ABB
ASEA BROWN BOVERI

April 30, 1993
LD-93-071

Docket 52-002

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

SUBJECT: System 80+™ Submittal #1 Design Descriptions and ITAAC

REFERENCE NRC Letter, D. Crutchfield to R. Matzie, dated 2/10/93

Dear Sirs:

Attached is Submittal #1 of the System 80+ Design Descriptions and associated ITAAC (Inspections, Tests, Analyses and Acceptance Criteria) which are submitted for review and approval.

ABB-CE has initiated an Integrated Review of the CESSAR-DC and Design Descriptions/ITAAC to ensure consistency among and within these documents. It is possible that changes to the attached material may be necessary should the review uncover any inconsistencies. It is our intention to incorporate such changes in our final amendment targeted for June 30, 1993.

Please feel free to query us as the staff evaluates this submittal. You may contact me or Mr. John Rec (203-285-2861) for assistance in this matter.

Very truly yours,

COMBUSTION ENGINEERING, INC.

G. D. Hess for

C. B. Brinkman
Acting Director
Nuclear Systems Licensing

cc: T. Boyce (NRC)
T. Wambach (NRC)
P. Lang (DOE)
J. Trotter (EPRI)
A. Heymer (NUMARC)
J. Egan (SPPT)
T. Crom (DE&S)
S. Stamm (SWEC)

D032

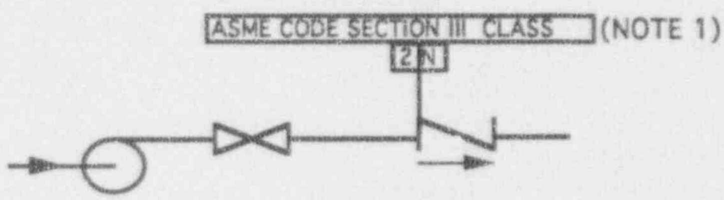
Combustion Engineering, Inc.
1000 Prospect Hill Road
Post Office Box 500
Windsor, Connecticut 06095-0500

Telephone (203) 688-1911
Fax (203) 285-9512
Telex 99297 COMBEN WSOR

1.3 FIGURE LEGEND AND ABBREVIATION LIST

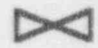
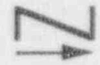
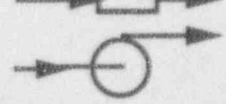
ASME Code Class Break

An ASME Code class break is identified by a single horizontal or vertical dashed line perpendicular to the designated location for the class break. As shown in the example.



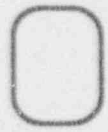
Instrumentation

Flow Instrument	F
Temperature Instrument	T
Radiation Instrument	R
Differential Pressure Instrument	PD
Pressure Instrument	P
Level Instrument	L
Current Instrument	I
Humidity Detector	H
Ultrasonic Instrument	U
Smoke Detector	SD
Sensor	S
Annunciator	A

Annunciator Symbols For:	
High High	HH
High	H
Low	L
Low Low	LL
<u>Valves</u>	
Gate Valve	
Globe Valve	
Check Valve	
Butterfly Valve	
Ball Valve	
Relief Valve	
Three Way Valve	
Valve Type Not Specified	
<u>Valve operators</u>	
Operator Of Unspecified Type	
Motor Operator	
Solenoid Operator	
Pneumatic Operator	
Pneumatic Operator Position Indications	
-Fails As Is	FAI
-Fails Closed	FC
-Fails Open	FO
Hydraulic Operator	
<u>Mechanical Equipment</u>	
Positive Displacement Pump	
Centrifugal Pump	

Pump Type Not Specified 

Sump Pump 

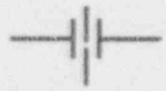
Tank 

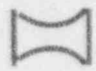
Filter 

Strainer 

Flexible Connection 

Delay Coil 

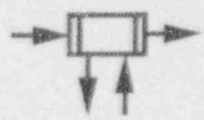
Orifice 

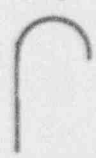
Venturi 

Compressor 

Air Distribution Device 

Vaneaxial Fan 

Heat Exchanger 

Vent  V

<u>Dampers</u>

Pressure Operated Damper 

Remotely operated Damper 

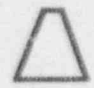
Fire Damper

Smoke Damper

Back Draft Damper

<u>Pump Drivers</u>

Turbine Drive



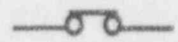
Motor Drive

<u>Electrical Equipment</u>

Circuit Breaker

Motor Operated Circuit Breaker

Disconnect Link



Multiplexer

Isolation

NOTE

1. The header, "ASME Code Section III Class", must appear at least once on each figure on which class breaks are shown, but need not appear at every class break shown on a figure.

$\boxed{\text{N}}$ Indicates Non ASME Code Section III.

ABBREVIATIONS

Abbreviation	Meaning
A/C	Air Conditioning
AFAS	Alternate Feedwater Actuation Signal
APC	Auxiliary Process Cabinet
APS	Alternate Protection System
CCS	Component Control System
CCW	Component Cooling Water
CCWLLSTAS	Component Cooling Water Low Level Surge Tank Actuation
CCWS	Component Cooling Water System
CEA	Control Element Assembly
CEDMCS	Control Element Drive Mechanism Control System
CEDM	Control Element Drive Mechanism
CET	Core Exit Thermocouple
CFR	Code of Federal Regulations
CH	Channel
CIAS	Containment Isolation Actuation Signal
CIV	Containment Isolation Valve
CPC	Core Protection Calculator
CRS	Control Room Supervisor
CSAS	Containment Spray Actuation Signal
CSB	Core Support Barrel
CSS	Containment Spray System
CVAP	Comprehensive Vibration Assessment Program
CVCS	Chemical and Volume Control System
DEMIN	Demineralized
DFSS	Diesel Fuel Storage Structure
DIAS	Discrete Indication and Alarm System
DPS	Data Processing System

ABBREVIATIONS (Continued)

Abbreviation	Meaning
DVI	Direct Vessel Injection
DWMS	Demineralized Water Makeup System
ECWS	Essential Chilled Water System
EFAS	Emergency Feedwater Actuation Signal
EFDS	Equipment and Floor Drainage System
EFW	Emergency Feedwater
EFWS	Emergency Feedwater System
EFWST	Emergency Feedwater Storage Tank
ENS	Emergency Notification System
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
ESF-CCS	Engineered Safety Features - Component Control System
FBOC	Fuel Building Overhead Crane
FHS	Fuel Handling System
FTS	Fuel Transfer System
HFE	Human Factors Engineering
HJTC	Heated Junction Thermocouple
HPN	Health Physics Network
HSI	Human-System/Interface
HVAC	Heating, Ventilating, Air Conditioning
HX	Heat Exchanger
IAS	Instrument Air System
ICI	In-Core Instrumentation
INIT	Initiation
INST	Instrumentation
IPSO	Integrated Process Status Overview
IRWST	In-containment Refueling Water Storage Tank

ABBREVIATIONS (Continued)

Abbreviation	Meaning
ITP	Interface and Test Processor
LOCA	Loss-of-coolant Accident
LTOP	Low Temperature Overrpressure Protection
LWMS	Liquid Waste Management System
MCR	Main Control Room
MDNBR	Minimum Departure From Nucleate Boiling Ratio
MOV	Motor Operated Valve
MSIS	Main Steam Isolation Signal
MSSS	Main Steam Supply System
NA	Nuclear Annex
NCW	Normal Chilled Water
NCWS	Normal Chilled Water System
NDE	Non-destructive Examination
NFE	New Fuel Elevator
NI Structures	Nuclear Island Structures
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PA	Public Address
PABX	Private Automatic Business Exchange
PCPS	Pool Cooling and Purification System
PPC	Plant Protection Calculator
PPS	Plant Protection System
PRA	Probability Risk Assessment
PSS	Process Sampling System
PSWS	Potable and Sanitary Water Systems
PZR	Pressurizer
RB	Reactor Building

ABBREVIATIONS (Continued)

Abbreviation	Meaning
RCGVS	Reactor Coolant Gas Vent System
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RDS	Rapid Depressurization System
RDT	Reactor Drain Tank
RM	Refueling Machine
RSP	Remote Shutdown Panel
RSR	Remote Shutdown Room
RT	Reactor Trip
RTSG	Reactor Trip Switchgear
RV	Reactor Vessel
SAFDL	Specified Acceptable Full Design Limits
SB	Shield Building
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
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SIS	Safety Injection System
SIT	Safety Injection Tank
SSW	Station Service Water
SSWS	Station Service Water System
TBCWS	Turbine Building Cooling Water System

ABBREVIATIONS (Continued)

Abbreviation	Meaning
TBSWS	Turbine Building Service Water System
TC	Thermocouple
TSC	Technical Support Center
UGS	Upper Guide Structure
UHS	Ultimate Heat Sink
VDU	Video Display Unit

2.1.1 NUCLEAR ISLAND STRUCTURES

Design Description

The Nuclear Island (NI) Structures house, protect, and support plant equipment and provide personnel and equipment access, support for systems and components under operating loads, radiation shielding, structural components to withstand loads due to design basis external and internal events, physical separation between Divisions of safety-related equipment, and barriers to minimize or prevent the release of radioactive materials.

The Basic Configuration of the NI Structures is as shown on Figures. 2.1.1-1 through 2.1.1-12.[1] The NI Structures are safety-related.

The NI Structures consist of the Reactor Building (RB) and the Nuclear Annex (NA). The RB and NA are further sub-divided into structures, buildings and areas. The RB and NA are structurally integrated on a common basemat which is embedded below the finished plant grade level.

The RB is a reinforced concrete and structural steel structure, which consists of the Shield Building (SB), the RB Subsphere, the Containment, and the Containment Internal Structures. The SB is composed of a reinforced concrete right cylinder with a hemispherical dome which encloses the Containment and is structurally connected to the NA. The area between the SB and the Containment is the RB Annulus. The RB Subsphere is located below the RB Annulus area and the Containment and is divided by a Divisional wall. Within the RB Subsphere, each Division is further divided, such that the RB Subsphere is separated into quadrants. The structural components of the RB Subsphere are structurally connected to the SB and support the Containment and Containment Internal Structures.

The Containment is a spherical, free-standing, welded steel structure. The Containment retains its integrity at the pressure and temperature conditions associated with the most limiting Design Basis Accident without exceeding the design leakage rate to the SB. Access to the Containment is provided through personnel air locks and an equipment hatch. Penetrations are also provided for electrical and mechanical components and for the transport of nuclear fuel.

The Containment Internal Structures are reinforced concrete and structural steel structures that support the reactor vessel and reactor coolant system. The primary shield wall supports and laterally surrounds the reactor vessel. The secondary shield wall laterally surrounds the primary shield wall and is structurally connected to the primary shield wall by reinforced concrete slabs and walls. The secondary shield wall also provides support for the polar crane. The Containment Internal Structures provide a reactor cavity area below the reactor vessel which can be flooded with

water. A gas vent path is provided between the reactor cavity and the free volume of the Containment.

The Containment and its penetrations, shown on Figures 2.1.1-1 through 2.1.1-12, are ASME Code Section III, Class MC.[2]

The Containment and its penetrations, shown on Figures 2.1.1-1 through 2.1.1-12, retain their pressure boundary integrity associated with the design basis pressure.

The Containment and its penetrations, shown on Figures 2.1.1-1 through 2.1.1-12, maintain the Containment leakage rate less than the maximum allowable leakage rate associated with the design basis pressure.

The NA consists of the Control Complex, the Diesel Generator Areas, the Fuel Handling Area, the Spent Fuel Storage Area, the Chemical and Volume Control System and Maintenance Area, and the Main Steam Valve Houses. The NA is a reinforced concrete and structural steel structure which is structurally connected to the SB. The NA laterally surrounds the RB and is divided by a Divisional wall.

The NI Structures provide the features which accommodate the static and dynamic loads and load combinations which define the structural design basis. The design basis loads are those loads associated with:

> Normal plant operation (including dead loads, live loads, lateral earth pressure loads, and equipment loads, including the effects of temperature and equipment vibration);
>
> External events (including rain, snow, wind, flood, tornado, tornado generated missiles, and earthquake); and
>
> Internal events (including flood, pipe rupture, equipment failure, equipment failure generated missiles, and fire).

The NI Structures, shown on Figures 2.1.1-1 through 2.1.1-12, are Seismic Category I, except as noted on Figure 2.1.1-12.

Flood doors, shown on Figures 2.1.1-1 through 2.1.1-12, have sensors with open and close status displays provided at a monitored location.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.1.1-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Nuclear Island Structures.

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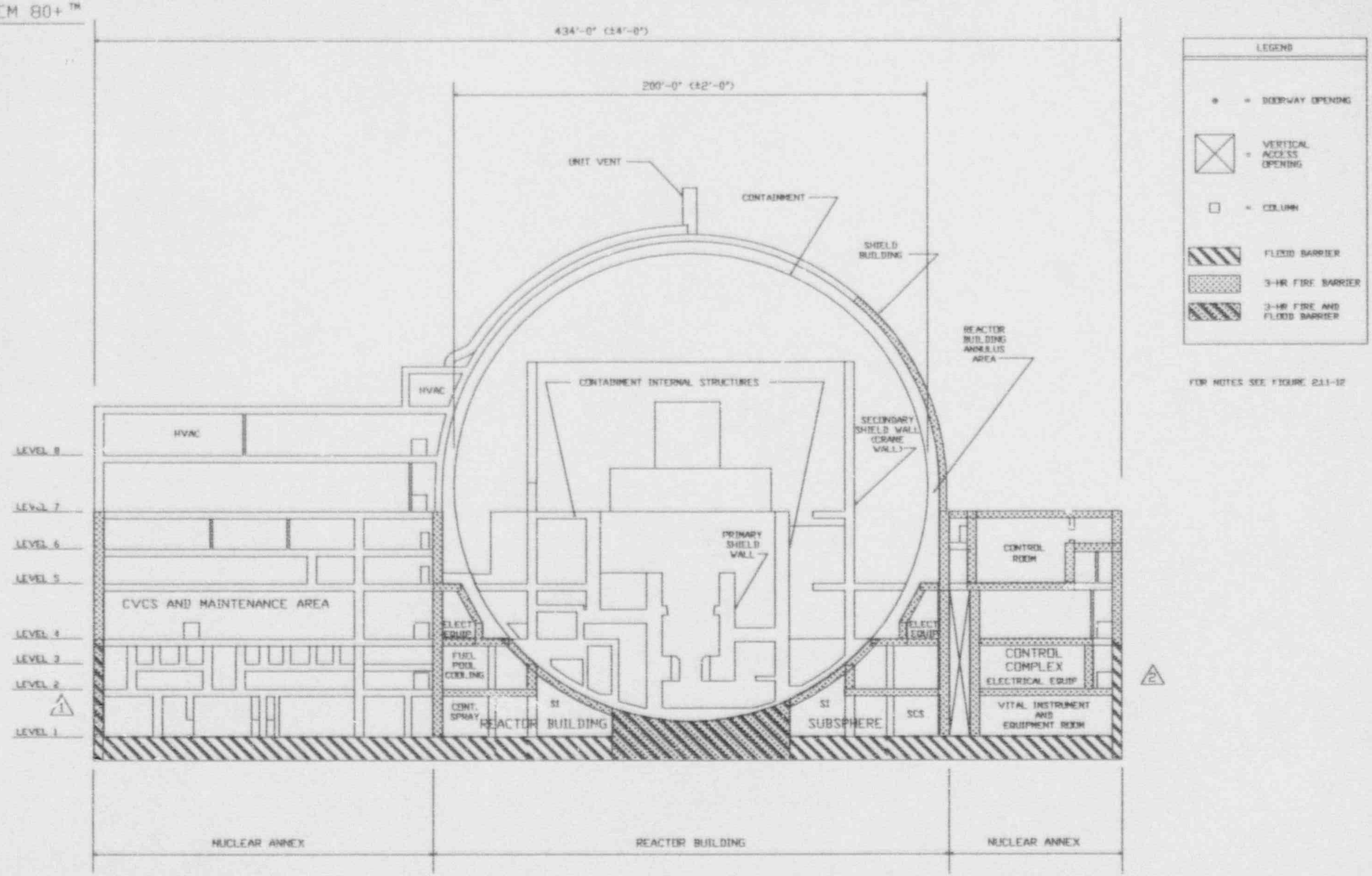
[1] The location of the NI Structures relative to the Turbine Building, the Component Cooling Water System Heat Exchanger Structure, the Diesel Fuel Storage Structures, and the Radwaste Building as described in Sections 2.1.2, 2.1.3, 2.1.4, and 2.1.5, respectively.

[2] Containment isolation devices are addressed in Section 2.4.5, Containment Isolation System.

SYSTEM 80+™



FOR NOTES SEE FIGURE 2.1.1-12

1. THE RADIOACTIVE WASTE STRUCTURE IS LOCATED ADJACENT TO THE NUCLEAR ANNEX
2. THE TURBINE BUILDING IS LOCATED ADJACENT TO THE NUCLEAR ANNEX

NUCLEAR ISLAND STRUCTURES
SECTION A-A

FIGURE 2.1.1-1

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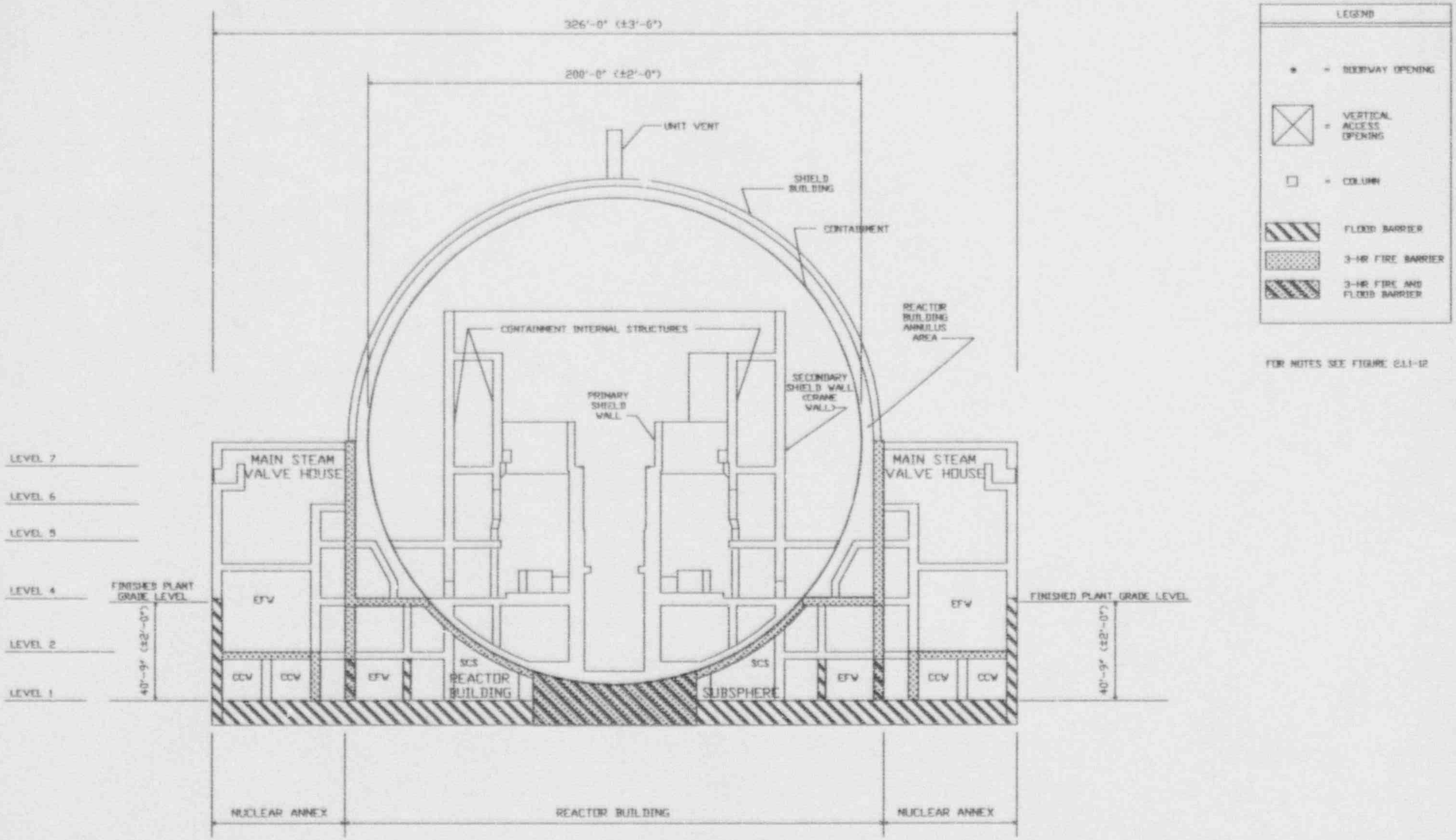
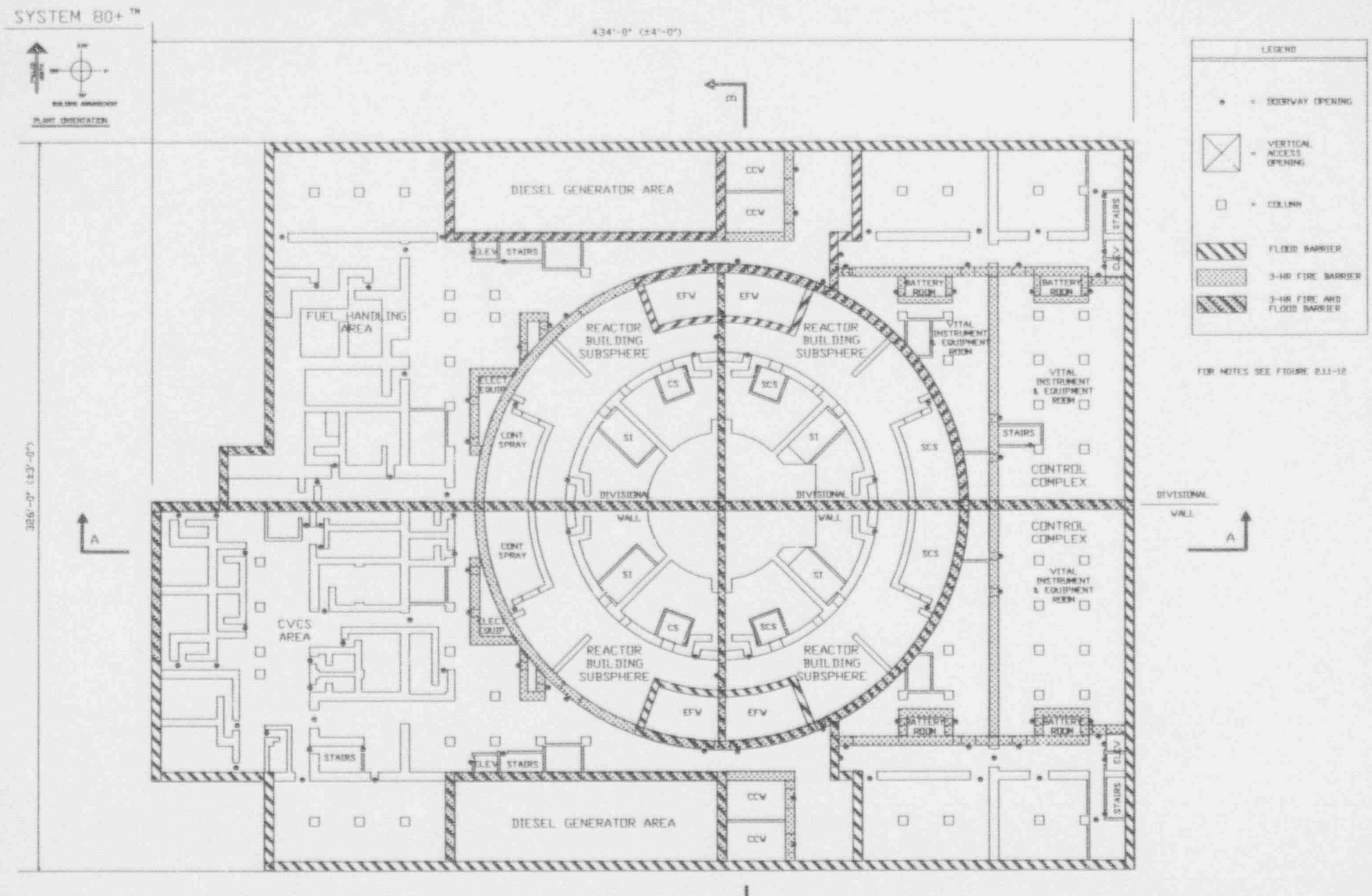


FOR NOTES SEE FIGURE 2.1.1-12

NUCLEAR ISLAND STRUCTURES
SECTION B-B

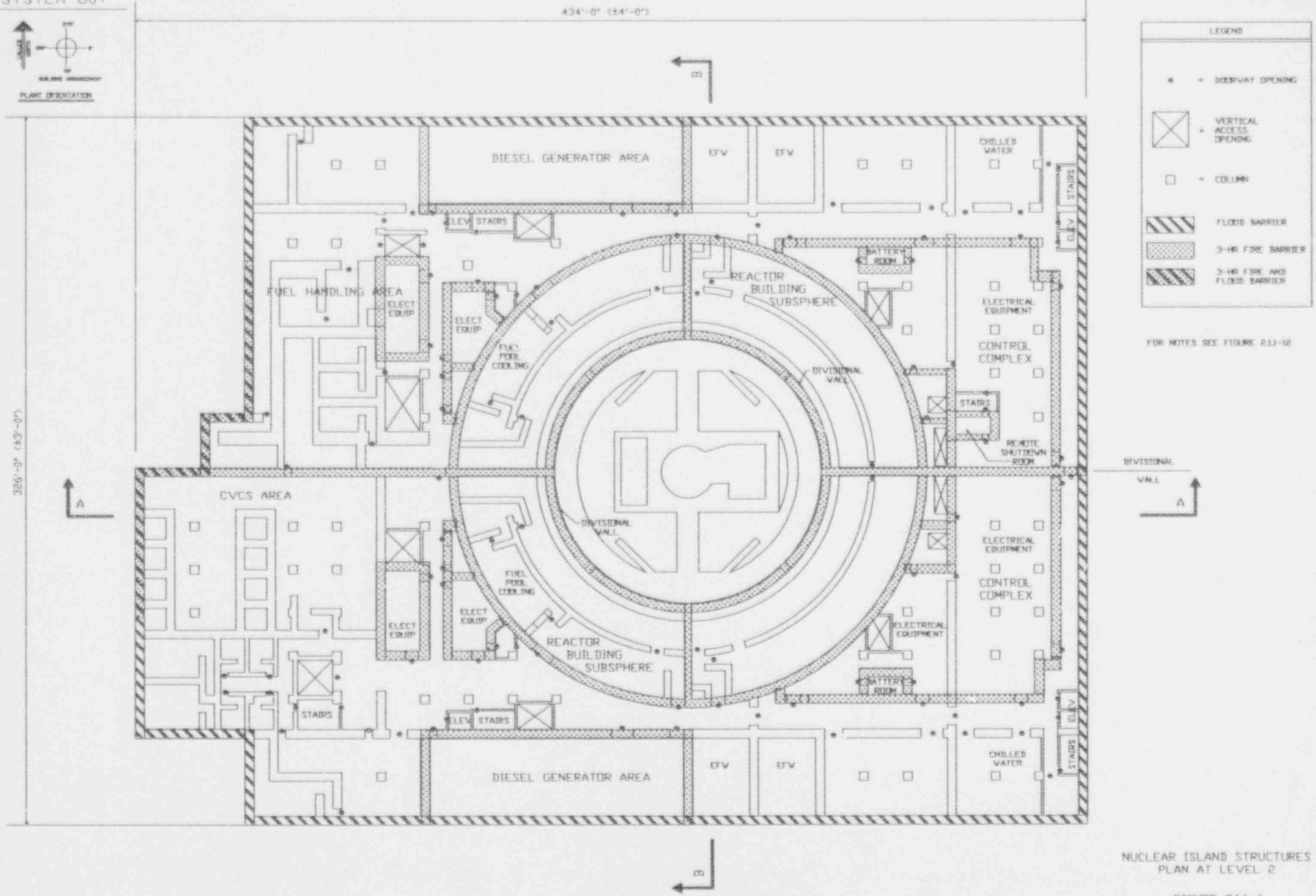
FIGURE 2.1.1-2



NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 1

FIGURE 2.11-3

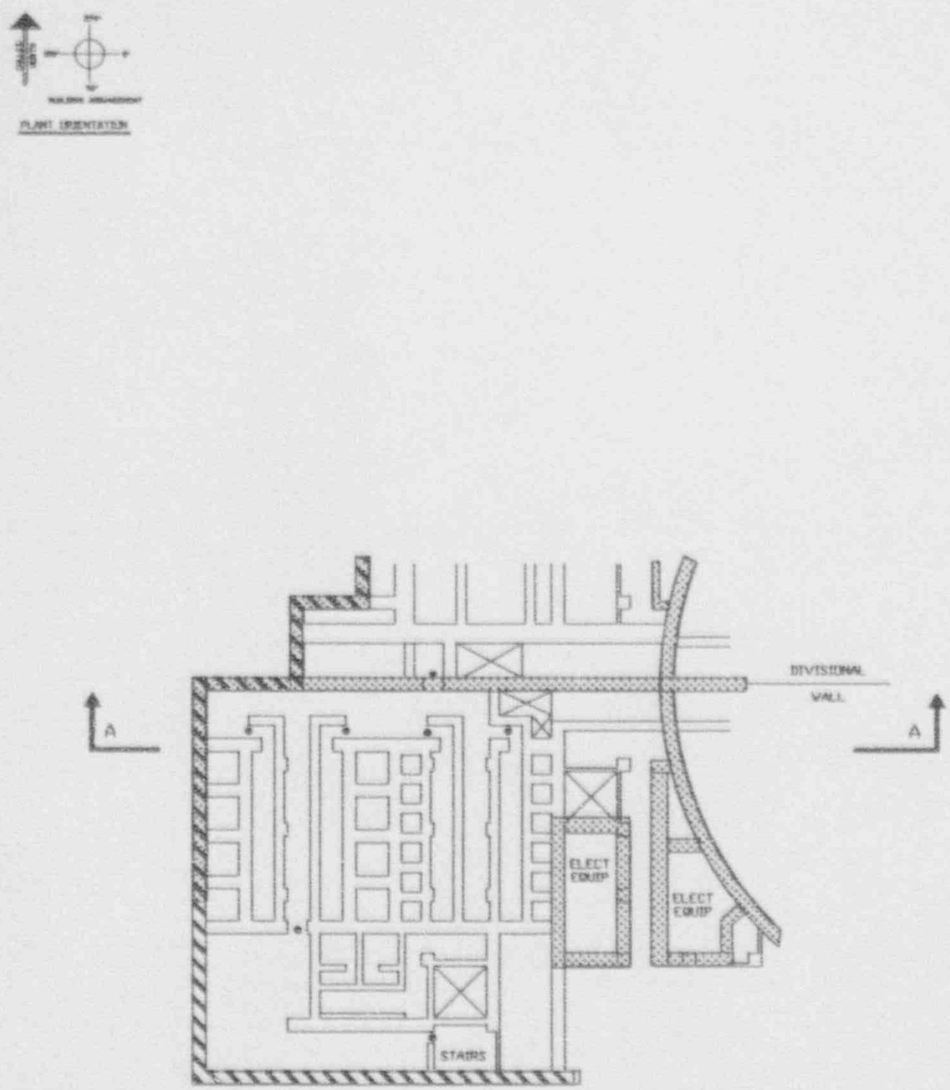
SYSTEM 80+™



NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 2

FIGURE 2.1.1-4

SYSTEM 80+ ™



LEGEND

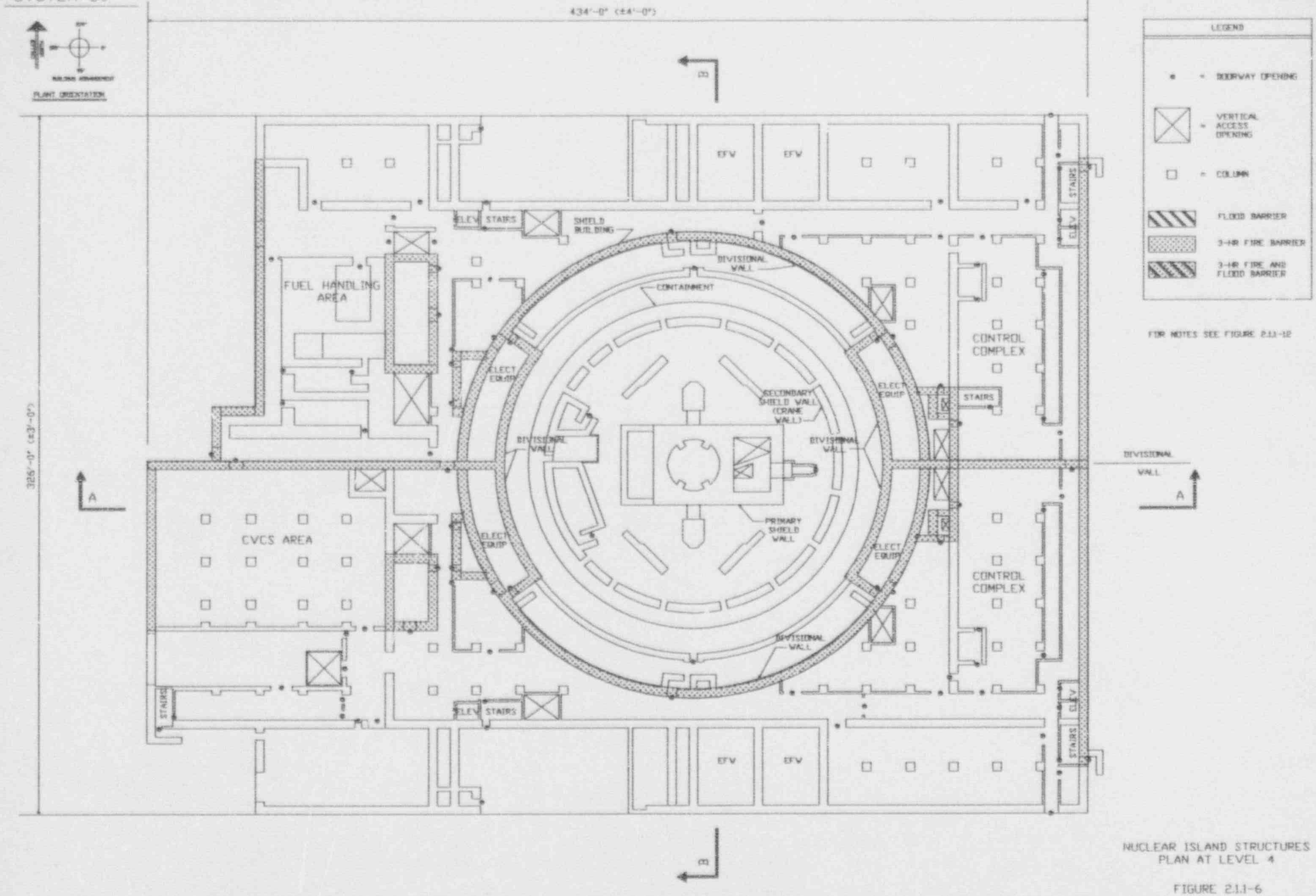
•	= DOORWAY OPENING
☒	= VERTICAL ACCESS OPENING
□	= COLUMN
	FLOOD BARRIER
	3-HR FIRE BARRIER
	3-HR FIRE AND FLOOD BARRIER

FOR NOTES SEE FIGURE 2.1.1-12

NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 3

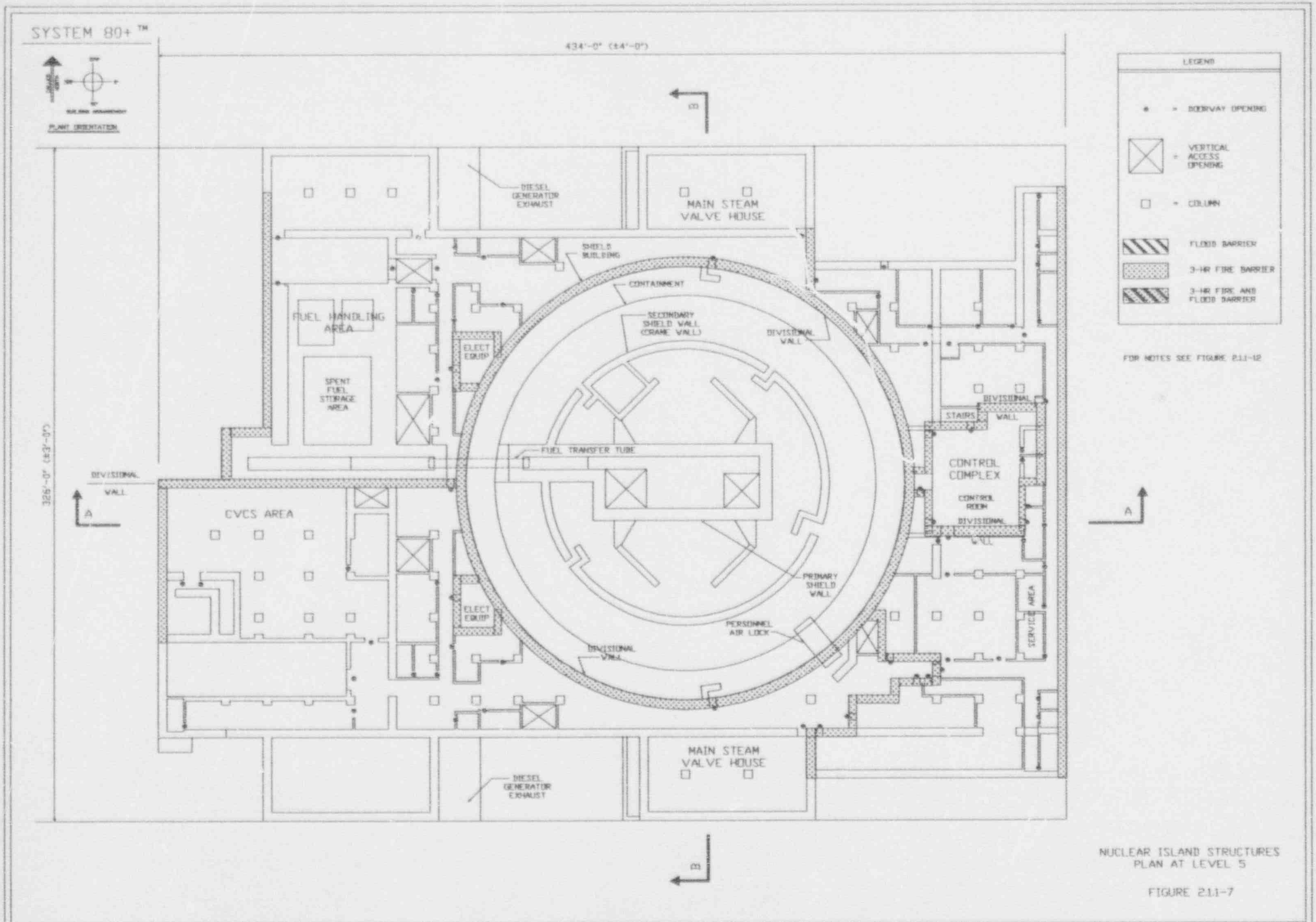
FIGURE 2.1.1-5

SYSTEM 80+ ™



NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 4

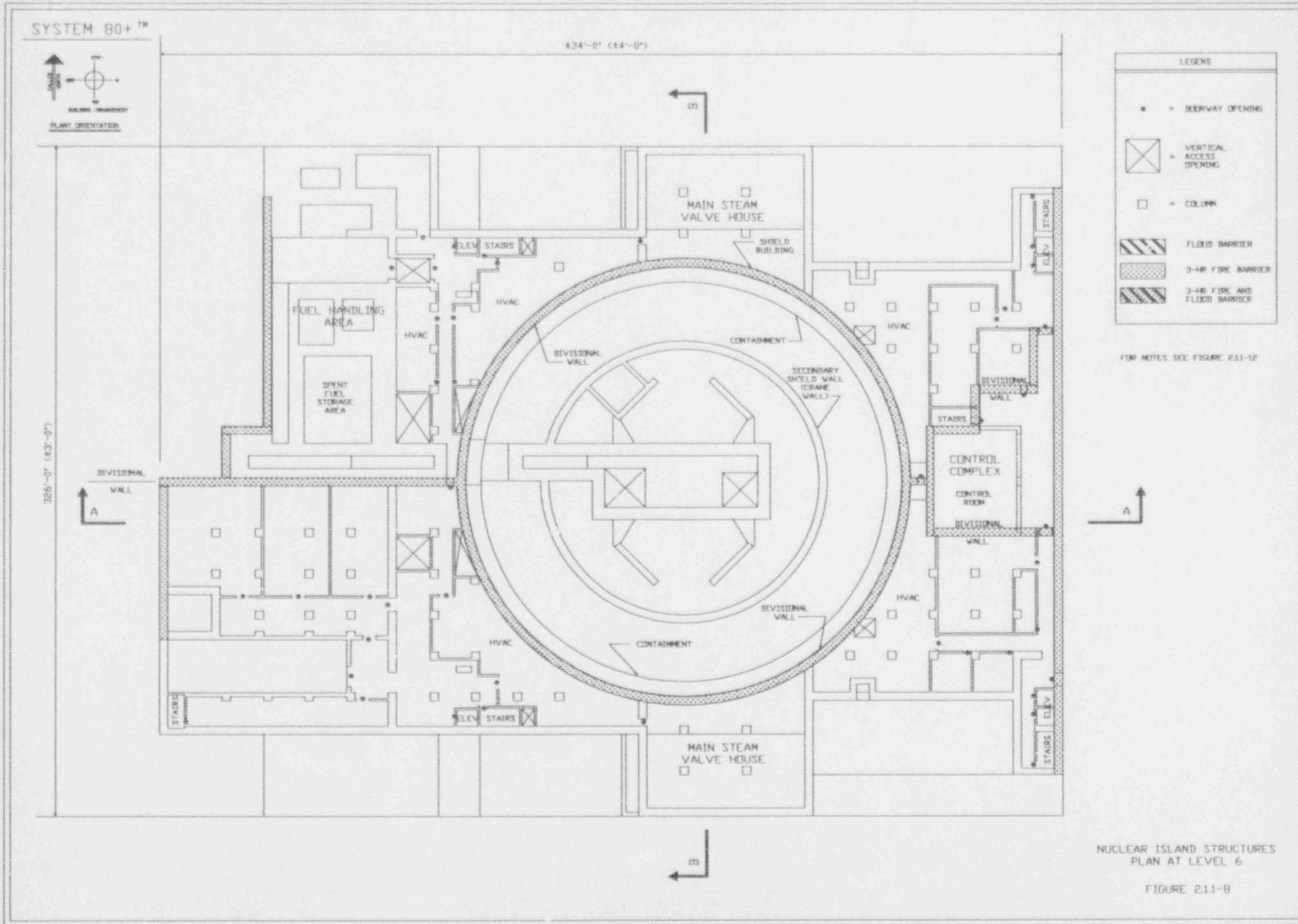
FIGURE 2.1.1-6



NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 5

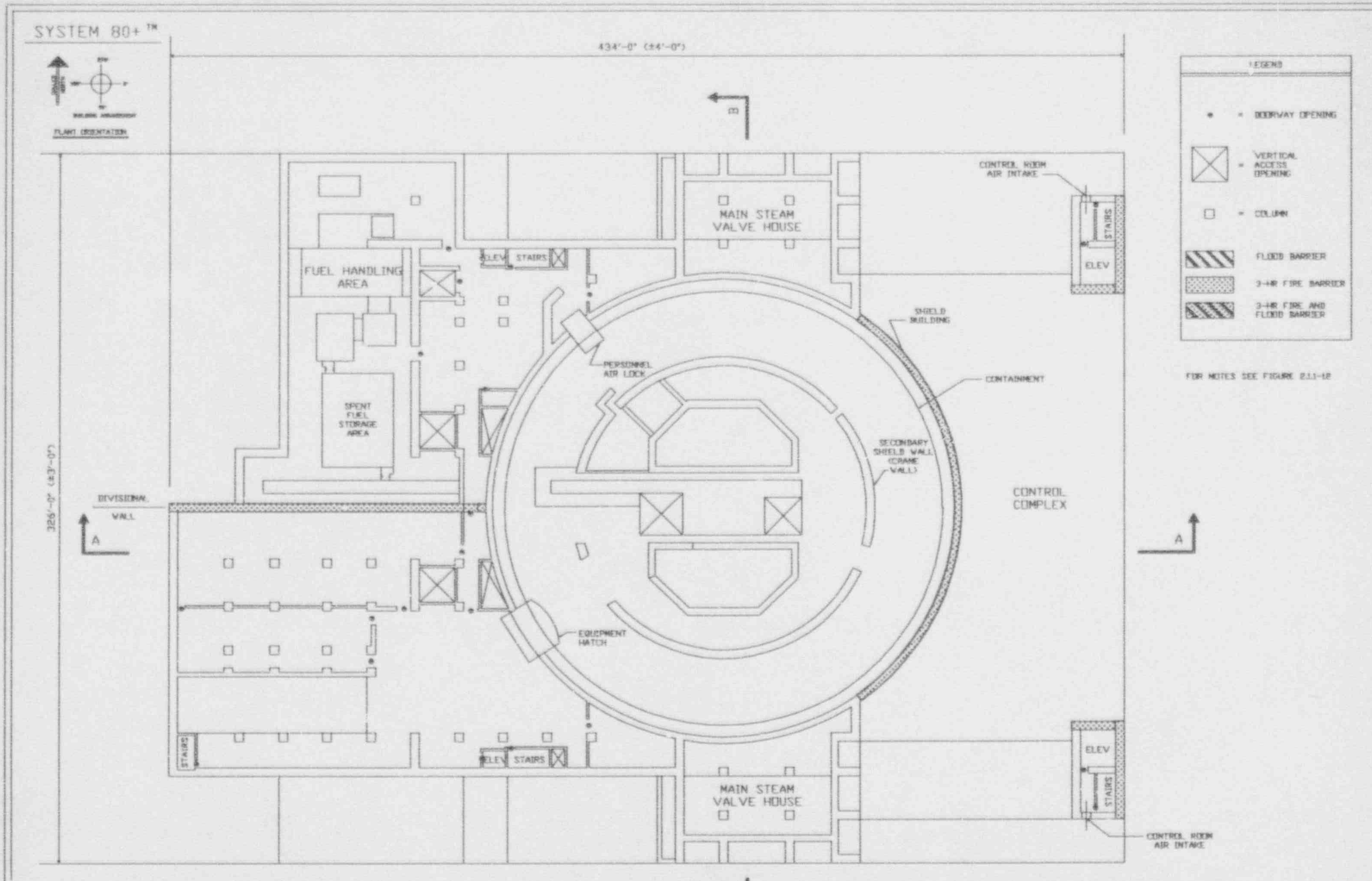
FIGURE 2.1.1-7

SYSTEM 80+™



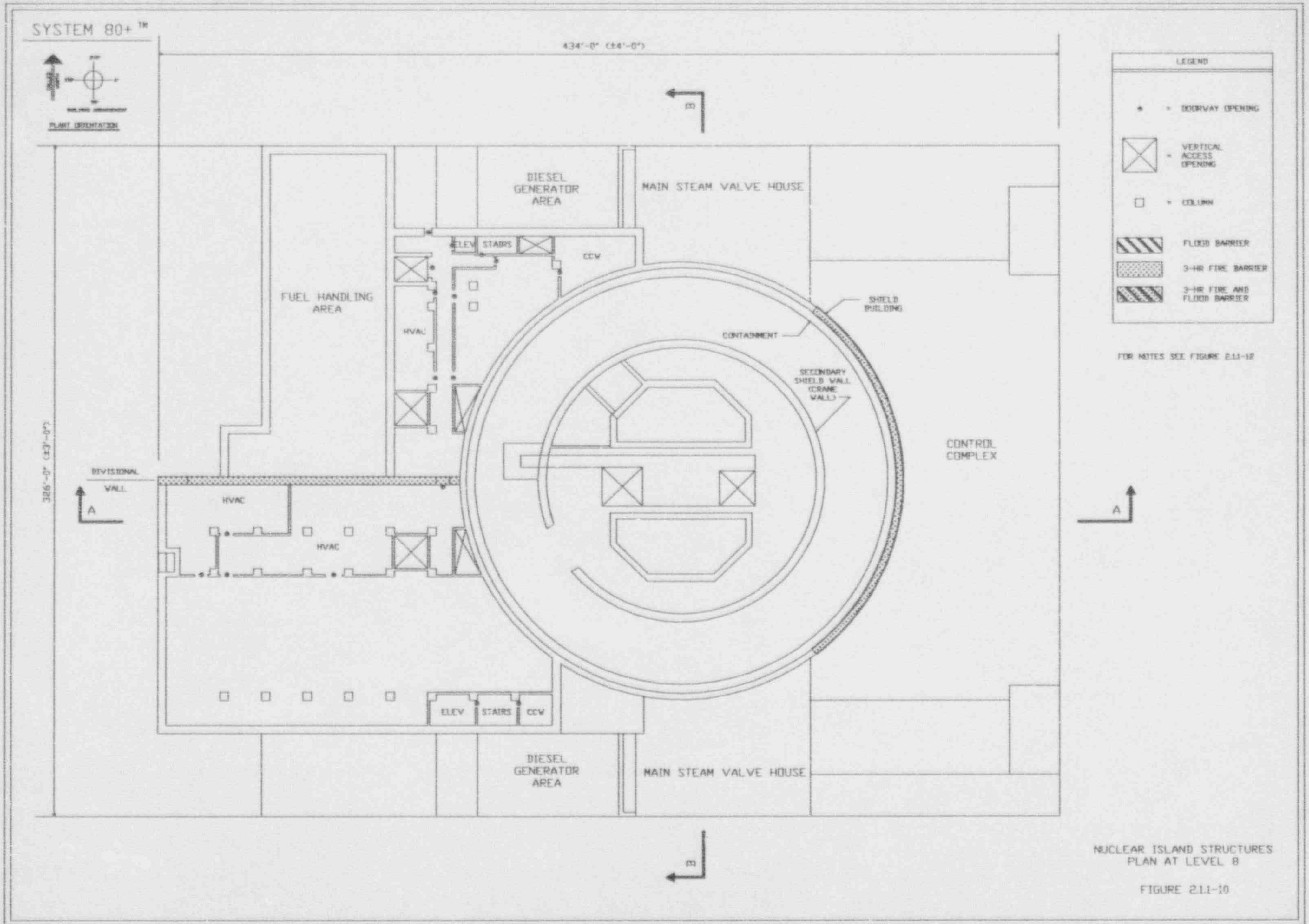
NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 6

FIGURE 2.1.1-8



NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 7

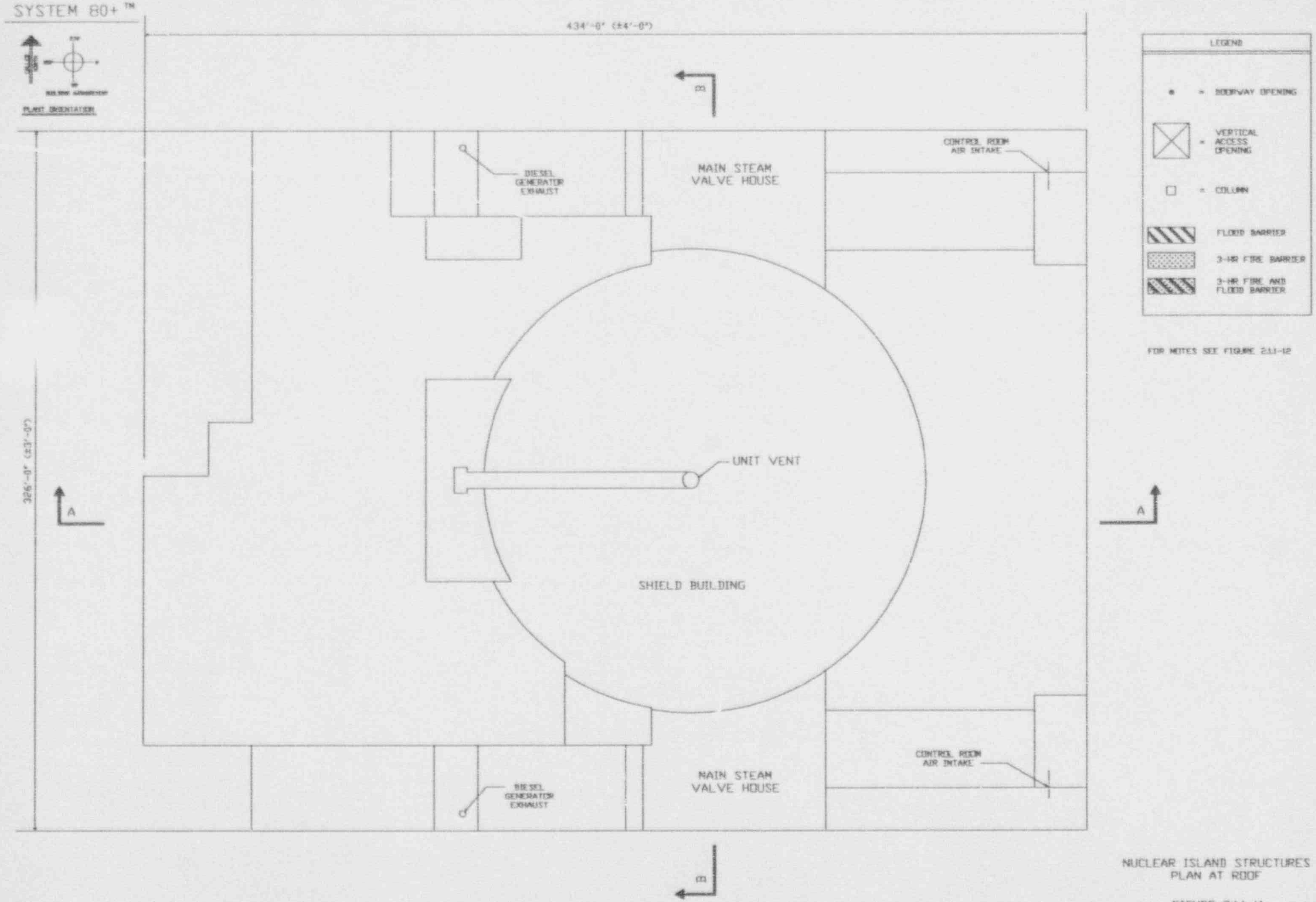
FIGURE 2.1.1-9



NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 8

FIGURE 2.1.1-10

SYSTEM 80+™

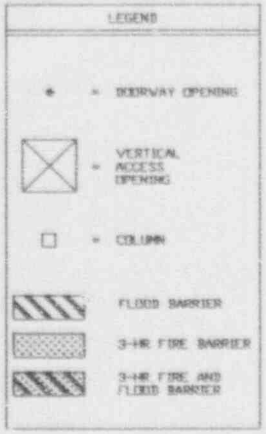


FOR NOTES SEE FIGURE 2.1.1-12

NUCLEAR ISLAND STRUCTURES
PLAN AT ROOF

FIGURE 2.1.1-11

SYSTEM 80+ ™



NOTES FOR FIGURES:

1. FLOOD DOORS ARE PROVIDED IN FLOOD BARRIERS, AND PENETRATIONS ARE SEALED UP TO THE EXTERNAL AND INTERNAL FLOOD LEVELS. SENSORS ARE PROVIDED ON FLOOD DOORS WITH OPEN AND CLOSE STATUS INDICATIONS AT A MONITORED LOCATION.

2. 3-HOUR FIRE RATED DOORS AND ELECTRICAL AND MECHANICAL PENETRATION SEALS ARE PROVIDED FOR OPENINGS IN THE 3-HOUR FIRE RATED BARRIERS.

3. THE FOLLOWING STRUCTURES, SYSTEMS, AND COMPONENTS DEPICTED ON THESE FIGURES ARE NOT SEISMIC CATEGORY I:
   - DOORWAY OPENINGS
   - VERTICAL ACCESS OPENINGS
   - STAIRS
   - ELEVATORS

ABBREVIATIONS:

BLDG .... BUILDING

CONT .... CONTAINMENT

ELECT .... ELECTRICAL

ELEV .... ELEVATOR

EQUIP .... EQUIPMENT

HR .... HOUR

MAINT .... MAINTENANCE

SYS .... SYSTEM

NUCLEAR ISLAND STRUCTURES
NOTES, LEGENDS
AND ABBREVIATIONS

FIGURE 2.1.1-12

TABLE 2.1.1-1

NUCLEAR ISLAND STRUCTURES
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the Nuclear Island Structures is as shown on Figures 2.1.1-1 through 2.1.1-12.	1. Inspections of the Basic Configuration of the as-built Nuclear Island Structures will be conducted.	1. For the structures shown on Figures 2.1.1-1 through 2.1.1-12, the Nuclear Island Structures conform with the Basic Configuration.
2.a) The Containment and its penetrations shown on Figures 2.1.1-1 through 2.1.1-12 are ASME Code Section III, Class MC.	2.a) Inspections, including non-destructive examination (NDE), of the as-built pressure boundary welds for the Containment and its penetrations will be conducted.	2.a) The inspections, including NDE, demonstrate that the requirements of ASME Code Section III, Class MC, for quality of the pressure boundary welds are met.
2.b) The Containment and its penetrations shown on Figures 2.1.1-1 through 2.1.1-12 retain their pressure boundary integrity associated with the design basis pressure.	2.b) A pressure test will be conducted on the Containment and its penetrations required to be pressure tested by ASME Code Section III.	2.b) The results of the pressure test on the Containment and its penetrations conform with the pressure testing criteria in ASME Code Section III.
2.c) The Containment and its penetrations shown on Figures 2.1.1-1 through 2.1.1-12 maintain the Containment leakage rate less than the maximum allowable leakage rate associated with the design basis pressure.	2.c) An inspection and leak rate tests on the Containment and its penetrations will be conducted.	2.c) The results of the inspection and leak rate tests demonstrate that the Containment leakage rate is less than or equal to 0.50 percent by weight of the original content of Containment air at the leakage rate test pressure during a 24 hour test period.

TABLE 2.1.1-1 (Continued)

NUCLEAR ISLAND STRUCTURES
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The Nuclear Island Structures are Seismic Category I, except as noted on Figure 2.1.1-12, and will withstand the structural design basis loads specified in the Design Description (Section 2.1.1).	3. A structural analysis will be performed which reconciles the as-built data with the structural design basis loads specified in the Design Description (Section 2.1.1).	3. A structural analysis report exists which concludes that the as-built Nuclear Island Structures will withstand the structural design basis loads specified in the Design Description (Section 2.1.1).
4. Flood doors, shown on Figures 2.1.1-1 through 2.1.1-12, have sensors with open and close status displays provided at a monitored location.	4. An inspection for existence of flood door sensors and open and close status displays will be conducted.	4. The flood door sensors and open and close status displays exist.

2.1.2 TURBINE BUILDING

Design Description

The Turbine Building is a non-safety related structure which houses the main turbine generator and provides housing and support for power conversion cycle equipment and auxiliaries. There is no safety-related equipment in the Turbine Building. The Turbine Building is located adjacent to the Nuclear Island (NI) Structures.

The Basic Configuration of the Turbine Building is as shown on Figure 2.1.2-1.

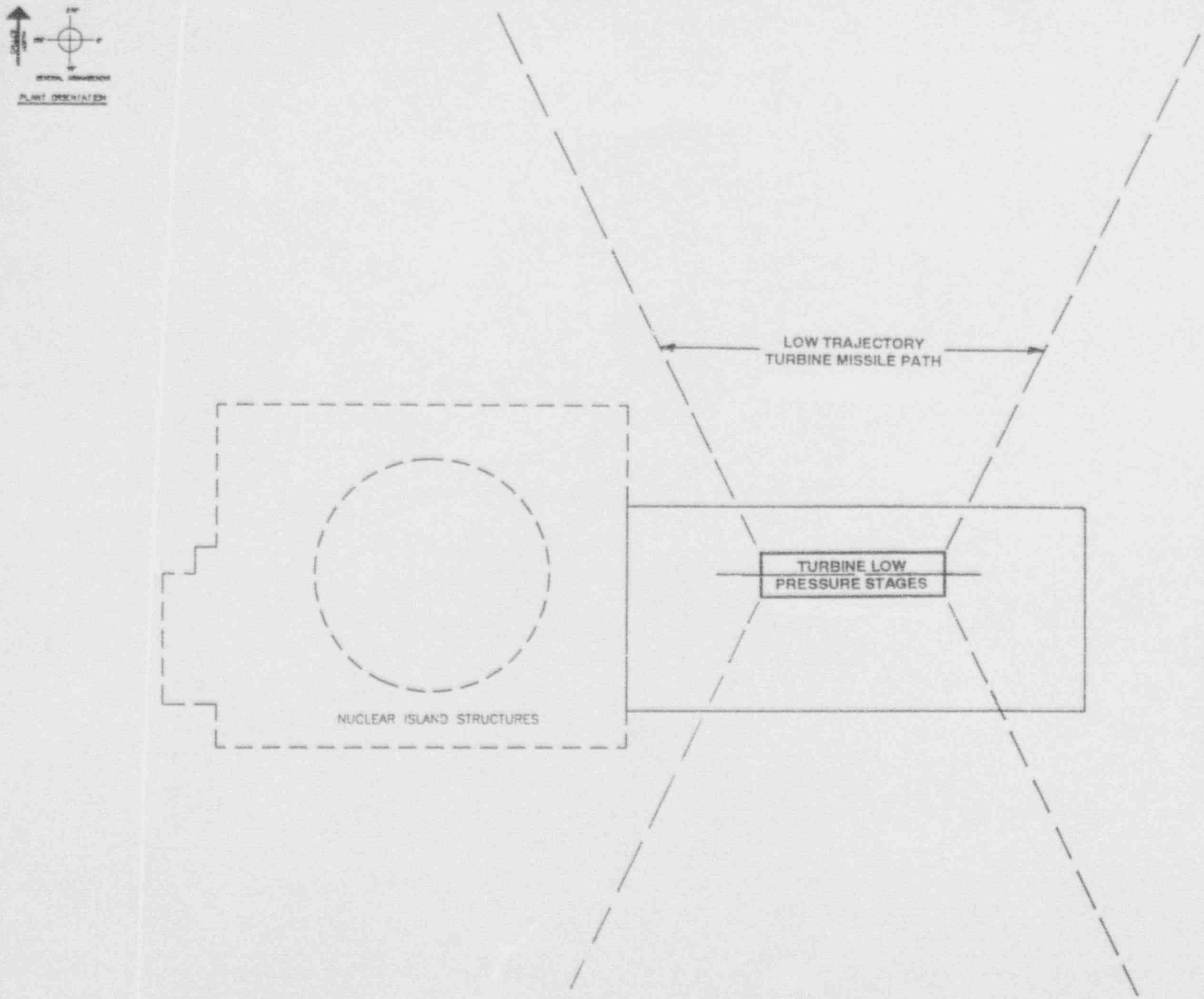
The Turbine Building contains a reinforced concrete turbine generator pedestal, a spread foundation, and a structural steel frame supporting bridge cranes, an operating floor, and a mezzanine.

The structural components of the Turbine Building accommodate design basis earthquake loads to the extent that the Turbine Building response to these loads cannot result in a loss of safety function of the adjoining NI Structures.

The turbine generator orientation and projected low trajectory turbine missile path are as shown on Figure 2.1.2-1.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.1.2-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Turbine Building.



TURBINE BUILDING
PLAN

FIGURE 2.1.2-1

TABLE 2.1.2-1

TURBINE BUILDING
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the Turbine Building is as shown on Figure 2.1.2-1.	1. Inspections of the Basic Configuration of the Turbine Building will be conducted.	1. For the structure shown on Figure 2.1.2-1, the as-built Turbine Building conforms with the Basic Configuration.
2. The structural components of the Turbine Building accommodate design basis earthquake loads to the extent that the Turbine Building response to those loads cannot result in a loss of safety function of the adjoining NI Structures.	2. A structural analysis of the Turbine Building will be performed.	2. A structural analysis report for the Turbine Building exists which concludes that structural components of the Turbine Building accommodate design basis earthquake loads to the extent that the Turbine Building response to these loads cannot result in a loss of safety function of the adjoining NI Structures.

2.1.3 COMPONENT COOLING WATER HEAT EXCHANGER STRUCTURE

Design Description

The Component Cooling Water (CCW) Heat Exchanger Structure houses and provides protection and support for component cooling water heat exchangers and supporting equipment. The CCW Heat Exchanger Structure is located outside the projected low trajectory turbine missile path.

The Basic Configuration of the CCW Heat Exchanger Structure is as shown on Figure 2.1.3-1. The CCW Heat Exchanger Structure is safety-related.

The CCW Heat Exchanger Structure provides personnel and equipment access, support for systems and components under operating loads, structural components to withstand loads due to design basis external and internal events, and physical separation between Divisions of safety-related equipment.

The CCW Heat Exchanger Structure is a separate reinforced concrete structure constructed of slabs and shear walls, and contains two Divisions of CCW Heat Exchangers and CCW components.

The CCW Heat Exchanger Structure provides features which accommodate the static and dynamic loads and load combinations which define the structural design basis. The design basis loads are those associated with:

> Normal plant operation (included dead loads, live loads, and equipment loads, including the effects of temperature and vibration);
>
> External events (including flood, wind, tornado, tornado generated missiles, earthquake, rain, and snow); and
>
> Internal events (including flood, pipe rupture, equipment failure, equipment failure generated missiles, and fire).

CCW piping enters and exits the CCW Heat Exchanger Structure through underground vaults. The CCW pipe vaults are routed underground from the CCW Heat Exchanger Structure to the CCW pipe chases located on either side of the Nuclear Island (NI) Structures.

The CCW Heat Exchanger Structure is Seismic Category I.

The CCW Heat Exchanger Structure provides physical separation of the Divisional CCW Heat Exchangers and supporting CCW components within the CCW Structure.

Inspections, Tests, Analyses, and Acceptance Criteria

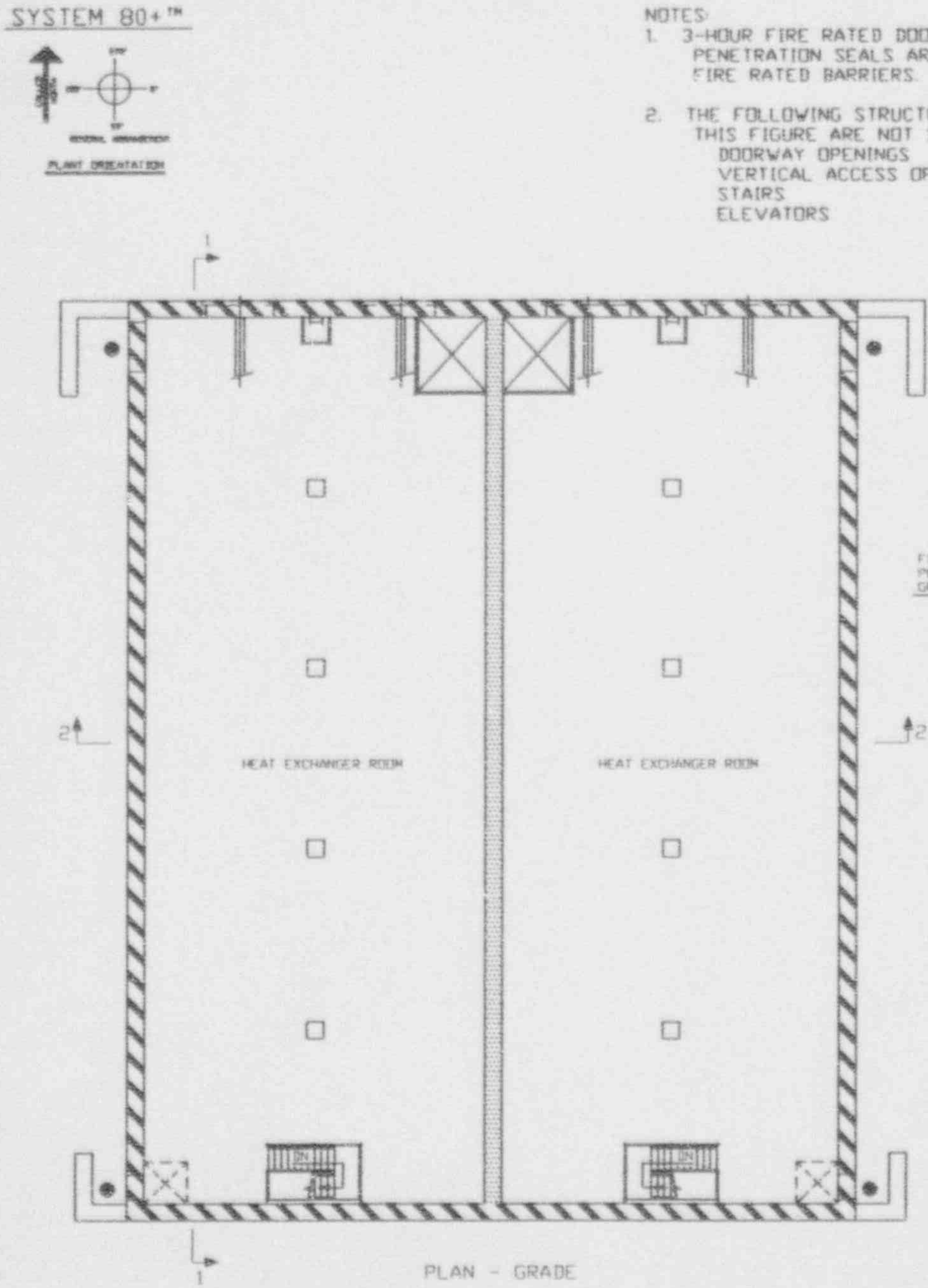
Table 2.1.3-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the CCW Heat Exchanger Structure.

SYSTEM 80+™

NOTES:
1. 3-HOUR FIRE RATED DOORS AND ELECTRICAL AND MECHANICAL PENETRATION SEALS ARE PROVIDED FOR OPENINGS IN THE 3-HOUR FIRE RATED BARRIERS.

2. THE FOLLOWING STRUCTURES, SYSTEMS, AND COMPONENTS DEPICTED ON THIS FIGURE ARE NOT SEISMIC CATEGORY I:
   DOORWAY OPENINGS
   VERTICAL ACCESS OPENINGS
   STAIRS
   ELEVATORS

LEGEND
- ● = DOORWAY
- ⊠ = VERTICAL ACCESS SHAFTS
- □ = COLUMN
- (hatched) = FLOOD BARRIER
- (shaded) = 3-HR FIRE BARRIER
- (hatched, heavy) = 3-HR FIRE AND FLOOD BARRIER



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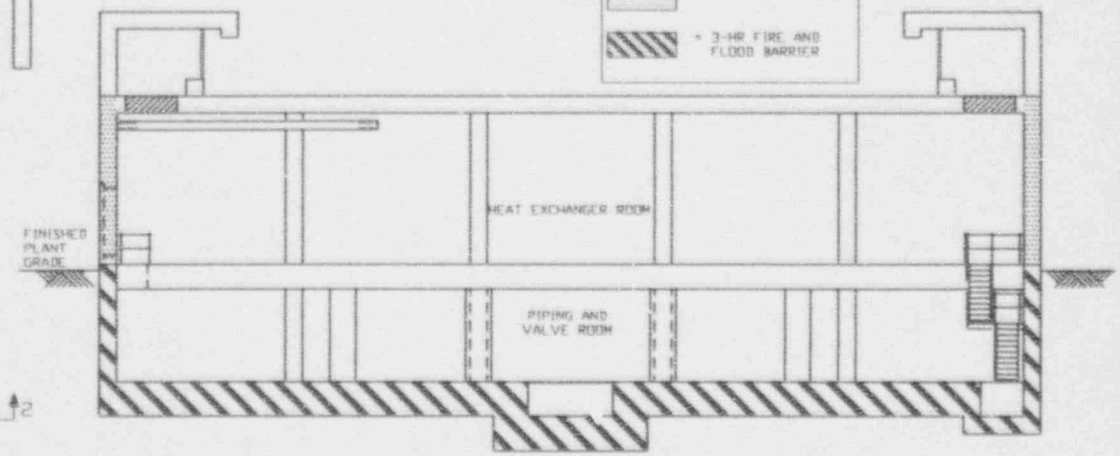

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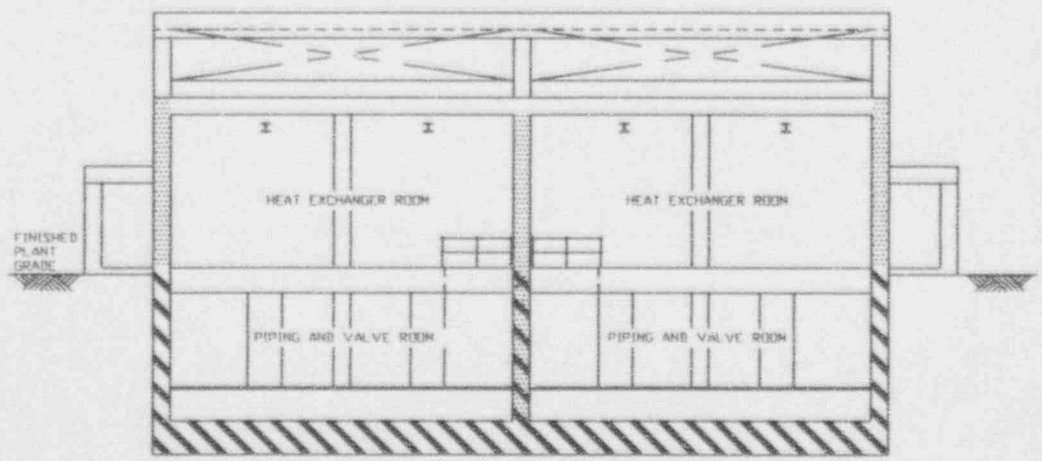

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CCW HEAT EXCHANGER STRUCTURE

FIGURE 2.3.1-1

TABLE 2.1.3-1

COMPONENT COOLING WATER HEAT EXCHANGER STRUCTURE
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the Component Cooling Water (CCW) Heat Exchanger Structure is as shown on Figure 2.1.3-1.	1. Inspections of the Basic Configuration of the CCW Heat Exchanger Structure will be conducted.	1. For the structure shown on Figure 2.1.3-1, the CCW Heat Exchanger Structure conforms with the Basic Configuration.
2. The CCW Heat Exchanger Structure is located outside the projected low trajectory turbine missile path.	2. An inspection of the location of the CCW Heat Exchanger Structure will be performed.	2. The CCW Heat Exchanger Structure is located outside the projected low trajectory turbine missile path.
3. The CCW Heat Exchanger Structure is Seismic Category I and withstands the structural design basis loads specified in the Design Description (Section 2.1.3).	3. A structural analysis will be performed which reconciles the as-built data with the structural design basis specified in the Design Description (Section 2.1.3).	3. A structural analysis report exists which concludes that the as-built CCW Heat Exchanger Structure withstands the structural design basis loads specified in the Design Description (Section 2.1.3).
4. The CCW Heat Exchanger Structure provides physical separation of the Divisional CCW Heat Exchangers and supporting CCW components within the CCW Structure.	4. An inspection of the as-built CCW Heat Exchanger Structure will be performed.	4. The CCW Heat Exchanger Structure has walls or fire barriers that separate the two Divisions of the CCW Heat Exchangers and CCW components in the CCW Heat Exchanger Structure.

2.1.4 DIESEL FUEL STORAGE STRUCTURE

Design Description

Two separate Diesel Fuel Storage Structures (DFSSs) house and provide protection and support for the diesel generator fuel oil storage tanks and associated piping and equipment. The DFSSs are not connected to the Nuclear Island (NI) Structures except by diesel fuel transfer piping.

The Basic Configuration of each DFSS is as shown on Figure 2.1.4-1. The DFSSs are safety-related.

The DFSSs are located outside the projected low trajectory turbine missile path.

Each Diesel Fuel Storage Structure provides personnel and equipment access, support for systems and components under operating loads, and structural components to withstand loads due to design basis external and internal events.

Each DFSS is a reinforced concrete vault containing two Fuel Storage Tank Areas and an attached equipment room and is constructed of slabs and shear walls. Each Fuel Storage Tank Area provides space for a diesel fuel oil storage tank and associated piping and pumps.

Each DFSS provides features which accommodates the static and dynamic loads and load combinations which define the structural design basis. The design basis loads are those associated with:

> Normal plant operation (including dead loads, live loads, lateral earth pressure loads and equipment loads, including the effects of temperature and vibration);
>
> External events (including flood, wind, tornado, tornado generated missiles, earthquake, rain, and snow); and
>
> Internal events (including flood, pipe rupture, equipment failure, equipment failure generated missiles, and fire).

The DFSSs are Seismic Category I.

The two DFSSs are physically separated by their placement on opposite sides of the NI Structures.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.1.4-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the DFSSs.

SYSTEM 80+™

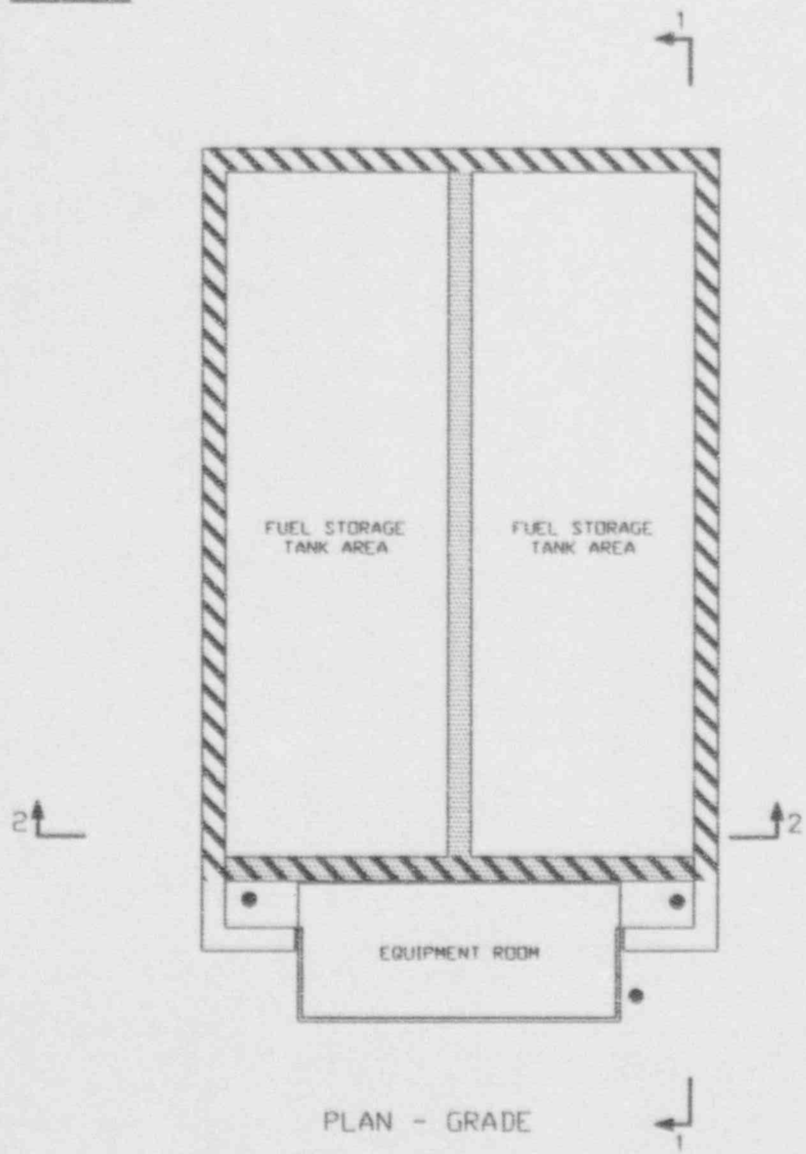
NOTES:

1. 3-HOUR FIRE RATED DOORS AND ELECTRICAL AND MECHANICAL PENETRATION SEALS ARE PROVIDED FOR OPENINGS IN THE 3-HOUR FIRE RATED BARRIERS.

2. THE FOLLOWING STRUCTURES, SYSTEMS, AND COMPONENTS DEPICTED ON THIS FIGURE ARE NOT SEISMIC CATEGORY I:
   - DOORWAY OPENINGS
   - VERTICAL ACCESS OPENINGS
   - STAIRS
   - ELEVATORS

LEGEND

- ● = DOORWAY
- ☒ = VERTICAL ACCESS SHAFTS
- □ = COLUMN
- = FLOOD BARRIER
- = 3-HR FIRE BARRIER
- = 3-HR FIRE AND FLOOD BARRIER



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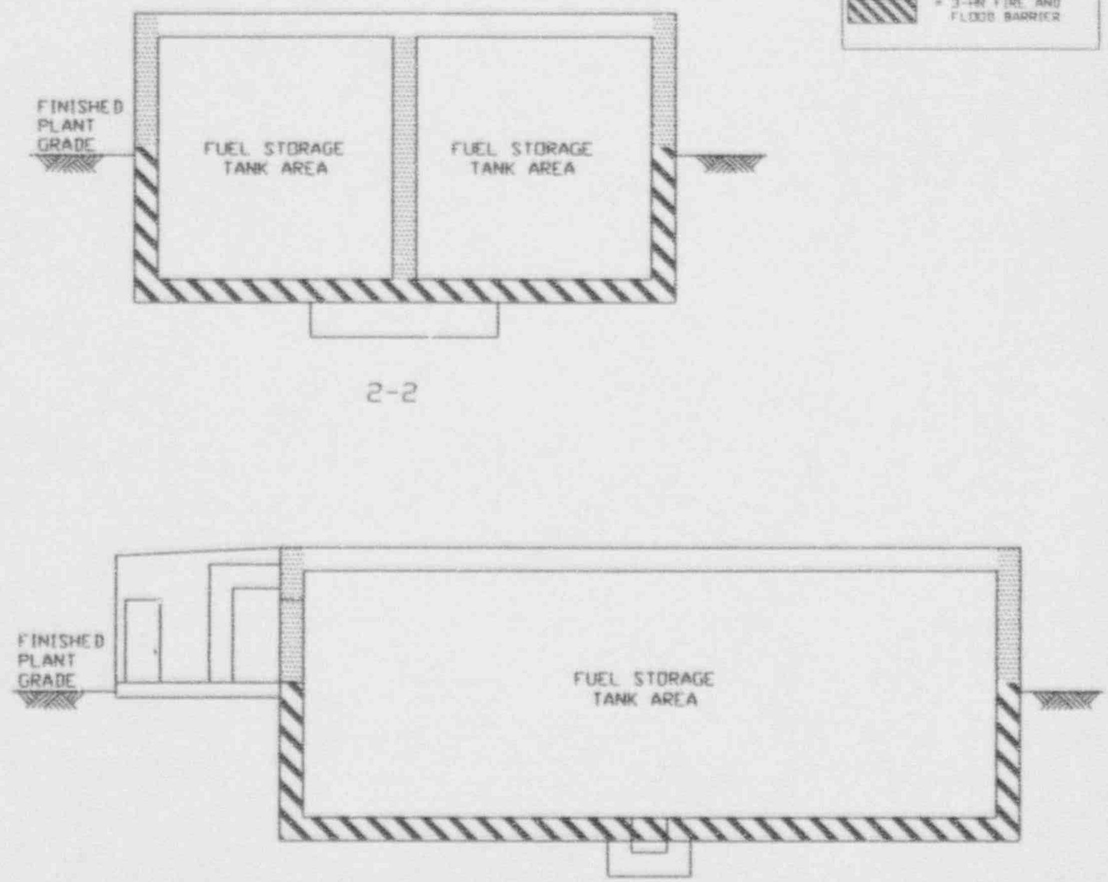

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DIESEL FUEL STORAGE STRUCTURE

FIGURE 2.1.4-1

TABLE 2.1.4-1

DIESEL FUEL STORAGE STRUCTURE
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of each Diesel Fuel Storage Structure is as shown on Figure 2.1.4-1.	1. Inspections of the Basic Configuration of each Diesel Fuel Storage Structure will be conducted.	1. For the structure shown on Figure 2.1.4-1 each Diesel Fuel Storage Structure conforms with the Basic Configuration.
2. The DFSSs are located outside the projected low trajectory turbine missile path.	2. An inspection of the location of the DFSSs will be performed.	2. The DFSSs are located outside the projected low trajectory turbine missile path.
3. Each Diesel Fuel Storage Structure is Seismic Category I and will withstand the structural design basis loads as specified in the Design Description (Section 2.1.4).	3. A structural analysis will be performed which reconciles the as-built data with the structural design basis as specified in the Design Description (Section 2.1.4).	3. A structural analysis report exists which concludes that each as-built Diesel Fuel Storage Structure will withstand the design basis loads as specified in the Design Description (Section 2.1.4).
4. The two DFSSs are physically separated by their placement on opposite sides of the NI Structures.	4. An inspection of the DFSSs will be performed.	4. The two DFSSs are separated by the Nuclear Island Structures.

2.1.5 RADWASTE BUILDING

Design Description

The Radwaste Building is a non-safety related structure that houses gaseous, liquid, and solid radioactive waste management structures, systems, and components and provides containment for liquid and solid radioactive waste materials. The Radwaste Building is located adjacent to the Nuclear Island (NI) Structures.

The Basic Configuration of the Radwaste Building is as shown on Figures 2.1.5-1 and 2.1.5-2.

The Radwaste Building consists of a reinforced concrete and structural steel structure.

The structural components of the Radwaste Building accommodate design basis earthquake loads such that the Radwaste Building response to these loads cannot result in a loss of safety function of the adjoining NI Structure. The Radwaste Building foundations and walls accommodate design basis earthquake loads such that the maximum liquid inventory in the building is contained.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.1.5-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Radwaste Building.

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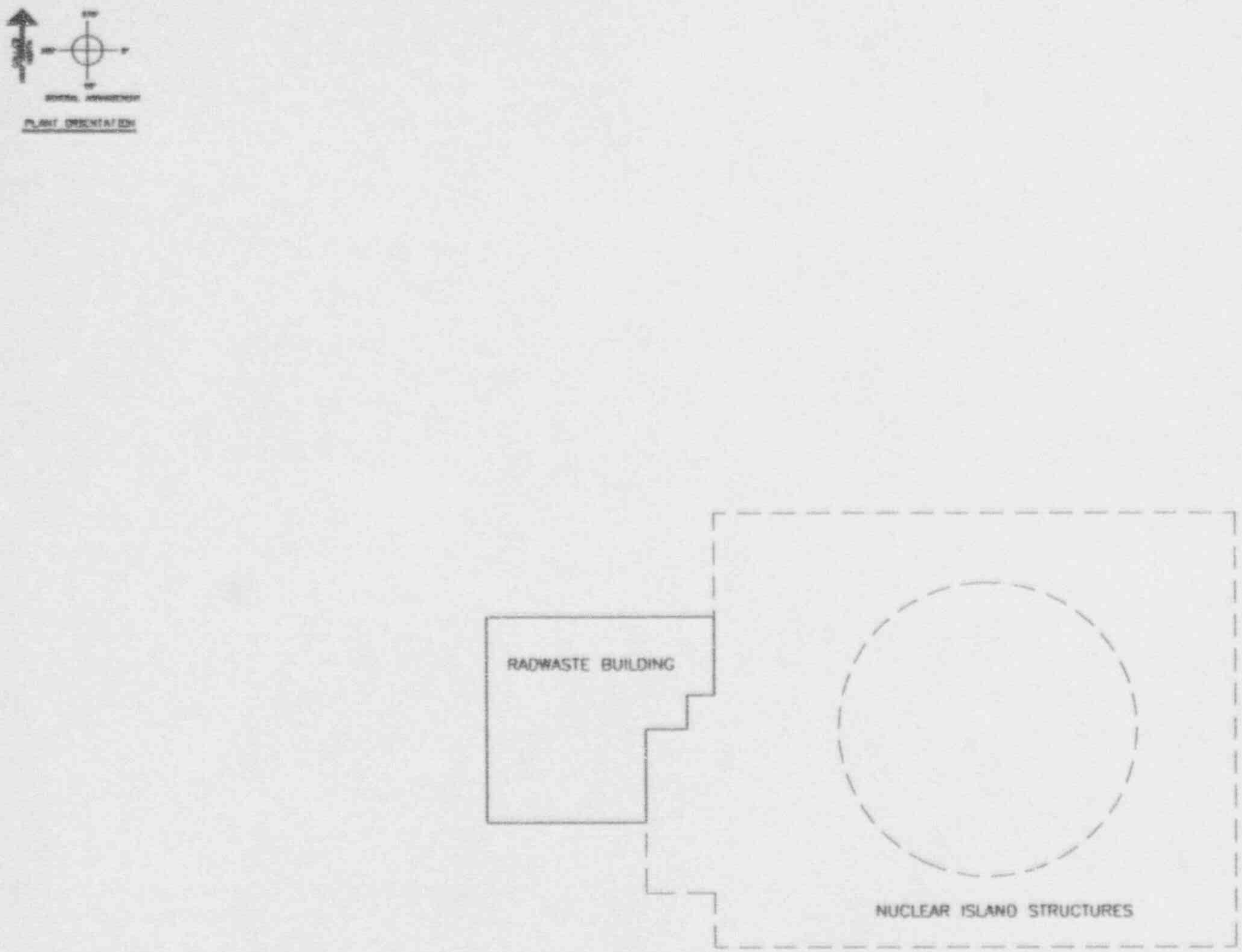

RADWASTE BUILDING
PLAN

FIGURE 2.1.5-1

SECTION

⚠ THE RADWASTE BUILDING IS LOCATED ADJACENT TO THE NUCLEAR ISLAND STRUCTURE

RADWASTE BUILDING
SECTION

FIGURE 21.5-2

TABLE 2.1.5-1

RADWASTE BUILDING
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the Radwaste Building is as shown on Figures 2.1.5-1 and 2.1.5-2.	1. Inspections of the Basic Configuration of the Radwaste Building will be conducted.	1. For the structure shown on Figures 2.1.5-1 and 2.1.5-2, the Radwaste Building conforms with the Basic Configuration.
2. The structural components of the Radwaste Building accommodate design basis earthquake loads such that the Radwaste Building response to these loads cannot result in a loss of safety function of the adjoining NI Structures.	2. A structural analysis of the Radwaste Building will be performed.	2. A structural analysis report for the Radwaste Building exists which concludes that structural components of the Radwaste Building accommodate design basis earthquake loads such that the Radwaste Building response to these loads cannot result in a loss of safety function of the adjacent NI Structures.
3. The Radwaste Building foundations and walls accommodate design basis earthquake loads such that the maximum liquid inventory in the building is contained.	3. A capacity analysis of the Radwaste Building will be performed using as-built liquid inventory data.	3. A capacity analysis report for the Radwaste Building exists which concludes that foundations and walls contain the maximum liquid inventory in the building.

2.1.6 REACTOR VESSEL INTERNALS

Design Description

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

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

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 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 fabricated in accordance with ASME Code Section III Subsection NG requirements and are qualified Seismic Category I. 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.

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 specifies the inspections, tests, analyses and associated acceptance criteria for the Reactor Vessel Internals.

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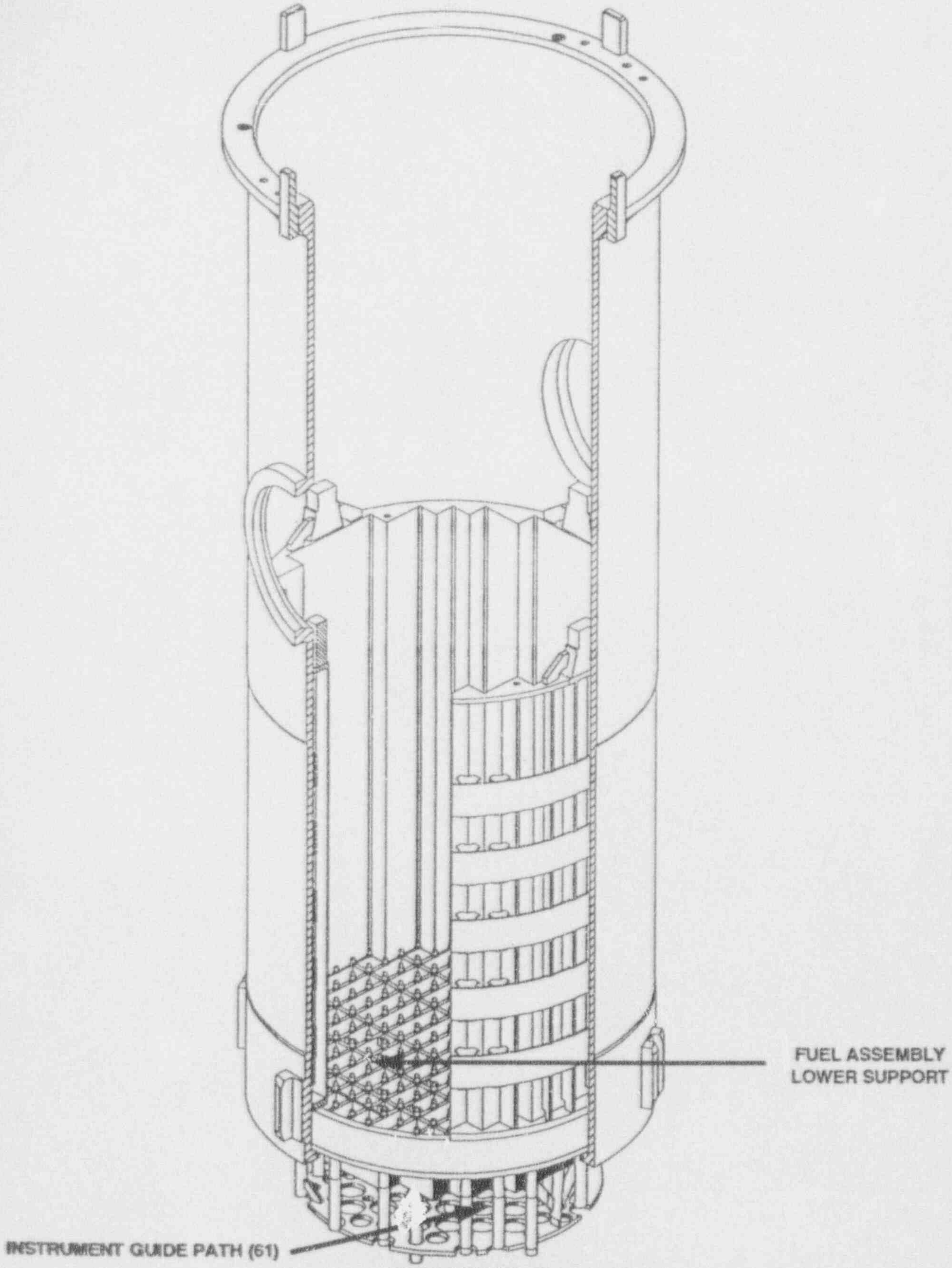

FIGURE 2.1.6-1
CORE SUPPORT BARREL ASSEMBLY

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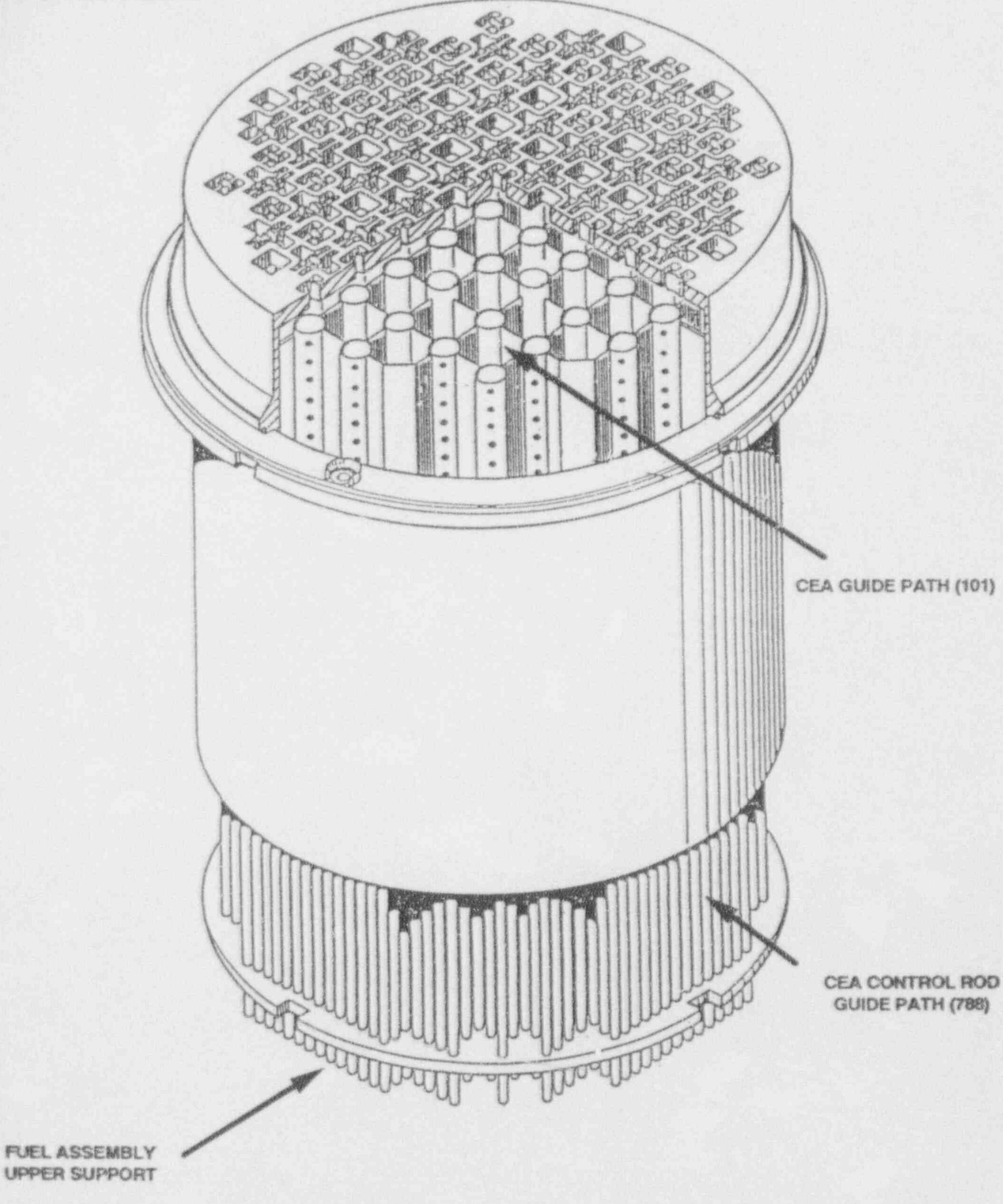


FIGURE 2.1.6-2
UPPER GUIDE STRUCTURE ASSEMBLY

TABLE 2.1.6-1

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 fabricated in accordance with ASME Code Section III Subsection NG requirements and are qualified Seismic Category I.	2. An 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. Tests 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. Tests and inspections results demonstrate that the Reactor Vessel Internals retain their integrity.

2.2.4 CONTROL ELEMENT DRIVE MECHANISM

Design Description

The safety related function of the Control Element Drive Mechanism (CEDM) is to release the Control Element Assembly (CEA) upon termination of electrical power to the CEDM. One hundred one CEDM can be installed.

Inspections, Tests, Analyses and Acceptance Criteria

None

The initial test program addressed in Section 2.11 will test the ability of the CEDM to release the CEA upon termination of electrical power to the CEDM.

2.7.1 NEW FUEL STORAGE RACKS

Design Description

The New Fuel Storage Racks provide on-site storage for New Fuel Assemblies. The New Fuel Storage Racks are safety-related.

The New Fuel Storage Racks are located in the Nuclear Island Structure 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 under normal operation and design postulated conditions.

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 qualified 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.

TABLE 2.7.1-1

NEW FUEL STORAGE RACKS
Inspection, Tests, Analysis and Acceptance Criteria

Design Commitment	Inspection, Test, Analysis	Acceptance Criteria
1. The New Fuel Storage Racks maintain the effective neutron multiplication factor less than the criticality limits during normal operation and postulated accident conditions.	1. Analysis will be performed to calculate the effective neutron multiplication factor.	1. The calculated effective neutron multiplication factor for the New Fuel Storage Racks is less than 0.95 for normal operation and accident conditions (less than 0.98 for immersion in a uniform density aqueous foam or mist of optimum moderation density).
2. The New Fuel Storage Racks are designed and fabricated in accordance with ASME Code Section III Subsection NF, Class 3 Component Supports requirements and are qualified Seismic Category I.	2. Inspections will be performed of the Fabrication Data Package, Certificate of Conformance and the Design Report Document.	2. The Fabrication Data Package, Certificate of Conformance and the Design Report Document exist, and conclude that the design requirements are met.

2.7.2 SPENT FUEL STORAGE RACKS

Design Description

The Spent Fuel Storage Racks provide on-site storage for 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 for normal operation and postulated accidents.

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 Racks are qualified 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.

TABLE 2.7.2-1

SPENT FUEL STORAGE RACKS
Inspection, Tests, Analysis and Acceptance Criteria

Design Commitment	Inspection, Test, Analysis	Acceptance Criteria
1. The Spent Fuel Storage Racks maintain the effective neutron multiplication factor less than the required criticality limits for normal operation and postulated accidents.	1. Analyses will be performed to calculate the effective neutron multiplication factor for the Spent Fuel Storage Racks under design basis conditions.	1. The calculated effective neutron multiplication factor is less than 0.95.
2. The Spent Fuel Storage Racks are designed and fabricated in accordance with the ASME Code Section III, Subsection NF, Class 3 Component Supports requirements and are qualified Seismic Category I.	2. Inspections will be performed of the Fabrication Data Package, Certificate of Conformance and Design Report Document.	2. The Fabrication Data Package, Certificate of Conformance and the approved Design Report Document exist and conclude that the design requirements are met.

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 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 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 qualified 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 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.

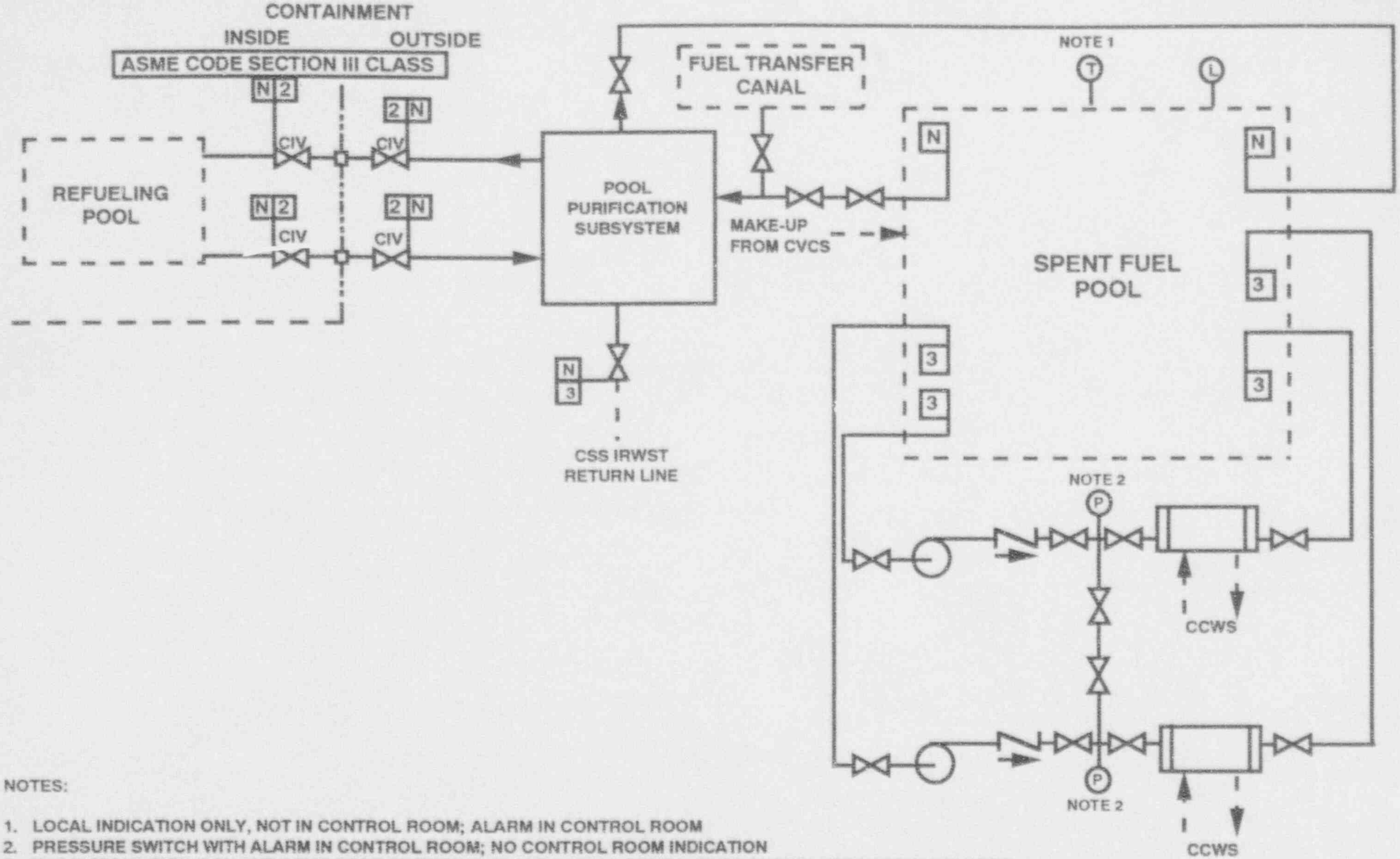
Water is supplied to each SFPCS pump at a pressure greater than the pump's required net positive suction head (NPSH).

The Class 1E loads shown on Figure 2.7.3-1 are powered from their respective Class 1E Division.

The two mechanical Divisions of the SFPCS are physically separated except for the cross-connect line between SFPCS pump discharge lines.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.3-1 specifies the inspections, tests, and analyses and associated acceptance criteria for the PCPS.



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 AND ASME CODE SECTION III CLASS 2 AND 3 COMPONENTS SHOWN ARE SAFETY RELATED. THE PUMPS AND INSTRUMENTATION SHOWN ARE POWERED FROM THEIR RESPECTIVE CLASS 1E DIVISION.

FIGURE 2.7.3 -1
POOL COOLING AND PURIFICATION SYSTEM

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. Inspections 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. Tests to measure SFPCS pump flow in each Division will be performed. Inspections and analyses to determine the heat removal capability of each SFPCS Division will be performed based on test data and as-built data.	2. Each PCPS Division will remove at least 67.25 million btu/hr from the spent fuel pool.
3. The PCPS includes provisions to prevent gravity draining of the spent fuel pool and the refueling pool.	3. Inspections 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 criteria in ASME Code Section III.

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

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.a) Displays of the PCPS instrumentation shown on Figure 2.7.3-1 are available as noted on the figure.	5.a) Inspection for the existence or retrieveability of instrumentation displays will be performed.	5.a) Displays of the instrumentation shown on Figure 2.7.3-1 are available as noted on the figure.
5.b) Controls exist in the Main Control Room to start and stop the spent fuel pool cooling SFP pumps.	5.b) Tests will be performed using the PCPS controls in the Main Control Room.	5.b) PCPS controls in the Main Control Room operate to start and stop the SFP pumps.
5.c) PCPS alarms shown on Figure 2.7.3-1 are provided as shown on the figure.	5.c) Tests of the PCPS alarms shown on Figure 2.7.3-1 will be performed using simulated signals.	5.c) The PCPS alarms shown on Figure 2.7.3-1 actuate in response to simulated signals.
6. Water is supplied to each SFP cooling pump at a pressure greater than the pump's required net positive suction head (NPSH).	6. Tests to measure SFP pump suction pressure will be performed. Inspections and analyses to determine NPSH available to each SFP pump will be performed based on test data and as-built data.	6. The available NPSH exceeds each SFP pump's required NPSH.
7. The Class 1E loads shown on Figure 2.7.3-1 are powered from their respective Class 1E Division.	7. Tests will be performed on the SFPCS system by providing a test signal in only one Class 1E Division at a time.	7. Within the SFPCS, a test signal exists only at the equipment powered from the Class 1E Division under test.
8. The two mechanical Divisions of the SFPCS are physically separated except for the cross-connect line between SFP pump discharge lines.	8. Inspections of as-built mechanical Divisions will be performed.	8. The two mechanical Divisions of the SFPCS are separated by a wall, or by a fire barrier, or by spatial separation in the spent fuel pool, except for the cross-connect line between SFP pump discharge lines.

2.7.8 CONDENSATE STORAGE SYSTEM

Design Description

The Condensate Storage System provides a source of condensate for makeup to the main condenser, is a source of startup feedwater to the steam generators, and provides a non-safety source of condensate to the Emergency Feedwater Storage Tanks.

The Basic Configuration is as shown on Figure 2.7.8-1. The Condensate Storage System is non-safety related.

The Condensate Storage System has a condensate storage tank, a condensate storage tank recycle pump, and associated valves, piping, and controls.

The Condensate Storage System provides makeup or receives excess condensate from the main condenser hotwell. The Condensate Storage System also serves to collect and store condensate from plant condensate drains.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.8-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Condensate Storage System.

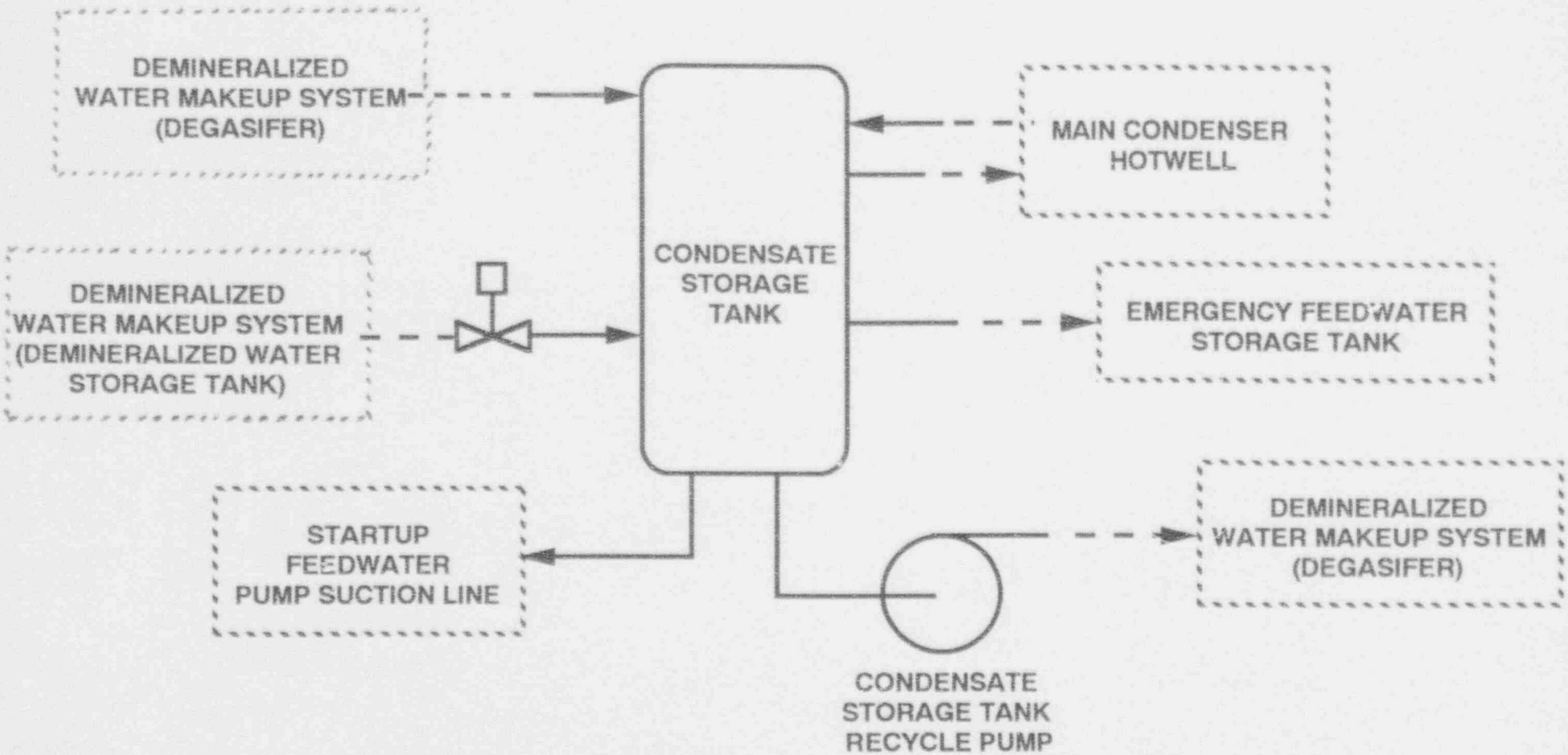


FIGURE 2.7.8-1
CONDENSATE STORAGE SYSTEM

TABLE 2.7.8-1

CONDENSATE STORAGE SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the Condensate Storage System is as shown on Figure 2.7.8-1.	1. Inspection of the as-built Condensate Storage System configuration will be conducted.	1. For the components and equipment shown on Figure 2.7.8-1, the as-built Condensate Storage System conforms with the Basic Configuration.

2.7.10 INSTRUMENT AIR SYSTEM

Design Description

The Instrument Air System (IAS) supplies compressed air to air-operated instrumentation, air-operated controls, and air-operated valves.

The Basic Configuration of the IAS is as shown on Figure 2.7.10-1. The IAS is a non-safety related system with the exception of the containment isolation valves and associated penetration piping.

IAS air compressors, air receivers, and dryer/filters are located in the Nuclear Annex.

The IAS supply lines extend to and end at the controller of the connected component.

Each IAS air compressor shown on Figure 2.7.10-1 is powered from a Permanent Non-safety Bus.

A display of the IAS instrumentation shown on Figure 2.7.10-1 exists in the Main Control Room or can be retrieved there.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.7.10-1 specifies the inspections, tests, analyses and associated acceptance criteria for the IAS.

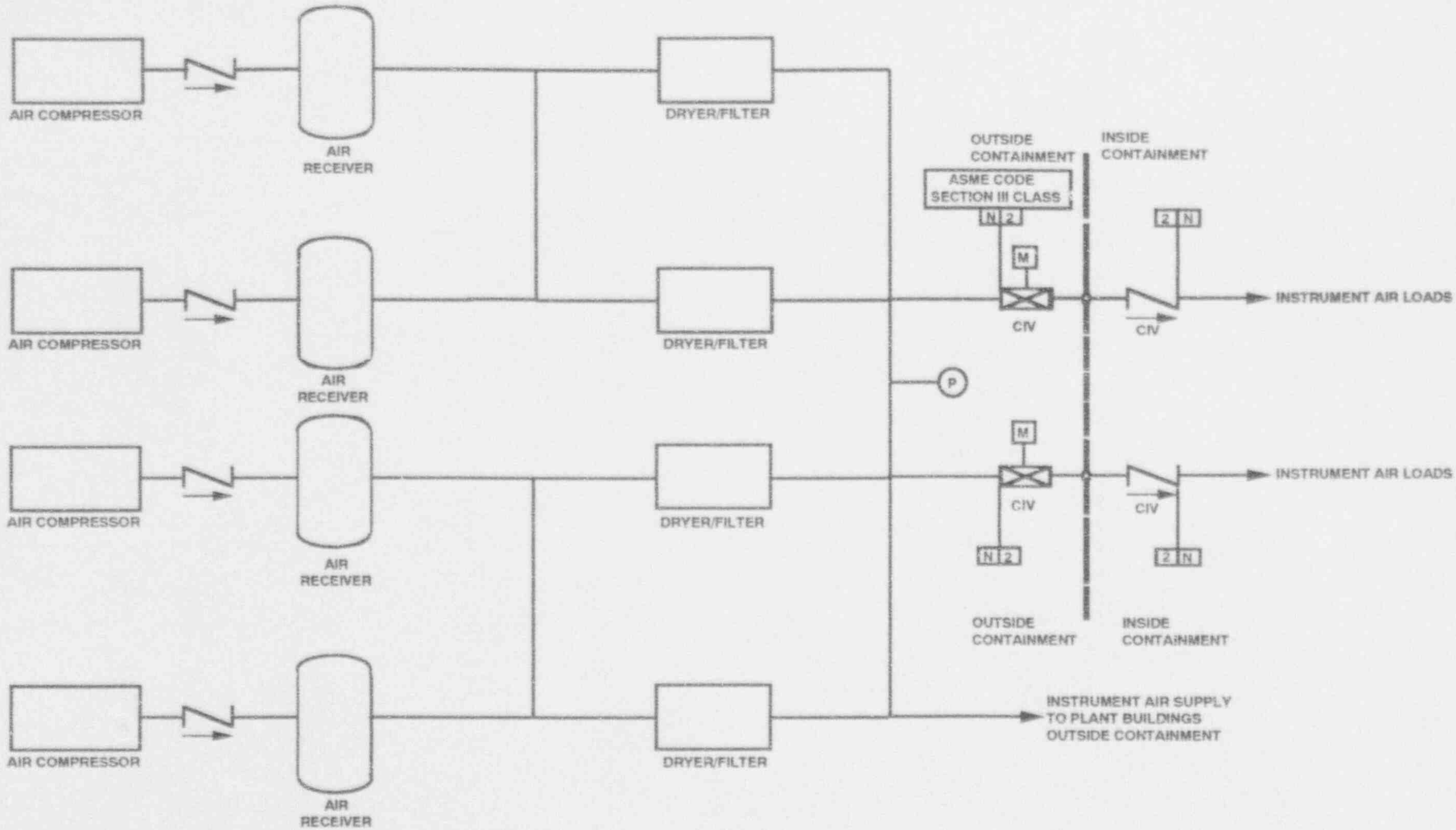


FIGURE 2.7.10-1
INSTRUMENT AIR SYSTEM

TABLE 2.7.10-1

INSTRUMENT AIR SYSTEMS
Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the IAS is as shown on Figure 2.7.10-1.	1. Inspections of the as-built IAS configuration will be conducted.	1. For the components and equipment shown on Figure 2.7.10-1, the as-built IAS conforms with the Basic Configuration.
2. A display of the IAS instrumentation shown on Figure 2.7.10-1 exists in the Main Control Room or can be retrieved there.	2. Inspection for the existence or retrieveability in the Main Control Room of instrumentation displays will be performed.	2. A display of the instrumentation shown on Figure 2.7.10-1 exists in the Control Room or can be retrieved there.
3. The IAS electrical loads shown on Figure 2.7.10-1 are powered from a Permanent Non-safety Bus.	3. Tests will be performed on the IAS by providing a test signal in the Permanent Non-safety Bus.	3. Within the IAS, a test signal exists at the equipment powered by the Permanent non-safety Bus under test.

2.7.11 TURBINE BUILDING COOLING WATER SYSTEM

Design Description

The Turbine Building Cooling Water System (TBCWS) provides cooling water to the non-safety related turbine plant auxiliary system components.

The Basic Configuration of the TBCWS is as shown on Figure 2.7.11-1. The TBCWS is non-safety related.

The TBCWS is a single closed loop cooling water system. The TBCWS has two heat exchangers, two pumps, one surge tank, piping, valves, and controls.

The TBCWS transfers heat from turbine building auxiliary system components to the Turbine Building Service Water System (TBSWS).

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.11-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the TBCWS.

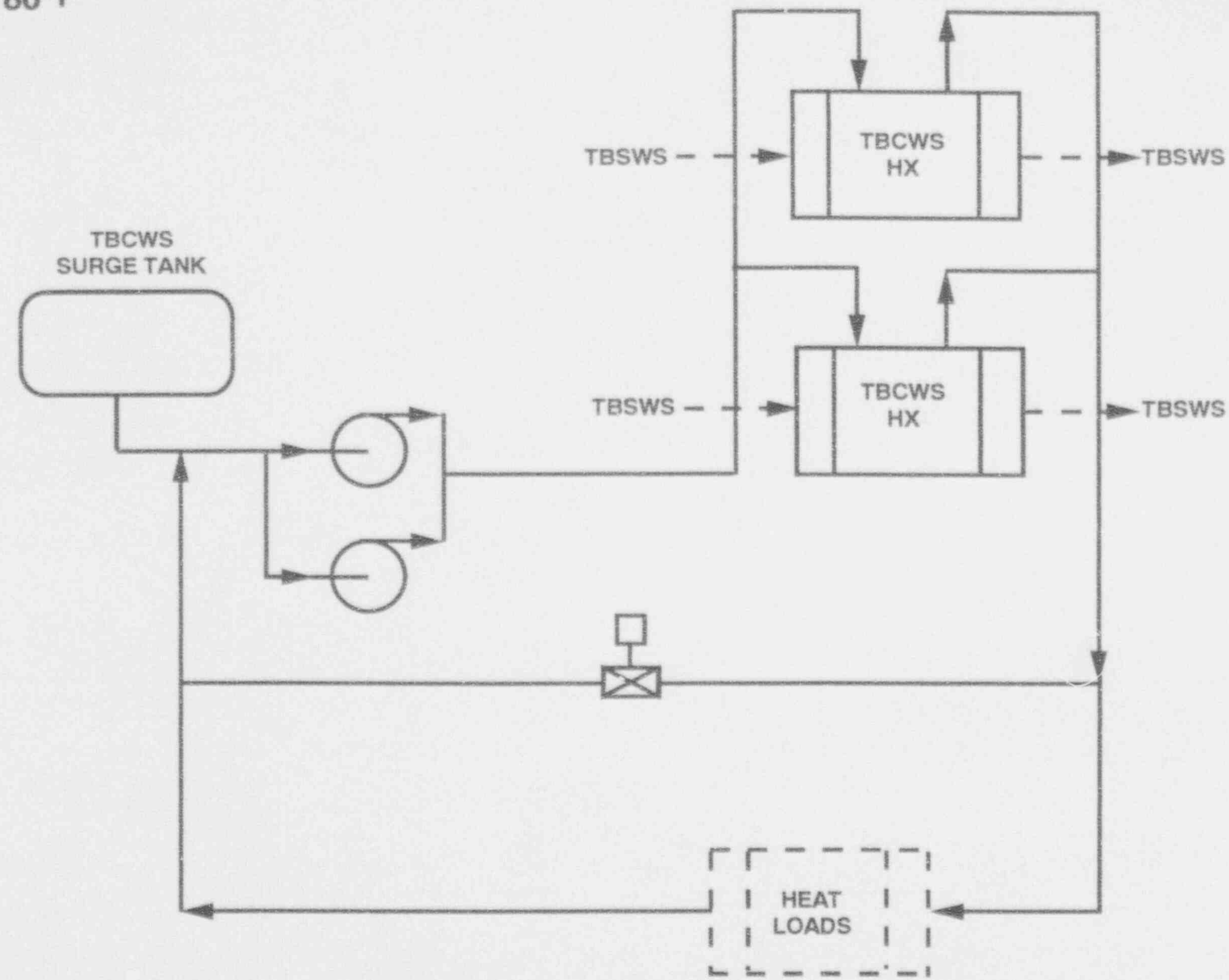


FIGURE 2.7.11-1
TURBINE BUILDING COOLING WATER SYSTEM

TABLE 2.7.11-1

TURBINE BUILDING COOLING WATER SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the TBCWS is as shown on Figure 2.7.11-1.	1. Inspection of the as-built TBCWS configuration will be conducted.	1. For the components and equipment shown on Figure 2.7.11-1, the as-built TBCWS conforms with the Basic Configuration.

2.7.14 TURBINE BUILDING SERVICE WATER SYSTEM

Design Description

The Turbine Building Service Water System (TBSWS) removes heat from the Turbine Building Cooling Water System (TBCWS) and transfers heat to the Condenser Circulating Water System.

The Basic Configuration of the TBSWS is as shown on Figure 2.7.14-1. The TBSWS is non-safety related.

The TBSWS has two pumps and associated piping, valves, and controls which provide cooling water to the TBCWS heat exchangers.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.14-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Turbine Building Service Water System.

SYSTEM 80+™

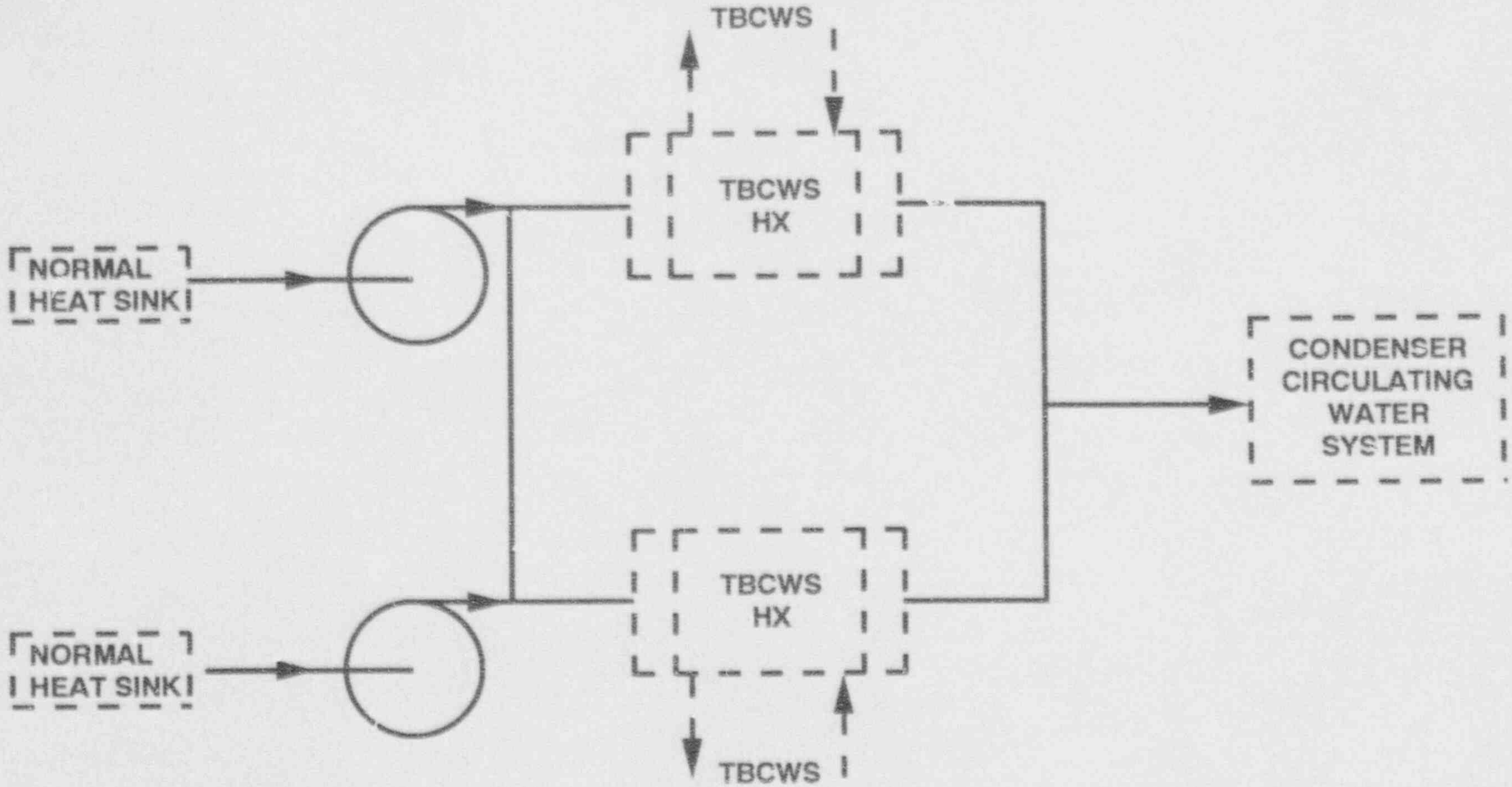


FIGURE 2.7.14-1
TURBINE BUILDING SERVICE WATER SYSTEM

TABLE 2.7.14-1

TURBINE BUILDING SERVICE WATER SYSTEM
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the Turbine Building Service Water System (TBSWS) is as shown on Figure 2.7.14-1.	1. Inspection of the as-built TBSWS configuration will be conducted.	1. For the components and equipment shown on Figure 2.7.14-1, the as-built TBSWS conforms with the Basic Configuration.

2.7.27 COMPRESSED GAS SYSTEMS

Design Description

The Compressed Gas Systems (CGS) are non-safety related systems which supply gases to equipment and instrumentation for cooling, purging, diluting, inerting, and welding. The CGS consists of high pressure gas cylinders and pressure regulators to control the pressure and distribution of the compressed gases. The compressed gas systems consist of some or all of the following separate subsystems:

A. N_2 System

B. H_2 System

C. O_2 System

D. CO_2 System

E. Argon/Methane System

F. Acetylene System

G. Argon System

The CGS gas cylinders are located in areas which contain no safety-related equipment.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.7.27-1 specifies the inspections, tests, analyses and associated acceptance criteria for the CGS.

TABLE 2.7.27-1

COMPRESSED GAS SYSTEMS
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The CGS gas cylinders are located in areas which contain no safety-related structures, systems, and components.	1. Inspections of the as-built plant arrangement will be performed.	1. The CGS gas cylinder storage areas contain no safety-related structures, systems, or components.

2.7.28 POTABLE AND SANITARY WATER SYSTEMS

Design Description

The Potable and Sanitary Water Systems (PSWS) are non-safety systems that provide process water for general plant use.

The Potable and Sanitary Water Systems are not within the scope of the certified design.

Interface Requirement

There are no interconnections between the Potable and Sanitary Water Systems and systems having the potential for containing radioactive material.

Inspections, Tests, Analyses, and Acceptance Criteria

None

2.8.9 CONDENSER CIRCULATING WATER SYSTEM

Design Description

The Condenser Circulating Water System is a non-safety interface system that provides cooling water for the turbine condensers and transfers heat from the Turbine Building Service Water System to the normal heat sink.

2.12.1 MAIN CONTROL ROOM [1]

Design Description

The Main Control Room (MCR) permits execution of MCR tasks to operate the plant and maintain plant safety. The MCR provides suitable workspace for continuous occupancy and use by MCR operators. 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.

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).

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	Auxiliary and Safety Consoles
Reactor Coolant System	Heating, Ventilation & Air Conditioning
Chemical & Volume Control System	Cooling Water Systems
Plant Monitoring & Control	Engineered Safety Features
Feedwater & Condensate Systems	Safety Monitoring
Turbine Control	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 MCR.

[1] (Nuclear Island Structures, ventilation, fire protection, communications, lighting, radiation protection and control panels are addressed in Sections 2.1.1, 2.7.17, 2.7.24, 2.7.25, 2.7.26, 3.2 and 2.12.3, respectively.)

TABLE 2.12.1-1

MCR MINIMUM INVENTORY

PARAMETER DESCRIPTION			
	Displays	Annunciators[1]	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 Diese· 3enerator 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	
Contrinment pressure	X	X	
Containment radiation		X	
CSAS actuation		X	X
Containment Spray flow	X		
Containment Spray pump on/off	X		X

TABLE 2.12.1-1 (Continued)

MCR MINIMUM INVENTORY

PARAMETER DESCRIPTION	Displays	Annunciators[1]	Controls
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	
Hot Leg inj valve position	X		X
IRWST level		X	
Main Control Room HVAC isolation dampers	X		X
Main Steam line area radiation monitor		X	
Main Steam safety valve position		X	
MSIS actuation	X	X	X
Nuclear Annex building ventilation radiation monitor		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 monitor		X	
Reactor Coolant gas vent valve position	X		X
Reactor power (NI)	X		
Reactor Trip (RPS)		X	X

TABLE 2.12.1-1 (Continued)

MCR MINIMUM INVENTORY

PARAMETER DESCRIPTION	Displays	Annunciators[1]	Controls
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 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 monitor		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.

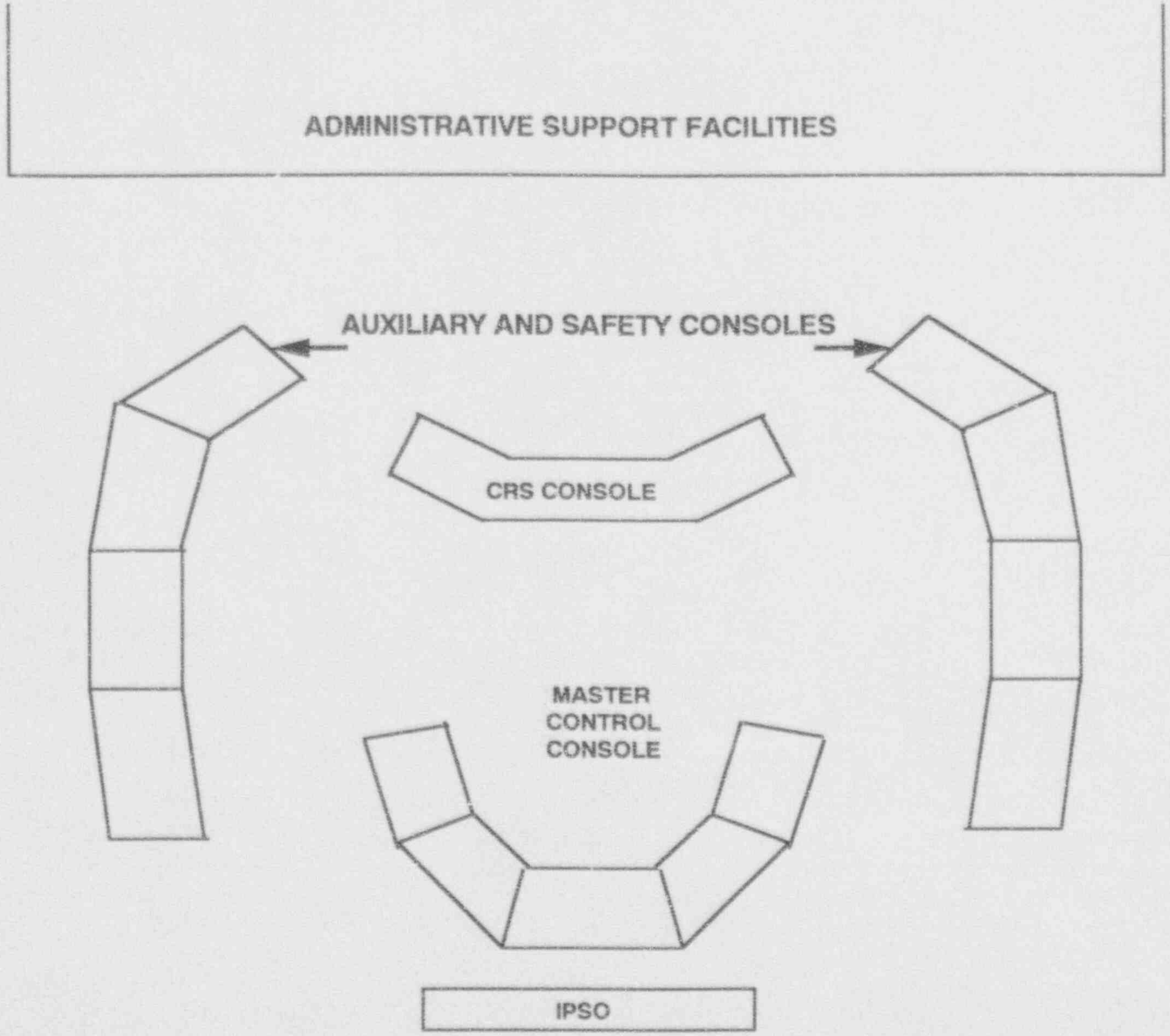


FIGURE 2.12.1-1 MAIN CONTROL ROOM

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 Main Control Room is as shown on Figure 2.12.1-1.	1. Inspections 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 Main Control Room conforms with the Basic Configuration.
2. The MCR makes available the annunciators, displays and controls identified in Table 2.12.1-1.	2. Human Factor 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 for continuous occupancy and use by MCR operators.	3. HFE suitability inspection against verification criteria will be performed.	3. Inspections demonstrate HFE suitability of the MCR workspace.
4. The MCR permits execution of MCR tasks to operate the plant and maintain plant safety.	4. Tests and analyses against the validation criteria using a fullsize dynamic mockup of the MCR consoles that simulates plant operational responses will be performed.	4. The test and analysis results demonstrate validation of MCR task execution.

2.12.2 REMOTE SHUTDOWN ROOM [1]

Design Description

The Remote Shutdown Room (RSR) permits execution of RSR tasks to place and maintain the plant in a safe shutdown condition. The RSR provides suitable workspace separate from the Main Control Room (MCR) for use by MCR operators in the event that the MCR becomes uninhabitable. The RSR makes available the annunciators, displays, and controls to shutdown the plant and maintain plant safety.

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 MCR operators perform RSR operations.

The RSR is located in the Nuclear Annex within fire and ventilation isolation boundaries.

Inspection, Test, Analyses, and Acceptance Criteria

Table 2.12.2-1 specifies the inspections, tests, analyses and acceptance criteria for the RSR.

[1] (Nuclear Island Structures, ventilation, fire protection, communications, lighting, radiation protection, and control panels are addressed in Sections 2.1.1, 2.7.17, 2.7.24, 2.7.25, 2.7.26, 3.2, and 2.12.3, respectively.)

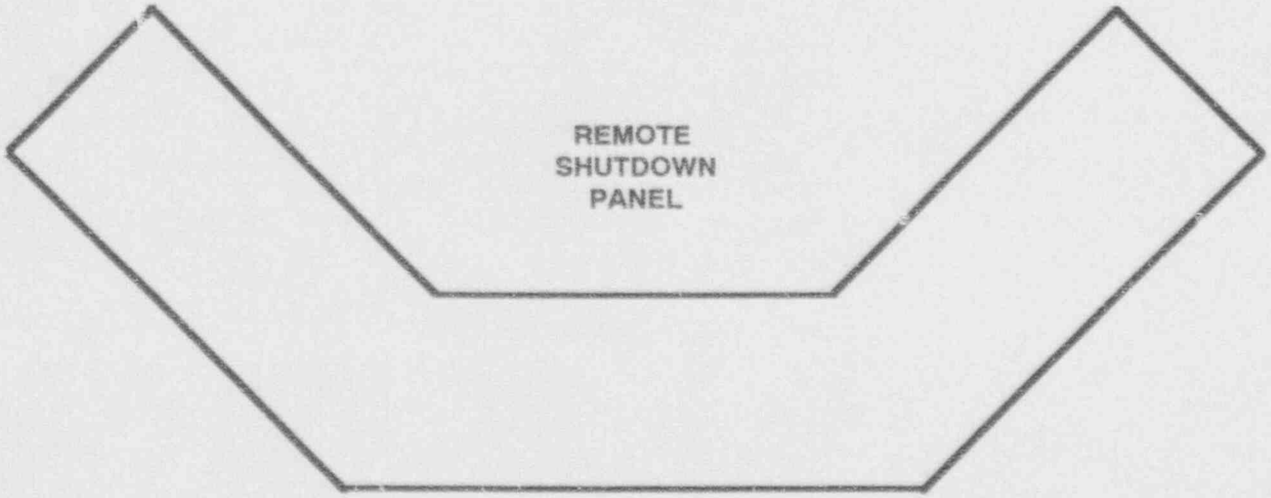


FIGURE 2.12.2-1 REMOTE SHUTDOWN ROOM

TABLE 2.12.2-1
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. Inspections 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 to shutdown the plant and maintain plant safety.	2. Human Factor 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 has capability for subsequent cold shutdown of the reactor through the use of suitable procedures.
3. The RSR provides suitable workspace for use by RSR operators.	3. HFE suitability inspection against verification criteria will be performed.	3. Inspections demonstrate HFE suitability of the RSR workspace.
4. The RSR permits execution of RSR tasks to shutdown the plant and maintain plant safety.	4. Tests and analyses against the validation criteria using a fullsize dynamic mockup of the Remote Shutdown Panel that simulates plant operational responses will be performed.	4. The test and analysis results demonstrate validation of RSR task execution.

2.12.3 CONTROL PANELS [1]

Design Description

Control Panels use standard features to provide suitable operator workstations for annunciators, displays and controls. Each Control Panel permits execution of the Control Panel's tasks to operate the plant and maintain plant safety.

The Basic Configuration of a Control Panel is as shown on Figure 2.12.3-1.

Control panels are provided in the Main Control Room (Section 2.12.1) and the Remote Shutdown Room (Section 2.12.2).

Control Panels with Class 1E instrumentation are qualified Seismic Category I.

Inspection, Test, Analyses, and Acceptance Criteria

Table 2.12.3-1 specifies the inspections, tests, analyses, and associated acceptance criteria that will be applied to each Control Panel.

[1] (Instrumentation and Control Systems are addressed in Section 2.5.)

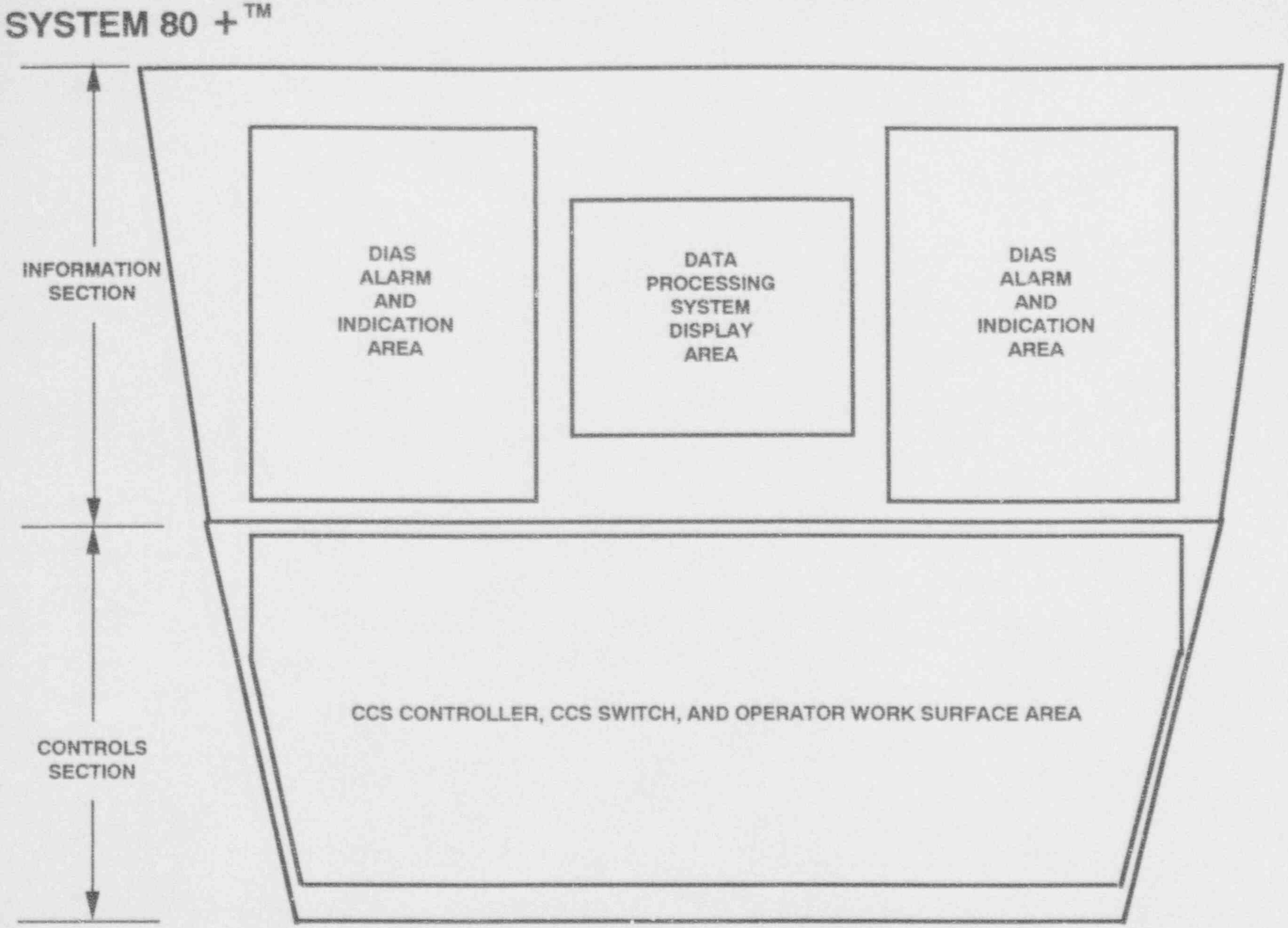


FIGURE 2.12.3 - 1
CONTROL PANEL LAYOUT CONFIGURATION

TABLE 2.12.3-1

CONTROL PANELS
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the Control Panel is as shown on Figure 2.12.3-1.	1. Inspections of each as-built Control Panel configuration will be conducted.	1. For the components and equipment shown on Figure 2.12.3-1, the as-built Control Panels conform with the Basic Configuration.
2. Control Panels provide suitable operator workstations.	2. Human Factor Engineering (HFE) suitability inspection against verification criteria will be performed.	2. Inspections demonstrate Human Factor Engineering (HFE) suitability of each Control Panel workstation.
3. Each Control Panel permits execution of the Control Panel's tasks to operate the plant and maintain plant safety.	3. Task Analysis of Control Panels tasks will be performed.	3. The Task Analysis results demonstrate that each Control Panel permits execution of the Control Panel's tasks.