 9.5.4 already describe this. Other Technical Specifications comments are that for SR 3.8.3.3 and 3.8.3.4, values of 500 gallons and 225 psig do not

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.6.2 (EDG/Mechanical Auxiliaries)

No.	Comments	Cat.	Resolution
4 (5)	(continued)	3	b. (continued) appear in CESSAR-DC. It is sufficient that they appear here in the Technical Specifications only, since they are typical values which are dependent on the specific diesel generator purchased and deal with operational concerns. c. Reviewer made comment that the COL Applicant item appearing on page 9.5-56 should be moved to page 9.5-57 and appear at the end of that section (9.5.4.5). Agree. See markup of CESSAR-DC pages 9.5-56 and 57.
5 (6)	Design Description (page 1, last par.) needs to be revised to include EDG load shedding features during loss of power or sustained bus under voltage condition (Refer CESSAR Section 8.3.1.1.4.6) and the time required for EDGs to be connected to the bus (≤ 20 seconds). ITAAC 10 needs to be revised to verify this.	1	Agreed. See markup of ITAAC 2.6.2.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.6.2 (EDG/Mechanical Auxiliaries)

No.	Comments	Cat.	Resolution
6 (7)	Design Description (page 2, para. 2) needs to be revised to include EDG load shedding features during loss of power and concurrent DBA condition (SIAS/CSAS/EFAS). ITAAC 12 needs to be revised to verify this.	1	Agreed. See markup of ITAAC 2.6.2.
7 (10)	The acceptance criteria for ITA 10 show +/- 10% tolerance for voltage and +/- 2% tolerance for frequency for the EDGs to automatically connect to its respective buses in $<$ 20 seconds. What is the basis for providing the above tolerances when the EDGs are required to attain rated voltage and frequency in $\leq$ 20 seconds?	1	Agreed. Replaced with NRC markup. See markup of ITAAC 2.6.2.

2.6.2 EMERGENCY DIESEL GENERATOR SYSTEM

DESIGN DESCRIPTION

The Emergency Diesel Generator (EDG) System is a safety-related system which has two diesel generators and their respective fuel oil, lube oil, engine cooling, starting air, and air intake and exhaust support systems. One EDG is connectable to the two Class 1E buses of an Electrical Power Distribution System (EPDS) Class 1E Division and the other EDG is connectable to the two Class 1E buses of the other EPDS Class 1E Division.

Each EDG and its support systems are physically separated from the other EDG and its support systems, and are located in physically separate areas of the Nuclear Island Structures. Portions of the EDG support systems which perform the safety function of starting and operating the EDG are classified ASME Code Class 3. The EDG generators are classified Class 1E. Class 1E equipment is classified Seismic Category I. The EDG engine and ASME Code Class 3 portions of its respective support systems are classified Seismic Category I.

The diesel fuel storage tanks for each of the two EDGs are located in physically separate diesel fuel storage structures. The underground fuel oil piping from each diesel fuel storage structure to its respective EDG day tank is classified Seismic Category I. Divisional separation is established by pipe routing and use of the Divisional wall.

The EDGs are sized to supply their load demands following a design basis accident which requires use of emergency power.

Each EDG has fuel storage capacity to provide fuel to its EDG for a period of no less than 7 days with the EDG supplying the power requirements for the most limiting design basis accident.

The starting air system receiver tanks of each EDG have a combined air capacity for 5 starts of the EDG without replenishing air to the receiver tanks.

The EDG combustion air intakes are separated from the EDG exhaust ducts.

Electrical independence is provided between Class 1E Divisions and between the Class 1E Divisions and non-Class 1E equipment.

load shedding of Class 1E buses in the affected Division

A loss of power to a Class 1E bus initiates an automatic start of the respective EDG, and automatic connection to the Class 1E buses in the affected Division. Following attainment of rated voltage and frequency, the EDG automatically connects to its respective Divisional buses. After the EDG connects to its respective buses, the non-accident loads are automatically sequenced onto the buses.

required

and load shedding of both Class 1E buses within the affected Division occurs.

SYSTEM 80+™

Each EDG receives an automatic start signal in response to a safety injection actuation signal (SIAS), a containment spray actuation signal (CSAS), or an emergency feedwater actuation signal (EFAS). An EDG does not automatically connect to its Divisional Class 1E buses, if the Divisional Class 1E buses are energized.

For a loss-of-power to a Class 1E medium voltage safety bus condition concurrent with a Design Basis Accident condition (SIAS/CSAS/EFAS), each EDG automatically starts, Following attainment of ~~rated~~ voltage and frequency, the EDG automatically connects to its respective buses, and loads are sequenced onto the buses.

required

When operating in a test mode, an EDG is capable of responding to an automatic start signal.

Displays of EDG voltage, amperage, frequency, watts, and vars instrumentation exist in the main control room (MCR) or can be retrieved there.

Controls exist in the MCR to manually start and stop each EDG. Controls exist at each EDG local control panel to manually start and stop its respective EDG.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.6.2-1 specifies the inspections, tests, analyses and associated acceptance criteria for the Emergency Diesel Generator System.

Insert New DC/ITA/AC #3

TABLE 2.6.2-1

EMERGENCY DIESEL GENERATOR SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the EDG System is as described in the Design Description (Section 2.6.2).	1. Inspection of the as-built EDG System will be conducted.	1. The as-built EDG System conforms with the Basic Configuration as described in the Design Description (Section 2.6.2).
2. Each EDG and its support systems are physically separated from the other EDG and its support systems, and are located in physically separate areas of the nuclear island structures.	2. Inspection of the as-built EDGs and EDG support systems will be performed.	2. The two EDGs and their respective support systems are located on opposite sides of the nuclear island structures and are separated by the Divisional wall.
~~3~~ 4. The diesel fuel storage tanks for each of the two EDGs are located in physically separate diesel fuel storage structures.	~~3~~ 4. Inspection of the as-built diesel fuel storage tank structures will be performed.	~~3~~ 4. The diesel fuel storage tanks for one EDG are located in a different structure from the diesel fuel storage tanks for the other EDG.
~~4~~ 5. The fuel oil piping from each diesel fuel storage structure to its respective EDG day tank is classified Seismic Category I. Divisional separation is established by pipe routing and use of the Divisional wall.	~~4~~ 5. Inspection of the as-built piping from each diesel fuel storage structure to its respective EDG day tank will be performed.	~~4~~ 5. The as-built fuel oil piping from each diesel fuel storage structure to its respective EDG day tank is classified Seismic Category I. Divisional separation is established by pipe routing and use of the Divisional wall.
~~5~~ 6. The EDGs are sized to supply their load demands following a design basis accident which requires use of emergency power.	~~5~~ 6. Analysis to determine EDG load demand, based on the as-built EDG load profile, will be performed.	~~5~~ 6. Analysis for the as-built EDGs exists and concludes that the EDGs' capacities exceed, as determined by their nameplate ratings, their load demand following a design basis accident which requires the use of emergency power.

TABLE 2.6.2-1 (Continued)

EMERGENCY DIESEL GENERATOR SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7 ~~6~~. Each EDG has fuel storage capacity to provide fuel to its EDG for a period of no less than 7 days with the EDG supplying the power requirements for the most limiting design basis accident.	7 ~~6~~. Inspection and analysis will be performed to determine fuel storage capacities and EDG fuel consumption.	7 ~~6~~. An analysis exists and concludes that each EDG has fuel storage capacity to operate the EDG for 7 days with the EDG supplying power during the most limiting design basis accident.
8 ~~7~~. The starting air system receiver tanks of each EDG have a combined air capacity for 5 starts of the EDG without replenishing air to the receiver tanks.	8 ~~7~~. Testing will be performed with the EDGs and their air start systems.	8 ~~7~~. Each EDG can be started 5 times without replenishing air to the receiver tanks.
9 ~~8~~. The EDG combustion air intakes are separated from the EDG exhaust ducts.	9 ~~8~~. Inspection of the as-built EDG air intakes and air exhaust will be performed.	9 ~~8~~. Each EDG's air intake and air exhaust is separated by distance and orientation. The air intakes and exhausts of the two EDGs are separated by the location of the EDGs on opposite sides of the nuclear island structures.
10 ~~9~~. Electrical independence is provided between Class 1E Divisions and between the Class 1E Divisions and non-Class 1E equipment.	10 ~~9~~.a) Testing will be performed on each EDG and support systems by providing a test signal in only one Class 1E Division at a time.	10 ~~9~~.a) A test signal exists only in the EDG and support systems Division under test.
	10 ~~9~~.b) Inspection of the as-installed Class 1E Divisions of the EDG System will be performed.	10 ~~9~~.b) Physical separation exists between Class 1E Divisions of the EDG system. Separation exists between Class 1E Divisions and non-Class 1E equipment in the EDG system.

and load sheds both Class 1E buses within the affected Division.

TABLE 2.6.2-1 (Continued)

EMERGENCY DIESEL GENERATOR SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
~~10~~ 11. A loss-of-power to a Class 1E medium voltage safety bus automatically starts its respective EDG. Following attainment of rated voltage and frequency, the EDG automatically connects to its respective Divisional buses. After the EDG connects to its respective buses, the non-accident loads are automatically sequenced onto the buses. required	~~10~~ 11. Testing for the actuation and connection of each EDG will be performed using a signal that simulates a loss-of-power.	~~10~~ 11. ~~As-built EDGs automatically start on receiving a loss-of-power signal, attain rated voltage (± 10%), and rated frequency (± 2%) in ≤ 20 seconds, automatically connect to their respective Divisional buses, and their non-accident loads are sequenced onto the buses.~~
~~11~~ 12. Each EDG receives an automatic start signal in response to a safety injection actuation signal (SIAS), a containment spray actuation signal (CSAS), or an emergency feedwater actuation signal (EFAS). An EDG does not automatically connect to its Divisional buses, if the Divisional Class 1E buses are energized.	~~11~~ 12. Testing for the actuation of each EDG will be performed using signals that simulate a SIAS, a CSAS, and a EFAS.	~~11~~ 12. Each EDG receives a start signal in response to each of the following simulated signals; a SIAS, a CSAS, and a EFAS, but does not automatically connect to its Divisional buses, if the Divisional buses are energized.

~~10~~ 11. As-built EDGs automatically start on receiving a LOOP signal and attain a voltage and frequency in ≤ 20 seconds which will assure an operating voltage and frequency at the terminals of the Class 1E equipment that is within the equipment's tolerance limits, automatically connect to their respective Divisional buses, and sequence their non-accident loads onto their Divisional buses.

and load shedding of both Class 1E buses within the affected Division occurs.

TABLE 2.6.2-1 (Continued)

EMERGENCY DIESEL GENERATOR SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
13 ~~12~~. For a loss-of-power to a Class 1E medium voltage safety bus condition concurrent with a Design Basis Accident condition (SIAS/CSAS/EFAS), each EDG automatically starts. Following attainment of rated voltage and frequency, the EDG automatically connects to its respective buses and loads are sequenced onto the buses. required	13 ~~12~~. Testing on the as-built EDG Systems will be performed by providing simulated SIAS/CSAS/EFAS and loss-of-power signals.	13 ~~12~~. In the as-built EDG Systems, when SIAS/CSAS/EFAS and loss-of-power signals exist, the EDG automatically starts, attains rated voltage and frequency and is connected to its Divisional buses within 20 seconds. Following connection, the automatic load sequence begins. Upon application of each load, the voltage on these buses does not drop more than 20% measured at the buses. Frequency is restored to within 2% of nominal, and voltage is restored to within 10% of nominal within 60% of each load sequence time interval. The SI, CS, and EFW loads are sequenced onto the buses in ≤ 40 seconds total time from initiating SIAS/CSAS/EFAS.
14 ~~13~~. When operating in a test mode, an EDG is capable of responding to an automatic start signal.	14 ~~13~~. Testing will be performed with each EDG in a test mode configuration. An automatic start signal will be simulated.	14 ~~13~~. When operating in a test mode, each EDG resets to its automatic control mode upon receipt of a simulated automatic start signal.

TABLE 2.6.2-1 (Continued)

EMERGENCY DIESEL GENERATOR SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
15.a) The EDG System displays identified in the Design Description (Section 2.6.2) exist in the MCR or can be retrieved there.	15.a) Inspection of the MCR will be conducted.	15.a) EDG System displays identified in the Design Description (Section 2.6.2) exist in the MCR or can be retrieved there.
15.b) Controls exist in the MCR and EDG local control panels to manually start and stop each EDG.	15.b) Tests will be performed using the EDG controls in the MCR and EDG local control panels.	15.b) EDG controls exist in the MCR and EDG local control panel to manually start and stop each EDG. After starting, the EDG remains in a standby mode, unless a LOOP signal exists.
3. The ASME Code Section III components of the EDG and its respective support systems identified in the Design Description (Section 2.6.2) retain their pressure boundary integrity under internal pressures that will be experienced during service.	3. A pressure test will be conducted on those components of the EDG and its respective support systems required to be pressure tested by ASME Code SEction III.	3. The results of the pressure test of ASME Code Section III components of the EDG and its respective support systems conform with the pressure testing acceptance criteria in ASME Code Section III.

9.5.4 DIESEL GENERATOR ENGINE FUEL OIL SYSTEM

9.5.4.1 Design Bases

9.5.4.1.1 Safety Design Bases

The Diesel Generator Engine Fuel Oil System is designed to provide for storage of a seven-day supply of fuel oil for each diesel generator engine and to supply the fuel oil to the engine, as necessary, to drive the emergency generator. The system is designed to meet the single failure criterion, and to withstand the effects of natural phenomena without the loss of operability.

All components and piping are located in a Seismic Category I structure (diesel generator building, diesel fuel storage structure) except for a portion of the piping from the fuel oil storage tanks to the day tank, which is seismically qualified and protected. All essential components and piping are fully protected from floods, tornado missile damage, internal missiles, pipe breaks and whip, jet impingement and interaction with non-seismic systems in the vicinity. The Diesel Generator Engine Fuel Oil System is designated as a vital system and components of the system are located within the plant's protected area.

9.5.4.1.2 Diesel Fuel Storage Structure

There are two Diesel Fuel Storage Structures, one on each side of the Nuclear Annex. Each consists of a reinforced concrete structure separated into two bays and an equipment room. The bays are separated from each other and from the equipment room by three-hour rated fire barriers. Each bay contains a diesel fuel oil tank, a tank vent, a sump, a sump pump and necessary piping. Each tank has a capacity of 45,000 gallons.
typically

The equipment room is a separate steel framed structure attached at the end of the tank bays. The equipment room contains a recirculation pump with simplex filter and piping, a fill connection with two strainers, a ventilation fan, and intake and exhaust dampers. Additional details of the Diesel Fuel Storage Structure are provided in Section 3.8.4.1.4. The structure general arrangement is shown on Figure 1.2-24. The following structural requirements ensure system adequacy:

A. The Diesel Fuel Storage Structure is a Seismic Category I, Safety Class 3 structure. The equipment room is non-nuclear safety, Seismic Category II.

B. The Seismic Category I portion of the Diesel Fuel Storage Structure is designed to withstand the effects of the following events:

9.5.4.5 Instrumentation Application

Each diesel generator engine is provided with sufficient instrumentation to monitor the operation of the fuel oil system. All alarms are separately annunciated on the local diesel engine control panel which also signals a general diesel trouble alarm in the control room. The fuel oil system is provided with the following instrumentation and alarms:

A. Fuel oil storage tanks

   1. Low level and high level annunciators.

   2. Technical specification low-low level alarm.

   3. Level indication, 0-100%.

   4. The capability for use of a stick gauge or similar means to measure the actual fuel oil level.

B. Fuel oil recirculation filter

   1. Inlet and outlet pressure indication.

C. Fuel oil day tank

   1. Fuel oil transfer valve control.

   2. High level alarm.

   3. Low level alarm.

   4. Level indication.

D. Fuel oil strainers (Engine-driven pump and motor-driven booster pump)

   1. High differential pressure alarm - Alerts the operator to take corrective action by manually switching over to the alternate clean strainer.

   2. Inlet and outlet pressure indication.

The COL Applicant will make available for NRC review, information on Diesel Generator Engine Fuel Oil System calibration frequencies.

→ Move to Clouded area on page 9.5-57

E. Fuel oil filter

1. High differential pressure alarm - Alerts the operator to take corrective action by manually switching over to the alternate clean filter.

2. Differential pressure indication.

3. Outlet pressure indication.

4. Low fuel oil pressure alarm.

F. Day tank retaining wall

1. High and low level drain valve and lube oil transfer pump control.

2. High-high level alarm.

More clouded
text from page 9.5-56 to here

ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time not met. <u>OR</u> One or more DGs with diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Conditions A, B, C, D, or E.	F.1 Declare associated DG inoperable.	Immediately

[45,000]

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each fuel storage tank contains ≥ ~~[*]~~ gallons of fuel.	31 days
SR 3.8.3.2	Verify lubricating oil inventory is ≥ [500] gallons.	31 days
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within, the limits of The Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program.
SR 3.8.3.4	Verify each DG air start receiver pressure is ≥ [225] psig.	31 days
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	31 days
SR 3.8.3.6	For each fuel oil storage tank: a. Drain the fuel oil; b. Remove the sediment; and c. Clean the tank.	10 years

~~* Value to be determined by system detail design.~~

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.6.3 (AC Instrumentation and Control Power System and DC Power System)

No.	Comments	Cat.	Resolution
1	Where is the legend provided for electrical symbols used in the CESSAR?	1	Agree. Symbology figure will be included in CESSAR-DC. See attached figure.
2	Design Description needs to be revised to show Design commitment 3.	1	Agree. See markup of ITAAC 2.6.3.
3	CESSAR Figures 8.3.2-1 and 8.3.2-2 show inverter power sources and regulated power supplies are connected in parallel via normally closed breakers for ESF-CCS and Process-CCS panels. Is there any interlock to prevent paralleling? Needs clarification.	1	Agree. A statement will be added to CESSAR-DC detailing the interlock feature between these two circuits. See markup of CESSAR-DC page 8.3-44.
4	Design Description, page 1, para. 5, states that "Each Class 1E ACI&C power supply is a constant voltage constant frequency inverter power supply unit." This information is not shown in Design Commitment 2. Design Commitment 2 and ITAAC need to be revised.	1	Agree. See markup of ITAAC 2.6.3.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.6.3 (AC Instrumentation and Control Power System and DC Power System)

No.	Comments	Cat.	Resolution
5	No ITAAC entry is provided to verify Design Description, page 1, para. 5, which states that the "alternate power source is a voltage regulating device which is supplied from the same ac power source as battery charger---."	1	Agree. See markup of ITAAC 2.6.3.
6	No ITAAC entry is provided to verify Design Description, page 1, para. 6, which states that "Each Class 1E power supply unit is synchronized, in both frequency and phase-----."	1	Agree. Resolved by #2 above.
7	Design Description needs to be revised to show Design Commitment 15.	1	Agree. See markup of ITAAC 2.6.3.
8	Design Commitment 4 needs to be revised as shown in the attachment to be consistent with the Design Description.	1	Agree. See markup of ITAAC 2.6.3.

CE 80+ ITAAC Independent Review Comments

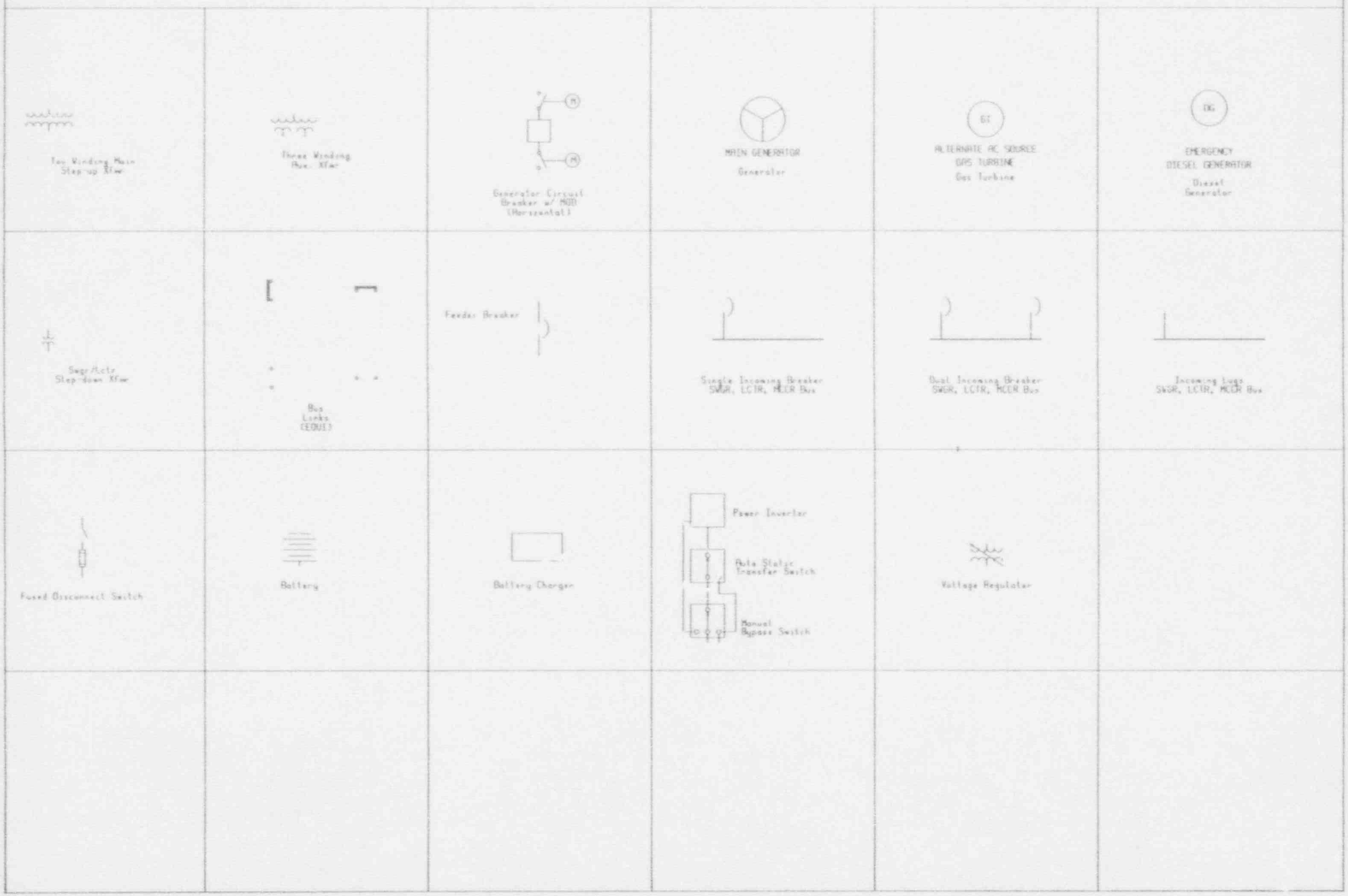
ITAAC No. 2.6.3 (AC Instrumentation and Control Power System and DC Power System)

No.	Comments	Cat.	Resolution
9	a. Determine whether the test described in ITA 19b can be performed before fuel load? b. If this test can be performed as written, then the acceptance criteria should be revised to verify the capacities of DC equipment such as battery charger, MCCs and DC distribution panels to operate the connected loads.	1	Agree. See markup of ITAAC 2.6.3.
10	Design Description needs to be revised to show Design Commitment 27.	1	Agree. See markup of ITAAC 2.6.3.
11	An ITAAC entry is needed to verify Class 1E DC Power System alarms and displays shown in page 3, last paragraph of the Design Description.	1	Agree. See markup of ITAAC 2.6.3.

Note: Markups also identify non-Class 1E loads that are connectable to the Class 1E buses and relocate the Seismic Category I statement to page 1.

2.6.3
Comment 1

SYSTEM 80+ ELECTRICAL ONE LINE SYMBOLS



2.6.3 AC INSTRUMENTATION AND CONTROL POWER SYSTEM AND DC POWER SYSTEM

DESIGN DESCRIPTION

See page 4

#

The AC Instrumentation and Control (I&C) Power System and DC Power System consist of Class 1E and non-Class 1E power systems. The non-Class 1E AC I&C Power System and DC Power System have non-Class 1E batteries, inverters, electrical distribution panels, and battery chargers. The non-Class 1E AC I&C Power System and DC Power System provide power to non-Class 1E equipment.

The Class 1E AC Instrumentation and Control (I&C) Power System (also referred to as the Vital AC I&C Power System) and the Class 1E DC Power System (also referred to as the Vital DC Power System) consist of Class 1E uninterruptible power supplies, their respective alternating current (AC) and direct current (DC) distribution centers, along with power, instrumentation and control cables to the distribution system loads. The Class 1E AC I&C Power System and the Class 1E DC Power System include the protection equipment provided to protect the AC and DC distribution equipment.

INSERT B

The Basic Configuration of the Class 1E AC Instrumentation and Control Power System and Class 1E DC Power System is as shown on Figures 2.6.3-1 and 2.6.3-2.

Class 1E AC Instrumentation and Control Power System

The Class 1E AC I&C Power System consists of two Division (Division I and II) and four Channel (A, B, C, D) uninterruptible power supplies, with their respective distribution panels.

Each Class 1E AC I&C power supply is a constant voltage constant frequency inverter power supply unit, which in normal operating mode receives Class 1E direct current (DC) power from its respective Class 1E DC distribution center. Each Class 1E inverter power supply unit also has capability to transfer from its respective Class 1E DC distribution center to an alternate source of alternating current (AC) power to directly supply the Class 1E AC I&C Power System loads.* This alternate power source is a voltage regulating device which is supplied power from the same AC power source as the battery charger associated with the Class 1E DC distribution center servicing the inverter power supply unit.

while maintaining continuity of power during transfer from the inverter power supply unit to the alternate power supply.

~~Each Class 1E inverter power supply unit is synchronized, in both frequency and phase, with its alternate power supply and maintains continuity of power during transfer from the inverter to the alternate power supply.~~

Each Class 1E inverter power supply unit is sized to provide power to its respective distribution center loads.

Insert B:

"The Containment Equipment Hatch Trolley, the Reactor Cavity Flood Valves, the Holdup Volume Flood Valves, and the Hydrogen Ignitors are the only electrical loads classified as non-Class 1E which are directly connectable to the Class 1E buses."

Class 1E inverter power supply units and their respective distribution centers are identified according to their Class 1E Division/Channel and are located in Seismic Category I structures and in their respective Division/Channel areas.

Independence is provided between Class 1E Divisions. Independence is provided between Class 1E Channels. Independence is provided between Class 1E Divisions/Channels and non-Class 1E equipment.

Class 1E AC I&C Power System distribution panels and their circuit breakers, disconnect switches and fuses are sized to supply their load requirements. Distribution panels and disconnect switches are rated to withstand fault currents for the time required to clear the fault from its power source. Circuit breakers and fuses are rated to interrupt fault currents.

Class 1E AC I&C Power System interrupting devices (circuit breakers and fuses) are coordinated so that the circuit interrupter closest to the fault opens before other devices.

Class 1E AC I&C Power System cables are sized to supply their load requirements and are rated to withstand fault currents for the time required to clear the fault from its power source.

The Class 1E AC I&C Power System supplies an operating voltage at the terminals of the Class 1E equipment which is within the equipment's voltage tolerance limits.

Class 1E AC I&C Power System cables and raceways are identified according to their Class 1E Division/Channel. Class 1E cables are routed in Seismic Category I structures and in their respective Division or Channel raceways.

Class 1E equipment is classified as Seismic Category I.

Class 1E DC Power System

The Class 1E DC Power System consists of two Divisional (Division I and II) and four Channel (A, B, C, D) batteries (2 Channel batteries per Division) with their respective DC electrical distribution panels and battery chargers. The Class 1E DC distribution system provides DC power to Class 1E DC equipment and instrumentation and control circuits.

Each Class 1E battery is sized to supply its Design Basis Accident (DBA) loads, at the end-of-installed-life, for a minimum of 2 hours without recharging.

Each Class 1E battery charger is sized to supply its respective Class 1E Division/Channel steady-state loads while charging its respective Class 1E battery.

Each Class 1E battery is provided with a normal battery charger supplied alternating current (AC) from a MCC in the same Class 1E Division as the battery.

Manual interlocked transfer capability exists within a Division between Class 1E DC distribution centers.

The Class 1E batteries, battery chargers and respective MCCs, DC distribution panels, disconnect switches, circuit breakers, and fuses are sized to supply their load requirements. The Class 1E batteries, battery chargers and respective MCCs, DC distribution panels, and disconnect switches are rated to withstand fault currents for the time required to clear the fault from its power source.

Class 1E DC Power System circuit breakers and fuses are rated to interrupt fault currents.

Class 1E DC Power System electrical distribution system circuit interrupting devices (circuit breakers and fuses) are coordinated so that the circuit interrupter closest to the fault is designed to open before other devices.

Class 1E DC Power System electrical distribution system cables are sized to supply their load requirements and are rated to withstand fault currents for the time required to clear the fault from its power source.

The Class 1E DC Power System electrical distribution system supplies an operating voltage at the terminals of the Class 1E equipment which is within the equipment's voltage tolerance limits.

Each Class 1E battery is located in a Seismic Category I structure and in its respective Division/Channel battery room.

Class 1E DC Power System distribution panels and MCCs are identified according to their Class 1E Division/Channel. Class 1E cables are routed in Seismic Category I structures and in their respective Division/Channel raceways.

Independence is provided between Class 1E Divisions. Independence is provided between Class 1E Channels. Independence is provided between Class 1E Divisions/Channels and non-Class 1E equipment.

The Class 1E DC Power System has the following alarms and displays in the main control room (MCR):

1) Alarms for battery ground detection.

2) Parameter displays for battery voltage and amperes.

3) Status indication for battery circuit breaker/disconnect position.

¶ Class 1E DC Power System cables are identified according to their Class 1E Division/Channel.

Class 1E equipment is classified as Seismic Category I.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.6.3-1 specifies the inspections, tests, analyses and associated acceptance criteria for the AC Instrumentation and Control Power System and DC Power System.

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TABLE 2.6.3-1

AC INSTRUMENTATION AND CONTROL POWER SYSTEM AND DC POWER SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the AC Instrumentation and Control Power System and the DC Power System is as described in the Design Description (Section 2.6.3).	1. Inspection of the as-built AC Instrumentation and Control Power System and the DC Power System configuration will be conducted.	1. The as-built AC Instrumentation and Control Power System and the as-built DC Power System conforms with the Basic Configuration as described in the Design Description (Section 2.6.3).
constant voltage, constant frequency 2. Each Class 1E inverter power supply unit in normal operating mode receives Class 1E direct current (DC) power from its respective DC distribution center. Each Class 1E inverter power supply unit also has capability to transfer from its respective Class 1E DC distribution center normal power source to an alternate source of alternating current (AC) power to directly supply the Class 1E AC I&C Power System loads.	2. Inspection of the as-built Class 1E inverter power supply unit will be conducted. constant voltage, constant frequency	2. Each Class 1E inverter power supply unit in normal operating mode receives Class 1E direct current (DC) power from its respective DC distribution center. Each Class 1E inverter power supply unit also has capability to transfer from its respective Class 1E DC distribution center normal power source to an alternate source of alternating current (AC) power to directly supply the Class 1E AC I&C Power System loads.
3. Automatic transfer between the normal and alternate power supplies for each Class 1E inverter power supply unit is provided and maintains continuity of power during transfer from the inverter power supply unit to the alternate power supply. Manual transfer between the normal and alternate power supplies for each Class 1E inverter power supply unit is also provided.	3. Testing on each as-built Class 1E inverter power supply unit will be conducted by providing a test signal in one power source at a time. A test of the manual transfer will also be conducted.	3. Each Class 1E inverter power supply unit automatically and manually transfers between its normal and alternate power sources and maintains continuity of power during transfer from the inverter to the alternate supply.

This alternate power source is a voltage regulating device which is supplied power from the same AC power source as the battery charger associated with the Class 1E DC distribution center servicing the inverter power supply unit.

AC INSTRUMENTATION AND CONTROL POWER SYSTEM AND DC POWER SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Each Class 1E inverter power supply unit is sized to provide ~~output~~ power to its respective distribution ~~panel~~ loads. Class 1E center	4. Analyses for each as-built Class 1E inverter power supply unit to determine the power requirements of its loads will be performed.	4. Analyses for each as-built Class 1E inverter power supply unit exist and conclude that each inverter power supply unit's capacity, as determined by its nameplate rating, exceeds its analyzed load requirements.
5. Class 1E inverter power supply units and their respective distribution panels are identified according to their Class 1E Division/Channel and are located in Seismic Category I structures and in their respective Division/Channel areas.	5. Inspection of the as-built Class 1E inverter power supply units and their respective distribution panels will be conducted.	5. The as-built Class 1E inverter power supply units and their respective distribution panels are identified according to their Class 1E Division/Channel and are located in Seismic Category I structures and in their Division/Channel areas.
6. In the Class 1E AC I&C Power System, independence is provided between Class 1E Divisions. Independence is provided between Class 1E Channels. Independence is provided between Class 1E Divisions/Channels and non-Class 1E equipment.	6.a) Testing on the Class 1E AC I&C Power System will be conducted by providing a test signal in only one Class 1E Division/Channel at a time.	6.a) A test signal exists only in the Class 1E Division/Channel under test in the Class 1E AC I&C Power System.
	6.b) Inspection of the as-built Class 1E Divisions/Channels in the Class 1E AC Power System will be conducted.	6.b) In the Class 1E AC I&C Power System, physical separation or electrical isolation exists between the Class 1E Divisions/Channels. Physical separation or electrical isolation exists between these Class 1E Divisions/Channels and non-Class 1E equipment. ~~Raceways containing Class 1E cables do not contain non-Class 1E cables.~~

TABLE 2.6.3-1 (Continued)

AC INSTRUMENTATION AND CONTROL POWER SYSTEM AND DC POWER SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. Class 1E AC I&C Power System interrupting devices (circuit breakers and fuses) are coordinated so that the circuit interrupter closest to the fault is designed to open before other devices.	10. Analysis for the as-built Class 1E AC I&C Power System to determine circuit interrupting device coordination will be performed.	10. Analysis for the as-built Class 1E AC I&C Power System circuit interrupting device coordination exists and concludes that the analyzed circuit interrupter closest to the fault will open before other devices.
11. Class 1E AC I&C Power System cables are sized to supply their load requirements.	11. Analysis for the as-built Class 1E AC I&C Power System cables to determine their load requirements will be performed.	11. Analysis for the as-built Class 1E AC I&C Power System exists and concludes that the capacities of the distribution system cables exceed, as determined by their cable ratings, their analyzed load requirements.
12. Class 1E AC I&C Power System cables are rated to withstand currents for the time required to clear the fault from its power source.	12. Analysis for the as-built Class 1E AC I&C Power System to determine fault currents will be performed.	12. Analysis for the as-built Class 1E AC I&C Power System cables exists and concludes that the distribution system cable current capacities exceed their analyzed fault currents for the time required, as determined by the circuit interrupting device coordination analysis, to clear the fault from its power source.

TABLE 2.6.3-1 (Continued)

AC INSTRUMENTATION AND CONTROL POWER SYSTEM AND DC POWER SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
13. The Class 1E AC I&C Power System supplies an operating voltage at the terminals of the Class 1E ~~utilization~~ equipment which is within the ~~utilization~~ equipment's voltage tolerance limits.	13. Analysis for the as-built Class 1E AC I&C Power System to determine voltage drops will be performed.	13. Analysis for the as-built Class 1E AC I&C Power System ~~voltage drops~~ exists and concludes that the analyzed operating voltage supplied at the terminals of the Class 1E equipment is within the equipment's voltage tolerance limits, as determined by their nameplate ratings.
14. Class 1E AC I&C Power System cables and raceways are identified according to their Class 1E Division/Channel. Class 1E cables are routed in Seismic Category I structures and in their respective Division or Channel raceways.	14. Inspection of the as-built Class 1E AC Power System cables and raceways will be conducted.	14. As-built Class 1E AC Power System cables and raceways are identified according to their Class 1E Division/Channel. Class 1E Divisional/Channel cables are routed in Seismic Category I structures and in their respective Division/Channel raceways.
15. Each Class 1E battery is provided with a normal battery charger supplied alternating current (AC) from a MCC in the same Class 1E Division as the battery.	15. Inspections of the as-built Class 1E DC Power System will be conducted.	15. Each Class 1E battery is provided with a normal battery charger supplied alternating current (AC) from a MCC in the same Class 1E Division as the battery.

AC INSTRUMENTATION AND CONTROL POWER SYSTEM AND DC POWER SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

batteries, battery chargers,

The batteries, battery chargers,

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
19. The Class 1E DC Power System batteries, battery chargers, MCCs, DC distribution panels, disconnect switches, circuit breakers, and fuses are sized to supply their load requirements.	19.~~a)~~ Analysis for the as-built Class 1E DC Power System electrical distribution system to determine the capacities of the battery, battery charger, MCCs, DC distribution panels, disconnect switches, circuit breakers, and fuses will be performed.	19.~~a)~~ Analysis for the as-built Class 1E DC Power System exists and concludes that the capacities of The batteries, battery chargers, MCCs, DC distribution panels, disconnect switches, circuit breakers, and fuses, as determined by their nameplate ratings, exceed their analyzed load requirements.
	24.b) ~~19.b)~~ Testing of the as-built Class 1E DC Power System ~~battery, battery charger, DC distribution panels, MCCs, and system circuit breakers, disconnect switches, and fuses~~ will be conducted by operating connected Class 1E loads at less than or equal to minimum allowable voltage and at greater than or equal to the maximum battery charging voltage.	~~19.b)~~ 24.b) Connected as-built Class 1E loads operate at less than or equal to the minimum allowable battery voltage and at greater than or equal to the maximum charging voltage.
20.a) The Class 1E batteries, battery chargers, DC distribution panels, MCCs, and disconnect switches are rated to withstand fault currents for the time required to clear the fault from its power source.	20.a) Analysis for the as-built Class 1E DC Power System to determine fault currents will be performed.	20.a) Analysis for the as-built Class 1E DC Power System exists and concludes that the capacities of the as-built Class 1E batteries, battery chargers, DC distribution panels, MCCs, and disconnect switches current capacities exceed their analyzed fault currents for the time required, as determined by the circuit interrupting device coordination analyses, to clear the fault from its power source.

TABLE 2.6.3-1 (Continued)

AC INSTRUMENTATION AND CONTROL POWER SYSTEM AND DC POWER SYSTEM

Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
20.b) Class 1E DC Power System circuit breakers and fuses are rated to interrupt fault currents.	20.b) Analysis for the as-built Class 1E DC Power System to determine fault currents will be performed.	20.b) Analysis for the as-built Class 1E DC Power System exists and concludes that the analyzed fault currents do not exceed the circuit breaker and fuse interrupt capacities, as determined by their nameplate ratings.
(circuit breakers and fuses)		
21. Class 1E DC Power System circuit interrupting devices are coordinated so that the circuit interrupter closest to the fault is designed to open before other devices.	21. Analysis for the as-built Class 1E DC Power System to determine circuit interrupting device coordination will be performed.	21. Analysis for the as-built Class 1E DC Power System circuit interrupting devices exists and concludes that the analyzed circuit interrupter closest to the fault is designed to open before other devices.
22. Class 1E DC Power System cables are sized to supply their load requirements.	22. Analysis for the as-built Class 1E DC Power System cables to determine their load requirements will be performed.	22. Analysis for the as-built Class 1E DC Power System cables exists and concludes that the Class 1E DC electrical distribution system cable capacities, as determined by cable ratings, exceed their analyzed load requirements.

TABLE 2.6.3-1 (Continued)

AC INSTRUMENTATION AND CONTROL POWER SYSTEM AND DC POWER SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
27. Class 1E DC Power System cables are identified according to their Class 1E Division/Channel.	27. Inspection of the as-built Class 1E DC Power System cables will be conducted.	27. As-built Class 1E DC Power System cables are identified according to their Class 1E Division/Channel.
28. Class 1E Division/Channel cables are routed in Seismic Category I structures in their respective Division/Channel raceways.	28. Inspection of the as-built Class 1E DC Power System cables and raceways will be conducted.	28. Class 1E Division/Channel cables are routed in Seismic Category I structures in their respective Division/Channel raceways.
29. In the Class 1E DC Power System, independence is provided between Class 1E Divisions. Independence is provided between Class 1E Channels. Independence is provided between Class 1E Divisions/Channels and non-Class 1E equipment.	29.a) Testing will be conducted on the as-built Class 1E DC Power System by providing a test signal in only one Class 1E Division/Channel at a time.	29.a) A test signal exists in only the Class 1E Division/Channel under test in the Class 1E DC Power System.
	29.b) Inspection of the as-built Class 1E DC Power System will be conducted.	29.b) In the as-built Class 1E DC Power System, physical separation or electrical isolation exists between Class 1E Divisions/Channels. Physical separation or electrical isolation exists between these Class 1E Divisions/Channels and non-Class 1E equipment. ~~Raceways containing Class 1E cables do not contain non-Class 1E cables.~~

AC INSTRUMENTATION AND CONTROL POWER SYSTEM AND DC POWER SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
30. The Class IE DC Power System displays identified in the Design Description (Section 2.6.3) exist in the MCR or can be retrieved there.	30. Inspection of the MCR will be conducted.	30. Class IE DC Power System displays identified in the Design Description (Section 2.6.3) exist in the MCR or can be retrieved there.

Add DC/ITA/AC #30

exceeds 2 hours and, as a minimum, permits operating the instrumentation and control loads associated with the turbine-driven emergency feedwater pumps for 8 hours. Battery installations are designed to meet the intent of IEEE Standard 484-1987 and are qualified using methodologies described in IEEE Standard 535-1986.

8.3.2.1.2.1.3 125V DC Vital Power Distribution Centers and Panelboards

A 125V DC distribution center is provided for each of the 125V DC Vital Power System load groups. Each distribution center supplies an independent channel of vital power, and is powered directly from an independent 125 volt battery and battery charger. Each of the distribution centers supplies one DC panelboard and one 125V DC - 120V AC static inverter. Each Division I and II distribution center also powers its respective emergency diesel generator.

8.3.2.1.2.1.4 120V AC Vital Instrumentation and Control Power System

The 120V AC Vital Instrumentation and Control Power System consists of four separate and independent 120 volt AC power panelboards, each powered from a 125 volt DC load group distribution center via a 125V DC - 120V AC static inverter. This power supply system is designed to provide an output frequency of 60 ±0.5 Hz and voltage regulation to within ±2% at full rated load for a load power factor greater than 0.8 (towards unity). Each 120 volt AC power panelboard supplies one channel of AC vital instrumentation and controls. A manual make-before-break bypass switch is provided to bypass the inverter for maintenance. An autostatic transfer switch is used to instantly transfer the load from the output of the inverter to the regulating transformer which is fed from a Class 1E 480V MCC. The 120V AC Vital Instrumentation and Control Power System is shown in Figure 8.3.2-2.

The inverters are sized to supply their maximum loads plus a margin of 15%.

The channelized portion of the AC vital instrumentation and control power system is an ungrounded system.

The 120V AC power feeds are provided to each redundant ESF-Component Control System to enhance their availability. One feed is from the Class 1E inverter via the vital I&C channel power panel as shown on Figure 8.3.2-2. The other feed is from the same Class 1E channel 480/120V AC regulated transformer. These power feeds are interlocked to prevent paralleling

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ITAAC No. 2.6.4 Containment Electrical Penetration

No.	Comments	Cat.	Resolution
1	Design description and ITAAC need to be revised as shown in the attached markup.	2	Agree. See markup of ITAAC 2.6.4.

2.6.4 CONTAINMENT ELECTRICAL PENETRATION ASSEMBLIES

DESIGN DESCRIPTION

Containment Electrical Penetration Assemblies are provided for electrical cables passing through the primary containment.

Containment Electrical Penetration Assemblies are classified as Seismic Category I.

Class 1E Division Containment Electrical Penetration Assemblies only contain cables of one Class 1E Division, and Class 1E Channel Containment Electrical Penetration Assemblies only contain cables of one Class 1E Channel.

Independence is provided between Division Containment Electrical Penetrations Assemblies. Independence is provided between Channel Containment Electrical Penetration Assemblies. Independence is provided between Containment Electrical Penetration Assemblies containing Class 1E cables and Containment Electrical Penetration Assemblies containing non-Class 1E cables.

Containment Electrical Penetration Assemblies are protected against ~~overcurrent~~.
currents which are greater than their continuous ratings.

Containment Electrical Penetration Assemblies are equipment for which paragraph number (3) of the "Verification for Basic Configuration for Systems" of the General Provisions (Section 1.2) applies.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.6.4-1 specifies the inspections, tests, analyses and associated acceptance criteria for the Containment Electrical Penetration Assemblies.

TABLE 2.6.4-1

CONTAINMENT ELECTRICAL PENETRATION ASSEMBLIES
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the Containment Electrical Penetration Assemblies is as described in the Design Description (Section 2.6.4).	1. Inspection of the as-built Containment Electrical Penetration Assemblies will be conducted.	1. The as-built Containment Electrical Penetration Assemblies conforms with the Basic Configuration described in the Design Description (Section 2.6.4).
2. Class 1E Division Containment Electrical Penetration Assemblies only contain cables of one Class 1E Division, and Class 1E Channel Containment Electrical Penetration Assemblies only contain cables of one Class 1E Channel.	2. Inspection of the as-built Division and Channel Containment Electrical Penetrations Assemblies will be conducted.	2. As-built Class 1E Divisional Containment Electrical Penetration Assemblies only contain cables of one Class 1E Division, and Class 1E Channel Containment Electrical Penetration Assemblies only contain cables of one Class 1E Channel.
3. Independence is provided between Division Containment Electrical Penetration Assemblies. Independence is provided between Channel Containment Electrical Penetration Assemblies. Independence is provided between Containment Electrical Penetration Assemblies containing Class 1E cables and Containment Electrical Penetration Assemblies containing non-Class 1E cables.	3. Inspection of the as-built Containment Electrical Penetration Assemblies will be conducted.	3. Physical separation exists between as-built Division Containment Electrical Penetration Assemblies. Physical separation exists between Channel Containment Electrical Penetration Assemblies. Physical separation exists between Containment Electrical Penetration Assemblies containing Class 1E cables and Containment Electrical Penetration Assemblies containing non-Class 1E cables.

CONTAINMENT ELECTRICAL PENETRATION ASSEMBLIES
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Containment Electrical Penetration Assemblies are protected against [redacted]currents which are greater than their continuous ratings	4. Analysis for the as-built Containment Electrical Penetration Assemblies will be performed.	4. Analysis exists for the as-built Containment Electrical Penetration Assemblies and concludes either (1) that the maximum [redacted]current of the circuits does not exceed the continuous rating of the Containment Electrical Penetration Assembly, or (2) that the circuits have redundant [redacted] protection devices in series and that the redundant [redacted]current protection devices are coordinated with the Containment Electrical Penetration Assembly's rated short circuit thermal capacity data and prevent [redacted]current from exceeding the continuous current rating of the Containment Electrical Penetration Assembly.

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ITAAC No. 2.6.5 (Alternate AC Source)

No.	Comments	Cat.	Resolution
1 (6)	ITAAC #4 needs to be revised to include verification of AAC source starting from the control room.	2	Agree. See markup of ITAAC 2.6.5.

TABLE 2.6.5-1

ALTERNATE AC SOURCE
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the AAC is as described in the Design Description (Section 2.6.5).	1. Inspection of the as-built AAC will be conducted.	1. The as-built AAC conforms with the Basic Configuration as described in the Design Description (Section 2.6.5).
2. The AAC can supply power to:	2. Testing on the as-built AAC will be conducted by connecting the AAC to:	2. The as-built AAC can supply power to:
a) the non-Class 1E permanent non-safety buses; or	a) the non-Class 1E permanent non-safety buses; and then	a) the non-Class 1E permanent non-safety buses; or
b) to a Class 1E Division through its associated non-Class 1E permanent non-safety bus.	b) to a Class 1E Division through its associated non-Class 1E permanent non-safety bus.	b) to a Class 1E Division through its associated non-Class 1E permanent non-safety bus.
3. The load capacity of the AAC is at least as large as the capacity of an EDG.	3. Inspection of the as-built AAC and EDGs will be conducted.	3. The as-built AAC load capacity is at least as large as the capacity of an EDG as determined by the AAC and EDG nameplate ratings.
4. The AAC displays and controls identified in the Design Description (Section 2.6.5) exist in the MCR or can be retrieved there.	4. Inspection of the MCR will be conducted.	4. AAC displays and controls identified in the Design Description (Section 2.6.5) exist or can be retrieved there. IN THE MCR

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ITAAC No. 2.7.1 (New Fuel Racks

No.	Comments	Cat.	Resolution
1	The design description states the racks are anchored to embedments. Supporting details could not be found in Section 9.1 of the CESSAR. The CESSAR should be appropriately supplemented.	3	Disagree. Section 9.1.1.2 states bolted anchorage to embedments. **NRC Staff concurs.**
2	The design description states there will be an initial storage for at least 121 new fuel assemblies (minimum value specified) whereas the CESSAR 9.1.1.3.1.3 states the criticality safety margins are maintained by limiting the capacity to 121 assemblies (a maximum value), please clarify.	3	Agree. The ITAAC design description and CESSAR-DC are revised as shown on the attached marked pages to clarify that the new fuel storage racks provide on-site storage for at least 121 new fuel assemblies. The SAR is also revised to identify that the criticality safety analyses have been performed for initial on-site storage of 121 fuel assemblies.

2.7.1 NEW FUEL STORAGE RACKS

Design Description

The New Fuel Storage Racks provide ~~[illegible]~~ on-site storage for at least 121 new fuel assemblies. The New Fuel Storage Racks are safety-related.

The New Fuel Storage Racks are located in the nuclear island structures in the new fuel storage pit.

The New Fuel Storage Racks support and protect new fuel assemblies. The New Fuel Storage Racks maintain the effective neutron multiplication factor less than the required criticality limits during normal operation and design postulated accident conditions.

The New Fuel Storage Racks are anchored to embedments at the bottom of the storage cavity.

The New Fuel Storage Racks are designed and constructed in accordance with ASME Code Section III, Subsection NF, Class 3 Component Supports requirements.

The New Fuel Storage Racks are designed to accommodate design basis loads and load combinations including the effects of impact of fuel assemblies on the racks and the impact due to postulated fuel handling accidents without losing the structural capability to maintain the fuel in a non-critical configuration.

The New Fuel Storage Racks are classified Seismic Category I.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.1-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the New Fuel Storage Racks.

9.0 AUXILIARY SYSTEMS

9.1 FUEL STORAGE AND HANDLING

9.1.1 NEW FUEL STORAGE

9.1.1.1 Design Bases

The following design bases are imposed on the storage of new fuel assemblies:

A. Accidental criticality shall be prevented for the most reactive arrangement of new fuel stored. For normal operation and postulated accident conditions identified in Section 9.1.1.3.1.1 A, C and D, K_{eff} shall be maintained less than 0.95. For the postulated accident condition identified in Section 9.1.1.3.1.1B, K_{eff} shall be maintained less than 0.98.

B. All requirements of Regulatory Guide 1.13 are met excluding those regarding the spent fuel pool water supply, since new fuel storage is dry. The new fuel storage area is designed to ensure that any light load, as described in Section 9.1.4.2.1, when handled over the fuel racks, will not exceed the design impact energy capacity of the rack if the load is postulated to fall from its operating height. In addition, all heavy loads, as described in Section 9.1.4.2.1, are prevented from travel over the new fuel racks by the use of mechanical and electrical interlocks on the cask handling hoist. The new fuel handling hoist incorporates load limiting devices to preclude fuel damage during handling.

C. The storage racks and facilities are (See Section 9.1.1.3.3) qualified as Seismic Category I per Regulatory Guide 1.29.

D. The new fuel storage racks ~~Storage is typically~~ provided on-site storage for at least 121 new fuel assemblies. This capacity, which represents 50% of the fuel assemblies in the core, envelops any reload batch size that would occur for refueling cycle lengths up to and including 24 months. W

E. The fuel handling equipment located in the new fuel storage area meets the requirements of ANS 57.1. The new fuel racks meet the requirements of ANS 57.3.

F. The New Fuel Storage Racks are designed to meet the requirements of SRP 3.8.4 Appendix D which addresses appropriate combinations of seismic and dropped loads with allowable stress/deformation limits.

F. No burnable poison shims or other supplemental neutron poisons (e.g., CEAs) are assumed to be present in the fuel assemblies.

9.1.1.3.1.3 Criticality Safety Margins for initial on-site storage

Criticality safety margins are maintained by:

A. Limiting the capacity to 121 fuel assemblies.

B. Defining an overall array configuration.

C. Providing adequate mechanical separation of fuel assemblies in the array, even under postulated accident conditions.

The mechanical separation provided is discussed in Section 9.1.1.2. In evaluating criticality safety, the three-dimensional Monte Carlo computer code KENO-IV (Reference 2) is used to perform the criticality calculations for the new fuel storage racks for the postulated accident condition of flooding with pure, unborated water for the full range of water densities. The calculations are performed for a typical repeating lattice unit of the new fuel storage racks and a fuel enrichment of 5.0 wt.% U-235, which envelops the design requirements for any fuel management scheme. The calculations include an allowance in K_{eff} for uncertainties due to deviations from nominal conditions (e.g., variations in water temperature) and calculational uncertainties. Including uncertainties, the maximum K_{eff} is less than 0.95 for flooding with pure, unborated water and less than 0.98 for immersion in a foam or mist of the optimum moderation density.

The rack structure provides a separation of at least 10 inches, which is greater than the separation between fuel assemblies within the rack, between the top of the active fuel and the top of the rack to preclude criticality in the event a fuel assembly is dropped into a horizontal position on the top of the rack.

The new fuel storage area is protected from the effects of missiles or natural phenomena as discussed in Section 3.5.

9.1.1.3.2 Compliance with Regulatory Guide 1.13

All requirements of Regulatory Guide 1.13 are met excluding those regarding the spent fuel pool water supply, since new fuel storage is dry. The new fuel storage area is designed to ensure that any light load, when handled over the fuel racks, will not exceed the design impact energy of the rack if the load is postulated to fall from its operating height. In addition, all heavy loads are prevented from travel over the new fuel racks by the use of mechanical and electrical interlocks on the cask handling hoist and the new fuel handling hoist incorporates load limiting devices to preclude fuel damage during handling.

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ITAAC No. 2.7.2 (Spent Fuel Racks

No.	Comments	Cat.	Resolution
1	Add the following statement to the design description: "Piping penetrations to the spent fuel pool are located to maintain a minimum level of water above the spent fuel pool." CESSAR 9.1.2.2.1 specifies that penetrations are at least 10 feet above the top of fuel assemblies.	1	Agree~~. Will be added to ITAAC 2.7.3.~~ with modification. The 10 foot requirement is in the acceptance criterion for ITAAC #3 of Table 2.7.3-1. NRC concurs that this is sufficient.
2	Revise the design description as marked-up. Neither Nuclear Island or spent fuel cooling CDM material address these aspects.	1	Partially agree. Only the support system will be included. See markup of DD 2.7.2.
3	CESSAR sections 9.1.2 and 9.1.2.3.4 state that the spent fuel storage is for "up to" 907 assemblies (a maximum) where the design description states "at least" 907 assemblies (a minimum). These descriptions should be consistent.	3	Agree. SAR and ITAAC wording will be revised to clarify. The ITTAC design description and CESSAR-DC are revised as shown on the attached marked pages to clarify that the spent fuel storage racks provide on-site storage for at least 907 spent fuel assemblies. The SAR is also revised to identify that the criticality safety analyses have been performed for initial on-site storage of ~~121~~ 907 fuel assemblies.
4	Revise the SFP (2.7.3) design description to ensure siphonic draining of pool is precluded.	1	Agree. Will be added to ITAAC 2.7.3. See markups to ITAAC 2.7.3.

2.7.2 SPENT FUEL STORAGE RACKS

Design Description

The Spent Fuel Storage Racks provide [redacted] on-site storage for at least 907 spent fuel assemblies. The Spent Fuel Storage Racks are safety-related.

The Spent Fuel Storage Racks are located in the nuclear island structures in the spent fuel pool.

The Spent Fuel Storage Racks are free standing structures that support and protect spent fuel assemblies. The Spent Fuel Storage Racks maintain the effective neutron multiplication factor less than the required criticality limits during normal operation and postulated accident conditions.

The Spent Fuel Storage Racks are designed and fabricated in accordance with ASME Code Section III, Subsection NF, Class 3 Component Supports requirements.

The Spent Fuel Storage Racks are designed to accommodate design basis loads and load combinations including the effects of impact of fuel assemblies on the racks and the impact due to postulated fuel handling accidents without losing the structural capability to maintain the fuel in a non-critical configuration.

The Spent Fuel Racks and support system are classified Seismic Category I.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.2-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Spent Fuel Storage Racks.

9.1.1.3.3 Seismic Classification

The New Fuel Storage Racks, Storage Vault, and the Rack Restraint System are qualified as Seismic Category I. The seismic category of other building components associated with handling fuel assemblies is noted in Table 3.2-1. Those items in the immediate vicinity of the new fuel storage area that are not qualified as Seismic Category I are designed such that their failure will not result in damage to the fuel racks or fuel (See Section 9.1.1.3.1.1.J).

9.1.1.3.4 Storage Capacity

Storage is typically provided for a total of 121 new fuel assemblies in two 50% density 11x11 racks.

9.1.2 SPENT FUEL STORAGE

9.1.2.1 Design Bases

The following design bases are imposed on the storage of fuel within the spent fuel pool:

A. Accidental criticality shall be prevented for the most reactive arrangement of stored spent fuel by avoiding a K_{eff} greater than 0.95. This design basis shall be met under any normal operation and postulated accident conditions identified in Section 9.1.2.3.1.1.

B. The requirements of Regulatory Guides 1.13, 1.29, 1.115, and 1.117 shall be met. The spent fuel pool area is designed to prevent a loss of water in the fuel pool from uncovering the fuel, prevent heavy loads from traversing over the fuel racks when the racks contain fuel assemblies, withstand the impact of a fuel assembly or a handling tool or a combination of both falling from the maximum handling elevation, incorporate components meeting the seismic classification designated in Table 3.2-1,and incorporate water level and radiation monitoring instrumentation.

C. The storage racks and facilities shall be Seismic Category I.

D. The spent fuel storage racks ~~Storage shall be~~ provide~~d~~ on-site storage for at least ~~up to~~ 907 spent fuel assemblies. All components within the area of the fuel racks meet the requirements of Table 3.2-1 to preclude rack damage. W

E. The racks shall not be anchored to the pool floor or wall. Clearances shall be allowed for rack tipping but the rack design and loading shall preclude rack overturning.

B. The rack is assumed to be filled to initial on-site storage capacity with fuel assemblies of the type whose criticality safety is evaluated with the spent fuel pool filled with water. |W

C. For normal operation, no credit is assumed for the boron normally found in the spent fuel pool water. For the flooded spent fuel pool criticality analysis, an optimum temperature is assumed for the water moderator. In evaluating the criticality limits of a dropped fuel assembly and tool accident, it is assumed that boron concentration in the spent fuel pool water is less than one-half of normal (see Section 9.1.3.1.4A) and well below the minimum defined by Technical Specifications.

D. An infinite fuel assembly array is assumed for the flooded spent fuel pool analysis.

E. Only one fuel assembly is assumed to be dropped in a fuel handling accident.

F. It is conservatively assumed that four rows of fuel rods are damaged during a fuel assembly handling accident.

G. Eighty-five percent (85%) of the actual burnup for a given initial enrichment is used for each fuel assembly in the spent fuel rack criticality analysis.

9.1.2.3.1.3 Criticality Safety Margins

Criticality safety margins for initial on-site storage are assured by: |W

A. Neglecting the neutron absorption effects associated with the boron normally in the spent fuel pool water during normal operations and assuming that spent fuel pool boron concentration during a fuel assembly drop accident is less than one-half of normal (see Section 9.1.3.1.4A) and well below the minimum defined by Technical Specifications.

B. When fuel is stored in the borated or mixed modes (freshly burned fuel assembly is inadvertently placed in Region II), the minimum boron concentration in the spent fuel pool water is that defined by Technical Specifications that apply whenever fuel is to be moved in the storage pool.

C. Limiting the capacity to 907 fuel assemblies. |W

In evaluating criticality safety, the two-dimensional transport code DOT-4 (Reference 1) is used to calculate the K_{eff} in the spent fuel storage racks for Region I and Region II for normal design conditions. The calculations are performed for a typical repeating lattice unit for Region I and Region II of the spent fuel storage racks. No credit is assumed for the boron normally found in the spent fuel pool water. For Region I, K_{eff} is calculated for fresh fuel with an enrichment of 5.0 wt.% U-235, with allowance for uncertainties due to deviation from nominal

9.1.2.3.4 Storage Capacity

Storage is provided for ~~up to~~ at least 907 spent fuel assemblies. This provides storage for approximately 10 years of unit operation.

9.1.2.3.5 Fuel Assembly Cooling

The spent fuel pool storage racks are designed to prevent extensive bulk boiling in the racks as well as maintain fuel cladding temperatures well below 650°F for the following collective conditions:

A. Natural convection water circulation within the spent fuel pool,

B. Maximum pool water temperature of 150°F at the fuel rack inlet flow passages, and

C. Maximum fuel pool heat load as described in Section 9.1.3.

9.1.2.3.6 Compliance with ANS 57.2

The design of the Spent Fuel Storage facility conforms with the guidelines of paragraph 5.4 of ANS 57.2. As an example, the facility incorporates monitoring systems to verify pool water temperature to insure adequate fuel assembly cooling, radiation detectors to determine if radiation levels exceed predetermined setpoints and alarms to notify plant personnel of abnormal conditions.

The features include:

A. A radiation monitor with audible alarm on the spent fuel handling machine adjacent to the operator control console.

B. Radiation monitors, including a continuous air monitor, within the spent fuel pool area. At least one monitor indicates and alarms in the control room.

C. Uninterruptible communications by the use of sound powered phones or a separate communication system.

D. Redundant alarm and actuation system for the pool ventilation system during those periods it is not in use.

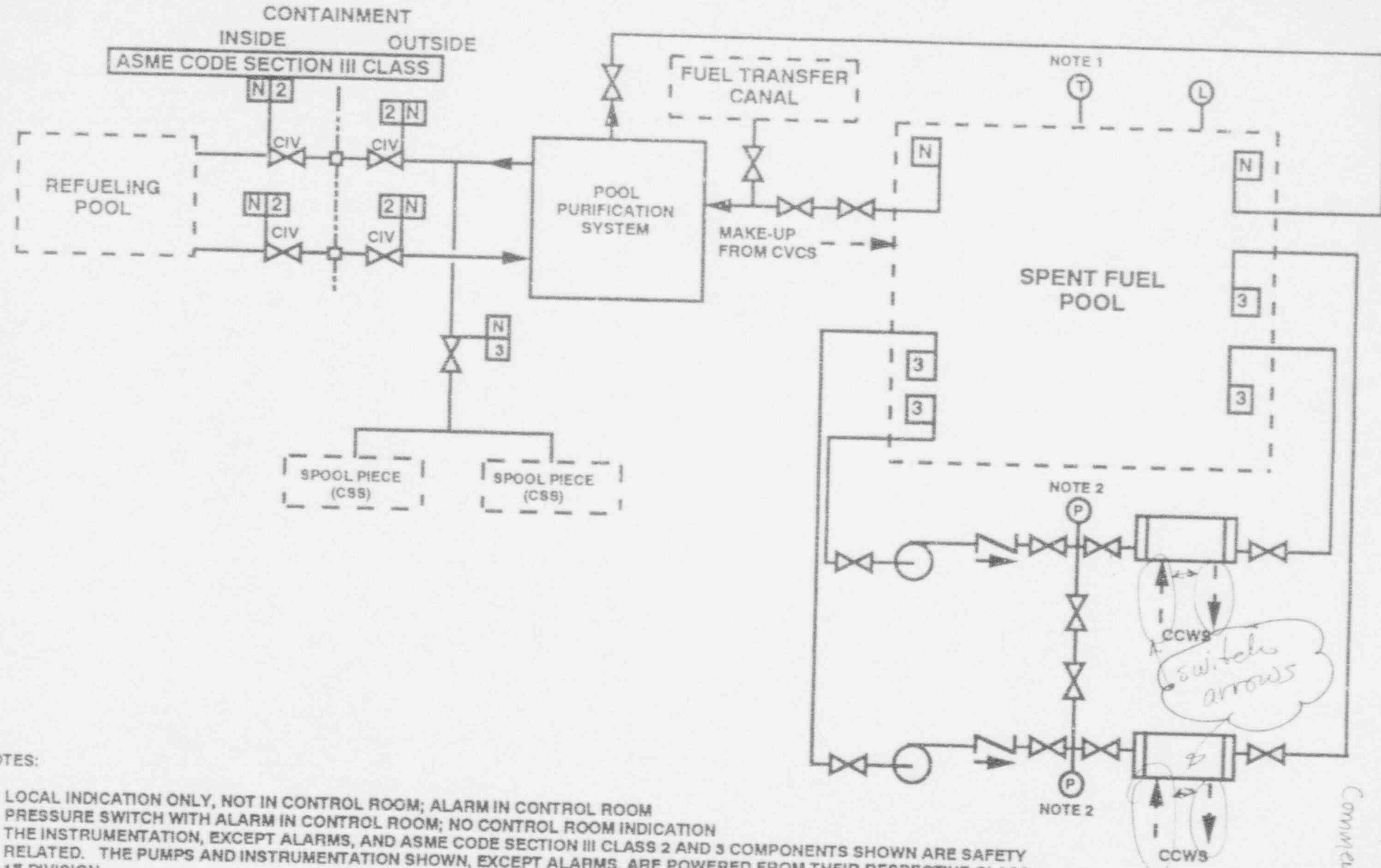
E. Ventilation sampling provisions.

To facilitate use, all monitoring systems are capable of being calibrated.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.3 (Pool Cooling and Purification System (PCPS)

No.	Comments	Cat.	Resolution
1	Figure 2.7.3-1: Reverse the flow directions of the CCWS to the SPF heat exchangers to counter-flow; to be consistent with P&ID 9.1-3 and Figure Legend page 3.	1	ABB-CE agrees. Figure 2.7.3-1 will be revised to show the flow direction of CCWS as counter-flow, as shown on the attached markup. **NRC Staff concurs.**



NOTES:

1. LOCAL INDICATION ONLY, NOT IN CONTROL ROOM; ALARM IN CONTROL ROOM
2. PRESSURE SWITCH WITH ALARM IN CONTROL ROOM; NO CONTROL ROOM INDICATION
3. THE INSTRUMENTATION, EXCEPT ALARMS, AND ASME CODE SECTION III CLASS 2 AND 3 COMPONENTS SHOWN ARE SAFETY RELATED. THE PUMPS AND INSTRUMENTATION SHOWN, EXCEPT ALARMS, ARE POWERED FROM THEIR RESPECTIVE CLASS 1E DIVISION.

2.7.3 Comment No. 1

FIGURE 2.7.3 -1
POOL COOLING AND PURIFICATION SYSTEM

12-31-93

2.7.3 POOL COOLING AND PURIFICATION SYSTEM

Design Description

The Pool Cooling and Purification System (PCPS) consists of a spent fuel pool cooling system (SFPCS) and a pool purification system. The SFPCS removes heat generated by the stored spent fuel assemblies in the spent fuel pool water. The pool purification system pumps spent fuel pool water, refueling pool water, and fuel transfer canal water through filters and ion exchangers.

The Basic Configuration of the PCPS is as shown on Figure 2.7.3-1. The SFPCS is safety-related and the pool purification system is non-safety-related.

The PCPS is located in the reactor building and nuclear annex.

The SFPCS has two Divisions, each with a spent fuel pool (SFP) pump, a SFP heat exchanger, and associated valves, piping, controls, and instrumentation. A cross-connect line with isolation valves between the SFP pump discharge lines is provided to allow either pump to be used with either heat exchanger.

Each SFPCS Division has the heat removal capacity to prevent boiling in the spent fuel pool with a full core offload of fuel assemblies and a ten year inventory of stored irradiated fuel. Heat from the spent fuel pool is transferred to the component cooling water system (CCWS) in the spent fuel pool cooling heat exchangers.

The PCPS includes provisions to prevent gravity and siphonic draining of the spent fuel pool and refueling pool. |

The ASME Code Section III Class for the PCPS pressure retaining components shown on Figure 2.7.3-1 is as depicted on the figure.

Safety-related equipment shown on Figure 2.7.3-1 is classified Seismic Category I.

Displays of the PCPS instrumentation shown on Figure 2.7.3-1 are available as noted on the Figure.

Controls exist in the main control room (MCR) to start and stop the spent fuel pool cooling pumps.

PCPS alarms shown on Figure 2.7.3-1 are provided as shown on the Figure.

Water is supplied to each SFPCS pump at a pressure greater than the pump's required net positive suction head (NPSH).

TABLE 2.7.3-1

POOL COOLING AND PURIFICATION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the PCPS is as shown on Figure 2.7.3-1.	1. Inspection of the as-built PCPS configuration will be conducted.	1. For the components and equipment shown on Figure 2.7.3-1, the as-built PCPS conforms with the Basic Configuration.
2. Each SFPCS Division has the heat removal capacity to prevent boiling in the spent fuel pool with a full core offload of fuel assemblies and a ten year inventory of stored irradiated fuel.	2. Testing to measure SFPCS pump flow in each Division will be performed. Inspection and analysis to determine the heat removal capability of each SFPCS Division will be performed based on test data and as-built data.	2. Each SFPCS Division will remove at least 67.25 million btu/hr from the spent fuel pool, with the spent fuel pool at 180°F and component cooling water supplied at 5000 gpm and 105°F.
3. The PCPS includes provisions to prevent gravity and siphonic draining of the spent fuel pool and the refueling pool.	3. Inspection of the PCPS suction and return line connections to the refueling pool and spent fuel pool will be performed.	3. Spent fuel pool cooling suction connections are located at least 10 feet above the top of the spent fuel. Anti-siphon devices are provided in the lines for spent fuel pool cooling return, spent fuel pool purification suction and return, and refueling pool suction and return.
4. The ASME Code Section III PCPS components shown on Figure 2.7.3-1 retain their pressure boundary integrity under internal pressures that will be experienced during service.	4. A pressure test will be conducted on those components of the PCPS required to be pressure tested by the ASME Code Section III.	4. The results of the pressure test of ASME Code Section III components of the PCPS conform with the pressure testing acceptance criteria in ASME Code Section III.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.5 (Station Service Water System (SSWS))

No.	Comments	Cat.	Resolution
1	Active valves delineated in CESSAR Section 9.2.1.2.1.8 and Table 9.2.1-3 are SSW strainer backwash MOVs (there are six MOVs per strainer, and 2 strainers per division), these are not included in the CDM. Also CDM Section 2.7.5 page 2, 1st para. discussed MOVs with active safety function but without any reference to a specific application, i.e., is this paragraph applicable to the MOVs as shown on Figure 2.7.5-1, but are not identified in CESSAR as "active"? Please clarify.	1	Disagree. The number of strainer backwash MOVs is dependent upon the type of strainer selected. Therefore, they are not included in the CDM. There are no active valves shown on the figure. The statement is for the backwash valves that are equipment dependent. The paragraph on Motor operated active valves will be deleted from the DD and the associated ITAAC will be deleted from the table. See markup of ITAAC 2.7.5.
2	CDM 2.7.5, page 2, under Interface Requirements: A. It was stated that "The UHS is capable of dissipating a heat load of at least $143*10^6$Btu/hr during the initial phase of a DBA."	1	Agree. Revise ITAAC for 134.2×10^6 Btu/hr. In addition the physical barrier to maintain separation will be revised to state "a physical barrier and fire barrier". See markup of ITAAC 2.7.5.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.5 (Station Service Water System (SSWS))

No.	Comments	Cat.	Resolution
2	(continued) B. CESSAR section 9.2.1.2.2.5 Emergency Operation and Ultimate Heat Sink section 9.2.5.2 System Description 3rd paragraph required the UHS to operate for a nominal 30 days "without any blowdown for salinity control". This requirement is not in the CDM. C. CESSAR section 9.2.5.1.3 stated that "The UHS shall meet Seismic Category I requirements." This requirement is not in the CDM.		Agreed with NRC to omit comment. Agreed with NRC to omit comment.
3 (4)	CESSAR comments: A. Section 9.2.1.5.3 D4 stated "SSW radiation monitors 1 and 2 low outlet flows." Is this statement correct? B. Table 9.2.1-3, sh 1 of 2: the last entry "SW-200" belongs to Division 2 which is on Sh 2 of 2. C. Table 18.5.4-1 sh 4 of 4 (also CDM Table 2.12.1-1) used "SSW HX" whereas CESSAR section 9.2 and CDM 2.7.5 used CCW HX to describe the same component.	1	A. Statement in CESSAR-DC is correct. B. Agree. See markup of CESSAR-DC Table 9.2.1-3. C. Agree. See markup to CESSAR-DC Table 18.5.4-1 and ITAAC Table 2.12.1-1.

Motor operated valves (MOVs) having an active safety function will open, or will close, or will open and also close, under differential pressure or fluid flow conditions and under temperature conditions.

Check valves shown on Figure 2.7.5-1 will open, or will close, or will open and also close, under system pressure, fluid flow conditions, or temperature conditions.

Interface Requirements

The Ultimate Heat Sink (UHS) transfers heat from the SSWS to the environment during operation, shutdown, refueling, and design basis accident conditions. The Ultimate Heat Sink is capable of dissipating a heat load of at least ~~143.0~~ 134.3 million BTU/hr during the initial phase of a design basis accident. The UHS is sized so that makeup water is not required for at least 30 days following a design basis accident. During this period of 30 days, the design basis temperatures of safety-related equipment are not exceeded.

Water is supplied to each SSWS pump at a net positive suction head (NPSH) greater than the pump's required NPSH.

The Station Service Water Pump Structure is classified Seismic Category I and provides ~~physical barriers~~ a physical barrier and fire barrier to maintain separation of SSWS mechanical Divisions.

The SSWS pump structure ventilation system is classified Seismic Category I, and its mechanical Divisions are separated by physical barriers.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.5-1 specifies the inspections, tests, analyses and associated acceptance criteria for the Station Servic Water System.

TABLE 2.7.5-1 (Continued)

STATION SERVICE WATER SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The two mechanical Divisions of the SSWS are physically separated.	5. Inspection of the as-built mechanical Divisions will be performed.	5. The two mechanical Divisions of the SSWS are separated by a Divisional wall or a fire barrier.
6.a) Displays of the SSWS instrumentation shown on Figure 2.7.5-1 exist in the MCR or can be retrieved there.	6.a) Inspection for the existence or retrieveability in the MCR of instrumentation displays will be performed.	6.a) Displays of the instrumentation shown on Figure 2.7.5-1 exist in the MCR or can be retrieved there.
6.b) Controls exist in the MCR to start and stop the station service water pumps, and to open and close those power operated valves shown on Figure 2.7.5-1.	6.b) Testing will be performed using the SSWS controls in the MCR.	6.b) SSWS controls in the MCR operate to start and stop station service water pumps, and to open and close those power operated valves shown on Figure 2.7.5-1.
~~7. Motor operated valves (MOVs) having an active safety function will open, or will close, or will open and also close, under differential pressure or fluid flow conditions and under temperature conditions.~~	~~7. Testing will be conducted to open, or close, or open and also close, MOVs having an active safety function under pre-operational differential pressure or fluid flow conditions and under temperature conditions.~~	~~7. Each MOV having an active safety function opens, or closes, or opens and also closes.~~
7 ~~8~~. Check valves shown on Figure 2.7.5-1 will open, or will close, or will open and also close under system pressure, fluid flow conditions, or temperature conditions.	7 ~~8~~. Testing will be conducted to open, or close, or open and also close, check valves shown on Figure 2.7.5-1 under system preoperational pressure, fluid flow conditions, or temperature conditions.	7 ~~8~~. Each check valve shown on Figure 2.7.5-1 opens, or closes, or opens and also closes.

TABLE 9.2.1-3

(Sheet 1 of 2)

ACTIVE VALVES, STATION SERVICE WATER SYSTEM

Valve Number	Safety Function	Valve Type	ASME Section III Code Class	Actuator Type
SW-100	Operate	Plug	3	Electric Motor
SW-101	Operate	Plug	3	Electric Motor
SW-102	Operate	Plug	3	Electric Motor
SW-103	Operate	Plug	3	Electric Motor
SW-104	Operate	Plug	3	Electric Motor
SW-105	Operate	Plug	3	Electric Motor
SW-106	Operate	Plug	3	Electric Motor
SW-107	Operate	Plug	3	Electric Motor
SW-108	Operate	Plug	3	Electric Motor
SW-109	Operate	Plug	3	Electric Motor
SW-110	Operate	Plug	3	Electric Motor
SW-111	Operate	Plug	3	Electric Motor
SW-1302	Operate	Swing Check	3	None
SW-1303	Operate	Swing Check	3	None
SW-200	Operate	Plug	3	Electric Motor

MOVE TO * ON TOP OF NEXT PAGE

TABLE 9.2.1-3 (Cont'd)

(Sheet 2 of 2)

ACTIVE VALVES, STATION SERVICE WATER SYSTEM

Valve Number	Safety Function	Valve Type	ASME Section III Code Class	Actuator Type
SW-201	Operate	Plug	3	Electric Motor
SW-202	Operate	Plug	3	Electric Motor
SW-203	Operate	Plug	3	Electric Motor
SW-204	Operate	Plug	3	Electric Motor
SW-205	Operate	Plug	3	Electric Motor
SW-206	Operate	Plug	3	Electric Motor
SW-207	Operate	Plug	3	Electric Motor
SW-208	Operate	Plug	3	Electric Motor
SW-209	Operate	Plug	3	Electric Motor
SW-210	Operate	Plug	3	Electric Motor
SW-211	Operate	Plug	3	Electric Motor
SW-2302	Operate	Swing Check	3	None
SW-2303	Operate	Swing Check	3	None

TABLE 18.5.4-1

(Sheet 4 of 4)

MCR MINIMUM INVENTORY OF FIXED POSITION
ANNUNCIATORS, DISPLAYS AND CONTROLS

CCW HX station service water

PARAMETER DESCRIPTION	Annunciators[1]	Displays	Controls
SCS HX Bypass Inlet & Outlet temperature (when SCS is in operation)		X	
SCS HX outlet valve position		X	X
SCS pump on/off		X	X
SCS/CSS pump suction cross-connect valve position		X	X
SCS/CSS pump discharge cross-connect valve position		X	X
SIAS actuation	X		X
SI flow		X	
SI pump on/off		X	X
SI throttling isolation valve position		X	X
Spent Fuel Pool level	X		
Startup Rate (NI)		X	
~~SSW HX~~ inlet isolation valve position		X	X
~~SSW HX~~ outlet isolation valve position		X	X
~~SSW HX~~ outlet flow	X		
SSW pump on/off		X	X
SG Blowdown sample radiation	X		
SG level	X	X	
SG pressure	X		
Vacuum Pump Activity	X		
Turbine Trip		X	X

(1) Annunciators are alarms and other alerting displays designed to direct operator attention.

MCR MINIMUM INVENTORY OF FIXED POSITION ANNUNCIATORS, DISPLAYS AND CONTROLS

CCW HX station service water

PARAMETER DESCRIPTION	Annunciators[1]	Displays	Controls
Reactor power (NI)		X	
Reactor Trip (RPS)	X		X
Reactor Vessel level	X	X	
SCS flow (while SCS is in operation)	X	X	
SCS isolation valve position (& LTOP)	X	X	X
SCS HX Bypass Valve position		X	X
SCS HX CCW supply/isolation valve position		X	X
SCS HX/Bypass Inlet & Outlet temperature (when SCS is in operation)		X	
SCS HX outlet valve position		X	X
SCS pump on/off		X	X
SCS/CSS pump suction cross-connect valve position		X	X
SCS/CSS pump discharge cross-connect valve position		X	X
SIAS actuation	X		X
SI flow		X	
SI pump on/off		X	X
SI throttling isolation valve position		X	X
Spent Fuel Pool level	X		
Startup Rate (NI)		X	
~~SSW HX~~ inlet isolation valve position		X	X
~~SSW HX~~ outlet isolation valve position		X	X
~~SSW HX~~ outlet flow	X		
SSW pump on/off		X	X
SG Blowdown sample radiation	X		
SG level	X	X	
SG pressure	X		
Vacuum Pump Activity	X		
Turbine Trip		X	X

[1] Annunciators are alarms and other alerting displays designed to direct operator attention.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.6 (Component Cooling Water System (CCWS))

No.	Comments	Cat.	Resolution
1 (2)	CESSAR Table 9.2.2-5: valve CC-130 is a CIV, and is currently listed as ASME Section III Code Class 3, whereas CIVs are Code Class 2 in accordance with CDM Section 2.4.5.	1	Agree. See markup of CESSAR-DC Table 9.2.2-5.
2 (3)	In the Refueling Mode, the letdown HX receives CCW flow in accordance with CDM Table 2.7.6-1 page 2, and CESSAR section 9.2.2.2.2.4 implied the same. Whereas CESSAR Table 9.2.2-3 sh 11 of 16 showed that the CCW flow is zero. Please resolve this discrepancy.	1	Disagree. In Table 2.7.6-1 it says that the letdown heat exchanger "can" recieve component cooling water flow. This means that component cooling water flow is not automatically isolated by an isolation valve in this mode. A control valve is provided to control component cooling water flow to this heat exchanger. The control valve is operated by the outlet temperature of the CVCS side of the letdown heat exchanger. **NRC Staff concurs.**
3 (4)	CESSAR comments: A. Section 9.2.2.2.1.9K - missing last digit of valve CC-24(?). B. Containment penetration numbers TO/FROM the Letdown HX:30,31; TO/FROM RCP 1A, 1B:32,33; TO/FROM RCP 2A,2B:34,35 have been identified in CESSAR Table 6.2.4-1 sh 6 of 15. These penetration numbers are not on the respective sheets of Figure 9.2.2-1.	1	Agree. Valve number will be revised to CC-243. See markup of CESSAR-DC page 9.2-32. Penetration numbers are not shown on P&IDs.

TABLE 9.2.2-5

(Sheet 1 of 3)

ACTIVE VALVES, COMPONENT COOLING WATER SYSTEM

Valve Number	Safety Function	Valve Type	ASME Section III Code Class	Actuator Type
CC-100	Close	Throttle	3	Pneumatic
CC-101	Close	Throttle	3	Pneumatic
CC-102	Close	Butterfly	3	Pneumatic
CC-103	Close	Butterfly	3	Pneumatic
CC-110	Open	Throttle	3	Pneumatic
CC-111	Operate	Butterfly	3	Electric Motor
CC-112	Open	Throttle	3	Pneumatic
CC-113	Close	Butterfly	3	Electric Motor
CC-114	Open	Butterfly	3	Electric Motor
CC-122	Close	Butterfly	3	Pneumatic
CC-123	Close	Butterfly	3	Pneumatic
CC-130	Close	Butterfly	~~3~~ 2	Electric Motor
CC-1302	Operate	Swing Check	3	None
CC-1303	Operate	Swing Check	3	None
CC-131	Close	Butterfly	2	Electric Motor
CC-136	Close	Butterfly	2	Electric Motor

These valves are pneumatically operated and are required to fail closed. These valves automatically close on an SIAS.

I. Component Cooling Water Pump Discharge Check Valves

Valves CC-1302, CC-1303, CC-2302, and CC-2303 are required to function during a safe plant shutdown. In the event that one of the pumps ceases to produce flow and pressure head, these valves prevent flow reversal through the non-operating pump.

J. Component Cooling Water Surge Tank Vacuum Breakers

The CCWS surge tank vacuum breakers are required to function during a safe plant shutdown.

K. Containment Isolation Valves

The following containment isolation valves close upon receipt of a Containment Isolation Actuation Signal (CIAS):

Supply to the letdown heat exchanger: CC-240, CC-241

Return from the letdown heat exchanger: CC-242, ~~CC-24~~ CC-243

The following containment isolation valves are automatically closed on a low-low CCW surge tank level:

CC-130, CC-131 - Supply to reactor coolant pumps 1A and 1B

CC-230, CC-231 - Supply to reactor coolant pumps 2A and 2B

CC-136, CC-137 - Return from reactor coolant pumps 1A and 1B

CC-236, CC-237 - Return from reactor coolant pumps 2A and 2B

These valves can be manually opened or closed from the control room.

L. Containment Penetration Piping Bypass Check Valves

Valves CC-1507, CC-1548, CC-2507, CC-2548, CC-2622 and CC-2628 provide overpressure protection for containment penetration piping to prevent damage when the piping is isolated.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.8 (Condensate Storage System

No.	Comments	Cat.	Resolution
1	Figure 2 7.8-1 showed a gate valve with an unspecified operator. CESSAR Figure 9.2.6-1 showed a diaphragm operated globe valve controlling the tank level from signals generated by a level transmitter. Please resolve this discrepancy.	1	Agree. The gate valve will be removed from Figure 2.7.8-1. Inclusion of this valve is not consistent with the level of detail for this non-safety system.

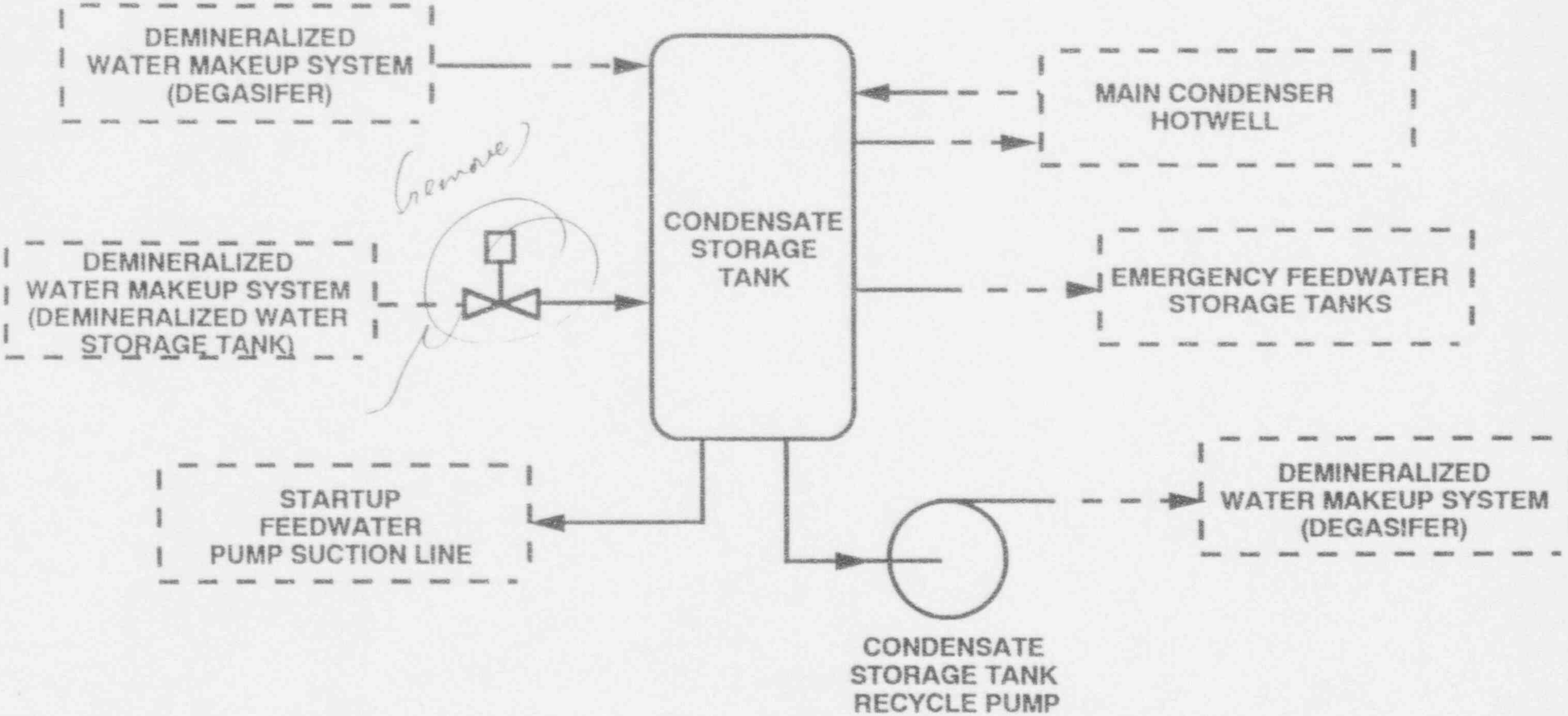


FIGURE 2.7.8-1
CONDENSATE STORAGE SYSTEM

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.9 (Process Sampling System)

No.	Comments	Cat.	Resolution
1	ITAAC item #4 is not covered in the Design Description.	1	ABB-CE agrees. ITAAC 4 will be deleted, as shown on the attached markup. The check valves shown on Figure 2.7.9-1 are ANSI 51.1 non-nuclear safety class, non-ASME Section III code class, and not subject to ASME Section XI testing. The check valves are shown on the Figure only to show the configuration of the relief valves in the system. The relief valves are a design improvement for resolution of the interfacing system LOCA issue.

TABLE 2.7.9-1 (Continued)

PROCESS SAMPLING SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Check valves shown on Figure 2.7.9-1 will open, or will close, or will open and also close under system pressure, fluid flow conditions, or temperature conditions.	4. Testing will be performed to open, or close, or open and also close check valves under preoperational differential pressure, fluid flow conditions, or temperature conditions.	4. Each check valve shown on Figure 2.7.9-1 opens, or closes, or opens and also closes.
4 ~~5~~. Valves with response positions indicated on Figure 2.7.9-1 change position to that indicated on the Figure upon loss of motive power.	4 ~~5~~. Testing of loss of motive power to these valves will be performed.	4 ~~5~~. These valves change position to the position indicated on Figure 2.7.9-1 on loss of motive power.
5 ~~6~~. The PASS can collect samples of reactor coolant and containment atmosphere.	5 ~~6~~. Testing of the PASS capability to obtain samples will be performed under preoperational conditions.	5 ~~6~~. Samples of reactor coolant and containment atmosphere are collected by the PASS.

Comment No. 1
2.7.9

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.11 (Turbine Building Cooling Water System (TBCWS))

No.	Comments	Cat.	Resolution
1	On Figure 2.7.11-1, show the TBCWS HX as counter-flow type to be consistent with the HX legend of page 1.3-3.	1	Agree. See markup.

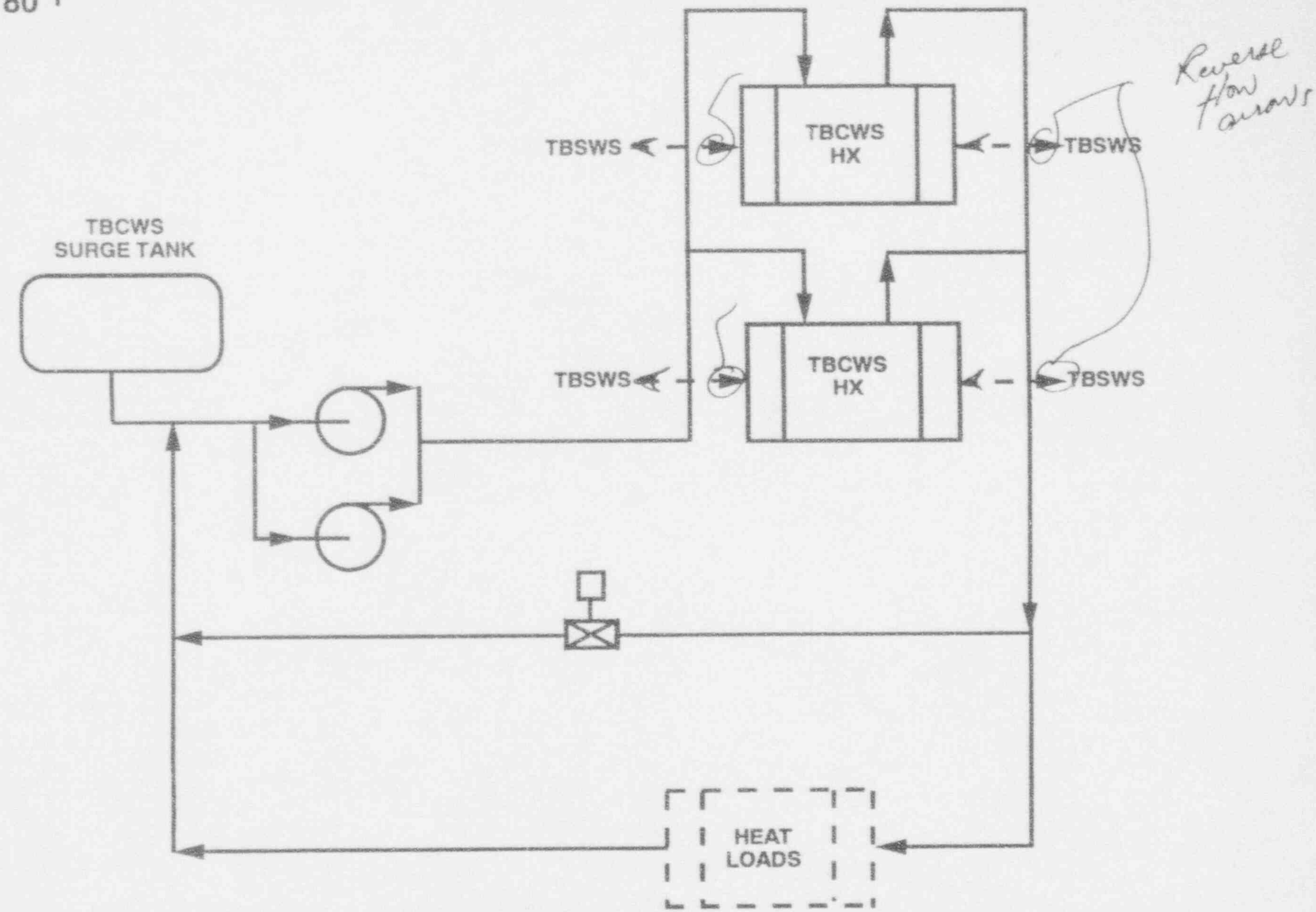
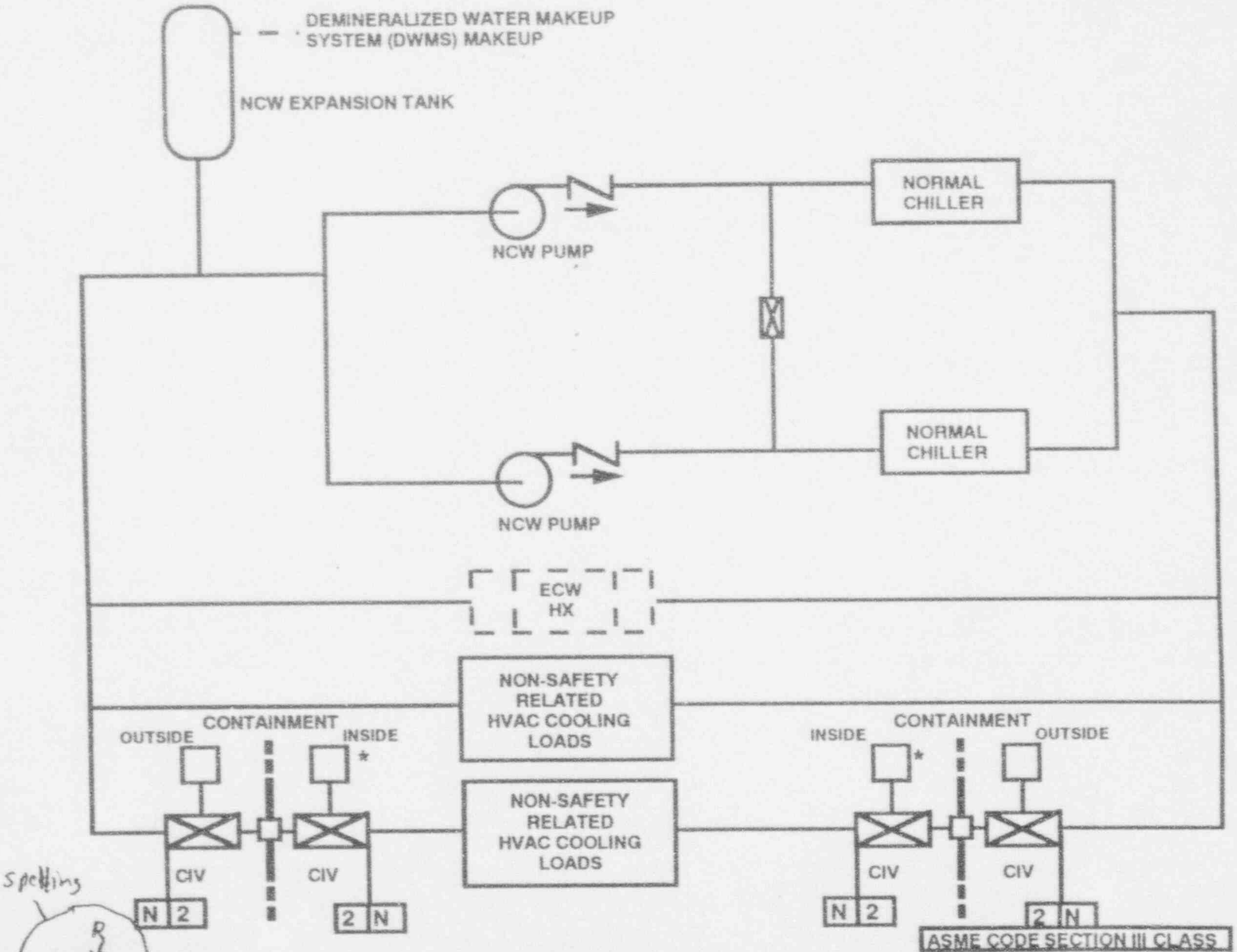


FIGURE 2.7.11-1
TURBINE BUILDING COOLING WATER SYSTEM

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.13 (Normal Chilled Water System)

No.	Comments	Cat.	Resolution
1	See attached markup.	3	Agree. See markup of ITAAC Figure 2.7.13-1.



NOTE:

* EQUIPMENT FOR WHICH PARAGRAPH NUMBER (3) OF THE "VERIFICATIONS FOR BASIC CONFIGURATION FOR SYSTEMS" OF THE GENERAL PROVISIONS (SECTION 1.2) APPLIES.

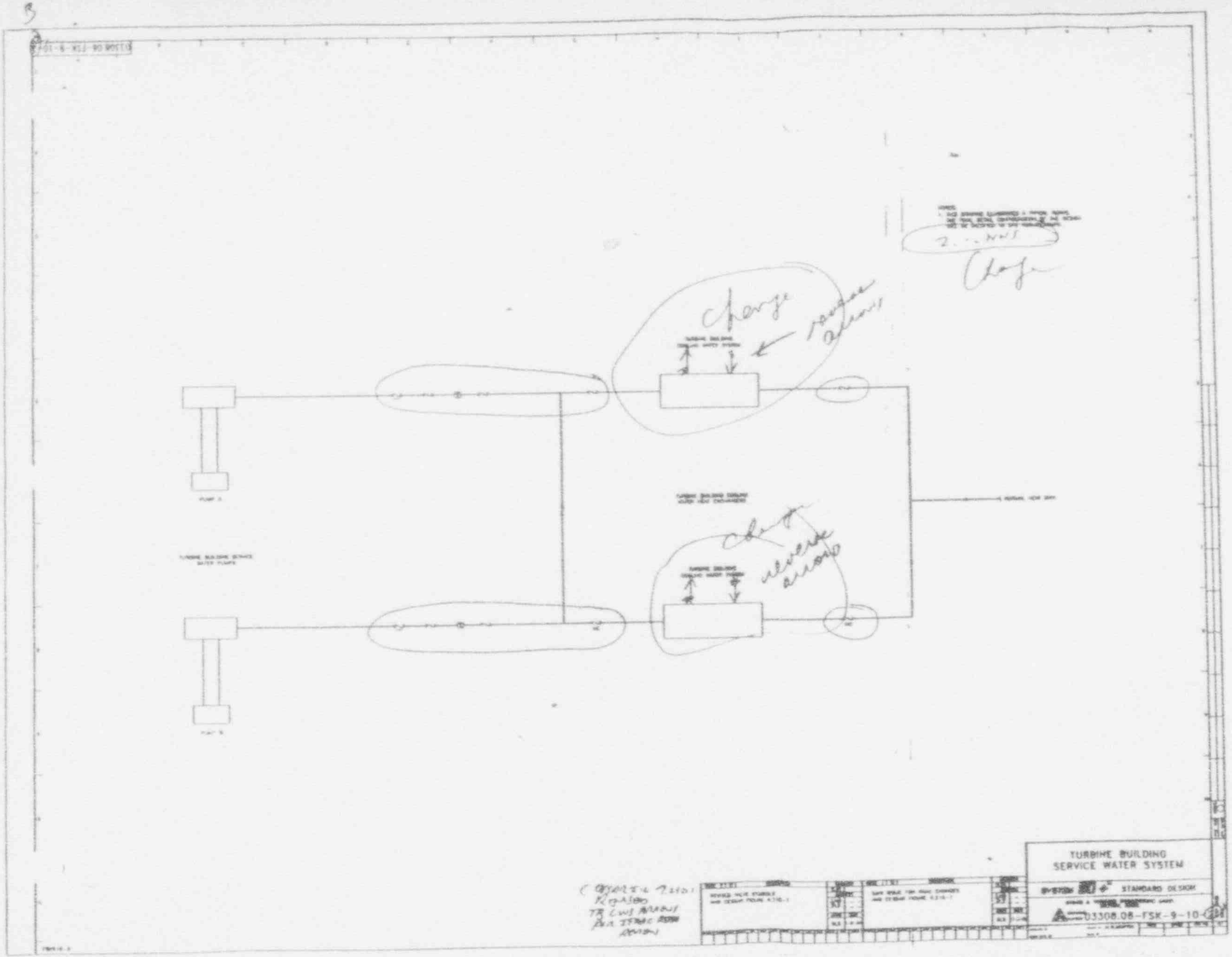
FIGURE 2.7.13-1
NORMAL CHILLED WATER SYSTEM
(ONE OF TWO DIVISIONS)

12-31-93

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.14 (Turbine Building Service Water System (TBSWS)) Page 1 of 1

No.	Comments	Cat.	Resolution
1	CESSAR Figure 9.2.10-1 used valve symbols that are not defined in accordance with the symbols and legends of Figures 1.7-1 and 1.7-2. Please clarify.	1	Agree. See markup.


TURBINE BUILDING
SERVICE WATER SYSTEM
STANDARD DESIGN
03308.08-FSK-9-10
Change
reverse arrows
Change
reverse arrows
Change

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.15 (Equipment and Floor Drainage)

No.	Comments	Cat.	Resolution
1	The system design bases (CESSAR 9.3.3.1) requires that the EFDS be capable of preventing a backflow of water that might exist from maximum flood levels resulting from external or system leakage to areas of the plant containing safety-related equipment. This should be added to the CDEM design description.	1	Disagree. This item is covered by basic configuration. The class 3 check valves on figure prevent backflow. **NRC Staff concurs**

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.16 (CVCS)

No.	Comments	Cat.	Resolution
1	Fig. 2.7.16-1 Note 1 should include ASME Section III, Class 3, components as safety-related items to be consistent with CESSAR Table 3.2-1 which lists CVCS components such as, volume control tank, charging pumps, seal injection heat exchanger, and valves as Safety Class 3.	1	ABB-CE ~~dis~~agrees with modification. The ASME Section III Class 3 portions of the CVCS are not safety-related, i.e., they do not perform a nuclear safety related function (reactor coolant pressure boundary or accident prevention/mitigation). ~~as defined by ANSI 51.1.~~ However, ABB-CE designs these portions of the CVCS to ASME Class 3 standards in accordance with Reg. Guide 1.26 Quality Group C. CESSAR-DC will be revised to delete Safety Class 3 designation for applicable CVCS components. **NRC Staff concurs.**

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.17 (Control Complex Ventilation System

No.	Comments	Cat.	Resolution
1	CESSAR Section 9.4.1.2, page 9.4-6 paragraph 1 absorber should be adsorber.	3	Agree. See markup of CESSAR-DC page 9.4-6.
2 (5)	Acceptance Criteria 8 does not use the 1/8" of water for the pressurization requirement as indicated in the CESSAR section 9.4.1.2, page 9.4-6 paragraph 4.	1	Agree. See markup of ITAAC Table 2.7.17-1.

adsorber

9.4.1.2 System Description

The main control room air-conditioning system consists of two Divisions. Each Division has an outside air intake, louver, tornado damper, dampers, filtration unit, an air conditioning unit with fan, ducting, instrumentation and controls. Each redundant air conditioning unit consists of filter, safety-related chilled water coil for heat removal, electric heating coil and fan for air circulation. Each of the filtration units consists of prefilter, electric heater, absolute (HEPA) filter, carbon ~~absorber~~, post filter (HEPA) and fan, along with ducts and valves and related instrumentation. Chilled water is supplied from the Essential Chilled Water System.

During normal operation, return air from the control room is mixed with a small quantity of outside air for ventilation, is filtered and conditioned in the control room air-conditioning unit, and is delivered to the control room through supply ductwork. Duct-mounted heating coils and humidification equipment provide final adjustments to the control room temperature and humidity for maintaining normal comfort conditions.

Each air inlet structure is provided with redundant radiation monitoring devices and a smoke detector. The designated MCR filtration units and ventilation fan start automatically on a Safety Injection Actuation Signal (SIAS) or high radiation signal. Upon failure of the designated filtration unit, the redundant filtration unit starts automatically. The MCR all filtration unit filters particulates and potential radioactive iodines from ~~a portion~~ of the return air, and delivers the filtered air to the inlet of the main air-conditioning unit.

CONDITIONING

The Technical Support Center air-conditioning system consists of an air-~~handling~~ unit, return air and smoke purge fans, and an emergency filter unit. The TSC is maintained at 1/8" water gauge positive pressure with respect to adjacent areas during post-accident conditions. A common supply air header and common outside air intake dampers are shared by the TSC and the control room to protect the TSC from the contaminants in the outside air intakes. The TSC can be isolated from the Main Control Room by using manual controls. The TSC is automatically isolated if control room pressurization falls below its design value.

The TSC is provided with shielding protection from direct radiation from an external radioactive cloud and internal radioactive sources. The combined effect of all radiation protection measures is designed to be adequate to limit the overall calculated radiation exposure to the personnel inside the TSC to the requirements of ~~GDC~~ 19. The computer room air-conditioning system consists of two 100% air-conditioning units and associated fans. Both the Technical Support Center and computer room air-handling systems are non-safety and non-seismic.

General Design Criteria

TABLE 2.7.17-1 (Continued)

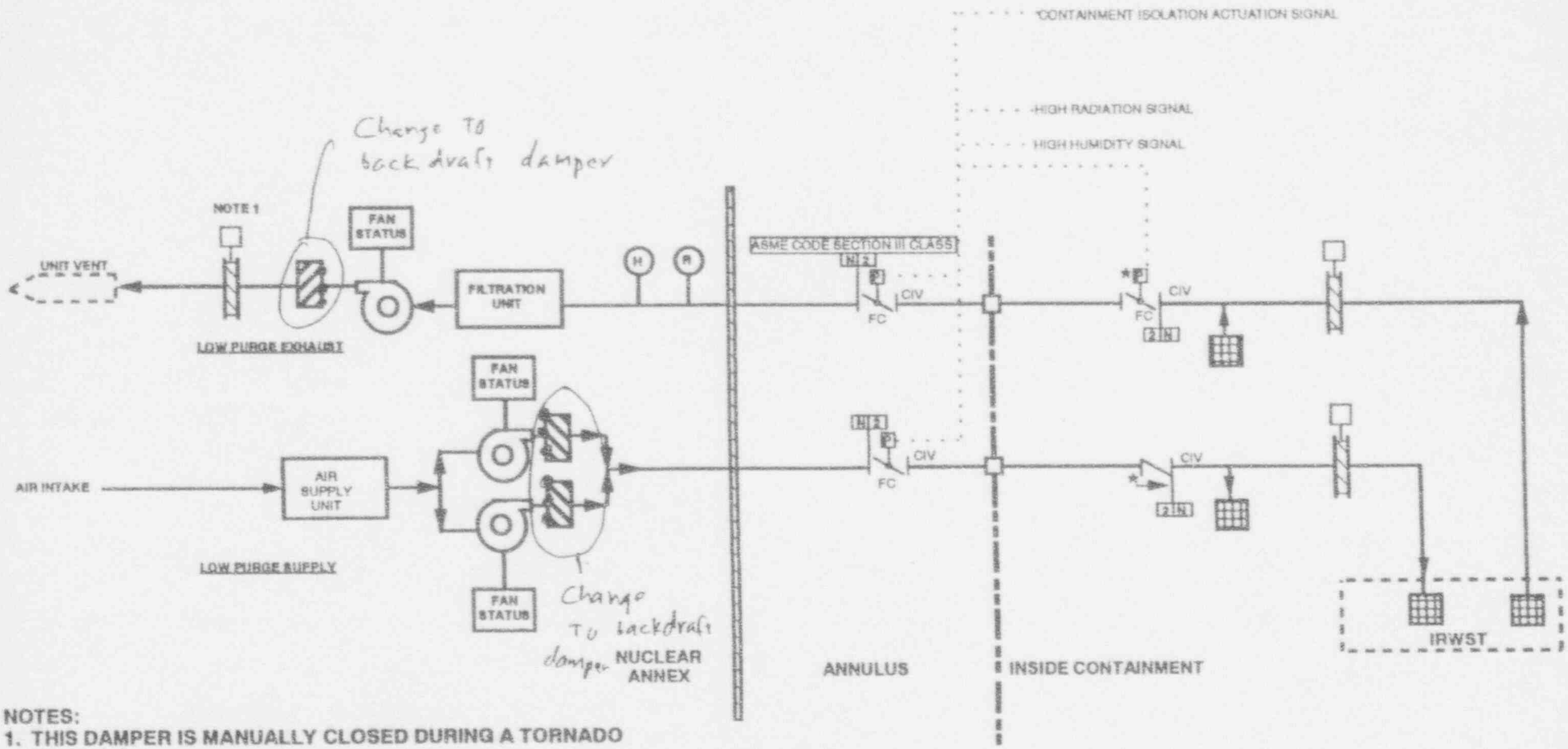
CONTROL COMPLEX VENTILATION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. Each MCR filtration unit and the TSC filtration unit remove particulate matter and iodine.	6. Testing and analysis will be performed on each MCR filtration unit and the TSC filtration unit to determine filter efficiencies.	6. The MCR and TSC filter efficiencies are greater than or equal to 95% for all forms of non-particulate iodine and greater than or equal to 99% for particulate matter greater than 0.3 micron.
7. The MCR is maintained at a positive pressure with respect to the adjacent areas.	7. Testing and analysis will be performed on the MCRACS.	7. The MCR is pressurized to at least 0.125 inches of water gauge relative to the adjacent areas with outside air supply no more than 2000 CFM.
8. The TSC can be pressurized with respect to the adjacent areas.	8. Testing will be performed on the TSC.	8 ~~9~~. The TSC can be maintained at a positive pressure with respect to the adjacent areas.
9. The designated MCR filtration unit starts automatically and the MCR air conditioning unit starts or continues to operate, if running, on receipt of a safety injection actuation signal (SIAS) or a high radiation signal. In addition, the dampers in the MCR circulation lines and the bypass lines reposition to establish the flow path through the MCR filtration units.	9. Testing will be performed on the MCR filtration units, MCR air conditioning units, and dampers using a signal that simulates a safety injection actuation signal (SIAS). The testing will be repeated for a signal that simulates a high radiation signal.	9. The MCR filtration units and MCR air conditioning units start on receipt of a signal that simulates a SIAS,or a signal that simulates high radiation,and dampers reposition to establish the flow path through the MCR filtration units.

CE 80+ ITAAC Independent Review Comments

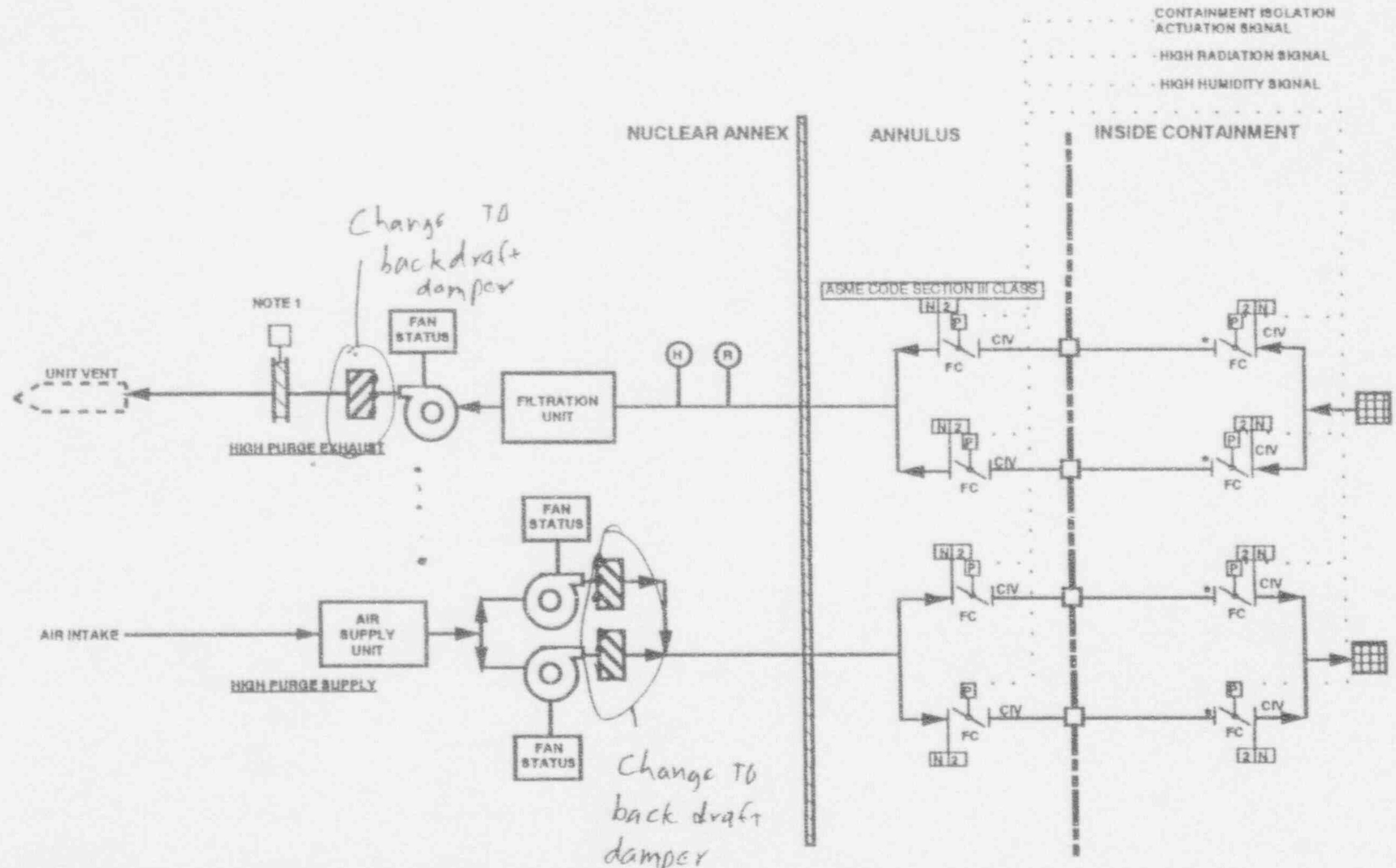
ITAAC No. 2.7.21 (Containment Purge Ventilation System

No.	Comments	Cat.	Resolution
1	Figure 2.7.21-1 and -2 should have back draft dampers on the discharge side of fans. CESSAR Figure 9.4.6.		Agree. See markup of ITAAC Figures 2.7.21-1 and 2.



NOTES:
1. THIS DAMPER IS MANUALLY CLOSED DURING A TORNADO WARNING.
2. * EQUIPMENT FOR WHICH PARAGRAPH NUMBER (3) OF THE " VERIFICATIONS FOR BASIC CONFIGURATION FOR SYSTEMS" OF THE GENERAL PROVISIONS (SECTION 1.2) APPLIES.
3. THE SAFETY-RELATED ELECTRICAL EQUIPMENT IS CLASS 1E.

FIGURE 2.7.21-1
CONTAINMENT PURGE VENTILATION SYSTEM (LOW PURGE)



NOTES:

1. THIS DAMPER IS MANUALLY CLOSED DURING A TORNADO WARNING.
2. * EQUIPMENT FOR WHICH PARAGRAPH NUMBER (3) OF THE " VERIFICATION FOR BASIC CONFIGURATION FOR SYSTEMS" OF THE GENERAL PROVISIONS (SECTION 1.2) APPLIES.
3. THE SAFETY-RELATED ELECTRICAL EQUIPMENT IS CLASS 1E.

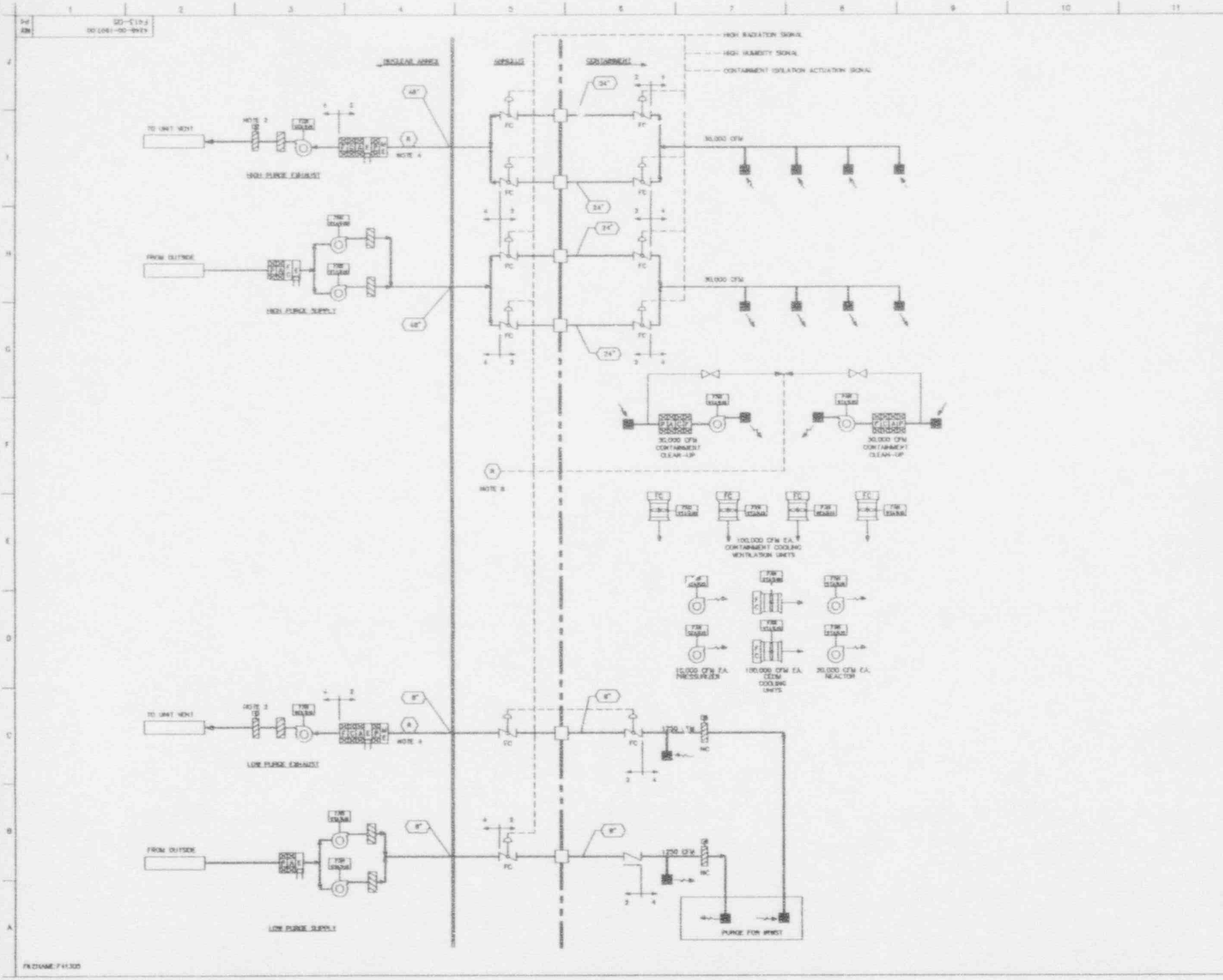
FIGURE 2.7.21-2
CONTAINMENT PURGE VENTILATION SYSTEM (HIGH PURGE)

12-31-93

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.22 (Containment Cooling and Ventilation System

No.	Comments	Cat.	Resolution
1	Figure 2.7.22-1 does not use the standard fan symbol.		Agree. The Containment Recirculation Cooling Fans and CEDM Fans will be revised to the axial vane symbol and the Containment Pressurizer Fans and Reactor Cavity Fans will be revised to the standard fan symbol. See revision to Containment Cooling and Ventilation System Air Flow Diagram and markup of ITAAC Figure 2.7.22-1.



NOTES:

1. FOR GENERAL NOTES AND DRAWING LEGENDS SEE DWGS F400-02-01 & F400-02-02.
2. THIS DAMPER IS REMOTELY AND MANUALLY CLOSED DURING A TORNADO WARNING.
3. THIS SYSTEM IS SAFETY CLASS NNS, EXCEPT FOR CONTAINMENT ISOLATION VALVES AND ASSOCIATED PIPING WHICH ARE SAFETY CLASS 2.
4. ON-LINE RADIATION MONITOR.
5. THE DUCTWORK FROM THE BUILDING EXIT UP TO AND INCLUDING THE ISOLATION DAMPER IS QUALIFIED FOR THE TORNADO DIFFERENTIAL PRESSURE.
6. RADIATION MONITOR IS AN OFFLINE MONITOR.

RELEASED FOR INFORMATION

PRELIMINARY

DUKE ENGINEERING & SERVICES, INC.
CHARLOTTE, N.C.

ABB COMBUSTION ENG. NUCLEAR POWER
COMBUSTION ENGINEERING, INC.

U.S. DEPARTMENT OF ENERGY
SYSTEM 80+™
DESIGN CERTIFICATION

DRAWING TITLE

AIR FLOW DIAGRAM
CONTAINMENT COOLING AND VENTILATION SYSTEM

DRAWING NO. 4148-00-1807.00
F413-05

REV P4

SCALE: NONE

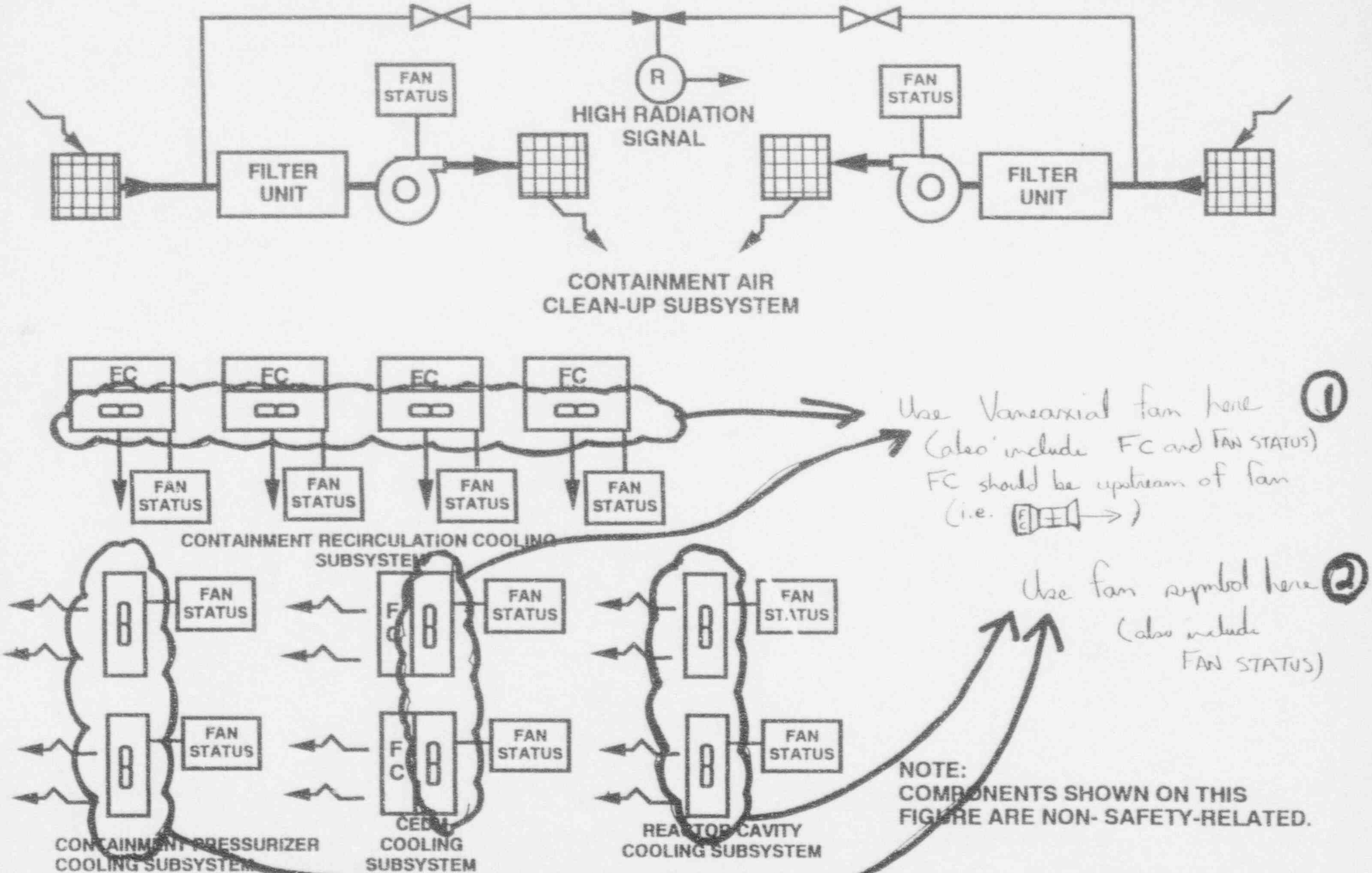


FIGURE 2.7.22-1
CONTAINMENT COOLING AND VENTILATION SYSTEM

12-31-93

FIGURE LEGEND (continued)

Centrifugal Pump

Pump Type Not Specified

Header

Tank

Filter

Strainer

Flexible Connection

Delay Coil

Orifice

NO CHANGES
TO THIS PAGE

Venturi

Compressor Or Fan a

Air Distribution Device

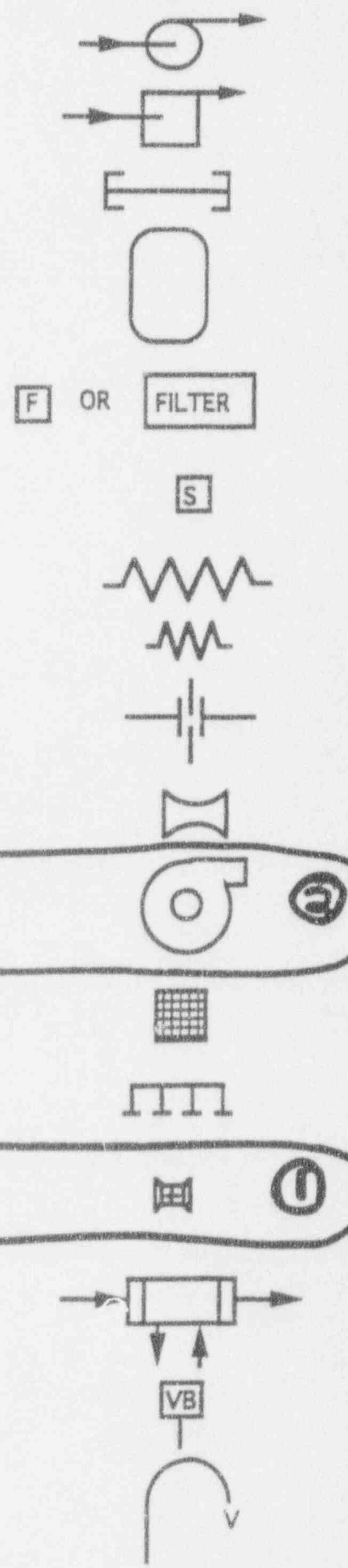
Air Distribution Header

Vaneaxial Fan f

Heat Exchanger

Vacuum Breaker

Vent



CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.23 (Nuclear Annex Ventilation)

No.	Comments	Cat.	Resolution
1 (2)	Paragraph 2 and 7 of the design description appear to be duplicative, this information needs to be stated in a more clear and coherent manner.	1	Agree. See markup of ITAAC 2.7.23.

2.7.23 NUCLEAR ANNEX VENTILATION SYSTEM

Design Description

The Nuclear Annex Ventilation System (NAVS) provides ventilation, cooling and heating to the nuclear annex and is located inside the nuclear annex. The exhaust and supply fans can be used for smoke removal.

The safety-related component cooling water system pump rooms and essential chilled water system pump and chiller rooms are cooled by the Essential Chilled Water System recirculating units.

The Basic Configuration of the NAVS is as shown on Figures 2.7.23-1 and 2.7.23-2. The NAVS is a non-safety-related system.

The NAVS has two Divisions. Each Division of the NAVS has a filtration unit, fans, ductwork, instrumentation, and controls.

Each division of the NAVS maintains its Division of the nuclear annex at a negative pressure relative to the outside atmosphere.

The two mechanical Divisions of the NAVS are physically separated.

~~The safety-related component cooling water system (CCWS) pump rooms and essential chilled water system pump and chilled rooms are cooled by the essential chilled water system recirculating units.~~

Displays of the NAVS instrumentation shown on Figures 2.7.23-1 and 2.7.23-2 exist in the main control room (MCR) or can be retrieved there.

Controls exist in the MCR to start and stop the NAVS filtration units and fans, and to open and close those power operated dampers shown on Figures 2.7.23-1 and 2.7.23-2.

In response to a high radiation signal, the filtration unit bypass dampers close and the filtration unit dampers open to route exhaust air through the filtration units.

The exhaust and supply fans can be used for smoke removal.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.23-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Nuclear Annex Ventilation System.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.24 (Fire Protection System)

No.	Comments	Cat.	Resolution
1	This section covers only the fire protection water supply and distribution system. The criteria and commitments related to passive fire mitigating features, are not discussed.	2	Disagree. Basic configuration of buildings and structures ITAAC covers 3 hour~~s~~ rated fire walls. Minteral insulated cable and containment penetration fire rating is Tier II material. The Fire Hazards analysis in the ITAAC demonstrates safe shutdown inside containment and the annulus. **NRC Staff concurs.**

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.26 (Lighting System)

No.	Comments	Cat.	Resolution
1	Design Description needs to be revised as shown in the attached markup.	1	Agree. Only system names will be capitalized (i.e., Emergency Lighting System, Security Lighting System, and Normal Lighting System). In addition the phrase "associated Class 1E circuits" will be revised to "associated circuits". See markup of ITAAC 2.7.26.
2	ITAAC entry is needed to verify that "the security lighting system provides illumination in isolation zones and outdoor areas within the plant protected perimeter." Refer 3rd para. of design description.	1	Agree. ITAAC will be added. See markup of ITAAC 2.7.26.
3	CESSAR Section 9.5.3 refers to standby non-safety AAC source as combustion turbines instead of gas turbines. The above CESSAR section needs to be revised to be consistent with SSAR Fig. 8.3.1-1 and CESSAR Section 8.3.3.1.1.5.	1	Agree. In addition the phrase "associated Class 1E circuits" will be revised to "associated circuits". See markup of CESSAR-DC pages 9.5-48 and 49.
4 (5)	Design commitment #4 states that "Class 1E DC self-contained battery operated lighting units are provided with a minimum 8 hour capacity." ITAAC does not verify this requirement as written. ITAAC #4 needs to be revised to include tests to verify the minimum 8 hour capacity of the batteries.	1	Agree. ITAAC 4 will be split. See markup of ITAAC 2.7.26.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.7.26 (Lighting System)

No.	Comments	Cat.	Resolution
5 (6)	The CESSAR Section 9.5.3 needs to be revised to show the separation and independent requirements of the lighting circuits.	2	Agree. See markups of CESSAR DC Section 9.5.3.
6 (7)	ITAAC #4 needs editorial change as shown in the attached mark-up.	1	Agree. See markups of ITAAC Tabie 2.7.26-1.

2.7.26 LIGHTING SYSTEM

Design Description

The Lighting System is a non-safety-related system that is used to provide illumination ~~at locations~~ in the plant and on the plant site. The Lighting System has a normal lighting system, a security lighting system, and an emergency lighting system.

The normal lighting system provides general illumination ~~at locations~~ in the plant.

The security lighting system provides illumination in isolation zones and outdoor areas within the plant protected perimeter. The security lighting system is powered from the permanent non-safety buses.

Class 1E

The Emergency Lighting System consists of conventional AC fixtures fed from Class 1E AC power sources and DC self contained battery operated lighting units. Class 1E DC self contained battery operated lighting units are provided with rechargeable batteries with a minimum 8 hour capacity. Class 1E DC self contained battery operated lighting units are supplied AC power from the same power source as the normal lighting system in the area in which they are located.

The emergency lighting system provides illumination in the vital areas that include the main control room (MCR), the technical support center, the operations support center, the remote shutdown room, and the stairway which provides access from the MCR to the remote shutdown room.

Emergency lighting in the MCR is provided such that at least two circuits of lighting fixtures are powered from different Class 1E Divisions. The emergency lighting in the MCR maintains minimum illumination levels in the MCR during emergency conditions including station blackout. The emergency lighting installations which serve the MCR are designed to remain operational following a design basis earthquake.

Lighting circuits which are connected to a Class 1E power source are treated as associated [redacted] circuits. ← ADD INSERT A

Class 1E equipment is classified as Seismic Category I.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.26-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Lighting System.

INSERT A:

Independence is maintained between Class 1E Divisions and between Class 1E Divisions and non-Class 1E equipment.

Class 1E or associated [redacted] lighting distribution system equipment is identified according to its Class 1E Division. Class 1E or associated [redacted] lighting distribution system equipment is located in Seismic Category I structures and in its respective Divisional areas.

Class 1E or associated [redacted] lighting system cables and raceways are identified according to their Class 1E Division. Class 1E or associated [redacted] lighting system cables are routed in Seismic Category I structures and in their respective Divisional raceways.

TABLE 2.7.26-1

LIGHTING SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

ADD INSERT B

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the Lighting System is as described in the Design Description (Section 2.7.26).	1. Inspection of the as-built Lighting System will be conducted.	1. For the Lighting System described in the Design Description (Section 2.7.26), the as-built Lighting System conforms with the Basic Configuration.
3 ~~2.~~ The security lighting system is powered from the permanent non-safety buses.	3 ~~2.~~ Testing will be performed on the security lighting by providing a test signal in the permanent non-safety buses.	3 ~~2.~~ Within the security lighting system, a test signal exists at the equipment powered by the permanent non-safety bus under test.
4 ~~3.~~ The emergency lighting system provides illumination in the vital areas that include the MCR, the technical support center, the operations support center, the remote shutdown room, and the stairway which provides access from the MCR to the remote shutdown room.	4 ~~3.~~ Inspection of the MCR, the technical support center, the operations support center, the remote shutdown room, and the stairway which provides access from the MCR to the remote shutdown room will be performed.	4 ~~3.~~ Emergency lighting is installed in the MCR, the technical support center, the operations support center, the remote shutdown room, and the stairway which provides access from the MCR to the remote shutdown room and emergency lighting provides illumination levels greater than or equal to 10 foot-candles in the MCR, technical support center, operations support center, and the remote shutdown panel room. Emergency lighting provides an illumination level greater than or equal to 2 foot candles in the stairway which provides access from the MCR to the remote shutdown room.

INSERT B:

2. The Security Lighting System provides illumination in isolation zones and outdoor areas within the plant protected perimeter.	2. Inspection and testing of illumination ~~of~~ in isolation zones and outdoor areas within the plant protected perimeter will be performed.	2. Security lighting is installed in isolation zones and outdoor areas within the plant protected perimeter. Security lighting provides illumination levels greater than 0.2 foot-candles when measured horizontally at ground level in these areas.

TABLE 2.7.26-1 (Continued)

LIGHTING SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5 ~~4.~~ Class 1E DC self contained battery operated lighting units are provided with rechargeable batteries with a minimum 8 hour capacity. Class 1E DC self contained battery operated lighting units are supplied AC power from the same power source as the normal lighting system in the area in which they are located.	5.a) ~~4.~~ Inspection of the as-built Class 1E DC self contained battery operated lighting units will be conducted. 5.b) Testing will be conducted by ~~provided~~ providing a test signal on electrical divisions that supply power to the normal lighting system.	5.a) ~~4.~~ Class 1E DC self contained battery operated lighting units are provided with rechargeable batteries with a minimum 8 hour capacity. 5.b) Class 1E DC self contained battery operated lighting units are supplied AC power from the same power source as the normal lighting system in the area in which they are located. Class 1E DC self contained battery operated lighting units are turned on when the normal lighting system in the area in which they are located is lost.
6 ~~5.~~ Emergency lighting in the MCR is provided such that at least two circuits of lighting fixtures are powered from different Class 1E Divisions.	6 ~~5.~~ Testing will be performed on the emergency lighting system in the MCR by providing a test signal in only one Class 1E Division at a time.	6 ~~5.~~ Within the MCR emergency lighting system, a test signal exists only at the equipment powered from the Class 1E Division under test.
7 ~~6.~~ The emergency lighting in the MCR maintains minimum illumination levels in the MCR during emergency conditions including station blackout.	7 ~~6.~~ Testing of the emergency lighting system will be performed under simulated station blackout conditions.	7 ~~6.~~ Under simulated station blackout conditions, the emergency lighting system in the MCR maintains illumination levels greater than or equal to 10 foot-candles.
8 ~~7.~~ Lighting circuits which are connected to a Class 1E power source are treated as associated ~~Class 1E~~ circuits.	8 ~~7.~~ Inspection of the associated ~~Class 1E~~ lighting circuits will be conducted.	8 ~~7.~~ The as-built associated ~~Class 1E~~ lighting circuits are identified as associated ~~Class 1E~~ circuits.

← ADD INSERT C

INSERT C:

9. Independence is maintained between Class 1E Divisions and between Class 1E Divisions and non-Class 1E equipment.	9.a) Testing on the Lighting System will be conducted by providing a test signal in only one Class 1E Division at a time.	9.a) A test signal exists only in the Class 1E Division under test in the Lighting System.
	9.b) Inspection of the as-built Class 1E Divisions in the Lighting System will be conducted.	9.b) In the Lighting System physical separation or electrical isolation exists between Class 1E Divisions. Physical separation or electrical isolation exists between these Class 1E Divisions and non-Class 1E equipment.
10. Class 1E or associated ~~Class 1E~~ lighting distribution system equipment is identified according to its Class 1E Division.	10. Inspection of the as-built Class 1E and associated ~~Class 1E~~ lighting distribution system equipment will be conducted.	10. The as-built Class 1E or associated ~~Class 1E~~ lighting distribution system equipment is identified according to its Class 1E Division.

INSERT C: (con't)

11. Class 1E or associated ~~Class 1E~~ lighting distribution system equipment is located in Seismic Category I structures and in its respective Divisional areas.	11. Inspection of the as-built Class 1E and associated ~~Class 1E~~ lighting distribution system equipment will be conducted.	11. The as-built Class 1E or associated ~~Class 1E~~ lighting distribution system equipment is located in Seismic Category I structures and in its respective Divisional areas.
12. Class 1E or associated ~~Class 1E~~ lighting system cables and raceways are identified according to their Class 1E Division.	12. Inspection of the as-built Class 1E and associated ~~Class 1E~~ lighting system cables and raceways will be conducted.	12. The as-built Class 1E or associated ~~Class 1E~~ lighting system cables and raceways are identified according to their Class 1E Division.
13. Class 1E or associated ~~Class 1E~~ lighting system cables are routed in Seismic Category I structures and in their respective Divisional raceways.	13. Inspection of the as-built Class 1E and associated ~~Class 1E~~ lighting system cables and raceways will be conducted.	13. The as-built Class 1E or associated ~~Class 1E~~ lighting system cables are routed in Seismic Category I structures and in their respective Divisional raceways.

The design of the plant lighting systems is in accordance with applicable industry standards for illumination fixtures, cables, grounding, penetrations, conduit, and controls.

All lighting fixtures and other components of the lighting system located in normally occupied areas or in areas containing safety equipment are supported so as to enhance the earthquake survivability of these components and to ensure, in particular, that they do not present a personnel or equipment hazard when subjected to a seismic loading of a design basis earthquake.

The normal lighting system is used to provide normal illumination under all plant operation, maintenance and test conditions. Table 9.5.3-1 summarizes typical illuminance ranges for normal lighting.

The security lighting system provides the illumination required to monitor isolation zones and all outdoor areas within the plant protected perimeter. The security lighting system complies with the intent of NUREG CR-1327.

The emergency lighting system is used to provide acceptable levels of illumination throughout the station and particularly in areas where emergency operations are performed, such as control rooms, battery rooms, containment, etc., upon loss of the normal lighting system.

Lighting circuits which are connected to a Class 1E power source are treated as associated ~~Class 1E~~ circuits.

ADD INSERT A

9.5.3.2 System Description

9.5.3.2.1 Normal Lighting System

The Normal Lighting System provides general illumination throughout the plant in the main control room and in accordance with illumination levels recommended by the Illuminating Engineering Society. Incandescent lighting is used in the Containment Building while incandescent, fluorescent and high intensity discharge lighting is provided in the remainder of the plant and on the plant site. Power for the Normal Lighting System is provided independently from the Normal Auxiliary Power System via dry-type transformers and lighting panelboards.

Indoor lighting is designed for continuous operation. Switching is by individual plant circuit breakers except in office areas. Outdoor lighting is controlled by photocells.

The normal lighting system is considered part of the plant permanent non-safety systems. As such, the normal lighting system is energized as long as power from an offsite power source or a standby non-safety source (Combustion Turbine) is available.

Emergency lighting in the main control room is provided such that at least two circuits of lighting fixtures are powered from different Class 1E divisions.

Normal system operation is not affected by the failure or unavailability of a single lighting transformer.

The circuits to the individual lighting fixtures are staggered as much as possible, with the staggered circuits fed from separate electrical divisions, to ensure some lighting is retained in a room in the event of a circuit failure.

9.5.3.2.2 Security Lighting System

The security lighting system is considered part of the permanent non-safety systems and is fed from the Alternate AC (AAC) Source (Combustion Turbine), which is located in a secure vital area for protection. Selected portions of the security lighting system essential to maintaining adequate plant protection are powered from a non-Class 1E battery power source.

The COL Applicant shall provide a security lighting system that will meet CCTV illumination requirements within camera viewing areas to permit prompt assessment of intrusion alarms.

The security lighting system is designed to provide a minimum illumination of 0.2 foot-candles when measured horizontally at ground level.

9.5.3.2.3 Emergency Lighting

Emergency lighting is located in vital areas throughout the plant as identified in Emergency Procedures and Hazards Analysis for safe-shutdown of the plant following an accident or hazard. Included in the vital areas will be the Control Room, Technical Support Center, Operations Support Center, the Remote Shutdown Panel Room, the stairway which provides access from the Control Room to the Remote Shutdown Panel room, Sample Room, Hydrogen Recombiner Rooms, Electrical System Areas, Main Steam Valve Houses, the Chemistry Labs, routes for personnel passage and egress, and other areas where operator access is required post-accident or hazard.

The emergency lighting system in the main control room is integrated with the normal lighting system. ~~and will be configured so that normal and emergency circuits will be staggered and fed from different safety divisions to ensure that lighting is retained in the event of a circuit failure.~~ The emergency lighting system in the main control room maintains ~~adequate~~ minimum illumination levels in the main control room during ~~all~~ emergency conditions, including station blackout. ~~The emergency lighting system in the main control room is powered from a Class 1E battery power source.~~

The emergency lighting installations which serve the main control room and other areas of the plant where safe shutdown operations may be performed are designed to remain functional during and after a design basis earthquake.

INSERT A: (Refer to page 9.5-48)

Independence is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E equipment.

The criteria for the physical identification and separation of lighting cables and circuits are in accordance with the criteria for physical identification and separation of Class 1E and non-Class 1E cables and circuits as discussed in Chapter 8, Electric Power. The criteria meet the intent of IEEE Standard 384 and Regulatory Guide 1.75.

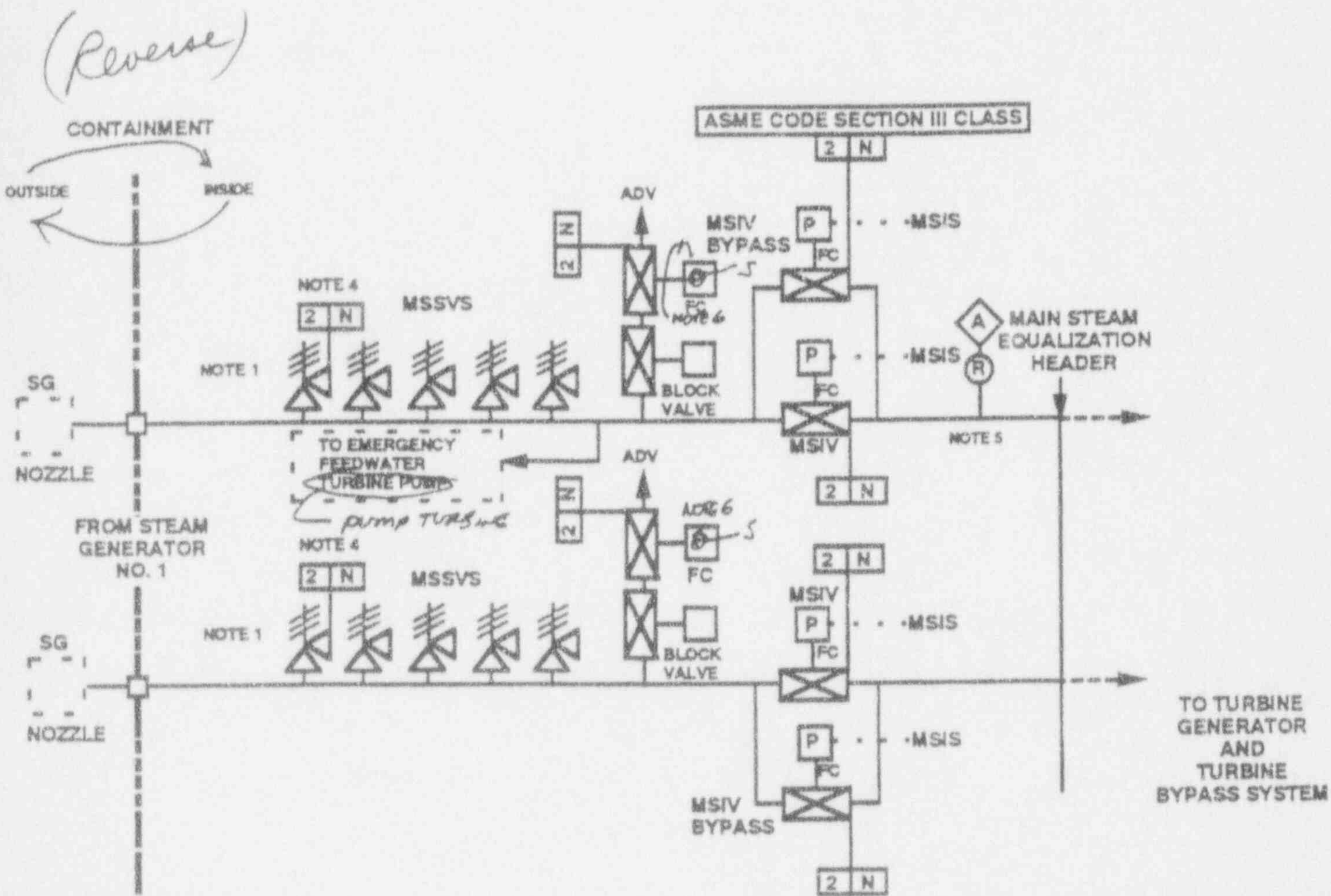
Class 1E or associated ~~Class 1E~~ lighting distribution system equipment is identified according to its Class 1E division. Class 1E or associated Class 1E lighting distribution system equipment is located in Seismic Category I structures and in its respective divisional areas.

Class 1E or associated ~~Class 1E~~ lighting system cables and raceways are identified according to their Class 1E division. Class 1E or associated ~~Class 1E~~ lighting system cables are routed in Seismic Category I structures and in their respective divisional raceways.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.8.2 (Main Steam Supply System

No.	Comments	Cat.	Resolution
1	See attached figure markup.	1	Agree. Will clarify the ADV operators. See markups.



NOTE:

1. NOT LESS THAN 5 MSSV WILL BE INSTALLED FOR EACH STEAMLINE.
2. ASME CODE SECTION III CLASS COMPONENTS SHOWN ON THE FIGURE ARE SAFETY-RELATED.
3. SAFETY-RELATED ELECTRICAL COMPONENTS AND EQUIPMENT SHOWN IN THIS FIGURE ARE CLASS 1E.
4. THE ASME CODE SECTION III CLASS BREAK OCCURS AT THE DISCHARGE OF EACH MSSV.
5. PRIMARY TO SECONDARY LEAKAGE MONITOR IS NOT SAFETY- RELATED

(ADD) 6. ADV IS MODULATING ELECTRICALLY OPERATED VALVE WITH INTERNAL SOLENOID PILOT

FIGURE 2.8.2-1
MAIN STEAM SUPPLY SYSTEM
(ARRANGEMENT SHOWN FOR ONE STEAM GENERATOR)

12-31-93

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.8.4 (Main Condenser Evacuation System

No.	Comments	Cat.	Resolution
1	See attached page for comments.	1	Agree. See markup of CESSAR-DC Section 10.4.2.2.

10.4.2 MAIN CONDENSER EVACUATION SYSTEM

10.4.2.1 Design Bases

The Main Condenser Evacuation System is designed to:

A. Remove air and other noncondensible gases from the condenser.

B. Maintain adequate condenser vacuum for proper turbine operation during startup and normal operation.

The system is designed to prevent uncontrolled release of radioactive material to the environment in accordance with 10CFR50, Appendix A, General Design Criteria (GDC) 60 and 64. System components conform to the requirements of Regulatory Guides 1.26 and 1.28 and Heat Exchange Institute (HEI) "Standards for Steam Surface Condensers." |

10.4.2.2 System Description

The Main Condenser Evacuation System is shown in Figure 10.4.2-1. |

and interconnecting piping

The Main Condenser Evacuation System consists of four skid mounted vacuum pumps which are used to pull a vacuum on the main condenser. The vacuum pumps are used for both hogging and holdings modes of condenser operation. The condenser evacuation system consists of four packaged/skid mounted vacuum pump units and interconnecting piping. Normally three vacuum pump units are in operation. The fourth pump unit is utilized as a maintenance spare. The vacuum pump units have two modes of operation, a hogging mode and a holding mode. The hogging mode is used to reduce the condenser pressure from atmospheric to approximately 5 to 10 in. Hg. absolute. The holding mode is used when these pressures are reached to reduce the condenser pressure to its operating value and then maintain the condenser operating pressure and provide deration capabilities during normal plant operations.

deaeration

The condenser evacuation system design provides a normally operating vacuum pump unit for each of the three condenser pressure zones and a common maintenance spare. Each operating vacuum pump unit is aligned to take suction from one of the three condenser pressure zones through two connections on the condenser shell. The normally operating vacuum pumps withdraw the air and noncondensible gases from the condenser shell, compress and discharge them through an individual line from the discharge nozzle of each vacuum pump unit to a common header routed to the unit vent.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.8.5 (Turbine Bypass System)

No.	Comments	Cat.	Resolution
1	Add acronym (SBCS) to the Abbreviation List of Section 1.3.	1	Agree. See markup of Section 1.3.
2 (4)	CESSAR Section 10.4.4.4 2nd paragraph: Replace "Turbine Bypass Control System (TBCS)" with "Steam Bypass Control System", and (TBCS) with (SBCS).	1	Agree. See markup of CESSAR-DC Section 10.4.4.4.

ABBREVIATION LIST (Continued)

Abbreviation	Meaning
RDS	Rapid Depressurization System
RDT	Reactor Drain Tank
RM	Refueling Machine
RPS	Reactor Protective System
RSP	Remote Shutdown Panel
RSR	Remote Shutdown Room
RSSH	Resin Sluce Slurry Header
RT	Reactor Trip
RTSG	Reactor Trip Switchgear
RV	Reactor Vessel
RVUH	Reactor Vessel Upper Head
RWBVS	Radwaste Building Ventilation System
SAFDL	Specified Acceptable Fuel Design Limit
SB	Shield Building
(insert) SBCS	Steam Bypass Control System
SBVS	Subsphere Building Ventilation System
SCS	Shutdown Cooling System
SDS	Safety Depressurization System
SFHM	Spent Fuel Handling Machine
SFP	Spent Fuel Pool
SFPCS	Spent Fuel Pool Cooling System
SG	Steam Generator
SGBS	Steam Generator Blowdown System
SGDT	Steam Generator Drain Tank
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SIS	Safety Injection System
SIT	Safety Injection Tank

I. The Turbine Bypass Valves are equipped with hand-wheels to permit manual operation at the valve location.

J. The Turbine Bypass Valves are arranged such that operation of any valves results in approximately equal blowdown from each steam generator.

10.4.4.3 Safety Evaluation

The valves in the turbine bypass system are designed to fail closed to prevent uncontrolled bypass of steam to the condenser. Should the bypass valves fail to open on command, the Main Steam safety valves provide main steam line overpressure protection. The power-operated atmospheric dump valves provide a means for controlled cooldown of the reactor. The Main Steam safety valves and power-operated atmospheric dump valves are described in Section 10.3.2.

Should the condenser not be available as a heat sink, an interlock will prevent opening, or if opened, will close the turbine bypass system valves. The Main Steam safety valves and power-operated atmospheric dump valves are used to control the load transient, if the bypass valves are disabled. Because the ASME Code Main Steam safety valves provide the ultimate overpressure protection for the steam generators, the turbine bypass system is defined as a control system and is designed without consideration for the special requirements applicable to protection systems. Failure of this system will have no detrimental effects on the Reactor Coolant System.

Operation of the turbine bypass system has no adverse effects on the environment since steam is bypassed to the condenser, the heat sink in use during normal operation.

This system is not required for the safe shutdown of the reactor and has no safety function.

10.4.4.4 Inspection and Testing Requirements

Preoperational and startup tests conform with the recommendations of NRC Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants."

A test will be conducted to verify opening of the Turbine Bypass Valves (TBVs) in response to a signal simulating turbine bypass from the Turbine Bypass Control System (TBCS). The objective of the test is to verify the function of the TBVs' response. Construction activities on the TBV, the TBCS, and their required support systems must be complete as a prerequisite. The test method consists of the application of a signal simulating turbine bypass to the controls of the TBVs, and recording of the opening of the TBVs as indicated by TBV valve stem travel indicator.

Steam

S

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.8.6 (Condensate and Feedwater Systems)

No.	Comments	Cat.	Resolution
1	Section 2.8.6 page 2 4th paragraph and Table 2.8.6-1 item 7 discussed ITAAC of MOVs with active safety function. According to CESSAR Table 3.9-15, none of the MOVs of Figure 2.8.6-1B belong in the "Active" category. Please provide rationale for the MOV discussion.	3	Agree. Will delete MOV statement. See markup of ITAAC 2.8.6.

SYSTEM 80+™

The two mechanical Divisions of the safety-related portions of the Feedwater System are physically separated.

Valves with response positions indicated on Figure 2.8.6-1B change position to that indicated on the Figure upon loss of motive power.

The MFIVs close on receipt of a main steam isolation signal (MSIS) or when remotely actuated from the control room.

Motor operated valves (MOVs) having an active safety function will open, or will close, or will open and also close, under differential pressure or fluid flow conditions and under temperature conditions.

Check valves shown on Figure 2.8.6-1B will open, or will close, or will open and also close, under system pressure, fluid flow conditions, or temperature conditions.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.8.6-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Condensate and Feedwater Systems.

TABLE 2.8.6-1 (Continued)

CONDENSATE AND FEEDWATER SYSTEMS
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. Motor operated valves (MOVs) having an active safety function will open, or will close, or will open and also close under differential pressure or fluid flow conditions and under temperature conditions.	7. Testing will be conducted to open, or close, or open and also close MOVs having an active safety function under preoperational differential pressure or fluid flow conditions and under temperature conditions.	7. Each MOV having an active safety function opens, or closes, or opens and also closes.
7, 8. Check valves shown on Figure 2.8.6-1B will open, or will close, or will open and also close under system pressure, fluid flow conditions, or temperature conditions.	7 8. Testing will be conducted to open, or close, or open and also close the check valves shown on Figure 2.8.6-1B under system preoperational pressure, fluid flow conditions, or temperature conditions.	7 8. Each check valve shown on Figure 2.8.6-1B opens, or closes, or opens and also closes.
8 9. Valves with response positions indicated on Figure 2.8.6-1B change position to that indicated on the Figure upon loss of motive power.	8 9. Testing of loss of motive power to these valves will be performed.	8 9. These valves change position to the position indicated on Figure 2.8.6-1B on loss of motive power.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.8.7 (Steam Generator Blowdown)

No.	Comments	Cat.	Resolution
1	CDM 8th paragraph and Table 2.8.7-1, item 4 should be supplemented with a statement that the valves also close upon receipt of a containment isolation actuation signal (CIAS) as described in CDM Table 2.4.5-2 (item 65/66) and CESSAR sections 10.4.8.1F and 10.4.8.3.	1	Disagree. Closure signals are in Section 2.4.5 per previous agreement. **NRC Staff concurs.**

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.8.8 (Emergency Feedwater System)

No.	Comments	Cat.	Resolution
1 (4)	Figure 2.8.8-1: Add symbols for alarms for the EFWST level instruments. They are more important than the temperature alarms.	1	Disagree. They are in the minimum inventory list in ITAAC 2.12.1 and, by previous agreement, don't need to be on the figure. **NRC Staff concurs.**

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.9.1 (Liquid Waste Management (mechanical aspects)

No.	Comments	Cat.	Resolution
1	Mark-up for design description and ITAAC is attached.	1	Agree. Inclusion of condensate demineralizer regenerate waste is a plant specific item
2	Editorial SSAR mark-up attached.	3	Agree with modification. Alternate rewrite provided.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.9.1 (Liquid Waste Management (Rad Prot aspects)) Page 1 of 1

No.	Comments	Cat.	Resolution
1	Markup for CESSAR Sections 11.1 and 11.2 are attached.	1	Agree. See markups.

2.9.1 LIQUID WASTE MANAGEMENT SYSTEM

Design Description

The Liquid Waste Management System (LWMS) is used to collect, segregate, store, process, sample, and monitor radioactive liquid waste. The LWMS is non-safety-related with the exception of the containment isolation valves and piping in between covered in Section 2.4.5.

The LWMS is located in the radwaste building.

The Basic Configuration of the LWMS is as shown on Figure 2.9.1-1.

The LWMS has four subsystems which process radioactive or potentially radioactive liquid waste. These four subsystems segregate liquid waste into high level waste, low level waste, laundry and hot shower/chemical waste, and the containment cooler condensate waste.

The high level waste subsystem has filters, demineralizers, provisions for batch sampling, and piping for recirculation of liquid waste for further processing.

The low level waste subsystem has filters, demineralizers, provisions for batch sampling, and piping for recirculation of liquid waste for further processing.

The laundry and hot shower/chemical waste subsystem has filters, demineralizers, provisions for batch sampling, and piping for ~~transfer of laundry and hot shower/chemical wastes to the low level waste subsystem~~ recirculation of liquid waste for further processing.

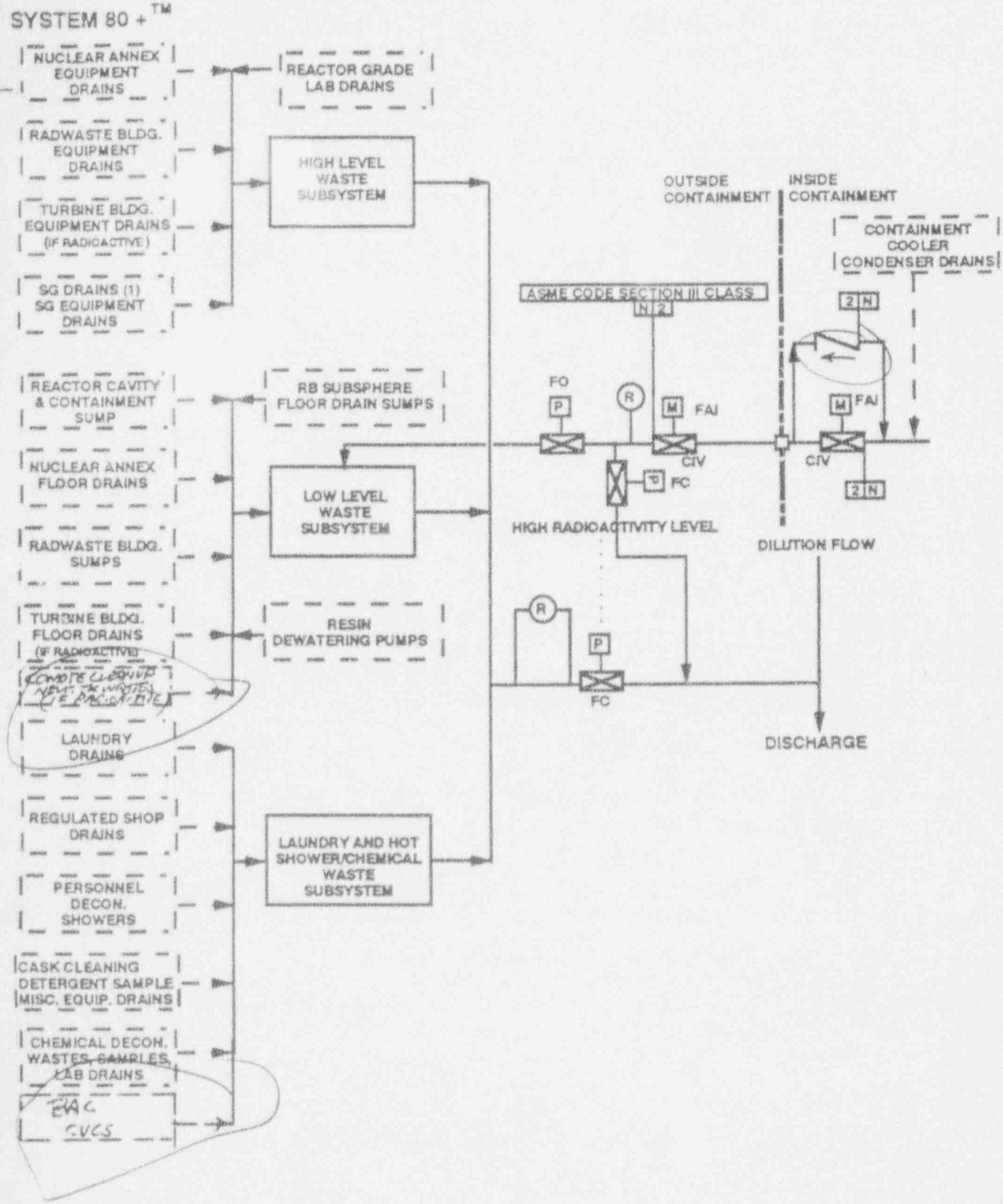
The containment cooler condensate subsystem has tanks to collect containment cooler condensate. The discharge from the tanks is monitored for radioactivity. Although not normally radioactive, this discharge can be diverted to the low level waste subsystem. The containment cooler condensate tank levels and discharge flow are also monitored by level and flow instrumentation.

The LWMS subsystems have collection and storage capacity to process waste volumes expected during normal operation and from anticipated operational occurrences.

Displays of the LWMS instrumentation shown on Figure 2.9.1-1 exist in the main control room (MCR) or can be retrieved there.

Controls exist in the MCR to open and close the power operated valves shown on Figure 2.9.1-1. X

The valves with the response position indicated on Figure 2.9.1-1 change~~s~~ position to X



NOTE
1. STEAM GENERATOR DRAINS MAY BE ROUTED TO EITHER HIGH LEVEL WASTE OR LOW LEVEL WASTE DEPENDING ON SAMPLE ANALYSIS

FIGURE 2.9.1-1
LIQUID WASTE MANAGEMENT SYSTEM

12-31-93

TABLE 2.9.1-1

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

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the Liquid Waste Management System (LWMS) is as shown on Figure 2.9.1-1.	1. Inspection of the as-built LWMS configuration will be conducted.	1. For the components and equipment shown on Figure 2.9.1-1, the as-built LWMS conforms with the Basic Configuration.
2. The ASME Code Section III LWMS components shown on Figure 2.9.1-1 retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A pressure test will be conducted on those components of the LWMS required to be pressure tested by ASME Code Section III.	2. The results of the pressure test of the ASME Code Section III components of the LWMS conform with the pressure testing acceptance criteria in ASME Code Section III.
3. The LWMS subsystems have collection and storage capacity to process waste volumes expected during normal operation and from anticipated operational occurrences.	3. Analysis of the as-built LWMS subsystems' processing capability will be performed.	3. An analysis exists which concludes the LWMS subsystems have collection and storage capacity to process waste volumes expected during normal operation and from anticipated operational occurrences.
4. Displays of the LWMS instrumentation shown on Figure 2.9.1-1 exist in the MCR or can be retrieved there.	4. Inspection for the existence or retrievability in the MCR of instrumentation displays will be performed.	4. Displays of the instrumentation shown on Figure 2.9.1-1 exist in the MCR or can be retrieved there.
5. Controls exist in the MCR to open and close the power operated valves shown on Figure 2.9.1-1.	5. Testing will be performed using the LWMS controls in the MCR.	5. LWMS controls in the MCR operate to open and close the power operated valves shown on Figure 2.9.1-1.

TABLE 2.9.1-1 (Continued)

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

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
X 6. The valves with the response position indicated on Figure 2.9.1-1 changes position to that indicated on the Figure upon loss of motive power.	6. Testing of loss of motive power to this valves will be performed.	6. ~~This~~ The valves change~~s~~ position to the position indicated on Figure 2.9.1-1 upon loss of motive power.
7. The radioactivity monitor provides a signal to terminate LWMS discharge when a specified radioactivity level is reached.	7. Testing of the as-built LWMS discharge controls will be performed using a signal which simulates radioactivity levels.	7. LWMS discharge is terminated in response to a signal simulating that the radioactivity level in the waste discharge line has reached a specified limit.

TABLE 11.1.1-3

TRITIUM ACTIVATION REACTIONS

	Reaction	Threshold Energy (MeV)	Cross Section [a]
1)	^{10}B (n, 2α)T	1.0	4.20(+1)mb[b]
2)	^{7}Li (n, nα)T	3.9	3.85(+2)mb
3)	^{6}Li (n, α) T	Thermal	9.45(+2) barns
4)	D (n, γ)T	Thermal	5.50(-1)mb
5)	^{11}B (n, T)^{9}Be	10.4	1.50(+1)mb

NOTES: (a) Threshold cross sections are from References 7 and 8. These are spectrum - averaged for neutrons of energy greater than indicated threshold energy.

(b) Number in parentheses denotes power of ten.

space

liquid effluent in the unrestricted area are within 10 CFR 20, Appendix B, Table ~~II~~ 2, Column 2 ~~maximum~~ permissible concentrations.

B. The system must contribute to meeting the performance design objectives in that it should not interfere with the normal station operation including anticipated operational occurrences.

The LWMS is a non-nuclear safety related system. It has no accident mitigation functions. The LWMS is designed in accordance with requirements in ANSI/ANS 55.2 and Regulatory Guide 1.143. This includes the following features:

1. The LWMS is designed with sufficient redundancy to tolerate a single major component failure and process radioactive liquid waste during normal operation, including anticipated occurrences.

2. The LWMS is designed with sufficient storage capacity and redundancy to accommodate an increase in demand during normal operation of the plant.

C. Releases of radioactive materials to the environment must be controlled and monitored in accordance with 10 CFR 50, Appendix A (General Design Criteria 60, 61 and 64).

The release of liquid waste requires an operator action. Prior to release through the plant discharge, radioactive liquid waste is sampled. The LWMS is also provided with a radiation monitor which monitors in the discharge line downstream from the Waste Monitor Tanks. In the event that the concentration of the discharge may exceed 10 CFR 20 limits, the radiation monitor would terminate the discharge. Section 11.5, Radiation Monitoring System, provides a detailed discussion regarding the radiation monitoring for the LWMS.

D. Accidental releases of radioactive materials from a single component of the LWMS must not result in offsite doses which exceed the guidelines of 10 CFR 20, Section 20.1301.

The LWMS and the Radwaste Building are designed so there is no liquid release to the environment due to a LWMS failure or leak. In addition, the LWMS is designed so that there is no possibility of gravity or syphon flow from the LWMS to the environment. This precludes an inadvertent release of radioactive liquid to the environment by this mechanism.

reasonably achievable offsite dose objectives. The dilution flow is provided by four centrifugal pumps. The pumps are sized such that any two pumps can provide a minimum of 100 CFS dilution flow to facilitate LWMS discharges.

11.2.2.2.8 Containment Cooler Condensate Tank

Two containment cooler condensate tanks are provided. The containment cooler condensate tank discharge will normally be routed to Industrial Waste Discharge since typically this stream has low activity. The capability to process this stream for processing as liquid waste will be provided.

The CCTs are fabricated of stainless steel.

11.2.2.2.9 Condensate Cleanup System Waste

The radioactive liquid waste water generated during regeneration of the condensate cleanup system polishers is collected in the neutralization tanks located in the Turbine Building. The contents of the neutralization tanks typically require no further processing and are discharged directly to the environment through a single designated discharge point. The neutralization tanks will be sampled prior to release.

From

X Separate piping is provided loop the neutralization tanks, which are located in the Turbine Building, to a common plant discharge header. A radiation monitor is provided downstream of the neutralization tank. Upon a receipt of radiation signal above the monitor setpoint, the discharge from the neutralization tanks will be terminated automatically. The operator would then sample the contents of the neutralization tanks and manually divert flow, as necessary based on the sampling results, to the Floor Drain Tank for processing in the low level waste subsystem of the LWMS prior to release to the environment.

A dike is provided around the neutralization tanks designed to be of sufficient height to contain maximum expected liquid inventory in these tanks. A dry sump is also provided to collect any spillage from the neutralization and route it to the LWMS for processing. Curbing and floor drains are provided in the regeneration area. This is discussed in Section 10.4.6.

11.2.2.2.10 Laundry and Hot Shower Tank

The laundry and hot shower and chemical waste subsystem (LHSCWS) is designed to provide the capability to terminate the discharge upon detection of high radiation in the discharge. The operator would then sample the detergent waste ~~collection~~ sample tank contents and manually divert flow to the low level subsystem for processing, as necessary, based on sampling results. Similarly, the condensate cooler tank discharge would be automatically terminated upon receipt of

Then either recirculate liquid waste to the collection tanks or LHSCWS for reprocessing by the LHS

11.2.6.1 Release Points

All discharges from the LWMS subsystems of detectable radioactivity are made through a common discharge header. The LWMS is designed with the capability to simultaneously discharge any or all of the radioactive liquid waste water from the LWMS subsystems' collection and/or waste monitor tanks and the condensate cleanup system neutralization tanks, as appropriate, through a single dedicated discharge point. The setpoints on each of the discharge lines will be determined and coordinated by the COL Applicant, as discussed in Section 11.5. The determination of the setpoints of the LWMS discharge radiation monitor, located downstream of the last possible point of input of radioactive liquid effluent from the respective LWMS collection or waste monitor tanks and radiation monitor located downstream of the condensate cleanup system neutralization tanks discharge, will be provided by the COL Applicant. The COL Applicant will develop the setpoints for radiation monitors on each of the discharge lines at the common plant discharge header for radioactive liquid effluents. Development of these setpoints is discussed in Section 11.5. All releases are monitored prior to dilution and discharge. Complete mixing of liquid waste with the dilution flow prior to discharge is assured by combining the two flows well upstream of the respective discharge point.

11.2.6.2 Dilution Factors

The dedicated liquid waste dilution flow can vary depending on the number of Liquid Waste Dilution Pumps that are operating. For the purpose of dose evaluations, an average dilution of 100 CFS is assumed for all release points for potentially radioactive liquid effluent. The 10 CFR 50, Appendix I analysis for the liquid pathways is based on a dilution flow of 100 cfs. This dilution flow may be comprised of dilution flow provided by the following sources as determined by the COL Applicant:

a. dilution pumps,
b. cooling tower blowdown, and/or
c. site specific dilution flow parameters (e.g., site specific hydrology);

but the discharge point is assumed to be located on a receiving water such that no significant recirculation occurs between the dilution flow intake and discharge.

The rate of radioactive liquid discharges will be based on the available dilution and concentrations of 10 CFR 20, Appendix B, Table ~~II~~.
2 Column 2

X

C. Results and Conclusions

The concentration of the liquid effluents at the plant discharge is shown in Table 11.2-5. The resultant concentration at the plant discharge is less than the EffluentS Concentrations specified in 10 CFR 20, Appendix B of Sections 20.1001 - 20.2402, Table 2, Column 2 ~~guidelines~~.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.9.2 (Gaseous Waste Management System - GWMS (Rad Prot aspects)) Page 1 of 1

No.	Comments	Cat.	Resolution
1	In Design Criteria #7 the word "terminal" should be changed to "terminate." A markup copy of the ITAAC is attached.	1	Agree. See attached markup.
2	A markup of the CESSAR Section 11.3 is attached. A consistent way to state the amount of fuel cladding defects needs to be establish, at least four different phrases were noted in the CESSAR, i.e., failed fuel rate, failed fuel, failed fuel defect, and failed fuel fraction, etc.	1	Agree. Consistent wording will be used. See attached markup.

TABLE 2.9.2-1 (Continued)

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

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. The radioactivity monitor provides a signal to terminat GWMS discharge when a specified radioactivity level is reached. terminate	7. Testing of the as-built GWMS discharge controls will be performed using a signal which simulates radioactivity levels.	7. GWMS discharge is terminated when the simulated radioactivity level in the discharge waste line reaches a specified limit.

11.3 GASEOUS WASTE MANAGEMENT SYSTEM

11.3.1 DESIGN BASES

11.3.1.1 Criteria and Evaluation

The GWMS is designed in accordance with the acceptance criteria defined in the Standard Review Plan, Section 11.3. The design criteria are the following:

A. Effluents normally released to unrestricted areas must meet the limiting requirements of 10 CFR 20 and meet the ALARA objectives of 10 CFR 50, Appendix I.

The GWMS continuously discharges effluent. Table 11.3-4 provides an estimate of the annual airborne effluent releases (Ci/yr) based on results from PWR-GALE. Assumptions used to calculate the annual release rate are discussed in Section 11.3.6. This estimated annual release rate is used to calculate the estimated annual dose to the maximum exposed individual. These results are listed in Table 11.3-5. This analysis assures that effluents during normal operation and anticipated operational occurrences meet 10 CFR 50, Appendix I objectives.

The GWMS is designed to ensure that normal releases to unrestricted areas are within 10 CFR 20, Appendix B of Sections 20.1001-20.2402, Table 2, Column 1 effluent concentrations based on the design basis source term. Section 11.3.8 provides a detailed discussion regarding the methodology used to calculate the concentration of the effluent at the Exclusion Area Boundary. The results of this analysis assure that the concentration of the effluent are within 10 CFR 20, Appendix B, Sections 20.1001-20.2402 Table 2, Column 1 ~~Effluent Concentrations~~ limits.

B. The system must contribute to meeting the performance design objectives in that it should not interfere with normal station operation including anticipated operational occurrences.

The GWMS is a non-nuclear safety related system. It has no accident mitigation functions. The GWMS is designed in accordance with ~~requirements~~ the guidance in ANSI/ANS 55.4, Regulatory Guide 1.143 and 1.140. This includes the following features:

1. The GWMS is designed to preclude a buildup of an explosive mixture of hydrogen and oxygen which could impact the operation of the plant.

2. The GWMS is designed with sufficient capacity and redundancy to accommodate an increase in demand during normal operation of the plant.

C. Releases of radioactive materials to the environment must be controlled and monitored in accordance with 10 CFR 50, Appendix A (General Design Criteria 60, 61 and 64).

20.1302

The GWMS is provided with radiation monitors which monitor the discharge from the charcoal adsorber beds upstream of the discharge to the Nuclear Annex Ventilation System. The GWMS discharge is automatically isolated if the discharge limit (10 CFR 20, Sections 20.1001-20.2402) will be exceeded. Section 11.5, Radiation Monitoring System, provides a detailed discussion regarding the radiation monitoring for the GWMS.

The COL Applicant will provide the operational setpoint for the termination of the gaseous waste management system discharge to the environment in the plant-specific offsite dose calculation manual (ODCM). This setpoint is based on the instantaneous dose rates in unrestricted areas due to the release of radioactive materials released via gaseous effluent. This setpoint ensures that the instantaneous dose rates offsite are less than the following:

Nobles Gases	500 mrem/yr total body; 3000 mrem/yr skin
Others	1500 mrem/yr to any organ

D. Accidental releases of radioactive materials from a single component of the GWMS must not result in offsite doses which exceed the guidelines of 10 CFR 20, Section 20.1301.

Section 11.3.7 provides a discussion of the analysis of a single component failure of the GWMS. The methodology used in this analysis is in accordance with Branch Technical ESTB-11-5 for the design basis source term. The results of this analysis confirm that the dose consequence of a single failure of a GWMS component is within the guidelines of 10 CFR 20, Section 20.1301.

dose limits

E. The system must also contribute to meeting the occupational exposure design objective by keeping operation and maintenance exposure ALARA.

The GWMS is designed in accordance with guidance provided in Regulatory Guide 8.8, ANSI/ANS-55.4, and Regulatory Guide 1.143 and 1.140. This ensures that the GWMS will meet ALARA objectives.

F. Protection will be provided to gaseous waste handling and treatment systems from the effects of an explosive mixture of hydrogen and oxygen in accordance with 10 CFR 50, Appendix A (General Design Criteria 3).

11.3.1.2 Codes and Standards

The GWMS is designed in accordance with the guidance of Regulatory Guide 1.143 from applicable regulatory positions (C.2, C.4, C.5 and C.6). These include:

A. The GWMS is designed and tested in accordance with regulatory position C.2 of Regulatory Guide 1.143.

   1. The GWMS is designed and tested to the codes and standards listed in Table 1 supplemented by regulatory positions C. 2.1.2 and C.4 ~~[illegible]~~

   2. Materials used for pressure retaining portions of the GWMS are designed in accordance with requirements specified in Section II of the ASME Boiler and Pressure Vessel Code. Materials used in the GWMS are compatible with the chemical, physical, and radioactive environment during normal and anticipated operating conditions. Malleable, wrought, or cast iron and plastics are not used in the GWMS.

      The GWMS is designed to preclude the buildup of an explosive mixture of hydrogen and oxygen. Gas analyzers are provided to monitor the concentration of hydrogen and oxygen in the GWMS. Alarms are provided locally in the Nuclear Annex and in the main control room to high alarm on 1% oxygen concentration.

   3. The Nuclear Annex houses the charcoal adsorber beds, which delay the release of radioactive gaseous waste from GWMS. The foundations and walls of structures housing the GWMS are designed to meet the requirements specified in regulatory position C.5. The Nuclear Annex is designed as a seismic Category I building and is designed to withstand a plant Safe Shutdown Earthquake (SSE).

B. The GWMS is designed and tested in accordance with regulatory position C.4 of the Regulatory Guide 1.143.

   1. The GWMS is housed in the Nuclear Annex. The GWMS is designed to control leakage. In addition, sufficient space is provided to facilitate access, operation, inspection, testing, and maintenance to maintain personnel exposures ALARA in accordance with Regulatory Guide 8.8 guidelines.

   2. A quality assurance (QA) program will be applied with the provisions as specified in regulatory position C.6 of Regulatory Guide 1.143.

or dry the charcoal. A charcoal guard bed is provided upstream of the adsorber beds to protect the beds from contamination and excessive moisture. The guard bed can be bypassed or purged and dried with nitrogen or reloaded if contaminated.

11.3.2.2.2 Cooler Condenser

A cooler condenser provides reduced temperature and reduces the moisture content of the gases. The cooling water supply comes from the chilled water system. The condenser is designed to take inlet gas flow of 8 SCFM of saturated water vapor at the maximum design temperature and discharge gases at 45°F.

11.3.2.2.3 Piping and Valves

Drain lines and valves are sized and continuously sloped to minimize the potential for plugging. Valves are of the packless metal diaphragm type and have bellows sealed stems to minimize leakage. All loop seals vent to a controlled vent system and equipment drains are closed or provided with loop seals to limit the escape of radioactive gases. The GWMS consists of welded piping to the greatest extent practicable. Flanged joints are kept to a minimum.

11.3.3 SAFETY EVALUATION

The GWMS has no plant safe shutdown or accident mitigation function. It is demonstrated in Section 15.7 that accidental releases, when evaluated on a conservative basis, are not expected to exceed the dose limits of 10 CFR 20.

11.3.4 INSPECTION AND TESTING REQUIREMENTS

The GWMS is tested to leak rate limits specified in ANSI/ANS 55.4. The sum of the leak rates from all individual components located within a zone does not exceed the zone totals in ANSI/ANS 55.4.

11.3.5 INSTRUMENTATION REQUIREMENTS

Table 11.3-3 provides a list of instrumentation for the GWMS. Additionally hydrogen detectors are provided in compartments containing off-gas systems under pressure and where hydrogen leakage may occur. Detection of hydrogen causes the GWMS to automatically shutdown. Normally, upon reaching a high level setpoint, an alarm annunciates. Instrumentation in contact with process streams is designed to minimize the potential for explosion. Manual override capability of automatic controls is provided where necessary to maintain system operability. For the equipment operated manually, remote manual hand switches with status lights are provided for all frequently operated valves and components. See Section 11.5.1.2.2 for description of Radiation Monitoring Systems interfaces with the Main Control Room.

The accident is described as an unexpected and uncontrolled release of radioactive Xenon and Krypton gases from the GWMS resulting from an inadvertent bypass of the main decay portion of the charcoal adsorber beds. It is assumed to take as long as 2 hours to isolate or terminate the release.

11.3.7.2 Analysis of Effects and Consequences

A. Bases

The bases for the estimated maximum offsite concentration of the gaseous effluent resulting from a leak or failure of the GWMS are as follows:

Insert A

1. The design basis airborne effluent source term is based on ~~1% failed fuel rate~~ in accordance with the Standard Review Plan Branch Technical Position (BTP) ESTB 11-5. The BTP ESTB 11-5 method adds the accident induced charcoal unit bypass leakage to the source term for normal operation; both accident source contributions are calculated based on a ~~1% failed fuel rate~~ assumption.

   Insert B

2. In the absence of site specific meteorological data and site Exclusion Area Boundary (EAB) information, the short-term 2-hour accident atmospheric dispersion factor, corresponding to a distance of approximately 0.5 miles from the station vent, is assumed to be 1.00×10^{-3} s/m^3. This is consistent with the dilution factors provided in Section 2.3.

Insert C

3. The sum of total estimated annual airborne effluent releases and the expected airborne effluent releases associated with the zero minute decay case are calculated by PWR-GALE and are multiplied by an isotope specific multiplication factor. This multiplication factor is calculated by the division of the ~~1% failed fuel RCS equilibrium concentration~~, calculated using the Combustion Engineering DAMSAM computer code and presented in Table 11.1.1-9, by the RCS equilibrium concentration calculated using PWR-GALE presented in Table 11.1.1-2, for ~~each isotope.~~

   Insert D

4. For isotopes with a ~~1% failed fuel rate calculated concentration~~ which is less than PWR-GALE results, the PWR-GALE concentration is used for conservatism. It is assumed that differences in the methodology used to calculate the reactor coolant concentrations are responsible for any differences observed in isotopic concentrations.

5. Particulates and radioiodines are assumed to be removed by pretreatment, gas separation, and intermediate radwaste treatment equipment. Therefore, only the whole body dose is calculated in this analysis.

B. Methodology

Insert E

To calculate the release of noble gases from the GWMS, the source term is based on the output from the computer code DAMSAM computer code. This code is used to calculate the reactor coolant equilibrium concentration with continuous degassing based on ~~1% failed fuel fraction~~ in accordance with Standard Review Plan Section 11.3. The resulting reactor coolant equilibrium concentration is divided by the reactor coolant concentration determined by PWR-GALE, using NUREG-0017, Revision 1 methodology, to yield a multiplication factor for each isotope. The total release of gaseous effluent for the zero minute decay case is calculated using PWR-GALE with BTP ESTB 11-5 alterations. The zero minute decay case releases are added to the normal operation source term and the sum for each radionuclide is multiplied by the multiplication factor, the 2-hour accident atmospheric dispersion factor, the total body dose factor, and a conversion factor to calculate whole body dose.

The methodology used to calculate the dose consequences for a GWMS failure, which is consistent with BTP ESTB 11-5, is as follows:

$$D=\sum K(i)\times Q(i)\times\frac{X}{Q}\times 7.25$$

Where: D = whole body dose (mrem)

$K(i)$ = the total-body dose factor given in Table B-1 of Regulatory Guide 1.109 for the ith isotope (mrem-m^3/pCi/yr)

$Q(i)$ = the noble gas nuclide accident release rate for the ith isotope (Ci/yr for 2 hours)

$$Q(i)=[R(i)_{Norm}+R(i)_0]\times MF(i)$$

$R(i)_{norm}$ = annual estimated airborne release rate for normal operation (Ci/yr) (Table 11.3-4)

$R(i)_0$ = annual estimate airborne release rate for zero minute decay case (Ci/yr)

MF = Multiplication Factor

$$MF = \frac{RCS(i)_{DAMSAM}}{RCS(i)_{CALS}}$$

X/Q = short-term 2-hour accident atmospheric dispersion factor at EAB (sec/m^3) = 1.00×10^{-3} (Section 2.3)

7.25 = conversion factor for 2 hour release (pCi-yr^2/Ci-event-sec)

C. Results and Conclusions

The calculated whole body dose at the exclusion area boundary is 49.3 mrem which is within the 500 mrem acceptance criterion specified in Standard Review Plan Section 11.3.

11.3.8 CONCENTRATION OF NORMAL EFFLUENTS

The Gaseous Waste Management System (GWMS) processes gaseous waste through a charcoal delay system which holds up noble gases and allows them to decay prior to release. The concentration at the exclusion area boundary during normal operation, including anticipated operating occurrences, was analyzed to verify it is less than 10 CFR 20, Appendix B, Table II, Column 1.

11.3.8.1 Analysis of Effects and Consequences

A. Bases

The bases for the estimated concentration of effluent are as follows:

1. The GWMS continuously discharges at a uniform rate at the design basis source term.

2. The design basis airborne effluent source term is based on ~~1% failed fuel rate~~ in accordance with the Standard Review Plan Section 11.3. It is assumed that the Reactor Coolant System (RCS) is continuously degassed by the CVCS during normal operating conditions. The reactor coolant equilibrium concentration is calculated using the Combustion Engineering DAMSAM computer code and is presented in Table 11.1.1-9.

Insert F

3. In the absence of site specific meteorological data and site Exclusion Area Boundary (EAB) information, the long-term annual average atmospheric dispersion factor, corresponding to a distance of approximately 0.5 miles from the station vent, is assumed to be 7.2×10^{-5} s/m³. This is consistent with the dilution factors assumed in Section 11.3.6.3.

4. The total estimated annual airborne effluent releases are multiplied by an isotope specific multiplication factor. This multiplication factor is calculated by the division of the ~~1% failed fuel RCS equilibrium concentration~~, calculated by the Combustion Engineering DAMSAM computer code, by the RCS equilibrium concentration, calculated using PWR-GALE, presented in Table 11.1.1-2, for each isotope.

Insert G

For isotopes with a ~~1% failed fuel rate calculated concentration~~ which is less than PWR-GALE results, the PWR-GALE concentration is used for conservatism. It is assumed that differences in the methodology used to calculate the reactor coolant concentrations are responsible for any differences observed in isotopic concentrations.

Insert H

5. Since DAMSAM does not calculate the concentration of tritium, the maximum calculated concentration of 1.00 µCi/gm is assumed for the 1% failed fuel source term for conservatism.

6. Since DAMSAM does not calculate the concentration of corrosion products, the PWR-GALE numbers are used. The concentration of these radionuclides should not be affected by the fraction of fuel defects.

B. Methodology

Insert I

To calculate the concentration at the exclusion area boundary, the source term is based on the output from the computer code DAMSAM computer code. This code is used to calculate the reactor coolant equilibrium concentration with continuous degassing based on ~~1% failed fuel fraction~~ in accordance with Standard Review Plan Section 11.3. The resulting reactor coolant equilibrium concentration is divided by the reactor coolant concentration determined by PWR-GALE, using NUREG-0017, Revision 1 methodology, to yield a multiplication factor for each isotope. The total annual release rate of gaseous effluent is multiplied by the multiplication factor and the average atmospheric dispersion factor to calculate the annual average concentration of the gaseous effluent at the exclusion area boundary. This

Independent Review Comments

Responses

1. Agree. ITAAC 2.9.2 has been updated, as attached.

2. The Section 11.3 of the CESSAR-DC has been updated, as attached, to use a consistent terminology when referring to the source term used to assess compliance with 10CFR20, Section 20.1302 limits.

3a-c. Agree. Section 11.3 of the CESSAR-DC has been revised, as attached.

3d. Partial agreement. Section 11.3.1.1.C, 2nd paragraph, 5th line has been updated as follows:

Replace "s 20.1001 - 20.2402" with "20.1302".

3e-f. Agree. The Section 11.3 of the CESSAR-DC has been updated, as attached.

3g. Partial agreement. Section 11.3.8.1.C has been updated as follows:

Replace "Maximum Permissible Concentrations" with "effluent concentrations" and replace "guidelines" with limits".

Insert A: Resolution of Comment 2 on ITAAC 2.9.2, Section 11.3.7.A.1 of the CESSAR-DC, 2nd line

design basis RCS equilibrium concentration resulting from fission product leakage into the RCS based on failure of 1% of power producing fuel

Insert B: Resolution of Comment 2 on ITAAC 2.9.2, Section 11.2.7.A.2 of the CESSAR-DC, 7th line

the design basis RCS equilibrium concentration

Insert C: Resolution of Comment 2 on ITAAC 2.9.2, Section 11.2.7.A.3 of the CESSAR-DC the design basis RCS equilibrium concentration

Insert D: Resolution of Comment 2 on ITAAC 2.9.2, Section 11.2.7.A.4 of the CESSAR-DC, 1st line

the design basis RCS equilibrium concentration, calculated by the Combustion Engineering computer code DAMSAM,

Insert E: Resolution of Comment 2 on ITAAC 2.9.2, Section 11.3.7.B of the CESSAR-DC

the design basis RCS equilibrium concentration

Insert F: Resolution of Comment 2 on ITAAC 2.9.2, Section 11.3.8.1.A.2 of the CESSAR-DC

design basis RCS equilibrium concentration resulting from fission product leakage into the RCS based on failure of 1% of power producing fuel

Insert G: Resolution of Comment 2 on ITAAC 2.9.2, Section 11.3.8.1.A.4 of the CESSAR-DC, 4th line

design basis RCS equilibrium concentration

Insert H: Resolution of Comment 2 on ITAAC 2.9.2, Section 11.3.8.1.A.4 of the CESSAR-DC, 2nd paragraph, 1st line

design basis RCS equilibrium concentration

Insert I: Resolution of Comment 2 on ITAAC 2.9.2, Section 11.3.8.1.A.5 of the CESSAR-DC

design basis RCS equilibrium concentration

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.9.3 (Solid Waste Management System (Rad Prot Aspects))

No.	Comments	Cat.	Resolution
1 (3)	In addition to other editorial comments, the statement in Section 11.4.7, paragraph D, is totally in error when used in reference to a "restricted or unrestricted" area. This statement should be changed to conform with the requirements of 10 CFR Part 20.1301 concerning dose limits for the public. A markup for CESSAR Section 11.4.1 is attached.	1	Agree with NRC resolution regarding Para D (doesn't need to be changed). Will revise SAR to state conformance to 10CFR20 limits. See attached markup.

11.4 SOLID WASTE MANAGEMENT SYSTEM

The Solid Waste Management System (SWMS) is designed to protect the plant personnel, the general public, and the environment by providing a means to collect, segregate, store, process, sample, and monitor solid waste. The SWMS processes both wet solid active waste and dry active waste for shipment to a licensed burial site.

11.4.1 DESIGN BASES

11.4.1.1 Criteria and Evaluation

The SWMS is designed in accordance with the following Standard Review Plan Section 11.4 acceptance criteria:

A. Releases of radioactive materials to the environment must be controlled and monitored in accordance with 10 CFR 50, Appendix A (General Design Criteria 60 and 64).

The SWMS is designed so that liquids removed during the dewatering process of wet solid waste are routed back to the Liquid Waste Management System (LWMS) to be processed prior to release to the environment. Non-clogging wire screens, such as Johnson screens, are provided on the Resin Storage Tanks and shipping containers to prevent an inadvertent discharge of resin beads to environment via the LWMS.

The gases collected in the dry active waste processing area are discharged via the Radwaste Building Ventilation System to the unit vent. The dry solids compactor is provided with an air filtration system which includes a HEPA filter. A fan draws air through the HEPA filter and exhausts gases, generated by compaction, through the Radwaste Building Ventilation System where the exhaust is filtered prior to release to the environment. This filtration system prevents a possible unfiltered release of airborne contamination to the environment.

Both of the above discharge paths are provided with monitors discussed in detail in Section 11.5.

B. Effluents normally released to unrestricted areas must meet the limiting requirements of 10 CFR 20, Appendix B of Sections 20.1001-20.2401, Table 2, Column 2. X

The liquid and gaseous effluents released during normal operation and anticipated operational occurrences to unrestricted areas are released through the LWMS and the Radwaste Building Ventilation System, respectively. ~~Section 11.2 provides a detailed discussion confirming compliance~~ X
with 10 CFR 20 Appendix B of Sections 20-1001-20.2401, Table X
2, Columns 2 and ~~for releases from the LWMS to the environment.~~ respectively. In addition, Section 11.3 provides an estimate

Liquid and gaseous effluents arising from the operation of the SWMS must meet the limiting requirements of in Tables 11.4-2 and 11.4-3, respectively. Table 11.4-4 lists the estimated burial volume and activity estimates for the various solid waste types that will be shipped for disposal from the System 80+. Radionuclide specific activities for each waste type are provided in Table 11.4-5.

11.4.4 SAFETY EVALUATION

The SWMS has no safe shutdown or accident mitigation function. ~~Finally, accidental releases from this system, will not exceed the limits of 10 CFR 20, Sections 20.1001-20.2402 of Appendix B, Table 2, Column 2.~~ Accidental releases due to a major component failure or SWMS leak will be contained in the Radwaste Building.

INSERT "A"

11.4.5 INSPECTION AND TESTING REQUIREMENTS

A Process Control Program appropriate to assure that the SWMS is operating as intended is developed prior to fuel loading. Procedures for each phase of system operation including resin transfer and batching help ensure that design objectives are met. Emphasis is placed on verifying instrumentation and remote functions important to these design objectives.

11.4.6 INSTRUMENTATION REQUIREMENTS

Instrumentation and indications important to the Design Basis of the SWMS are as follows:

A. Level Indicators

High level indication will be provided to prevent overflow of tanks during fill and resin transfer/sluice operations. These indications will be read in the radwaste control room. Also, video observation of all fill processes is included.

Densitometers are provided on the spent resin storage tanks and used to verify correct resin-to-water ratio when a batch of bead resin is to be solidified.

B. Flow and Pressure Indicators

Pump discharge flow and suction metering as well as pump discharge pressure indication will be provided to properly control the bed transfer process.

C. Radiation Monitoring

Area radiation monitors will be provided as discussed in Section 11.5.

Insert "A"

Liquid and gaseous effluents arising from the operation of the SWMS must meet the limiting requirements of 10CFR20, Appendix B of Sections 20.1001-20.2402, Table 2, Columns 1 and 2, respectively. Accidental liquid waste releases arising from the operation of the SWMS will also not exceed the limiting requirements of 10CFR20, Appendix B of Sections 20.1001-20.2402, Table 2, Column 2.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.9.4 (Process and Effluent Rad Mon & Sampling Sys - PERMSS (Rad Prot Aspects))

No.	Comments	Cat.	Resolution
1	It is noted that the CESSAR refers to this system as the Radiation Monitoring System (RMS) and at times as the PERMSS. Consistency should be established. A markup of CESSAR Section 11.5 is attached.	1	Disagree. RMS in Section 11.5 of the CESSAR-DC refers to the radiation monitoring portion of the PERMSS not the entire system. **NRC Staff concurs.**
2	The first paragraph in the DD is an inaccurate description of the system's capabilities. The third paragraph is superfluous but gives a better description of the PERMASS/RMS capabilities. A markup of the DD/ITAAC is attached.	1	Disagree. The first paragraph is accurate. **NRC Staff concurs.**
3	ITAAC Design Commitment 8.b (channel separations) were not given any inspections/tests or acceptance criteria to meet.	1	Design Commitment 8.b will be deleted as agreed with NRC staff. See attached markup.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.9.4 (Process and Effluent Rad Mon & Sampling Sys - PERMSS (Rad Prot Aspects))

No.	Comments	Cat.	Resolution
4	Certain TSC (CESSAR Sect. 13.3.3.1.6) and EOF (CESSAR Sect. 13.3.3.2.5 monitors were not listed in the Table 2.9.4-2 of the DD/ITAAC. Also, the CVCS gas stripper "Effluent" monitor referenced in CESSAR Sections 9.3.4.5.5.1 and 14.2.12.1.20 is not listed in the Table. Recommend that all CESSAR referenced process, area, and effluent monitors be placed in the DD/ITAAC Table 2.9.4-2. A DD/ITAAC markup is provided.	1	Disagree. The TSC is supplied with ventilation via the control room ventilation system which is provided with a process radiation monitor that isolates the most contaminated intake. The TSC is also provided with an area radiation monitor, listed in Table 11.5-4, to monitor direct gamma radiation and provide visual and audible alarms, as necessary, to alert the operators to adverse radiation conditions during occupancy. The EOF ventilation system monitor is listed in Table 11.5-3 and shown in Table 2.9.4-2 under airborn radiation monitors. The monitors are listed in Table 2.9.4-2 **NRC Staff concurs.**
5	ITAAC #5 uses the word "exceeds" to determine if a monitor trips when it is supposed to. The monitor should trip when it "reaches" the set point, not some time after it exceeds that point. A markup of the ITAAC is attached.	1	Agree. See attached markup.

TABLE 2.9.4-1 (Continued)

PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.a) Independence is provided between Class 1E Divisions, and between Class 1E Divisions and non-Class 1E equipment, in the PERMSS.	8.a) Inspection of the as-installed Class 1E Divisions of the PERMSS will be performed.	8.a) Physical separation exists between Class 1E Divisions in the PERMSS. Physical separation exists between Class 1E Divisions and non-Class 1E equipment in the PERMSS.
~~8.b) Independence is also provided between Class 1E Channels and between Class 1E Channels and non-Class 1E equipment in the PERMSS.~~		

TABLE 2.9.4-1 (Continued)

PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Each safety-related area radiation monitor channel monitors the radiation level in its assigned area, and indicates its respective Main Control Room (MCR) alarm and local audible and visual alarm (if provided) when the radiation level ~~exceeds~~ reaches a preset level.	5. Testing of each channel of the safety-related area radiation monitors will be conducted using simulated input signals.	5. MCR and local alarms are initiated when the simulated radiation level ~~exceeds~~ reaches a preset limit.
6. The following PERMSS safety-related instrumentation shall be provided: a. control room intake radiation monitor (2/intake), b. high range containment area radiation monitor (2), c. containment atmosphere radiation monitor (particulate channel only), d. primary coolant loop radiation monitors (2).	6. Inspection of the as-built system will be conducted.	6. The as-built PERMSS conforms with the design description.
7. The PERMSS safety-related instrumentation (the control room intake radiation monitors, high range containment area radiation monitors, containment atmosphere radiation monitor (particulate channel), and the primary coolant loop radiation monitors) are classified Seismic Category I.	7. Seismic analyses of the as-built PERMSS safety-related instrumentation will be performed.	7. An analysis report exists which concludes that the PERMSS safety-related instrumentation (the control room intake radiation monitors, high range containment area radiation monitors, containment atmosphere radiation monitor (particulate channel), and the primary coolant loop radiation monitors) are classified Seismic Category I.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.10 (Technical Support Center - TSC (Rad Prot. Aspects))

No.	Comments	Cat.	Resolution
1 (3)	A markup of the CESSAR Section 13.3 is attached.	1	Agree. See attached markup.

13.3.3.1.6 Habitability

TSC personnel are protected from radiological hazards, including direct radiation and airborne radioactivity from in-plant sources under accident conditions, to the same degree as control room personnel, so far as the maximum permissible radiation exposure is concerned while the TSC is habitable. Applicable criteria are specified in 10 CFR 50 General Design Criterion 19, NRC Standard Review Plan 6.4, and NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.B.2.

To ensure adequate radiological protection of TSC personnel, radiation monitoring systems are provided in the TSC. These systems continuously indicate radiation dose rates and airborne radioactivity concentrations inside the TSC while it is in use during an emergency. These monitoring systems shall include local alarms with trip levels set to provide early warning to TSC personnel of adverse conditions that may affect the habitability of the TSC. Detectors are able to distinguish the presence or absence of radioiodines at concentrations as low as 10^{-7} microcuries/cc.

If the TSC becomes uninhabitable, the TSC plant management function can be performed in the control room. Reference Section 6.4 TSC for habitability details. Control Building HVAC is discussed in Section 9.4.1.

13.3.3.1.7 Communications

The TSC is the primary onsite communications center for the nuclear power plant during an emergency. It has reliable voice communications to the control room, the OSC, the emergency operations facility (EOF), and the NRC. The primary functions of this voice communication system are plant management communications and the immediate exchange of information on plant status and operations. Provisions for communications with State and local operations centers are provided in the TSC to provide early notification and recommendations to offsite authorities prior to activation of the EOF.

The TSC voice communications facilities includes means for reliable primary and backup communication. The TSC voice communications will include private telephones, commercial telephones, radio networks, and intercommunication systems as appropriate to accomplish the TSC functions during emergency operating conditions. The licensee provides a means for TSC telephone access to commercial telephone common-carrier services that may be susceptible to loss of power during emergencies. The licensee ensures that spare commercial telephone lines to the plant are available for use by the TSC during emergencies.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.12.1 (Main Control Room (System Aspect))

No.	Comments	Cat.	Resolution
1	It should be stated in the design description that I&C not listed in the table that are required for the safe operation of the plant are included in the individual system descriptions.	1	ABB-CE agrees with the first suggestion. See attached markup. However, the Class 1E designation is not appropriately applied to the ABB-CE minimum inventory items. The inventory items represent operating information and control. Whether or not such information and control capabilities are provided on 1E-qualified hardware is not significant to its users, and is not a human factors issue. Furthermore, fixed position displays (e.g., DIAS) may utilize both 1E and non-1E sensor inputs, so that such distinctions are not possible.

CE 80+ ITAAC Independent Review Comments

ITAAC No 2.12.1 (Main Control Room (Human Factors Aspect))

No.	Comments	Cat.	Resolution
1	The design description defines the functional organization of the MCR. The ITAAC do not address the functional organization of the MCR. Consider deleting the functional organization description as it may be premature to define it prior to design validation.	2	ABB-CE is willing to remove the functional organization statements from the MCR design description, if necessary, but this is not considered desirable. On the other hand, adding a corresponding Design Commitment to ITAAC is inappropriate, because objective acceptance criteria will be difficult or impossible to formulate. Leave as is. **NRC Staff concurs.**
2	Deleted		
3 (4)	Design Commitment 2, column 2, states that an availability verification inspection of the as-built MCR will be performed. The inspection is inadequately defined. Provide additional detail to ensure an inspection consistent with that described in SSAR Section 18.9.1.	1	In general, it is basic to ITAAC that details are located in Tier 2. As noted by the reviewer, specific details on the availability verification process are provided in SSAR section 18.9.1. Also, since human factors is Tier 2 "star" material (so that associated SSAR changes receive full NRC review), there is added assurance that availability verification will proceed as described in the SSAR. However, since the availability criteria are process-dependent, they are not sufficiently objective for ITAAC inclusion. The ITAAC should be left as-is. **NRC Staff concurs.**

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.12.1 (Main Control Room (Human Factors Aspect))

No.	Comments	Cat.	Resolution
4 (5)	Design commitment 3, column 3. Typ - Change the word "used" to "use".	3	Agree. See markup.
5 (6)	Design commitment 3, column 2 states that a suitability inspection against verification criteria will be performed. The inspection and criteria are inadequately defined. The suitability inspection and verification criteria should be defined consistent with SSAR Section 18.9.2 to ensure an adequate suitability verification is conducted.	1	Suitability criteria are identified in lower level documents as specified in the SSAR. Also, see MCR comment 3 (4), above. **NRC Staff concurs.**
6 (8)	Design commitment 4, column 2 states that testing and analysis will be performed against validation criteria. The validation criteria are not defined in any manner. It is not possible to assess the appropriateness and adequacy of the criteria. The validation criteria should be defined consistent with validation objectives described in SSAR Section 18.9.3.1.		Validation criteria are identified in lower level documents as specified in the SSAR. Also, see MCR comment 3 (4), above. **NRC Staff concurs.**
7 (9)	Design commitment 4, column 2 indicates that the validation facility will dynamically represent the operating characteristics and responses of the System 80+ design. It is not clear that interface dynamics are considered part of the System 80+ design. The validation facility should represent the MCR interface dynamics of the System 80+ design.		Agreed. "MCR interface characteristics" will be added to the dynamic features specified. See attached markup.

2.12.1 MAIN CONTROL ROOM [1]

Design Description

The Main Control Room (MCR) permits execution of MCR tasks performed by MCR operators to operate the plant and maintain plant safety. The MCR provides suitable workspace and environment for continuous occupancy and use by MCR operators when the MCR is used for Plant Control. The MCR makes available the annunciators, displays, and controls to operate the plant and maintain plant safety, including at least those annunciators, displays and controls identified in Table 2.12.1-1.

Other annunciators, displays, and controls for systems operation are described in the design descriptions for the respective systems.

The Basic Configuration of the MCR is as shown on Figure 2.12.1-1.

The MCR contains the master control console, the auxiliary console, the safety console, the control room supervisor (CRS) Console, administrative support facilities, and the integrated process status overview (IPSO).

Control panels with Class 1E instrumentation are classified Seismic Category I.

The MCR is located in the nuclear annex within fire and ventilation isolation boundaries.

MCR consoles are organized functionally according to the following:

Master Control Console

Reactor Coolant System
Chemical & Volume Control System
Plant Monitoring & Control
Feedwater & Condensate Systems
Turbine Control

Auxiliary and Safety Consoles

Heating, Ventilation & Air Conditioning
Cooling Water Systems
Engineered Safety Features
Safety Monitoring
Secondary Auxiliaries
Switchyard
Electrical Distribution

The CRS console provides a workstation from which the CRS coordinates MCR operations. Administrative support facilities provide office workspace. The IPSO provides safety parameter display information at a fixed location that can be viewed from the MCR consoles and administrative support facilities.

Inspection, Test, Analyses, and Acceptance Criteria

Table 2.12.1-2 specifies the inspections, tests, analyses, and associated acceptance criteria for the Main Control Room.

[1] (Nuclear island structures, ventilation, fire protection, communications, lighting, and radiation protection are addressed in Sections 2.1.1, 2.7.17, 2.7.24, 2.7.25, 2.7.26, and 3.2 respectively.)

SYSTEM 80+™ TABLE 2.12.1-1

MCR MINIMUM INVENTORY OF FIXED POSITION ANNUNCIATORS, DISPLAYS AND CONTROLS

PARAMETER DESCRIPTION	Annunciators[1]	Displays	Controls
Offsite Bus voltage status	X		
120 VAC Vital load center voltage status	X	X	
125 VDC Vital load center voltage status	X	X	
24 KV Main Turbine Generator output breaker position	X	X	X
4.16 KV Class 1E bus breaker positions (supply & crossover)		X	X
4.16 KV Class 1E voltage status	X	X	
4.16 KV Diesel Generator output breaker position		X	X
4.16 KV Diesel Generator start control		X	X
4.16 KV Diesel Generator synchroscope		X	X
4.16 KV Reserve Aux Xfmr output voltage status	X		
480 VAC Class 1E voltage status	X	X	
Annulus ventilation control setpoint		X	X
Annulus ventilation damper position		X	X
Annulus ventilation fan on/off		X	X
Atmospheric dump valve position		X	X
CEA position	X		
CET temperature		X	
CIAS actuation	X		X
CIAS success monitor	X	X	
CCW HX inlet valve position		X	X
CCW HX outlet valve position		X	X
CCW HX outlet flow	X		
CCW pumps on/off		X	X
CCW surge tank level	X		
Containment hydrogen level (when analyzer is in operation)	X	X	
Containment pressure	X	X	
Containment radiation	X		
CSAS actuation	X		X
Containment Spray flow		X	

SYSTEM 80+™ TABLE 2.12.1-1 (Continued)

MCR MINIMUM INVENTORY OF FIXED POSITION ANNUNCIATORS, DISPLAYS AND CONTROLS

PARAMETER DESCRIPTION	Annunciators[1]	Displays	Controls
Containment Spray pump on/off		X	X
Containment Spray pump discharge valve position		X	X
Containment temperature	X	X	
DVI valve position		X	X
EFAS actuation	X		X
EFW flow control valve position		X	X
EFW header flow		X	
EFW motor-driven pump on/off		X	X
EFW pump suction pressure	X		
EFW steam-driven pump on/off		X	X
EFW-to-SG isolation valve position		X	X
EFW Storage Tank level	X		
INJECTION Hot Leg Inj valve position		X	X
IRWST level	X		
Main Control Room HVAC isolation dampers		X	X
Main Steam radiation (Area monitors & Line monitors)	X		
SG Main Steam safety valve position	X		
MSIS actuation	X	X	X
Nuclear Annex building ventilation radiation	X		
Pzr Backup Heaters on/off		X	X
Pzr Level	X	X	
Pzr Pressure	X	X	
Rapid Depressurization valve position		X	X
RCP on/off		X	X
RCS Cold Leg temperature		X	
RCS Hot Leg temperature		X	
RCS subcooling margin	X	X	
Reactor Building subsphere ventilation radiation	X		
Reactor Coolant gas vent valve position		X	X

SYSTEM 80+™ TABLE 2.12.1-1 (Continued)

MCR MINIMUM INVENTORY OF FIXED POSITION ANNUNCIATORS, DISPLAYS AND CONTROLS

PARAMETER DESCRIPTION	Annunciators[1]	Displays	Controls
Reactor power (NI)		X	
Reactor Trip (RPS)	X		X
Reactor Vessel level	X	X	
SCS flow (while SCS is in operation)	X	X	
SCS isolation valve position (& LTOP)	X	X	X
SCS HX Bypass Valve position		X	X
SCS HX CCW supply/isolation valve position		X	X
SCS HX/Bypass Inlet & Outlet temperature (when SCS is in operation)		X	
SCS HX outlet valve position		X	X
SCS pump on/off		X	X
SCS/CSS pump suction cross-connect valve position		X	X
SCS/CSS pump discharge cross-connect valve position		X	X
SIAS actuation	X		X
SI flow		X	
SI pump on/off		X	X
SI throttling isolation valve position		X	X
Spent Fuel Pool level	X		
Startup Rate (NI)		X	
CCWS ~~SSW~~ HX inlet isolation valve position		X	X
CCWS ~~SSW~~ HX outlet isolation valve position		X	X
CCWS ~~SSW~~ HX outlet flow	X		
SSW pump on/off		X	X
SG Blowdown sample radiation	X		
SG level	X	X	
SG pressure	X		
Vacuum Pump Activity	X		
Turbine Trip		X	X

[1] Annunciators are alarms and other alerting displays designed to direct operator attention.

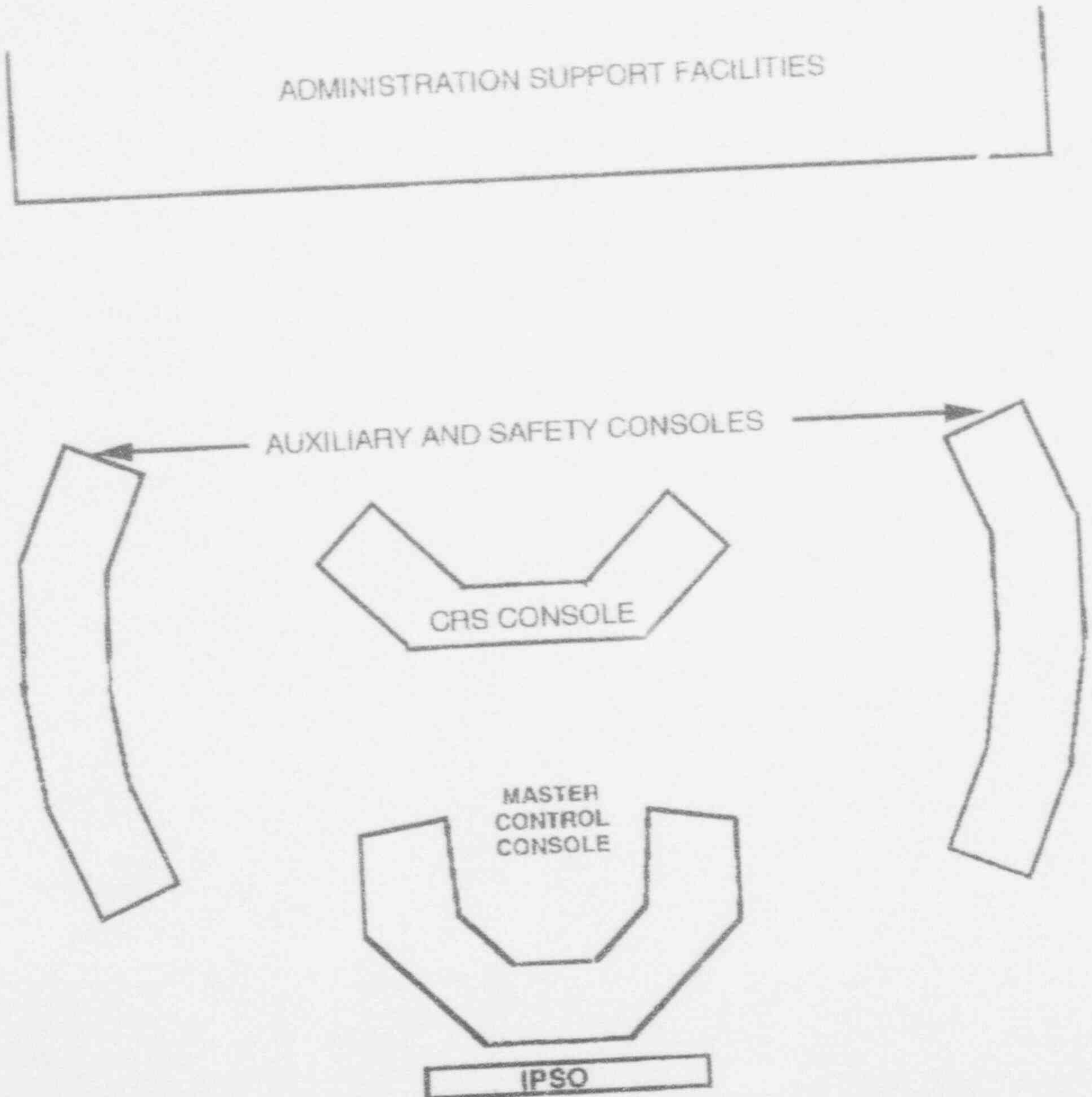


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FIGURE 2.12.1-1 MAIN CONTROL ROOM

TABLE 2.12.1-2

MAIN CONTROL ROOM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the MCR is as shown on Figure 2.12.1-1.	1. Inspection of the as-built MCR configuration will be conducted.	1. For the components and equipment shown on Figure 2.12.1-1, the as-built MCR conforms with the Basic Configuration.
2. The MCR makes available the annunciators, displays and controls identified in Table 2.12.1-1.	2. Human Factors Engineering (HFE) availability verification inspection of the as-built MCR will be performed.	2. The MCR makes available the annunciators, displays and controls identified in Table 2.12.1-1.
3. The MCR provides suitable workspace and environment for continuous occupancy and use by MCR operators when the MCR is used for plant control.	3. HFE suitability inspection against verification criteria will be performed.	3. The MCR workspace and environment are determined to be suitable for used by MCR operators.
4. The MCR permits execution of MCR tasks performed by MCR operators to operate the plant and maintain plant safety.	4. Testing and analysis against the validation criteria using a facility that physically represents the MCR configuration and dynamically represents the operating characteristics and responses of the System 80+ design will be performed.	4. The test and analysis results demonstrate validation of MCR task execution by MCR operators to operate the plant, and maintain plant safety.

THE MCR INTERFACE CHARACTERISTICS AND...

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.12.2 (Remote Shutdown Room (System Aspects)) Page 1 of 1

No.	Comments	Cat.	Resolution
1 (5)	It should be stated that the Remote Shutdown Panel is safety-related and Class 1E, or the portions of the RSR which are safety-related and Class 1E should be described. The 4th paragraph does not do this.	1	Disagree. See response to Main Control Room (System Aspects) comment 1, second paragraph regarding Class 1E designation.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.12.2 (Remote Shutdown Room (Human Factors Aspects))

No.	Comments	Cat.	Resolution
1	The design description states that "The RSR makes available the annunciators, displays, and controls to achieve and maintain prompt shutdown ... including at least those ... identified in Table 2.12.2-1." (Emphasis added) Design commitment 2 is to make available only those in Table 2.12.2-1. The design commitment should be revised to indicate that the RSR makes available the annunciators, etc. to achieve and maintain prompt shutdown etc. Identify Table 2.12.2-1 as a minimum list.	1	Table 2.12.2-1 for the RSR is a minimum inventory but is not presently labeled as such. This will be revised. See attached markup.
2	Design commitment 2, column 2, states that an availability verification inspection of the as-built RSR will be performed. The inspection is inadequately defined. Additional detail should be provided to ensure an inspection consistent with that described in SSAR Section 18.9.1.	1	In general, it is basic to ITAAC that details are located in Tier 2. As noted by the reviewer, specific details on the availability verification process are provided in SSAR section 18.9.1. Also, since human factors is Tier 2 "star" material (so that associated SSAR changes receive full NRC review), there is added assurance that availability verification will proceed as described in the SSAR. However, since the availability criteria are process-dependent, they are not sufficiently objective for ITAAC inclusion. The ITAAC should be left as-is. **NRC Staff concurs.**

CE 80+ ITAAC Independent Review Comments

ITAAC No. 2.12.2 (Remote Shutdown Room (Human Factors Aspects))

No.	Comments	Cat.	Resolution
3	Design commitment 3, column 2 states that a suitability inspection against verification criteria will be performed. The inspection and criteria are inadequately defined. The suitability inspection and verification criteria should be defined consistent with SSAR section 18.9.2 to ensure an adequate suitability verification is conducted.	1	Suitability criteria are identified in lower level documents as specified in the SSAR. Also, see RSR comment 2, above. **NRC Staff concurs.**
4 (5)	Design commitment 4, column 2 states that testing and analysis will be performed against validation criteria. The validation criteria are not defined in any manner. It is not possible to assess the appropriateness and adequacy of the criteria. The validation criteria should be defined consistent with validation objectives described in SSAR Section 18.9.3.1.	1	Validation criteria are identified in lower level documents as specified in the SSAR. Also, see RSR comment 2, above. **NRC Staff concurs.**
5 (6)	Design commitment 4, column 2 indicates that the validation facility will dynamically represent the operating characteristics and responses of the System 80+ design. It is not clear that interface dynamics are considered part of the System 80+ design. The validation facility should represent the MCR interface dynamics of the System 80+ design.	1	Agreed. "MCR interface characteristics" will be added to the dynamic features specified. See markup.

2.12.2 REMOTE SHUTDOWN ROOM [1]

Design Description

The Remote Shutdown Room (RSR) permits execution of RSR tasks performed by RSR operators to place and maintain the plant in a safe shutdown condition. The RSR provides suitable workspace and environment separate from the main control room (MCR) for use by RSR operators in the event that the MCR becomes uninhabitable. The RSR makes available the annunciators, displays, and controls to achieve and maintain prompt shutdown of the plant and maintain safe shutdown conditions including at least those annunciators, displays, and controls identified in Table 2.12.2-1. The RSR provides capability for RSR operators to perform RSR tasks to achieve subsequent cold shutdown of the plant.

The Basic Configuration of the RSR is as shown on Figure 2.12.2-1.

The RSR contains the Remote Shutdown Panel. The Remote Shutdown Panel provides a workstation from which RSR operators perform RSR operations.

Control panels with Class 1E instrumentation are classified Seismic Category I.

The RSR is located in the nuclear annex within fire and ventilation isolation boundaries.

Inspection, Test, Analyses, and Acceptance Criteria

Table 2.12.2-2 specifies the inspections, tests, analyses and acceptance criteria for the RSR.

CONTROLS EXIST IN THE RSR TO STOP THE REACTOR COOLANT PUMPS (RCPs), TRIP THE REACTOR, CONTROL THE EMERGENCY FEED WATER (EFW) STEAM-DRIVEN PUMP TURBINE SPEED, AND TO START AND STOP THOSE OTHER PUMPS, OPEN AND CLOSE THOSE VALVES, AND ENERGIZE OR DE-ENERGIZE THOSE PRESSURIZER HEATERS LISTED IN TABLE 2.12.2-1.

SUPPORTS DESIGN COMMITMENT #5 IN TABLE 2.12.2-2

[1] (Nuclear Island Structures, ventilation, fire protection, communications, lighting, and radiation protection are addressed in Sections 2.1.1, 2.7.17, 2.7.24, 2.7.25, 2.7.26, and 3.2)

SYSTEM 80+™

TABLE 2.12.2-1

MINIMUM INVENTORY OF
AVAILABLE RSR ANNUNCIATORS, DISPLAYS AND CONTROLS

PARAMETER DESCRIPTION	Annunciators[1]	Displays	Controls
Reactor Power (Neutron Logarithmic Power)		X	
RCS Cold Leg Temperature		X	
RCS Hot Leg Temperature		X	
PZR Level		X	
PZR Pressure		X	
Reactor Trip (RPS)		X	X
SG Level		X	
SG Pressure		X	
CVCS Charging Flow		X	
CVCS Charging Pressure		X	
Boric Acid Storage Tank Level		X	
IRWST Level		X	
EFW Steam-Driven Pump Suction Pressure	X	X	
EFW Motor-Driven Pump Suction Pressure	X	X	
EFW Steam-Driven Pump Discharge Pressure		X	
EFW Motor-Driven Pump Discharge Pressure		X	
EFW Steam-Driven Pump Turbine Inlet Pressure		X	
EFW Steam-Driven Pump Flow		X	
EFW Motor-Driven Pump Flow		X	
EFW Steam-Driven Pump Recirculation Flow		X	
EFW Motor-Driven Pump Recirculation Flow		X	
EFW Storage Tank Level	X	X	
EFW Steam-Driven Pump Turbine Speed		X	X
EFW Turbine Trip and Throttle (Stop) Valve Open/Close Position (Trip/Reset)	X	X	X
Ultimate Heat Sink Status		X	X
4.16 KV Diesel Generator Status (Emergency)		X	

SYSTEM 80+™ TABLE 2.12.2-1 (Continued)

MINIMUM INVENTORY OF
AVAILABLE RSR ANNUNCIATORS, DISPLAYS AND CONTROLS

PARAMETER DESCRIPTION	Annunciators[1]	Displays	Controls
Reactor Coolant Pump Trip		X	X
PZR Backup Heaters (Groups 1 & 2) On/Off		X	X
Atmospheric Dump Valve Position		X	X
ADV Block Valve Position		X	X
PZR Auxiliary Spray Valve Position		X	X
Reactor Coolant Gas Vent Valve Position		X	X
CVCS Charging Pump On/Off		X	X
Letdown Isolation Valve Position		X	X
RCP Seal Bleedoff Valve Position		X	X
MSIS Actuation		X	X
EFW Motor Driven Pump On/Off		X	X
EFW Steam Driven Pump On/Off		X	X
EFW Flow Control Valve Position		X	X
EFW-to-SG Isolation Valve Position		X	X
EFW Steam Supply Bypass Valve Position		X	X
EFW Steam Supply Isolation Valve Position		X	X
PZR Pressure Control Setpoint		X	X
SG Pressure Control Setpoint		X	X
SCS Suction Line Isolation Valve Interlock Status		X	
SCS HX/Bypass Inlet & Outlet Temperature (when SCS is in operation)		X	
SCS Flow	X	X	
SCS HX Bypass Valve Position		X	X
SCS Pumps On/Off		X	X
SIT Pressure		X	
SIT Vent Valve Position		X	X
SIT Isolation Valve Position		X	X

SYSTEM 80+™ TABLE 2.12.2-1 (Continued)

Minimum inventory of AVAILABLE RSR ANNUNCIATORS, DISPLAYS AND CONTROLS

PARAMETER DESCRIPTION	Annunciators[1]	Displays	Controls
SCS Isolation Valve Position		X	X
SCS HX Outlet Valve Position		X	X
SCS Warmup Bypass Valve Position		X	X
SI Flow[2]		X	
SI Discharge Header Pressure[2]		X	
SI Pump On/Off[2]		X	X
SI Throttling Isolation Valve Position[2]		X	X

[1] Annunciators are alarms and other alerting displays designed to direct operator attention.

[2] Indication for two discharge headers only

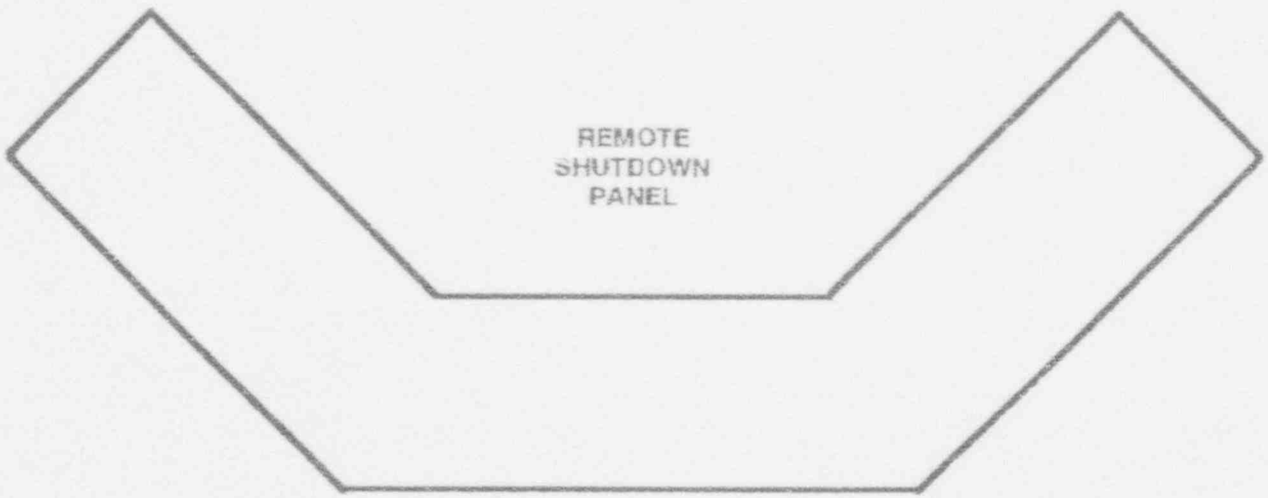

FIGURE 2.12.2-1 REMOTE SHUTDOWN ROOM

TABLE 2.12.2-2
REMOTE SHUTDOWN ROOM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the RSR is as shown on Figure 2.12.2-1.	1. Inspection of the as-built RSR configuration will be conducted.	1. For the components and equipment shown on Figure 2.12.2-1, the as-built RSR conforms with the Basic Configuration.
2. The RSR makes available the annunciators, displays and controls identified in Table 2.12.2-1.	2. Human Factors Engineering (HFE) availability verification inspection of the as-built RSR will be performed.	2.a) The as-built RSR makes available the annunciators, displays, and controls necessary to achieve and maintain prompt hot shutdown of the reactor.
		2.b) The as-built RSR provides capability for RSR operators to perform RSR tasks to achieve subsequent cold shutdown of the plant.
3. The RSR provides suitable workspace and environment for use by RSR operators.	3. HFE suitability inspection against verification criteria will be performed.	3. The RSR workspace and environment are determined to be suitable for use by RSR operators.
4. The RSR permits execution of RSR tasks performed by RSR operators to shutdown the plant and maintain safe shutdown conditions.	4. Testing and analysis against the validation criteria using a facility that physically represents the RSR configuration and dynamically represents ... ... THE RSR INTERFACE CHARACTERISTICS AND ... ... the operating characteristics of the System 80+ design will be performed.	4. The test and analysis results demonstrate validation of RSR task execution by RSR operators to achieve and maintain safe shutdown conditions.
5. Controls exist in the RSR to stop the RCPs, trip the reactor, control the EFW steam-driven pump turbine speed, and to start and stop those other pumps, open and close those valves, and energize or denenergize those pressurizer heaters listed in Table 2.12.2-1.	5. Testing will be performed using the controls in the RSR.	5. Controls in the RSR operate to stop the RCPs, trip the reactor, control the EFW steam-driven pump turbine speed, and to start and stop those other pumps, open and close those valves, and energize or de-energize those pressurizer heaters listed in Table 2.12.2-1.

Resolves 2.3.2 SCS Comment No. 3 (SRXB)

CE 80+ ITAAC Independent Review Comments

ITAAC No. 3.1 (Piping Design)

No.	Comments	Cat.	Resolution
1	Page 1: Add "Code, Section III," after ASME in the first sentence.	1	Agree with adding "Code". See markup. **NRC Staff concurs.**
2	Add a definition for Seismic Category II piping.	1	Agree. See markup.

Marked pages include changes previously agreed to with the staff.

3.1 PIPING DESIGN

Design Description

Code

The requirements for piping design in this section apply to ASME Class 1, 2 and 3 piping that is classified as Seismic Category I unless otherwise noted.

Piping classified as Seismic Category I is required to withstand the effects of a safe shutdown earthquake (SSE), maintain dimensional stability, and remain functional. Seismic Category I piping, structures, systems and components assure: (1) the integrity of the reactor coolant pressure boundary, and (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Seismic Category I piping is designed to meet the requirements of the ASME Code, Section III. Seismic Category II piping is designed and constructed such that the SSE will not cause failure in a manner to adversely affect a safety system or result in an incapaciting injury to a control room occupant. Applicable piping loads due to pressure, gravity, thermal expansion, seismic excitation, wind, tornado, fluid transients, thermal stratification, missiles, and postulated pipe breaks are considered in the piping analyses. Analytical methods and load combinations used for analysis of piping systems will be referenced or specified in the ASME Code certified stress report. Computer programs used for piping system dynamic analysis shall be benchmarked.

The as-built ASME Code Section III piping will be reconciled with the piping design requirements described herein. The as-built reconciliation will be documented in the as-built piping report.

Piping systems are designed to reduce the potential for effects of erosion/corrosion, and to reduce the potential for waterhammer and steam hammer. Piping system supports for Seismic Category I and II piping systems are designed to meet the requirements of the ASME Code Section III, Subsection NF. Pipe loads applied to attached equipment are shown to be less than the equipment allowable loads.

and fabrication processes   to ensure that the system is

For those piping systems using ferritic materials as permitted by the design specification, the materials will be chosen ~~[illegible]~~ not 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 material and fabrication process will be selected to reduce the possibility of cracking during service. Chemical, fabrication, handling, welding, and examination requirements that reduce the potential for cracking shall be employed.

Piping which does not perform a safety related function but whose structural failure or interaction could degrade the functioning of a Seismic Category I structure, system or component to an unacceptable safety level or could result in an incapacitating injury to an occupant of the control room is classified as Seismic Category II.

Piping systems classified as ASME Code Section III Class 1, 2, or 3 are designed to maintain dimensional stability and functional integrity under design loadings expected to be experienced during a 60 year design life.

Design of piping systems provides for clearances between adjacent piping, components, and other structures when the piping moves due to design static, dynamic, and thermal loadings.

The following piping systems are designed to meet leak-before-break (LBB) criteria:

Reactor coolant system hot leg piping, reactor coolant pump (RCP) suction piping and RCP discharge piping,

Surge line,

Main steam lines inside containment from the steam generator to the anchor at the containment penetrations.

Shutdown cooling lines inside containment from the reactor coolant system to the anchor at the containment penetration, and

Direct vessel injection lines inside containment from the reactor vessel to the safety injection tank and the anchor at the containment penetration.

LBB acceptance criteria are established and LBB evaluations are performed for each piping system designed to meet LBB criteria. For each piping system qualified for LBB, the as-built piping and materials will be reconciled with the bases for the LBB acceptance criteria.

and environmental and cracks

Structures, components, ~~equipment~~ and systems required for safe shutdown are protected from the dynamic effects of postulated pipe breaks in Seismic Category I and non-nuclear safety-related (NNS) piping systems where consideration of these dynamic effects is not eliminated by LBB. Design of features which protect these items consider, as applicable, pipe whip, water spray, jet impingement, flooding, compartment pressurization, and environmental conditions in the area where the piping is located.

Each postulated pipe crack and break shall be documented in a pipe break analysis report.

Structures, systems, and components that are required to be functional during and following an SSE are protected against the effects of spraying, flooding, pressure, and temperature due to postulated pipe breaks and cracks in Seismic Category I and NNS piping systems.

Inspections, Tests, Analyses and Acceptance Criteria

Table 3.1-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Piping Design.

TABLE 3.1-1

PIPING DESIGN
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The as-built piping is reconciled with the as-designed piping configurations.	1. A reconciliation analysis using the as designed and as-built information will be performed.	1. An as-built piping stress report exists and concludes that the as-built piping has been reconciled with the documents used for design. The as-built piping is reconciled with the piping design requirements described in the piping design description. 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.
2. Piping systems classified as ASME Code Section III Class 1, 2, or 3 are designed to maintain dimensional stability and functional integrity under design loadings expected to be experienced during a 60-year design life.	2. Inspection for the existence of ASME design reports will be performed.	2. ASME design reports for piping systems classified as ASME Code Section III Class 1, 2, or 3 exist and conclude that the design complies with the requirements of the ASME Code, Section III.
3. For each piping system qualified for LBB, the as-built piping and materials will be reconciled with the bases for the LBB acceptance criteria.	3. For each piping system qualified for LBB, an inspection of the LBB evaluation report will be performed.	3. A LBB evaluation report exists which documents that leak-before-break acceptance criteria are met by the as-built piping and piping materials.

TABLE 3.1-1 (Continued)

PIPING DESIGN
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Structures, components~~, equipment~~ and systems required for safe shutdown are protected from the dynamic and environmental effects of postulated pipe breaks and cracks in Seismic Category I and non-nuclear safety-related (NNS) piping systems where consideration of these dynamic effects is not eliminated by LBB. Each postulated pipe crack and break shall be documented in a pipe break analysis report.	4. For piping systems with postulated pipe breaks, an inspection of the pipe break analysis report will be performed. An inspection of the as-built high and moderate energy pipe break mitigation features will be performed.	4. A pipe break analysis report exists and concludes that Seismic Category I structures, systems, and components ~~classified as ASME Code Section III Class 1, 2, or 3~~ remain functional after postulated pipe breaks. The pipe break analysis report includes the results of inspections of high and moderate energy pipe break mitigation features (including spatial separation).

ASME Code — American Society of Mechanical Engineers Boiler and Pressure Vessel Code

SYSTEM 80+™

ABBREVIATION LIST

Abbreviation	Meaning
AAC	Alternate AC Source
A/C	Air Conditioning
ADV	Atmospheric Dump Valve
AFAS	Alternate Feedwater Actuation Signal
ALMS	Acoustic Leak Monitoring System
APC	Auxiliary Process Cabinet
APS	Alternate Protection System
ASME	American Society of Mechanical Engineers
ATM	Atmosphere
AVS	Annulus Ventilation System
BAC	Boric Acid Concentrator
CCCT	Containment Cooler Condensate Tank
CCS	Component Control Syst[illegible]
CCVS	Control Complex Ven[illegible]stem
CCW	Component Cooling W[illegible]
CCWHXSVS	CCW Heat Exchanger Structure Ventilation System
CCWLLSTAS	Component Cooling Water Low Level Surge Tank Actuation
CCWS	Component Cooling Water System
CEA	Control Element Assembly
CEACP	CEA Change Platform
CEAE	CEA Elevator
CEDM	Control Element Drive Mechanism
CEDMCS	Control Element Drive Mechanism Control System
CET	Core Exit Thermocouple
CFM	Cubic Feet Per Minute
CFR	Code of Federal Regulations
CFS	Cavity Flooding System
CGCS	Combustible Gas Control System

CE 80+ ITAAC Independent Review Comments

ITAAC No. 3.2 (Radiation Protection)

No.	Comments	Cat.	Resolution
1	The DD does not properly address what type of exposure the plant is designed for. A markup copy of the DD/ITAAC is attached.	1	Agree with NRC resolution that existing words are acceptable.

CE 80+ ITAAC Independent Review Comments

ITAAC No. 5.0 (Site Parameters)

No.	Comments	Cat.	Resolution
1	As this section does not have an ITAAC, revise the site parameters text to include the following (equivalent to Section 4.0) "An application for a combined license (COL) that references the System 80+ Certified Design must describe how the actual site location characteristics are bounded by the site parameters."	2	Agree with modification. See markup. Note: Attached Figures 5.0-1 and 5.0-2 fulfill a commitment in letter LD-93-178 to provide such figures scaled to 0.3 g.

5.0 SITE PARAMETERS

This section presents the parameters encompassed in the System 80+™ Certified Design.

An applicant selecting a site for the construction of the System 80+ Certified Design shall either describe how the actual site location characteristics are enveloped by the Site Parameters in this section, or demonstrate that those site characteristics not bounded by the Site Parameters do not invalidate the certified design commitments in Sections 1.0, 2.0, 3.0, and 4.0.

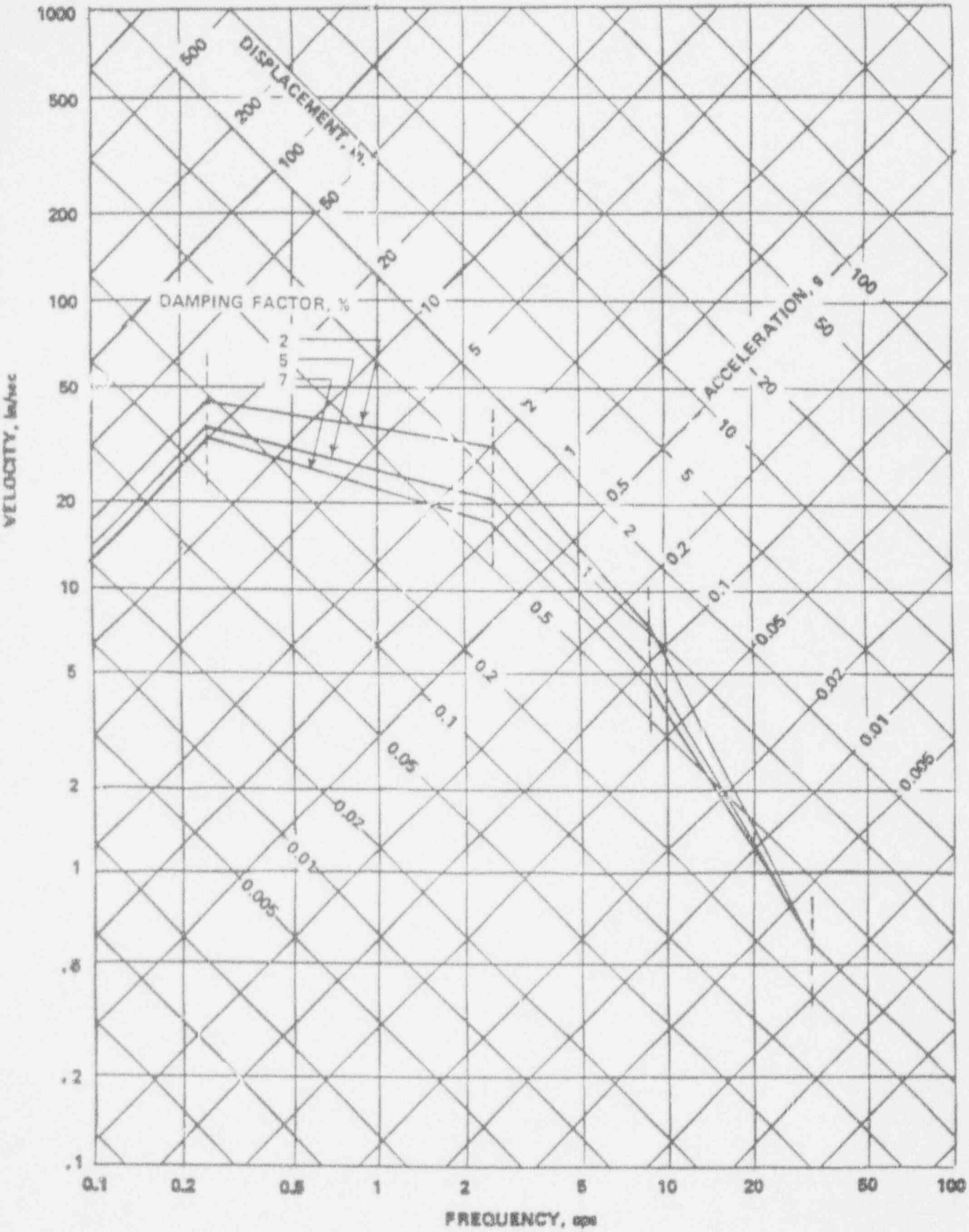

FIGURE 5.0-1
HORIZONTAL FREE FIELD SURFACE SPECTRA AT FINISHED PLANT GRADE LEVEL

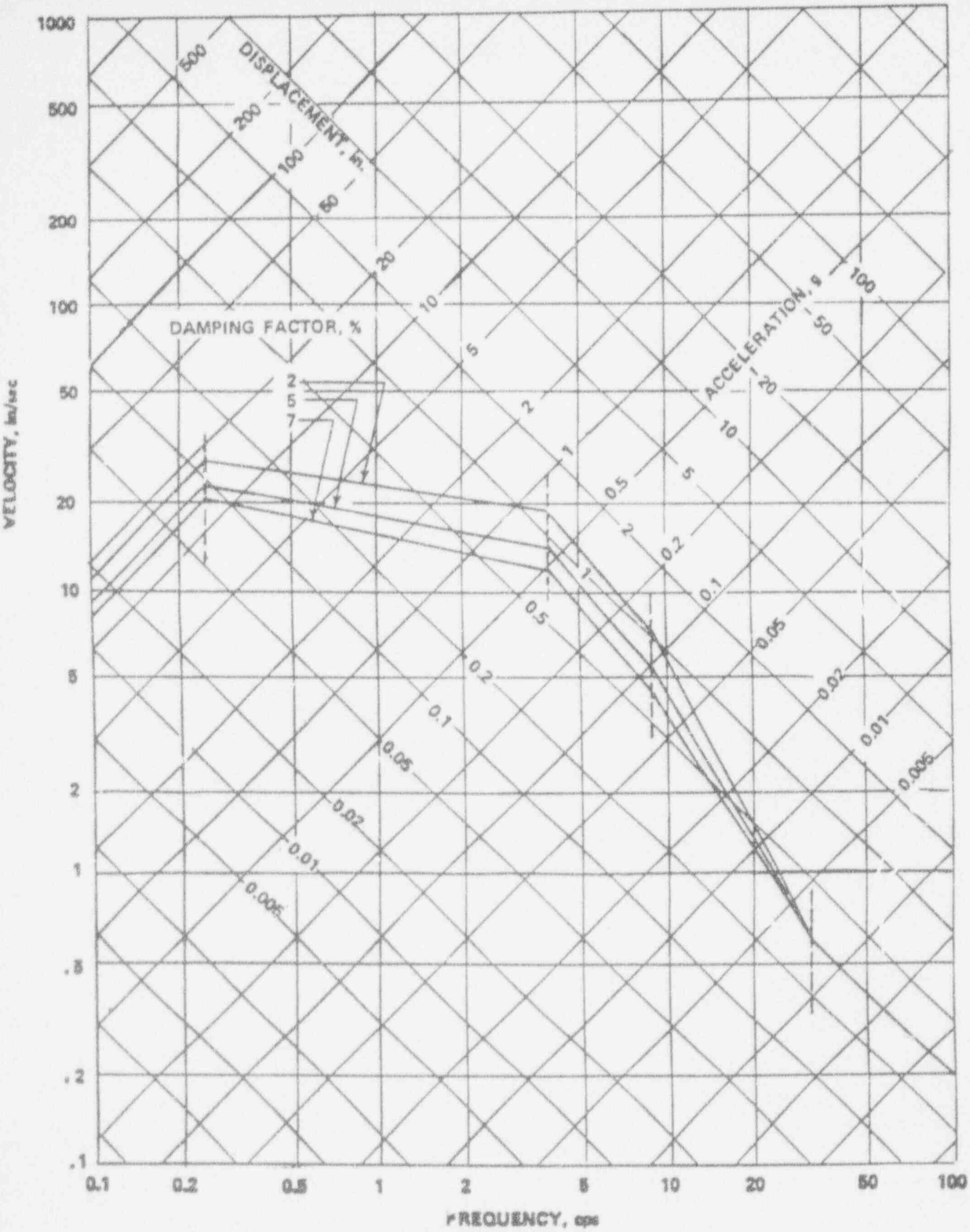

FIGURE 5.0-2
VERTICAL FREE FIELD SURFACE SPECTRA AT FINISHED PLANT GRADE LEVEL

These nine ITAAC/CESSAR-DC comments are not part of the NRC Independent Review. These comments were provided to ABB CE by R. Emch/J. Lee and were resolved independently. The attached markups represent the resolution agreed to with R. Emch and the NRC supporting staff.

ITAAC items discussed in 2/15/94 telephone call with John Rec, Robert Ellis, and George Hess of ABB-CE and Rich Emch and Jay Lee of NRC

1. Include recirc flow value of 4000 cfm in item 7 of ITAAC Table 2.7.17-1.

2. Include test and analysis in item 15 of ITAAC Table 2.7.17-1. Use results of leakage test in analysis to ensure control room doses are acceptable as described in SAR 6.4.

3. How to handle X/Q values for control room dose analysis? Maybe put in site parameters, or Tier 2*, or mention X/Qs in analysis statement in 2. above.

4. New Rad Pro DAC words in item 4.(c) of Table 3.2-1 to address ACRS concern. New words proposed attached.

5. Steam generator drain discharge monitor was deleted from Table 2.9.4-2. SER says this is an effluent monitor; if so needs to stay in ITAAC table.

6. ITAAC for OSC need to be addressed, probably with TSC in ITAAC 2.10.

7. CE did not specify bounding accident X/Q in site parameters the way GE did, do you want to reconsider?

8. CE used dilution factor of 1.49E-4 for BAST tank rupture analysis; put in ITAAC probably in site parameters.

9. Include verification analysis in ITAAC that effective spray volume of containment is 82% or greater.

Resolution

1. ABB CE agrees. See markup of Table 2.7.17-1.
2. ABB CE agrees. See markup of Table 2.7.17-1.
3. NRC withdrew comment. No changes required.
4. ABB CE agrees. See markup of Table 3.2-1 (Item 4.(c))
5. This drain path was eliminated in the design. No changes required. NRC concurs.
6. ABB CE agrees. See markup of ITAAC 2.10.
7. ABB CE declined to change the X/Q parameters. NRC concurs.
8. NRC agreed that there is sufficient coverage in CESSAR DC. A Tier 1 Commitment is not required.
9. ABB CE agreed to add a commitment related to containment spray coverage. See markup of ITAAC 2.4.6.

RESPONSE TO
ACRS COMMENTS ON
ABWR DAC

not a CE issue

1. Use of term "Vital Area":

   The term was retained to be consistent with the terminology used in NUREG 0737 item II.B.2 and the ABWR SSAR to ensure that it is understood that the areas referred to here are the same areas listed in the SSAR to meet the criteria in 0737.

   In response to the last ACRS comment on this, GE revised the DAC to define the vital areas each time the term is used. It is possible to delete the term from the DAC but there is a small risk that a future shielding designer may connect these areas to the vital areas listed in the SSAR.

2. Distinction between rooms that require infrequent access and those that seldom require access:

   The terminology was selected to be consistent with the SRP. We haven't experienced any confusion or problems with current plants, licensed under the same criteria.

   In response to the last ACRS comment on this, the parenthetical examples (such as ...) were added to clarify the intent.

   If we try to provide a specific frequency of entry to define infrequent we are going beyond the SRP and GE will probably want to negotiate what the frequency is.

   If the term "seldom" is the problem, we (with GE's agreement) could revise (c) to read;

   > For rooms where access is not anticipated to perform scheduled maintenance or surveillance (such as the backwash receiving tank room)...

TABLE 2.7.17-1 (Continued)

CONTROL COMPLEX VENTILATION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Item 1

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. Each MCR filtration unit and the TSC filtration unit remove particulate matter and iodine.	6. Testing and analysis will be performed on each MCR filtration unit and the TSC filtration unit to determine filter efficiencies.	6. The MCR and TSC filter efficiencies are greater than or equal to 95% for all forms of non-particulate iodine and greater than or equal to 99% for particulate matter greater than 0.3 micron.
7. The MCR is maintained at a positive pressure with respect to the adjacent areas.	7. Testing and analysis will be performed on the MCRACS.	7. The MCR is pressurized to at least 0.125 inches of water gauge relative to the adjacent areas with outside air supply no more than 2000 CFM and a recirculation flow of at least 4000 CFM.
8. The TSC can be pressurized with respect to the adjacent areas.	8. Testing will be performed on the TSC.	9. The TSC can be maintained at a positive pressure with respect to the adjacent areas.
9. The designated MCR filtration unit starts automatically and the MCR air conditioning unit starts or continues to operate, if running, on receipt of a safety injection actuation signal (SIAS) or a high radiation signal. In addition, the dampers in the MCR circulation lines and the bypass lines reposition to establish the flow path through the MCR filtration units.	9. Testing will be performed on the MCR filtration units, MCR air conditioning units, and dampers using a signal that simulates a safety injection actuation signal (SIAS). The testing will be repeated for a signal that simulates a high radiation signal.	9. The MCR filtration units and MCR air conditioning units start on receipt of a signal that simulates a SIAS, or a signal that simulates high radiation, and dampers reposition to establish the flow path through the MCR filtration units.

TABLE 2.7.17-1 (Continued)

ITEM 2

CONTROL COMPLEX VENTILATION SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
13.a) Displays of the CCVS instrumentation shown on Figure 2.7.17-1 exist in the MCR or can be retrieved there.	13.a) Inspection for the existence or retrieveability in the MCR of instrumentation displays will be performed.	13.a) Display of the instrumentation shown on Figure 2.7.17-1 exist in the MCR or can be retrieved there.
13.b) Controls exist in the MCR to start and stop the MCR filtration units and the TSC filtration unit, and to open and close the isolation dampers shown on Figures 2.7.17-1, 2.7.17-2 and 2.7.17-3.	13.b) Tests will be performed using the CCVS controls in the MCR.	13.b) CCVS controls in the MCR operate to start and stop the MCR filtration units and the TSC filtration unit and air conditioning unit, and to open and close the power operated isolation dampers shown on Figures 2.7.17-1, 2.7.17-2 and 2.7.17-3.
14. Components with response positions indicated on Figure 2.7.17-1 change position to that indicated on the figure upon loss of motive power.	14. Testing of loss of motive power to these components will be performed.	14. These components change position to the position indicated on Figure 2.7.17-1 on loss of motive power.
15. The leakage through MCRACS intake ductwork is less than the maximum allowable for the associated design.	15. The ductwork will be pressure tested for leakage. Analysis of the dose to the MCR operators will be performed.	15. The results of leak rate tests And analysis demonstrate that the leakage through ductwork is less than the maximum allowable for the associated design.
16. The fire dampers in the CCVS HVAC ductwork can close under design air flow conditions.	16. A type test will be performed to demonstrate that the dampers can close under design air flow conditions.	16. A test and analysis report exists that concludes the fire dampers can close under design air flow conditions.

TABLE 3.2-1 (Continued)

ITEM 4

RADIATION PROTECTION
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. (Continued)		4. (Continued)
		a) For normally occupied areas of the plant, (i.e., those areas requiring routine access to operate and maintain the plant), equilibrium concentrations of airborne radionuclides will be a small fraction of the Derived Air Concentrations in NRC dose regulations.
		b) For areas that require infrequent access (such as for non-routine equipment maintenance), the ventilation system shall be capable of reducing radioactive airborne concentrations to the Derived Air Concentration in NRC dose regulations during the periods that occupancy is required.
		c) For rooms ~~that seldom require~~ where access is not anticipated to perform scheduled maintenance, plant design shall provide features to reduce airborne contamination spread to other areas of lower contamination.

ITEM 6

2.10 TECHNICAL SUPPORT CENTER AND OPERATIONS SUPPORT CENTER

Design Description

The Technical Support Center (TSC) performs a non-safety-related function and is located adjacent to the main control room (MCR) in the nuclear annex. The TSC provides facilities for management and technical support to plant operations during emergency conditions.

The TSC is located less than or equal to two minutes walking time from the MCR.

The TSC has floor space of at least 75 square feet per person for a minimum of 25 persons.

The TSC has radiation detection equipment for monitoring radiation levels within the TSC when the TSC is in use.

The TSC has means for voice communication to the MCR, to on-site emergency support facilities, and to off-site via dedicated or commercial telephone networks.[1]

Displays of the information from the discrete indication and alarm system (DIAS) and the data processing system (DPS) exist in the TSC or can be retrieved there.[2]

The Operations Support Center (OSC) performs a non-safety related function and is located in the nuclear island structures. The OSC provides an assembly area separate from the MCR and TSC where operations support personnel can assemble in an emergency.

The OSC has equipment for voice communication with the MCR and the TSC.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.10-1 specifies the inspections, tests, analysis, and associated acceptance criteria for the Technical Support Center, and Operations Support Center.

[1] Communication Systems are addressed in Section 2.7.25.

[2] Display information from the DIAS and DPS is addressed in Section 2.5.3.

TABLE 2.10-1

TECHNICAL SUPPORT CENTER
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.a) The TSC is located less than or equal to two minutes walking time from the MCR.	1.a) A test of walking time from the TSC to the MCR will be performed.	1.a) The TSC can be reached in less than or equal to two minutes walking time from the MCR.
1.b) The TSC has floor space of at least 75 square feet per person for a minimum of 25 persons.	1.b) Inspection of the TSC will be performed.	1.b) Floor space of at least 1875 sq. ft. is provided in the TSC.
1.c) The TSC has radiation detection equipment for monitoring radiation levels within the TSC when the TSC is in use.	1.c) An inspection of the radioactivity detection equipment in the TSC will be performed.	1.c) Radiation detection equipment to monitor radiation levels within the TSC is available in the TSC.
1.d) The TSC has means for voice communications to the MCR, to on-site emergency support facilities, and to off-site via dedicated or commercial telephone networks.	1.d) An inspection of the TSC will be performed.	1.d) Communications equipment is installed, and voice transmission and reception are accomplished.
2. Displays of information from the DIAS and the DPS exist in the TSC or can be retrieved there.	2. Inspection for the existence or retrievability in the TSC of the information from the DIAS and the DPS will be performed.	2. Displays of information from the DIAS and the DPS exist in the TSC or can be retrieved there.
3. The OSC is located in the nuclear island structures.	3. Inspection of the location of the OSC will be performed.	3. The OSC is located in the nuclear island structures.
4. The OSC has equipment for voice communication with the MCR and the TSC.	4. Testing of the equipment for voice communication will be performed.	4 Communications equipment is installed and voice transmission and reception are accomplished.

 ITEM 9

2.4.6 CONTAINMENT SPRAY SYSTEM

Design Description

The Containment Spray System (CSS) is a safety-related system which removes heat and reduces the concentration of radionuclides released from the fuel from the Containment atmosphere and transfers the heat to the component cooling water system following events which increase Containment temperature and pressure. The CSS can also remove heat from the in-containment refueling water storage tank (IRWST).

The CSS is located in the reactor building subsphere and Containment.

The Basic Configuration of the CSS is as shown on Figure 2.4.6-1.

The CSS consists of two Divisions. Each CSS Division has a CSS pump, a CSS heat exchanger, valves, piping, spray headers, nozzles, controls, and instrumentation.

Each CSS Division has the heat removal capacity to cool and depressurize the containment atmosphere, such that containment design temperature and pressure are not exceeded following a loss of coolant accident (LOCA) or a main steam line break (MSLB).

Each CSS Division has the capacity to reduce the concentration of radioactive material in the containment atmosphere such that the design basis accident dose criteria are not exceeded.

The CSS limits the maximum flow in each Division.

The CSS pump and the Shutdown Cooling System (SCS) pump in the same Division are connected by piping and valves such that the SCS pump in a Division can perform the pumping function of the CSS pump in that Division. The piping and valves in the cross-connect line between the SCS pump suction and the CSS pump suction permit flow in either direction.

A flow recirculation line around each CSS pump provides a minimum flow recirculation path.

The CSS pumps can be flow tested during plant operation.

The ASME Code Section III Class for the CSS pressure retaining components shown on Figure 2.4.6-1 is as depicted on the Figure.

The safety-related equipment shown on Figure 2.4.6-1 is classified Seismic Category I.

CSS pressure retaining components shown on Figure 2.4.6-1, except the shell side of the heat exchangers, have a design pressure outside Containment of at least 900 psig.

Displays of the CSS Instrumentation shown on Figure 2.4.6-1 exist in the main control room (MCR) or can be retrieved there. Controls exist in the MCR to start and stop the CSS pumps, and to open and close those remote-operated valves shown on Figure 2.4.6-1. CSS alarms shown on Figure 2.4.6-1 are provided in the MCR.

Water is supplied to each CSS pump at a pressure greater than the pump's required net positive suction head (NPSH).

The Class 1E loads shown on Figure 2.4.6-1 are powered from their respective Class 1E Division. The CSS pump motor and the SCS pump motor in each Division are powered from different Class 1E buses in that same Division.

Independence is provided between Class 1E Divisions and between Class 1E Divisions and non-Class 1E equipment in the CSS.

The two mechanical Divisions of the CSS are physically separated.

The CSS pumps are started upon receipt of a containment spray actuation signal (CSAS), except when the CSAS is aligned to the SCS pump in the same Division. The isolation valves to the CSS spray headers and nozzles are opened on receipt of a containment spray actuation signal (CSAS).

Motor operated valves (MOVs) having an active safety function will open, or will close, or will open and also close under diffential pressure or fluid flow conditions, and under temperature conditions.

Check valves shown on Figure 2.4.6-1 will open, or will close, or will open and also close under system pressure, fluid flow conditions, or temperature conditions.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.4.6-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Containment Spray System.

TABLE 2.4.6-1

CONTAINMENT SPRAY SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. Each CSS Division has the capacity to reduce the concentration of radioactive material in the containment atmosphere such that the design basis accident dose criteria are not exceeded.	3. Inspection of the CSS spray headers will be performed.	3. Each CSS Division has spray headers and nozzles as follows: At least 168 noggles at plant elevation of at least 225 feet, at least 121 noggles at plant elevation of at least 197 feet, and at least 40 noggles at plant elevation of at least 141 feet.

(note to typist: insert and renumber other paragraphs)

Notes;

1. TUBE SIDES ARE ASME CODE SECTION III CLASS 2 AND SHELL (CCW) SIDES ARE ASME CODE SECTION III CLASS 3.
2. SAFETY-RELATED ELECTRICAL COMPONENTS AND EQUIPMENT SHOWN ON THIS FIGURE ARE CLASS 1E. ALARMS AND PRESSURE AND CURRENT INSTRUMENTS ARE NOT SAFETY-RELATED AND NOT CLASS 1E.
3. THE ASME CODE SECTION III CLASS 2 AND 3 PRESSURE RETAINING COMPONENTS SHOWN ARE SAFETY-RELATED

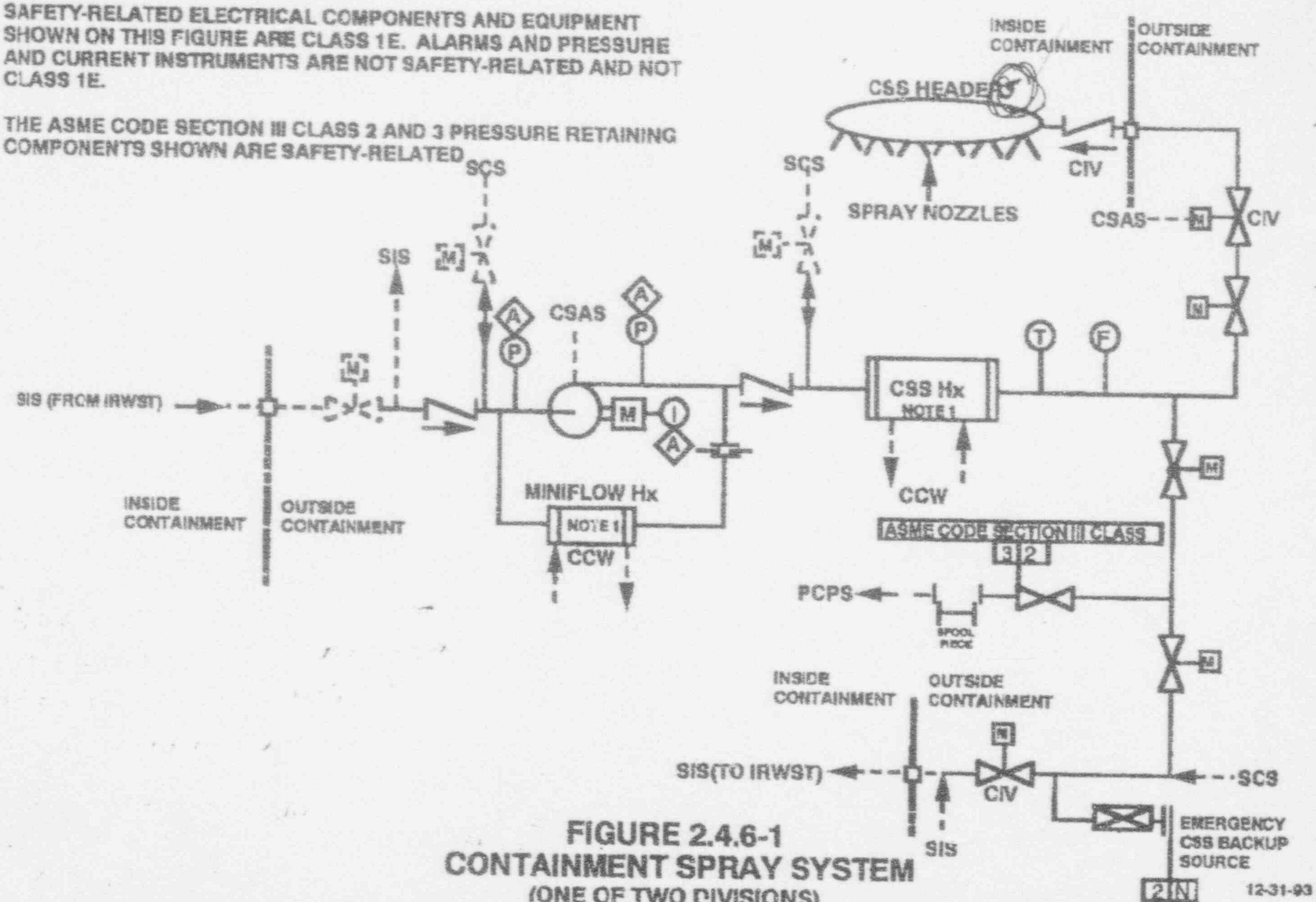

FIGURE 2.4.6-1
CONTAINMENT SPRAY SYSTEM
(ONE OF TWO DIVISIONS)