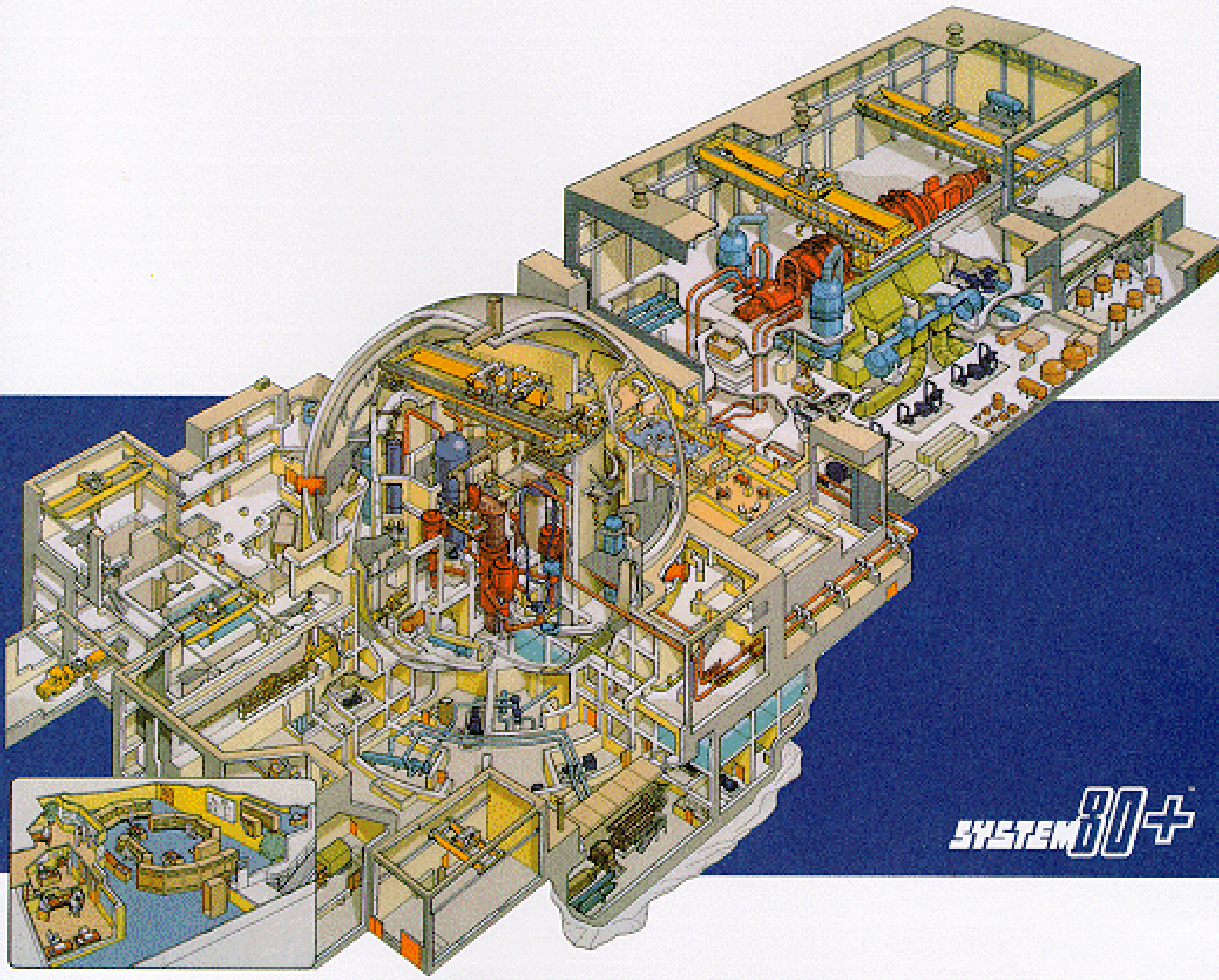
DOE/SF/19682--T1-Vol.1

DOE Plutonium Disposition Study
Pu Consumption in ALWRs

Contract No. DE-AC03-93 SF19682



A Final Report

by

ABB-Combustion Engineering
Windsor, Connecticut

May 15, 1993

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DOE/SF/19682--T1-Vol.1

DOE PLUTONIUM DISPOSITION STUDY

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ASEA BROWN BOVERI

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LIST OF ACRONYMS AND TERMINOLOGY

Generally, terms and abbreviations are defined at the time they are first introduced in the following report. The more commonly found terms and abbreviations which are appropriate to the System 80+™ station design and Plutonium Disposition Study are compiled below as a convenient reference for the reviewer. In the interest of practical application, note very possible scientific term or abbreviation is listed.

ABB-CE	Asea Brown Boveri-Combustion Engineering
ADV	Atmospheric Dump Valve
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
APR	All Plutonium Reactor
APS	Alternate Protection System
ATWS	Anticipated Transient Without Scram
AVS	Annulus Ventilation System
BNWL, BNL	Battelle Northwest Laboratories
BPR	Burnable Poison Rod
CAS	Central Alarm Station
CBC	Critical Boron Concentration
CCTV	Closed Circuit Television
CCW(S)	Component Cooling Water (System)
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
CEG	Cost Estimating Guidelines
CIS	Commonwealth of Independent States
CM	Corrective Maintenance
CPC	Core Power Calculator
CRT	Cathode Ray Tube
CS	Containment Spray (System)
CSAS	Containment Spray Actuation Signal
CSB	Core Support Barrel
CVCS	Chemical and Volume Control System
D&D	Decontamination & Decommissioning
DE&S	Duke Engineering and Services
DIAS	Discrete Indication and Alarm System
DIT	Discrete Integral Transport
DNB	Departure from Nucleate Boiling
D-O	Destruction Deployment Option
DPS	Data Processing System
DVI	Direct Vessel Injection
ECA	Energy Conversion Area
EFPD	Effective Full Power Days
EFW(S)	Emergency Feedwater (System)
EFWST	Emergency Feedwater Storage Tank
EOP	Emergency Operating Procedures
EPRI	Electric Power Research Institute
ESFAS	Emergency Safety Features Actuation Signal

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LIST OF ACRONYMS AND TERMINOLOGY (Continued)

ESW(S)	Essential Service Water (System)
EWG	Electric Wholesale Generator
FEA	Fuel Element Assembly
FMEF	Fuels and Material Examination Facility
FPF	Fuel Pin Fabrication
FRS	Fuel Receiving and Storage
FTC	Fuel Temperature Coefficient
FTFF	Fuel and Target Fabrication Facility
GVR	Gas-to-Volume Ratio
GWD	Gigawatt-Days
GWMS	Gaseous Waste Management System
HACTS	Head Area Cable Tray Structure
HEPA	High Efficiency Particulate Air (Filter)
HJTC	Heated Junction Thermocouple
HPSI	High Pressure Safety Injection
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation and Air Conditioning
IAEA	International Atomic Energy Agency
ICI	In-Core Instrumentation
ID	Intrusion Detection
INPO	Institute of Nuclear Power Operations
IPP	Independent Power Producer
IPSO	Integrated Process Status Overview
IRWST	In-Containment Refueling Water Storage Tank
ITAAC	Inspection, Test, Analysis Acceptance Criteria
LCO	Limiting Conditions of Operation
LDB	Licensing Design Basis
LOCA	Loss of Coolant Accident
LTOP	Low Temperature Over Pressurization
LWMS	Liquid Waste Management System
MAA	Material Access Area
MFIV	Main Feedwater Isolation Valve
MOPS	Moisture Preseparators
MOX	Mixed Oxide
MRS	Material Receiving and Storage
MSIV	Main Steam Isolation Valve
MSSA	Master Safeguards and Security Agreement
MST	Multiple Stud Tensioner
MT, MTU	Metric Ton (Uranium)
MTC	Moderator Temperature Coefficient
MWD	Megawatt-Days
MW(t)	Megawatt-Thermal
NEPA	National Environmental Policy Act
OBE	Operating Basis Earthquake
O&M	Operation and Maintenance
ONM	Other Nuclear Materials

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LIST OF ACRONYMS AND TERMINOLOGY (Continued)

PBRC	Plutonium Burner Reactor Complex
PBRF	Plutonium Burner Reactor Facility
PDR	Plutonium Disposition Reactor
PDS	Plutonium Disposition Study
PM	Preventive Maintenance
PRA	Probabilistic Risk Assessment
PRF	Permeation Reduction Factor
PSN	Project Summary Network
PVNGS	Palo Verde Nuclear Generating Station
RCB	Reactor Containment Building
RCM	Reliability Centered Maintenance
RCP	Reactor Coolant Pump
RCGV	Reactor Coolant Gas Vent
RCS	Reactor Coolant System
RD	Requirements Document
RDT	Reactor Drain Tank
RPCS	Reactor Power Cutback System
RPS	Reactor Protection System
RSPT	Reed Switch Position Transmitter
SAF	Secure Automated Fabrication
SAS	Secondary Alarm Station
SBO	Station Black Out
SC	Shutdown Cooling (System)
SCRUPS	Special Cross-Under Pipe Separators
SCV	Steel Containment Vessel
SDS	Safety Depressurization System
SF-O	Spent Fuel Deployment Option
SF-1	Spent Fuel Deployment Option - 1 Reactor
SF-2	Spent Fuel Deployment Option - 2 Reactors
SFS	Spent Fuel Storage
SG	Steam Generator
SGR	Self Generated Recycle
SI(S)	Safety Injection (System)
SIT	Safety Injection Tank
S-O	Spiking Deployment Option
SMB	Safety Margin Basis
SNM	Special Nuclear Materials
SRS	Savannah River Site
SSE	Safe Shutdown Earthquake
SSSP	Site Safeguards and Security Plan
SWEC	Stone and Webster Engineering Corporation
SWMS	Solid Waste Management System
T&Q	Training and Qualification (Program)
TTDP	Tritium Target Development Program
UGS	Upper Guide Structure
URD	Utility Requirements Document

LIST OF ACRONYMS AND TERMINOLOGY (Continued)

WANO	World Association of Nuclear Operators
WNP, WNPP	Washington Nuclear Power Project

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FOREWORD

This report is organized into seven major sections comprising two volumes and follows the general organization of topics given in the DOE Plutonium Disposition Study Requirements Document. A cross-reference of the Requirements Document and areas of study in this report are presented in Table 1 and Table 2, which follow.

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Table 1

Pu Consumption Requirements Cross-Reference

Requirements Document	Final Report Section
4.0 Information	
4.1 Safeguards	III.J
4.2 Safety and Environmental	II.I, II.J, V
4.3 Technology Base	II.A through G, TABLE II-1, III.A through I
4.4 Infrastructure	III.H, VII.A through C
4.5 Costs	VI.A through E
4.6 Schedules	IV, VI.F
5.0 Other Considerations	
5.1 Deployment Strategies	VII.A, B, C
5.2 Challenges	VII.D

Table 2

Pu Consumption - Display of Engineering, Technical and Cost Data: Cross-Reference to Report

Requirements Document			Final Report	
FIGURE NO.	**DESCRIPTION**	**RD NUMBER**	**REPORT SECTION**	**FIGURE OR TABLE**
A-1	Occupational Exposure vs Time	4.2.3	III.K	Table III.K-1
A-2	MW_t Installed Capacity vs Time	4.3.1.2	III.K	Table III.K-2
A-3	Kg of Plutonium Stockpile Reduction is Time	4.3.1.4	III.K	Table III.K-3
A-4	KW_{hrs} Electricity Produced vs Time	4.3.1.5	III.K	Table III.K-4
A-5	Kg of Strategic Material vs Time	4	III.K	Table III.K-5
A-6	Input Kg Fresh Fuel vs Time	4.3.1.4	III.K	Table III.K-6
A-7	On-Site Spent Fuel Storage vs Time	4.3.5.3	III.K	Table III.K-7
A-8	Overall Radioactive Waste Generated vs Time	4.3.5.5	III.K	Table III.K-8
A-9	Output, Kg - Actinide vs Time	4.3.5.5	III.K	Table III.K-9
A-10	Output, Kg - Fission Product vs Time	4.3.5.5	III.K	Table III.K-10
A-11	Cash Flow vs Time	4.5.3	VI.B	Figures VI.B-1, 3, 5, 7, 9
A-12	R&D Expenditure vs Time	4.5.1.1	VI.B	Figures VI.B-11, 12
A-13	Pre-Operating Costs vs Time	4.5.1.1	VI.B	Figures VI.B-13, 14, 15, 16
A-14	Capital Costs vs Time	4.5.1.2	VI.B	Figures VI.B-2, 4, 6, 8, 10
A-15	O&M Costs vs Time	4.5.1.3	VI.C	Table VI.C-1
A-16	Revenues from Sale of Electric Power	4.5.1.5	VI.E	Tables VI.E-1 through 5

I. EXECUTIVE SUMMARY

The Department of Energy (DOE) has contracted with Asea Brown Boveri-Combustion Engineering (ABB-CE) to provide information on the capability of ABB-CE's System 80+ Advanced Light Water Reactor (ALWR) to transform, through reactor burnup, 100 metric tonnes (MT) of weapons grade plutonium (Pu) into a form which is not readily useable in weapons. This information is being developed as part of DOE's Plutonium Disposition Study, initiated by DOE in response to Congressional action.

ABB-CE has undertaken this study supported by sub-contractors Battelle Northwest Laboratories, Duke Engineering & Services, Inc., Ebasco Services Inc. and Stone & Webster Engineering Corporation. This team with a broad base of nuclear experience and active working relationships with ABB-CE on the ALWR and on the New Production Reactor Program has been able to rapidly mobilize and conduct this study guided by the DOE Requirements Document.

Table I-1 summarizes the results of the studies performed to-date. The table summarizes the three requested alternatives (spiking, spent fuel and destruction) in compliance with the Requirements Document and on two alternates to demonstrate the capability of a single and dual System 80+ installation. We believe this demonstrates an excellent capability and flexibility of the System 80+ ALWR design to meet and exceed mission objectives.

The single System 80+ plant shown as the Optimized Spent Fuel - 1 in Table I-1 shows a seven year schedule to start of operation, a full Mixed-Oxide (MOX) loading, a capability of handling the 100MT to spiking in 15 years and spent fuel in 60 years. The dual System 80+ plant has a seven year schedule for the first unit, a full MOX loading, a capability of handling the 100MT to spiking in 7.5 years and spent fuel in 30 years.

The following missions outlined in the Requirements Document; Plutonium disposition, electric power production, and tritium production capability are all within the capability of System 80+ and will require only minimum modifications. The Pu disposition capability exists primarily because System 80, from which the System 80+ ALWR evolved, was designed in the 1970's with a reactor capable to accept a 100% Mixed Oxide (MOX) fuel loading in what was referred to as an All Plutonium Reactor (APR). System 80 and System 80+ represent the only existing designs for a 100% MOX core with the resulting capability three times (3x) that of other Light Water Reactor designs.

The design of the System 80+ ALWR, a 3931MWt (1350MWe) Evolutionary Advanced Pressurized Water Reactor is a complete nuclear plant design which is in the final stages of licensing review by the Nuclear Regulatory Commission (NRC). The Nuclear Steam Supply System is a traditional two-loop arrangement with two steam generators, two hot legs and four cold legs each with a reactor coolant pump. The steam generators and pressurizer have been increased in size to provide additional primary and secondary side inventory with the resulting increases in operating margins, thus decreasing potential challenges to the safe operation of the plant. The reactor design of System 80+ has had minimum change since the original System 80 design that is currently in operation at the Palo Verde three unit site in Arizona. One of the original design objectives of System 80 was the capability to use large Pu loadings - up to and including an APR core. The features which were designed into System 80 plants to accommodate an all Pu core included increased control rod worth; greater boration capability; increased decay heat removal capability; and improved reactor internals, core support barrel, and fuel handling and storage equipment. The most challenging design consideration was the provision for an increased control rod worth. The need for additional control rod worth motivated a unique design feature of the System 80 reactor - the tubesheet upper guide structure. The System 80 design, with its upper guide structure support barrel and lower tube sheet shrouds for individual control rods, results in control rod access to all assemblies of the reactor core. This innovative concept was developed, tested, designed, manufactured, installed and has been in successful operation for over eight years at the Palo Verde site.

The System 80+ ALWR has been under development at ABB-CE since 1986 as a part of the DOE design certification program. The complete 25 volume CESSAR-DC (Combustion Engineering Standard Safety Analysis Report - Design Certification), has been undergoing review by the NRC since 1987. The draft Safety Evaluation Report was issued in October 1, 1992 and a major step towards closure of all issues was the submittal by ABB-CE of over 8800 amended pages in March, 1993. The status of System 80+ is best reflected in a recent letter from Thomas E. Murley of the NRC who states "Due to extensive amount of information in the current safety analysis report and the DSER, the staff considers the DSER for System 80+ essentially equivalent to the draft Final Safety Evaluation Report (DFSER) for the General Electric Company Advanced Boiling Water Reactor design. The staff is confident that once the remaining open items are resolved and once the Tier 1 design description and ITAAC are submitted, the staff will be able to document its findings in a FSER and to issue a final design approval within a reasonable time." (A copy of this letter is included at the end of this section.)

Since the System 80 reactor design is proven in service at Palo Verde, and the System 80+ is nearing completion of NRC review with no major open tasks, it is an excellent base for the Pu and tritium missions. This report documents the studies that ABB-CE has completed which validate the capability of System 80+ to accommodate a 100% MOX fuel loading. ABB-CE continues to use conservative groundrules, such as maintaining critical core coefficients within the limits proven in operation at Palo Verde, to maintain the well proven and documented design basis. ABB-CE has incorporated recent fuel developments such as the use of Erbium burnable poison, in its MOX fuel design. This results in excellent core characteristics well within those whose performance has been proven. The use of Erbium up to 2.5 weight percent has been approved for ABB-CE reactors by NRC.

The various core loadings reviewed for this Pu disposition study demonstrate the flexibility built into the System 80+ reactor. The spiking, spent fuel and destruction alternatives as well as the option to produce tritium can be accommodated with only minor modifications to the design. The basic configuration for a uranium core is modified only by burnable poisons and by tritium targets to accomplish the various missions.

The MOX fuel fabrication required for the Pu mission is mechanically similar to that for uranium fuel with the added considerations of safeguards and radiological hazards of Pu. The designed, manufactured and constructed Secure Automated Fabrication (SAF) line is an existing fuel fabrication line within the Fuel Material Examination Facility (FMEF) at

Hanford. While this facility is expandable, it may be limited for the spiking mission. Should the location be other than Hanford (e.g., Savannah River), transportation aspects would have to be reviewed from a safeguards and economics viewpoint. However, FMEF does represent a valuable resource as input to studies and design of a MOX fuel Fabrication line and facility.

The spent fuel handling, storage and ultimate disposal are similar to these activities for a commercial LWR. For all the alternates and deployment option that have been examined, wet storage for fifteen full cores is contemplated. At the MOX loading of 6.7% Pu fissile, the 100MT would be accommodated for the life of the plant with no other fuel required to fuel the plant. From a safeguards viewpoint once the fuel is fabricated and delivered to the reactor plant it does not have to be shipped off-site until later. When the fuel is removed it can be shipped and disposed of in the High Level Waste Repository similar to spent fuel from a commercial LWR.

The basic conservative, flexible design of the System 80+ reactor allows it to accommodate tritium targets without any significant re-design. The tritium targets developed for DOE under the Light Water New Production Reactor program can be substituted for burnable poison rods to produce the desired rates of tritium. Several cases have been explored with 32 tritium target rods per assembly producing contract goal tritium. With the 32 targets per assembly, the power output would decrease slightly to 3410 MW thermal. A reduced number of tritium targets results in a higher power capability.

The System 80+ design is the only Evolutionary Advanced <u>Pressurized</u> Light Water Reactor (PWR) in the United States. Evolutionary Pressurized Water Reactors have been adapted worldwide as the plants for the next generation of power reactors. France, Germany and Korea have selected <u>PWR</u> technology based on every measure of comparison with other reactor types (i.e., load factor, forced outage rate, waste produced, radiation exposure, O&M costs etc.) System 80+ has been designed to incorporate improvements based on the detailed criteria that worldwide utilities established for new plants, the EPRI ALWR Utility Requirements Document. Feedback from operating experience and full compliance with the US NRC regulations have resulted in more than 5500 separate PWR requirements included in the ALWR utility requirements document. These requirements embody four main ideas: safety, simplicity, ample design margin and full integration of the human operator.

As an indication of its <u>immediate</u> commercial viability, it should be noted that the System 80+ standard design is being bid by ABB-CE in Taiwan this year.

This study and efforts put into the Deployment Options provides encouragement that the potential, for private investment in all or portions of the Complex is realistic and should be pursued. While interest in the private sector is dependent upon the need for power, power distribution, and ability to sell the power, it is also a function of the risk involved. System 80+ offers to investors a design that is licensable, known, mature and with a capacity factor that can be <u>defended</u> based on proven experience, all of which are so vital to financial commitment.

In summary, the proven operating record of the System 80 design, safety enhancements for System 80+™ that have resulted in a licensing application review almost completed by NRC, close adherence to the ALWR Utility Requirements document, design capabilities for Pu applications and the high 3931 MW thermal rating of the plant, all combine to make System 80+ the least cost, least-risk, fastest way to have a Pu disposition reactor on-line to meet the Pu disposal mission.

TABLE I-1

SUMMARY COMPARISON OF ABB-CE SYSTEM 80+ PLUTONIUM DISPOSITION OPTIONS

	SPIKING	SPENT FUEL	DESTRUCTION	OPTIMIZED SPENT FUEL-1	OPTIMIZED SPENT FUEL-2
Reactor Type	APWR	APWR	APWR	APWR	APWR
Lead Designer	ABB-CE	ABB-CE	ABB-CE	ABB-CE	ABB-CE
MWe/Reactor	1350 Gross 1256 Net	1350 Gross 1256 Net	1350 Gross 1256 Net	1350 Gross 1256 Net	1350 Gross 1256 Net
MWt/Reactor	3800	3800	3800	3800	3800
Number of Reactor Modules per Plant	1	1	1	1	1
Number of Plants Required (or Reactors)	1	4	4	1	2
Coolant Type	Water	Water	Water	Water	Water
Moderator Type	Water	Water	Water	Water	Water
Size of Site Area (Acres)	200	500	500	200	300
Cycle Exposure EFPD	39	274	274	274	274
Cycle Length (Days)	90	365	365	365	365
Percent Core Refueled per Cycle	100	100	100	100	100
Capacity Factor (%)	43	75	75	80	90
Discharge Exposure (MWD/MT)	1500	42,200	42,200	45,000	45,000
Average Occupational Exposure (P-Rem/YR)	200	364	364	97	194

TABLE I-1 (Continued)					
SUMMARY COMPARISON OF ABB-CE SYSTEM 80+ PLUTONIUM OPTIONS					
	SPIKING	SPENT FUEL	DESTRUCTION	OPTIMIZED SPENT FUEL-1	OPTIMIZED SPENT FUEL-2
Timeframe to Start of Operations (YRS)					
a) Fuel Fabrication	6	6	6	6	6
b) Reactor	7	7	7	7	7
Req'd Fuel Fabrication Rate MT Pu/YR	25.67 4 Core/YR	6.67 1 Core/YR	6.67 1 Core/YR	1.67-6.67 0.25 to 1 Core/YR	3.33-6.67 0.5 to 1 Core/YR
Spent Fuel Storage Capacity Req'ts. Cells	5000	5000	5000	1250	2500
Pu Stockpile Reduction (MT/YR) (Normal Operations)	20.00	6.67	6.67	1.67-6.67	3.33-6.67
Pu Stockpile Disposition Duration (YRS) (After Start of Operations)	4	18	18	60	30
Net Pu Destruction (%)	4	27	61	27	27
Cost of Construction and Engineering ($) Millions	3,720	9,704	9,704	3,470	5,570
Avg. Operational Cost Data ($/YR)					
a) O&M Costs (Nonfuel)	97.50 M	243.00 M	243.00 M	92.8 M	126.2 M
b) Revenue Generated	160 M	1.300 M	1.300 M	323 M	640 M

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 6, 1993

Docket No. 52-002

Mr. Regis Matzie, Vice President
Nuclear Systems Development
ABB-Combustion Engineering
1000 Prospect Hill Road
Windsor, Connecticut 06095-0500

Dear Mr. Matzie:

SUBJECT: THE NUCLEAR REGULATORY COMMISSION (NRC) REVIEW STATUS OF THE ABB-COMBUSTION ENGINEERING (ABB-CE) SYSTEM 80+ DESIGN

The NRC staff released its draft safety evaluation report (DSER) on the ABB-CE System 80+ design to the public on October 1, 1992. This DSER identified 637 open items that required additional information from ABB-CE for the staff to complete its review. In addition, it was noted that ABB-CE had not yet submitted the Tier 1 design description and the inspections, tests, analyses, and acceptance criteria (ITAACs).

Both ABB-CE and the NRC staff have made significant progress since the DSER was issued. Of the 637 open items, 119 have been closed and completely resolved, 234 have attained technical resolution and are now classified as confirmatory (i.e., ABB-CE must submit the agreed to resolution formally on the docket or as an amendment to CESSAR-DC). Thus there are 284 open items. The majority of these open items require only additional NRC staff review and no further technical work by ABB-CE. In addition, significant progress is being made on the development of acceptable Tier 1 design descriptions and ITAAC through ABB-CE meetings with the staff and ABB-CE's participation in industry activities. None of the remaining open items appears to present an obstacle to final design approval and eventual certification rulemaking of the ABB-CE System 80+ design.

Due to the extensive amount of information in the current safety analysis report and the DSER, the staff considers the DSER for System 80+ essentially equivalent to the draft final safety evaluation report (DFSER) for the General Electric Company Advanced Boiling Water Reactor design. The staff is confident that once the remaining open items are resolved and once the Tier 1 design description and ITAAC are submitted, the staff will be able to document its findings in an FSER and to issue a final design approval within a reasonable time.

Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

cc: See next page

II. TECHNICAL DESCRIPTION

This section presents a technical description of the various elements of the System 80+ Standard Plant Design upon which the Plutonium Disposition Study was based. The System 80+ Standard Design is fully developed and directly suited to meeting the mission objectives for plutonium disposal. The base UO_2 plant design is discussed here. Adaptations for plutonium are given in Section III which follows. Table II-1 compares the characteristics for UO_2, plutonium disposal and tritium production options.

A. INTRODUCTION

The System 80+ Plutonium Disposition Complex is based on ABB-CE's System 80+ which is a complete Nuclear Power Plant currently in the final stages of licensing review by the Nuclear Regulatory Commission (NRC). The System 80+ plant layout (described in detail in Section II B) has evolved over the past few years based on previous designs, human factors considerations, compliance with the EPRI ALWR Utility Requirements Document and review by several Architect-Engineers and numerous representatives of operating power plants.

The general arrangement of the site groups the Reactor Building with its spherical containment and the Nuclear Annex which houses reactor safety, auxiliary and control systems and equipment into a single complex situated on a common foundation mat, thereby providing for excellent overall seismic stability, ground water resistance, and interbuilding access.

Separate structures have been provided to house the radwaste facilities, station services, the component cooling water heat exchangers, and station administrative services. This allows ready access to these areas for parallel construction without interference.

Site Layout

The evolution of the site layout from the System 80+ Nuclear Power Plant to the Plutonium Disposition Complex will involve location of the Fuel and Target Fabrication Facility, expansion of the spent fuel pool and location of the tritium extraction process. The resulting site layouts are shown in the enclosed Figures II-A-1, II-A-2, and II-A-3 for the single, dual and four plant arrangements. The System 80+ Nuclear Power Plants are shown as slide along plants with sharing of some site facilities (warehouses, heat sink etc.) Section B which follows describes the buildings and structures in more detail.

Fuel Fabrication and Processing

Figure II-A-4 shows a process overview of the complex for this study based on direction from the DOE. All facilities have been located at the same site as the reactor.

The MOX fuel fabrication plant (FTFF) will have capacity to produce 100 MT of MOX fuel per year. Plutonium will be received as PuO_2 to purity and physical properties specifications. The plutonium will be received in sealed PuO_2 containers. The PuO_2 containers will be stored in the storage vault. The uranium will be received as depleted

UO_2. DOE will supply depleted UF_6 that will be converted to UO_2 by a commercial fuel supplier.

The Fuel Fabrication Facility will also be required for target fabrication for the tritium production options. Target pins will consist of lithium aluminate $LiAlO_2$, pellets in the shape of annular rings contained in Type 316 stainless steel cladding. A zirconium sleeve that serves as an oxygen getter will be inserted inside of the lithium aluminate pellets. On the outside, the lithium aluminate pellets will be surrounded by a sleeve of zircaloy that serves as a tritium getter. This sleeve, in turn, will be surrounded by the cladding. The inner and outer surfaces of the cladding will be coated with an aluminide barrier to prevent the escape of tritium.

Site Design Parameters

When the specific location of the Plutonium Disposition Reactor is identified the System 80+ envelope of plant site design parameters will be examined to ensure that the selected site is within the parameters shown in Table II.A-1. The envelope of System 80+ offers flexible siting parameters and no problem is anticipated in designing the additional facilities (fuel fabrication etc) to these parameters.

TABLE II.A-1

(Sheet 1 of 3)

ENVELOPE OF PLANT SITE DESIGN PARAMETERS

Ground Water

Maximum Level:	2 feet below grade

Flood (or Tsunami) Level[1]

Maximum Level:	1 foot below grade

Precipitation (for Roof Design)

Maximum rainfall rate:	19.4 in/hr. and 6.2 in/5 min.[2]
Maximum snow load:	50 lb/sq. ft.

Design Temperatures

Ambient

1% Exceedance Values

Maximum:	100°F dry bulb/77°F coincident wet bulb
	80°F wet bulb (non-coincident)
Minimum:	-10°F

0% Exceedance Values (Historical Limit excluding peaks < 2 hours)

Maximum:	115°F dry bulb/80°F coincident wet bulb
	81°F wet bulb (non-coincident)
Minimum:	-40°F
Station Service Water Inlet:	95°F[3]
Condenser Circulating Water Inlet:	≤100°F

TABLE II.A-1 (Cont'd)

(Sheet 2 of 3)

ENVELOPE OF PLANT SITE DESIGN PARAMETERS

Extreme Wind

Basic Wind Speed:	110 mph
Importance Factors:	1.0[4]/1.11[5]

Tornado[6]

Maximum tornado wind speed:	330 mph
Rotational Speed:	260 mph
Translational velocity:	70 mph
Radius:	150 ft
Maximum pressure differential:	2.4 psi
Rate of pressure drop:	1.7 psi/sec
Missile spectra:	per SRP 3.5.1.4 Spectrum II

Soil Properties

Minimum Bearing Capacity (demand):	15 ksf (static)
Minimum Shear Wave Velocity:	500 ft/sec
Liquefaction Potential:	None (at site-specific SSE level)

Seismology

SSE Peak Ground Acceleration (PGA):	0.30 g

Aircraft Hazards

Plant to airport distance	5mi. $<$ D $<$ 10mi. with annual operation less than $500D^2$ or D $>$ 10mi. with an annual operation less than $1000D^2$ (D = distance in miles)
Plant to edge of military training routes	D $>$ 5mi. with an annual operation less than 1000 flights (D = distance in miles)
Plant to edge of Federal airway, holding pattern, or airport	D $>$ 2mi. (D = distance in miles)

TABLE II.A-1 (Cont'd)

(Sheet 3 of 3)

ENVELOPE OF PLANT SITE DESIGN PARAMETERS

Meteorology

Short-term dilution factor	X/Q 1.0×10^{-3}; EAB = 500 meters
Long-term dilution factor	X/Q 2.2×10^{-5}; LPZ = 3000 meters

NOTES:

1. Probable maximum flood level (PMF), as defined in ANSI/ANS-2.8, "Determining Design Basis Flooding at Power Reactor Sites."

2. Maximum value for 1 hour 1 sq. mile PMP with ratio of 5 minutes to 1 hour PMP of .32, as found in National Weather Service Publication HMR No. 52.

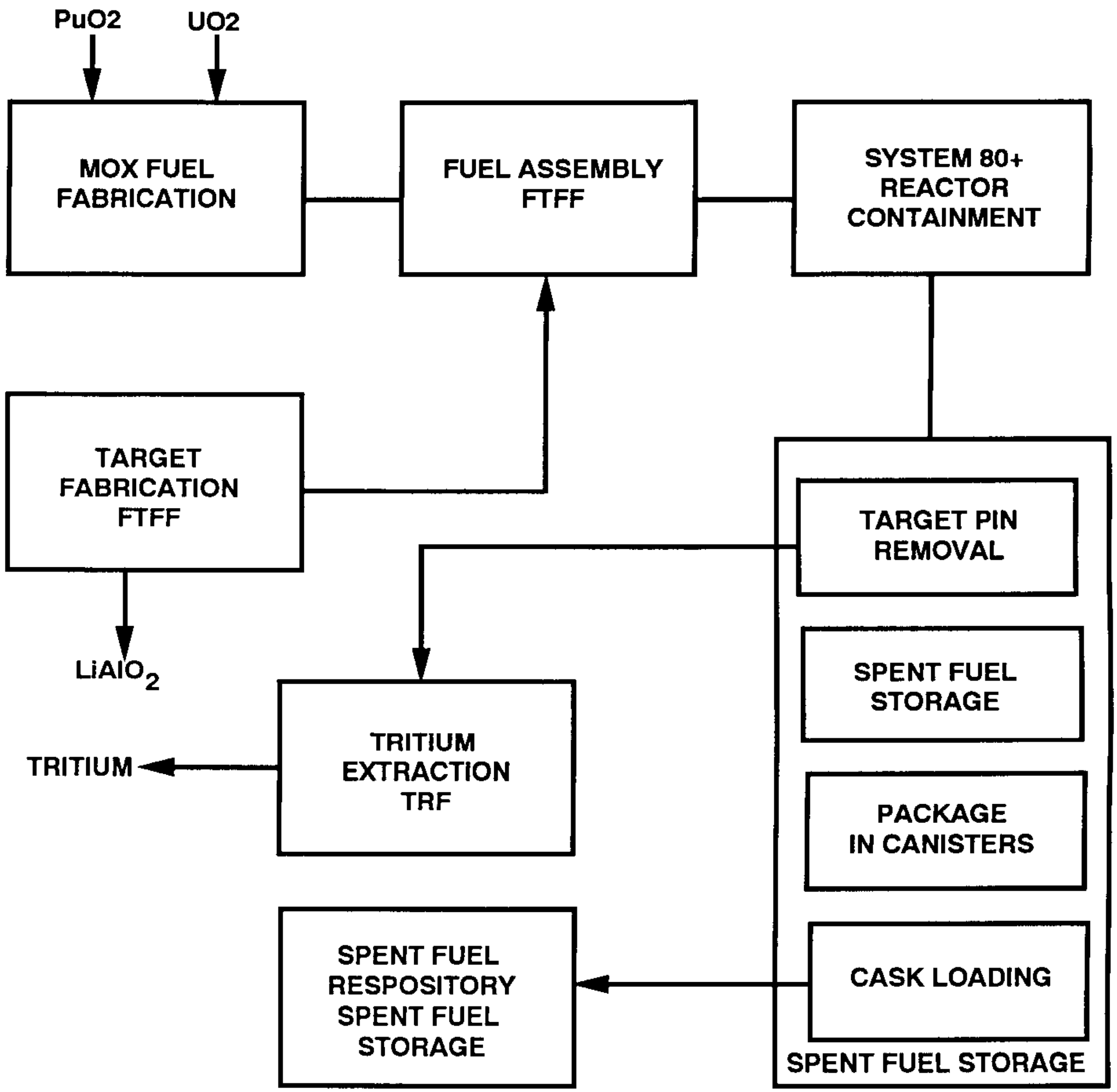
3. Maximum normal power and normal shutdown temperature of the Station Service Water System Intake based on one percent exceedance meteorologic conditions.

4. 50-year recurrence interval; value to be utilized for design of non-safety-related structures only.

5. 100-year recurrence interval; value to be utilized for design of safety-related structures only.

6. 10,000,000-year tornado recurrence interval, with associated parameters based on the NRC's interim position on Regulatory Guide 1.76. Pressure effects associated with potential offsite explosions are assumed to be non-controlling for the design.

FIGURE II.A-4
SYSTEM 80 + PROCESS OVERVIEW OF THE PLUTONIUM DISPOSAL COMPLEX



B. POWER PLANT DESCRIPTION

1. General

The System 80+ Advanced Light Water Reactor consists of a complete nuclear power plant. The plant structures include:

<u>Nuclear Island Structures</u>
- Reactor Building
  - Shield Building
  - Steel Containment Vessel
  - Internal Structure
  - Subsphere
- Nuclear Systems Annex
  - Fuel Pool Area (Including Auxiliary Fuel Storage Facility)
  - Control Area
  - Diesel Generator Area
  - Main Steam Valve House
  - Maintenance/Outage Area

<u>Turbine Island Structures</u>
- Turbine Building

<u>Other Plant Structures</u>
- Fuel Oil Tanks and Pumphouse
- Service Water Pumps & CCW Heat Exchanger Bldg.
- Ultimate Heat Sink (UHS)
- Radwaste Building
- Circulating Water Pumphouse
- Plant Cooling Tower
- Auxiliary Boiler Building
- Station Services Building
- Administration Building
- Security Building
- Station Structures for Pu Consumption
- Warehouse
- Yard Structures
- Fire Tanks & Pump House
- Fuel & Target Fabrication Facility
- Tritium Recovery Building
- Auxiliary Fuel Storage Facility

These buildings and structures are shown on the plot plans presented in Section II.A and discussed in more detail in the subsection which follow. Figures II-B-1 through II-B-3 illustrate the arrangements of the Nuclear Island.

The general arrangement of the site groups the Reactor Building with its spherical containment and the Nuclear Annex which houses reactor safety, auxiliary and control systems and equipment into a single complex situated on a common foundation mat, thereby providing for excellent overall seismic stability, ground water resistance, and interbuilding access.

Separate structures have been provided to house the radwaste facilities, station services, the component cooling water heat exchangers, and station administrative services. This allows ready access to these areas for parallel construction without interference.

The locations of the principal plant features provide for a high degree of separation of safety components from potential hazards. For example, the turbine rotating axis is selected to direct missiles away from all Category I structures. Maintenance convenience is a major consideration in the development of this plant. This includes single piece steam generator removal, condenser retubing access, diesel generator removal, and many other considerations.

The dual spherical steel containment is based on the design used for the cancelled Duke Power Company Cherokee/Perkins* plants and provides additional operating floor space for maintenance and layout. Also, the inherent strength in the spherical arrangement provides additional design margin for any possible future regulatory concerns relating to severe accident scenarios. The mitigation of severe accidents was explicitly considered in the design of the containment. For example, the large containment volume provided and the internal containment arrangement promotes post accident hydrogen mixing and dilution. The Reactor Shield Building forms an additional barrier to provide a minimum accident release potential when compared to current designs.

The design complies with environmental discharge and other requirements, including those regarding appearance, noise, emissions effluents and drains.

The layout of the plant considers both internal and external hazards. The primary protection for external hazards is achieved through the structural arrangement that provides spatial separation of redundant safety equipment.

Buildings, structures, and components have been designed such that their failures or interactions will not cause an unacceptable failure of safety related features. This has been accomplished in one of the following ways:

- Through physical separation from safety related features by a distance sufficient to prevent unacceptable damage to safety related features, or

- By designs than can withstand loads to the extent required to prevent unacceptable damage to safety related features, or

- By the use of barriers or other protective features to protect safety related features from failures or interactions to the extent required to prevent unacceptable damage to safety related features.

* Construction of the Cherokee 1 containment was well along when the plant was cancelled.

When safety related and non-safety related buildings, structures, or components are integrally connected, the non-safety related feature is accounted for in the model when determining the design of the safety related feature.

Safety related buildings, structures, and equipment subject to hazards, either internally and externally generated, has been designed for or protected from such hazards sufficient to prevent unacceptable damage.

The plant arrangement is such as to minimize the impact from a potential turbine missile by defining the orientation of the turbine longitudinal axis towards the containment and by spatial separation of safety equipment.

Wind-generated missiles are considered in the design of reinforced concrete walls for each structure which houses safety-related equipment.

The station layout provides a centralized Main Control Room from which all day-to-day plant operations are controlled and monitored. The control room is located in the Nuclear Annex adjacent to the Reactor Building, between the Reactor Building and the Turbine Building. Batteries, cable shaft and switchgear areas are located below the control room. A separate Remote Shutdown Room, and Technical Support Center are located in the vicinity of the central Control Complex. The capability of bringing the plant to a cold shutdown from outside of the control room is provided through the use of controls in the Remote Shutdown Room.

The ultimate heat sink shown in Figure II-A-1 is a single passive independent cooling water pond. However, it is recognized that site-specific conditions may require the use of two ponds to meet Regulatory Guide 1.27. The design brackets alternative ultimate heat sinks which may be specified for a particular site if environmental restrictions limit the use of a cooling pond or if an alternative water supply is more reliable. Acceptance alternate ultimate heat sinks are an ocean, a large lake, a large river, a lake and a cooling pond, a river and a cooling pond, or a cooling tower and cooling pond.

Each safety train is separated from the others by physical barrier or distance where impractical, e.g., inside containment. Diesel generators are located on opposite sides of the Reactor Building. Major cable runs are separated by concrete barriers between trains with individual runs separated in accordance with USNRC Regulatory Guideline 1.75.

The emergency feedwater system contains two storage tanks, each with a minimum useable volume of 350,000 gallons.

The HVAC intakes are positioned to preserve the functioning of the emergency diesel generators and the occupancy of the Main Control Room and other safety-related areas which are required to be manned during operation or shutdown of the station.

The fire zones and barriers are determined by conducting detailed analysis of hazards during the final design of interior structures. Entry to the site is strictly controlled. Plant and buildings are located relative to security fencing or barriers such that the risk of unauthorized access or interference is minimized.

2. Reactor Building

The Reactor Building's function is to provide a suitable environment to house the Reactor and its associated safety systems, and to prevent the uncontrolled release of radioactivity to the outside environment. The Reactor Building also shields the Containment from external hazards and environmental effects. As an integral part of these functions, it forms a part of the engineered safety systems of the plant.

The fundamental design objective is to ensure that the reactor can safely be shut down and adequately cooled in the event of a design basis fault and to provide assurances that the radiation levels at the site boundary will not exceed the permitted levels. This objective is accomplished by insuring that the structures will withstand all internal and external loading conditions that may reasonably be expected to occur during the life of the plant. This includes both the short and long term effects following design basis accidents, e.g., loss of coolant accidents. Meeting these objectives protects the public health and safety as well as the owners' investment.

These objectives are met by providing a reinforced concrete Shield Building housing a free-standing Spherical Steel Containment Vessel (SCV). The Shield Building is designed to protect the SCV against externally generated environmental and missile loads.

The System 80+ Reactor Building is composed of the following major parts, which are all Category I structures:

- Shield Building
- Steel Containment Vessel (SCV)
- Internal Structure
- Subsphere

Each of these parts function with the others to meet the design objectives.

a. <u>Shield Building</u>

The Shield Building provides protection from the outside environment. It is a reinforced concrete structure composed of the foundation and exterior walls and designed in accordance with ACI 349 and 350. The foundation is approximately ten feet thick and is integral with the adjacent Category 1 structures. The exterior walls are composed of a 210 ft. diameter right circular cylinder with a hemispherical dome. The exterior wall thickness varies from 4 ft. for the cylindrical portion to 3 ft. for the hemispherical dome. The overall building height is approximately 216 ft.

A 5-foot annular space is provided between the SCV and Shield Building for structural separation and to provide access to the containment and penetrations for testing and inspection.

b. <u>Steel Containment Vessel (SCV)</u>

The SCV is a spherical steel shell pressure vessel constructed of formed plate segments welded together. The diameter of the SCV is 200 ft. The plate nominal thickness is 1.75

inches. The vessel plate is thickened around penetrations to compensate for the opening. Where there is a cluster of penetrations in the same plate segment, the entire segment is fabricated out of thicker plate and transition tapered to 1.75 in at the edges. The additional thickness depends upon the nominal size, thickness and location of the penetration sleeve and is in accordance with ASME Code requirements.

The materials are in accordance with Article NE-2000 of Subsection NE, "Class MC Components," of the ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components." The containment plate material is SA537 Class 2.

The SCV is an independent, freestanding structure above grade elevation. Below grade, the SCV is encased between the base slab of the Internal Structures and a reinforced concrete pedestal extending from the Shield Building foundation. No shear connections are provided between the containment plate and the Shield Building foundation or the base slab of the Internal Structures. The lateral loads due to seismic forces, etc., will be transferred to the foundation concrete by friction and bearing. No credit is taken for the reduction of shell stresses from the concrete encasement of the SCV; therefore, the SCV plate thickness in the embedded zone is the same as in the free zone. In the transition region (where the SCV emerges from the concrete encasement), a compressible material is provided to eliminate excessive bearing loads on the concrete as well as to reduce the secondary stresses in the SCV at this location. The containment plate is also thickened in this region to accommodate potential corrosion of the plant and to accommodate the additional stress in this deflection area during events that pressurize the containment.

Penetration assemblies in the encased region are surrounded by a compressible material. The compressible material significantly reduces the shear transfer from the SCV to the concrete by the penetration assemblies. The compressible material is designed to permit movement of the SCV plate for all design loadings based on a conservative assumption that the plate is free to move between the two layers of concrete. The weld connection between the SCV and the penetration sleeve is designed to withstand the nominal shear required to compress the compressible material. The compressible material has a useful life equal to or greater than the design life of the plant.

The major penetrations to the containment are the equipment hatch and two personnel airlocks. The equipment hatch is composed of a cylindrical sleeve in the containment shell and a dished head 22 ft. in diameter with mating bolted flanges. Each personnel airlock has double doors with an interlocking system to prevent both doors being opened simultaneously. Remote indication is provided to indicate the position of each door. Double seals are provided on each door. These penetrations are designed, fabricated, tested and stamped in accordance with Section III, Subsection NE of the ASME Boiler and Pressure Vessel Code. Provisions are made for performing the required inservice pressure and leak rate tests.

A fuel transfer tube penetration sleeve is provided for the transfer tube for transfer of fuel between the fuel pool and the Reactor Building. The fuel transfer penetration sleeve has a double gasketed blind flange in the Reactor Building.

Mechanical penetrations are treated as fabricated piping assemblies meeting the requirements of ASME III, Subsection NE, and Subsection NC. The process line and fluid heads making up the pressure boundary are consistent with the system piping materials; fabrication, inspection, and analysis requirements are as required by ASME III, Subsection NC. All welds on the process pipe are accessible for inspection in accordance with ASME Section XI. High energy lines and selected engineered safety system and auxiliary lines require the "Hot Penetration" assembly which features the exterior guard pipe for protection of other penetrations in the building annular space. Other lines are treated as cold penetrations since a leak into the annular space would not cause a personnel hazard or damage other penetrations in the immediate area.

Medium voltage electrical penetrations through the SCV for reactor coolant pump power use sealed bushings for conductor seals. The assemblies incorporate dual seals along the axis of each conductor. The low voltage power, control, and instrumentation cables enter the SCV through penetration assemblies which are designed to provide two leak-tight barriers in series with each conductor. All electrical penetrations are designed to maintain the SCV integrity for Design Basis Accident conditions, including pressure, temperature, and radiation. Double barriers permit testing of each assembly as required to verify that the SCV integrity is maintained.

c. Internal Structure

The Internal Structure is a group of reinforced concrete structures that support the Reactor Vessel and primary system. The Internal Structure also provides biological shielding for the containment interior.

The primary shield wall encloses the Reactor Vessel and provides protection from internal missiles. The primary shield wall provides biological (radiation) shielding and is designed to withstand the temperatures and pressures following LOCA. In addition, the primary shield wall provides structural support for the Reactor Vessel. The primary shield wall is greater than 6 ft thick.

The secondary shield wall (crane wall) provides support for the polar crane and protects the SCV from internal missiles. In addition to providing biological shielding from the reactor coolant loops and equipment, the crane wall also provides structural support for pipe restraints and platforms at various levels. The crane wall is a right cylinder with an inside diameter of 130 ft and a height of 118 ft from its base. The crane wall is 4 ft thick.

The refueling pool, when filled with borated water, forms a pool above the Reactor Vessel to facilitate the fuel handling operation without exceeding the acceptable levels of radiation inside the Containment. The Reactor Vessel flange is permanently sealed to the pool floor to prevent leakage of refueling water into the reactor cavity. The refueling pool has the following subcompartments:

- Storage area for upper guide structure.
- Storage area for core support barrel.
- Refueling canal.

The fuel transfer tube connects the refueling pool to the Spent Fuel Pool. The shield walls that form the refueling cavity are greater than 6 ft thick.

The operating floor provides personnel access for operating functions and provides biological shielding. Inside the crane wall, the operating floor is a reinforced concrete slab with covered hatches aligned with hatches in the two lower slabs to facilitate maintenance activities. Outside the crane wall, the operating floor consists of steel grating. There are also reinforced concrete floor slabs that connect the crane wall and the primary shield wall. The concrete slabs provide lateral stability to the steam generator and pressurizer enclosures as well as connecting the primary shield wall and the crane wall.

d. Subsphere

The area of the Reactor Building below grade elevation and outside of the containment makes up the region referred to as the Subsphere. The Subsphere allows efficient use of space for locating safety equipment adjacent to the SCV and eliminating excessive piping while allowing maximum access to the containment for locating penetrations. This subsphere houses most of the engineered safety systems external to containment.

3. Nuclear Systems Annex

The Nuclear Systems Annex houses those reactor auxiliary and control systems not located within the Reactor Building. It consists of the Fuel Pool Area, the Control Area, Emergency Diesel Generator Areas, the Main Steam Valve Houses, and Maintenance/Outage areas.

The Nuclear Systems Annex is classified as safety-related and Seismic Category I and is supported on a reinforced concrete foundation mat, common to the Reactor Building. The exterior walls are reinforced concrete. Interior floors are reinforced concrete supported on concrete framing. Reinforced concrete walls separate equipment and piping systems to provide biological shielding, floor protection, and protective missile barriers, where required.

The roof drainage system discharges to the storm and waste water system. The Fuel Pool and Maintenance/Outage Area, floor and equipment drains which are potentially contaminated, are sent to the radioactive liquid waste system via the aerated drain system.

a. Fuel Pool Area

The fuel storage areas for the plutonium disposition plant are similar to the facilities in the System 80+ plant. However, the plutonium disposition mission requires that the configuration be modified to meet the special requirements of the mission. The System 80+ plant provides storage of new and spent fuel assemblies in the fuel storage area of the nuclear annex. New UO_2 fuel assemblies are stored in dry racks and spent fuel assemblies are stored under water in racks that provide the capability for placing a fuel assembly in each cell location when poison inserts are used in the cells. The plutonium disposition plant may have the entire complement of fuel assemblies for the plant operating life on site and irradiated before any fuel assemblies are shipped from the plant. This will require larger under water storage capacity than is typically available at commercial plants.

The requirement for the quantity of storage will be met by eliminating the new fuel dry storage area and maximizing the capacity of the underwater pool in the Nuclear Annex. Due to criticality of the plutonium fuel assemblies, the fuel assemblies will be stored in a checkerboard pattern that uses every other cell location. Approximately 1250 assemblies will be stored in the Nuclear Annex. The remaining assemblies will be stored in the auxiliary fuel storage facility which is connected to the Nuclear Annex. The storage area in the Nuclear Annex maintains the envelope of the commercial plant area to minimize the impact on the System 80+ design. The auxiliary facility is located adjacent to the annex and is located on a separate base mat.

The plutonium disposition plant also includes the capability to produce tritium. The System 80+ fuel handling system design accommodates the production of tritium. The cask laydown area in the System 80+ plant is designed to support reconstitution (repair) of irradiated fuel assemblies. The installation and recovery of the tritium targets requires similar technology for the disassembly and reassembly of the fuel. As the work is done in the cask laydown area, tritium targets can be loaded directly into a cask for shipment. Both the Nuclear Annex and auxiliary fuel storage facilities include cask laydown areas. Receipt and shipment of fuel assemblies can be performed currently with tritium recovery operations.

The new and spent fuel storage facilities as well as systems and components directly related to the fuel storage and handling are contained in the fuel storage area of the Nuclear Annex. Due to the number of fuel assemblies that require storage, additional new and spent fuel storage is located in the auxiliary fuel storage facility. The storage of fuel assemblies in the Nuclear Annex is maximized while maintaining the size of the System 80+ base mat.Both of these fuel storage areas are located in safety-related Seismic Category I structures.

The fuel storage areas are designed to store the fuel assemblies for fifteen full reactor cores, which represents a total of 3,615 fuel assemblies. Additional storage locations are allocated for failed fuel assemblies and temporary storage of containerized spent control element assembly rods and spent ICI assembly segments. As the entire quantity of fuel assemblies may be irradiated before full burnup is attained on any of the assemblies, the entire storage capacity is located under water in the fuel pools.

The fuel pools are stainless steel-lined reinforced concrete structures within the fuel storage facilities. The fuel racks are a series of monolithic, honeycomb structure modules, the elements of which are the individual storage cells. The fuel assemblies are placed in a checkerboard pattern in the modules. Cell locations not intended for use are permanently blocked to prevent insertion of fuel assemblies. The fuel storage rack modules are described further in Section II F.

The fuel pool in the Nuclear Annex is connected to the refueling cavity in the reactor building by a transfer tube assembly and transfer system. A second transfer tube and transfer system connect the Nuclear Annex fuel pool to the fuel pools in the auxiliary fuel storage facility. The auxiliary fuel storage facility is located off of the base mat. The connecting transfer tube assembly accommodates differential building settlement and seismic motions.

The transfer system components, including the fuel upenders, are located in canals that are connected by gates to the fuel storage pools. The gates are hinged to close towards the transfer canals so the water pressure in the fuel pool helps to seal the gates when the canals are drained for equipment servicing. Fuel cask laydown areas are also located adjacent to the fuel storage pools. These laydown areas are also connected to the fuel storage pools by gates similar to those used for the transfer canals. A cask washdown area is located adjacent to the cask laydown area to permit the fuel shipping cask to be cleaned before it is removed from the fuel storage area. Depending on the final arrangement of the auxiliary fuel storage, multiple cask laydown areas will be provided for each pool area.

The gates connecting the fuel storage pools to the adjacent pools such that the bottom of the gate is located above the top of the fuel rack modules to ensure water coverage is maintained over the fuel assemblies in the event of a gate failure. Pipe connections to the pools are designed to prevent gravity draining of the pools.

The new fuel assemblies are expected to arrive in shipping casks that are similar to the available PWR spent fuel shipping casks. The new fuel casks are located in the cask laydown area which is then filled with water. This permits the new fuel assemblies to be removed from the cask and loaded in the fuel racks using the spent fuel handling machines.

Space is provided in the cask laydown area to permit subsidiary fuel activities to be performed. These activities include irradiated and new fuel inspection, fuel reconstitution, and replacement of tritium targets. The benefits of performing the work is the cask laydown area are the area can be drained for equipment set up or repair and it is not necessary to move the equipment over the fuel storage pool. This eliminates the concerns of moving heavy loads or the crane over the stored fuel assemblies.

The shipping cask areas are serviced by a bridge crane with a 150 ton main hoist and a ten ton auxiliary hoist. The crane is used to unload the shipping cask from the transport vehicle in the receiving bay and to move the cask through the cask decontamination area to the laydown area. The crane and building design allows movement of the shipping cask without lifting the cask above the operating floor. Hardstops prevent movement of the crane over the fuel storage racks. The bridge crane can also be used to set up equipment to perform subsidiary fuel activities in the cask laydown area. A jib crane is located over the cask laydown area to support subsidiary fuel activities.

The fuel storage pools, transfer canals and cask laydown areas are serviced by one or more spent fuel handling machines. A separate machine is provided for each fuel storage pool. The spent fuel handling machines replicate the refueling machine to the maximum extent possible to permit operating training prior to entering the reactor building for refueling. The spent fuel handling machines are described in Section II F.

The fuel shipping cask decontamination area is approximately sixteen feet by sixteen feet in area. The area is lined with stainless steel and the floor is sloped to direct water used for cask cleaning to a floor drain. The floor drain is connected to the sump system to direct the effluent to the radioactive liquid waste system. The floor areas in the fuel storage areas and the equipment in these areas could be potentially contaminated during operations. Therefore, the floor drains and equipment drains are piped to a sump via the aerated drains system and are pumped from the sump to the radioactive liquid waste system.

The fuel storage area purification and cooling system is described in Section II E.

b. Control Area

The Control Area is a five-story complex, which houses the Main Control Room, the safety-related battery rooms, cable shafts, switchgear areas, control area ventilation, technical support center, computer room and Remote Shutdown Room. The Control Area is isolated from the adjacent areas by reinforced concrete walls to provide environmental and hazard separation.

The Main Control Room itself will be maintained at a slight positive pressure to prevent the ingress of radioactive contamination during and following a postulated accident. Air intakes are provided from each side of the Nuclear Systems Annex to prevent both intakes from being exposed to a common environmental hazard.

c. Main Steam Valve Houses

The Main Steam Valve Houses are on opposite sides, outside the Shield Building. The Valve Houses provide shielding for the main steam and feedwater lines. The main steam safety, power operated relief, and isolation valves are located in the Main Steam Valve House. The Emergency Feedwater (EFW) Storage Tanks are located beneath the main steam and feedwater penetration areas. The Valve House provides environmental and missile protection for these systems.

d. Maintenance/Outage Area

The Maintenance/Outage Area is located adjacent to the Fuel Pool Area. This area houses reactor auxiliary equipment including that of the chemical and volume control system, a component cooling water surge tank and Nuclear Systems Annex HVAC equipment. Access is provided for both personnel and equipment to/from all areas of the Nuclear Systems Annex through dedicated passage space.

4. Turbine Island Structure

The function of the Turbine Building is to house the components of the steam power plant, including the turbine, the generator, and associated systems for power generation.

The Turbine Building is located so that the Reactor Building is on the projection of the turbine shaft, on the high-pressure turbine side. This allows the pipework and cable routing to be optimized and also minimizes the risk of damage to safety-related equipment in the Nuclear Island by missiles from the turbine or generator, in the event of an accident.

The Turbine Building has three main floors above the ground level: the ground floor, mezzanine, and operating floor. Besides these main floors, there are several smaller service floors.

The Turbine Building houses the components of the steam power plant, including turbine generator, condenser system, preheater system and general service water systems as well as the turbine building heating and ventilation system. The main feedwater pumps, the deaerator/ feedwater tank, and the condensate pumps, the deaerator/feedwater tank, and the condensate cleaning plant are arranged in an annex to the main Turbine Building.

The turbine and generator are arranged in the main bay of the Turbine Building. The space around the turbine is capable of accommodating the dismantled parts of the HP-turbine, one LP-turbine cylinder and the generator during maintenance. The LP and HP feedwater heaters are arranged vertically between the turbine generator and the feedwater pump annex. They can be handled easily by the crane. LP heating of the first stage is integrated in the condenser neck. The turbine oil and governor fluid tank are located outboard of the HP side of the turbine foundation on the Turbine Building ground floor. The generator auxiliaries are arranged in the area below the generator. The encapsulated single phase busbars leave the turbine hall at the transverse rear wall of the building. The main Turbine Building crane is capable of lifting the heaviest part, the generator stator. The track and the main opening is at the rear end of the building. A part load crane which travels below the main crane is also provided in order to facilitate and to expedite maintenance.

For personnel access, four stairways are provided, and two personnel/light equipment elevators. The ventilation air enters the building through intake louvers above ground and leaves the building via roof mounted exhaust fans equipped with silencers.

The Turbine Building is constructed to conventional standards. The turbine building is 360 ft long, and 164 ft wide. Inlet and outlet cooling water pipes are located on the open longitudinal side of the building.

The Turbine Building is a braced steel structure and consists of AISC Type 2 steel framing. The Turbine Building roof is supported with a truss, which spans across the Turbine-Generator bay. The roof structure has a horizontal truss system to transmit design loads to the braced walls. The steel-framed structure has insulated metal siding, metal roof decking, and built-up roofing. The Turbine-Generator is supported at the operating floor level on a reinforced concrete foundation plate, which is spring-mounted on reinforced concrete columns; the columns are supported by a mat foundation. The turbine pedestal structure is isolated from the main building frame. The main building frame structure is supported on a conventional reinforced concrete mat foundation.

The Turbine Building is classified as non-safety related, and will be designed and constructed in accordance with the codes and standards established for non-seismic structures.

5. **Other Plant Structures**

a. <u>Category I Structures</u>

(1) **Fuel Oil Pumphouse/Tanks**

The diesel generator fuel oil storage area consists of one set of two seven-day fuel oil storage tanks and an above-ground reinforced concrete fuel oil pump house for each diesel.

(2) **Component Cooling Water (CCW) Heat Exchanger Structure**

The CCW Heat Exchanger structure is a reinforced concrete building, located close to the Ultimate Heat Sink to minimize the amount of Station Service Water piping. The structure houses four component cooling water heat exchangers and supporting equipment. The CCW heat exchanger structure provides physical barriers to maintain divisional separation of CCWS components and piping for fire, single failure, pipe whip and seismic interaction effects.

(3) **Station Service Water (SSW) Pump Structure**

The SSW Pump Structure houses four pumps and supporting equipment. The construction and location are dependent on the type of UHS selected for the plant site. The structure provides physical barriers to maintain divisional separation of SSWS components.

(4) **Ultimate Heat Sink (UHS)**

The UHS selected for the steady is a safe shutdown pond. The size is nominally based on the EPRI ALWR utilities requirements document. The selection of the UHS may vary based on site selection. Options include a safe shutdown pond, a spray pond, and mechanical draft cooling towers. For the multiple unit site it is recognized that size usually limits the ability to locate individual ponds. It is therefore anticipated that spray array would be necessary.

The UHS is designed to accommodate the heat loads resulting from all normal and abnormal station operating conditions while maintaining the temperature of the service water system within its design basis limit.

b. <u>Non Category I Structures</u>

(1) **Radwaste Building**

The radwaste building is classified as nonsafety-related. The Radwaste building houses and protects components of the following systems: solid waste, liquid waste (portions), primary grade water (portions). It also provides structures necessary to the operation of these systems, including radiation shields where required.

The Radwaste Building additionally provides the means for retention and cleanup of spillage from vessels containing potentially contaminated fluids, provides a decontamination area and solid waste storage area. It is a four-story structure located adjacent to the fuel building. It is basically a steel-framed structure with a reinforced concrete substructure such that spillage from a pipe or tank rupture will be retained within the structure. The exterior walls are insulated metal siding. The roof is steel decking covered by asphalt and gravel roofing.

A 30 ton (metric) bridge crane is provided for solid waste drum handling. A 12 ton (metric) monorail is also provided.

The roof drainage system discharges to the storm and waste system. The building floor and equipment drains are potentially contaminated and are piped to a sump via the aerated drains system and are then pumped to the radioactive liquid waste system.

b. Plant Cooling Tower and Circulating Water Pumphouse

The plant cooling tower consists of one natural draft hyperbolic cooling tower per unit. The tower consists of a reinforced concrete cooling tower shell, concrete basin, foundation and support columns, PVC corrugated sheet metal fill, water distribution system, drift eliminators, lightning protection system and aircraft warning system.

The circulating water pumphouse is located adjacent to the natural draft cooling tower. The building houses the circulating water and turbine plant service water pumps. Separate pump bays are provided for each of the pumps and the bays and flow are constructed of reinforced concrete.

The bottom of the structure is an extension of the cooling tower cold water basin. It is constructed of reinforced concrete. Above grade the building is a steel framed structure with a steel deck roof covered by asphalt and gravel roofing. Walls are insulated metal siding. The structure includes a crane for maintenance of equipment.

c. Station Services Buildings

The Station Services Building houses the following station equipment and systems:

Water Treatment
Work Shops
Compressed Air Equipment

The Station Services Building is classified as nonsafety-related.

The building is a one-story steel-framed structure with a steel deck roof covered by asphalt and gravel roofing. Walls are insulated metal siding.

Roof drainage and clean floor drainage discharge to the storm and waste water system.

d. Administration Building

This building is founded at grade on a reinforced concrete foundation. It is a two-story metal-sided building with a structural steel frame and has asphalt and gravel roofing.

The building is located immediately outside the perimeter fence at the entrance to the plant, near the security building.

Air conditioning and heating are provided to meet specified design temperatures.

e. Security Building

This single-story building is founded at grade on a reinforced concrete foundation. It is a masonry building with asphalt and gravel built-up roofing.

The Security Building is located along the perimeter fence at the entrance to the plant and is near the administration building.

The guard work area within the Security Building, which controls access to the protected area, is constructed with the walls, doors, ceiling, floor, and any windows in the walls and in the doors, of bullet-resisting construction.

The Security Building ventilation system is provided to maintain the building within design temperature limits.

f. Warehouse

This single-story building is founded at grade on a reinforced concrete foundation. It is a metal enclosed building with a structural steel frame and has asphalt and gravel roofing.

The approximate building dimensions are 395 ft long by 230 ft and 26 ft high.

g. Fire Protection Pumphouse/Tanks

The Fire Pumphouse is equipped with two fire pumps, one electric-motor driven, and one diesel-engine driven. The building is of reinforced concrete construction.

Roof drainage consists of roof sloped to drains, leaders, and underground drain piping discharging to the storm and waste water system.

The diesel fire pump is diked to contain major spills from the diesel fuel oil day tank. An overflow siphon and sump pump are arranged to discharge drainage to an oil separator prior to entering the storm and waste water system.

h. Auxiliary Boiler Building

The Auxiliary Boiler Building houses the auxiliary boiler and is located directly adjacent to the Turbine Building and is used to provide steam during startup and shutdown periods.

i. Miscellaneous Yards Structures

Other miscellaneous yard structures are provided within the plant boundary. These structures include the sewerage treatment plant, transformer yard, switchyard, relay houses and the gas turbine facility. Also a rail system is provided around the site perimeter with loading/unloading areas at various buildings to facilitate maintenance and operation activities.

(5) Additional Station Structures for Pu Consumption

The System 80+ plant design for surplus plutonium disposal and tritium production would be expanded to include structures for fuel and target fabrication, fuel assembly, irradiation in System 80+ reactor(s), disassembly of the spent fuel to remove the target pins, processing the target pins to extract tritium, storage of the spent fuel in the reactor basin, encapsulating the spent fuel pins in canisters for repository disposal, and preparation for disposal in the geologic repository. The design of these structures would need to be further developed once final mission requirements are established. Figures II A-1 through 4 presented earlier illustrate how these added structures can be easily accommodated into the base design.

TECHNICAL DESCRIPTIONS

TABLE II-B-1 (Sheet 1)
MAJOR BUILDING FUNCTIONS

Building	Building Function (below)					
	1	2	3	4	5	6
Nuclear Island Structures						
Reactor Building	X	X	X	X	X	X
Nuclear Systems Annex	X	X	X	X	X	X
Fuel Pool Area (including auxiliary fuel storage facility)		X	X	X		
Control Area			X	X	X	X
Diesel Generator Area	X			X	X	X
Main Steam Valve House	X			X	X	
Maintenance/Outage Area	X	X	X	X	X	
Turbine Island Structures						
Turbine Building	X			X		
Other Plant Structures						
Fuel Oil Tanks and Pumphouse	X			X	X	X
Service Water Pumps & CCW Heat Exchanger Bldg.	X			X	X	X
UHS	X			X		X
Radwaste Building	X	X	X	X		
Circulating Water Pumphouse	X		X	X	X	

TECHNICAL DESCRIPTIONS

TABLE II-B-1 (Sheet 2)
MAJOR BUILDING FUNCTIONS

Building	Building Function (below)					
	1	2	3	4	5	6
Other Plant Structures (Continued)						
Plant Cooling Tower	X					
Station Services Building	X					
Administration Building	X					
Security Building	X					
Warehouse	X					
Miscellaneous Yard Structures	X					
Fire Tanks & Pump House	X					

TECHNICAL DESCRIPTIONS
TABLE II-B-1 (Sheet 3)
MAJOR BUILDING FUNCTIONS

(1) House and support plant in a safe and suitable environment

(2) Prevent the uncontrolled release of radioactivity to the environment and contain the activity within the building

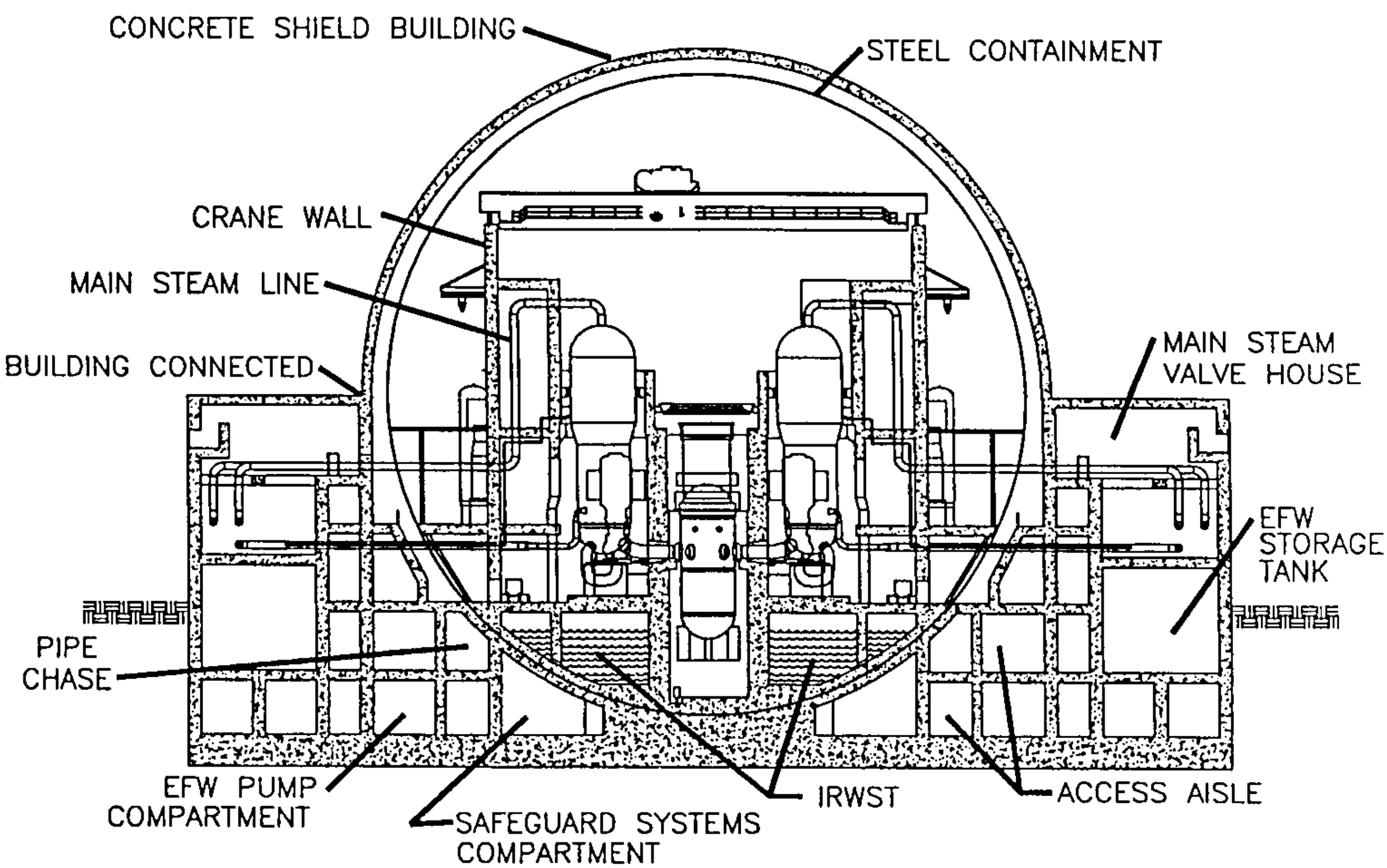
(3) Provide shielding to the operating staff

(4) Protect the plant within the building from external hazards and environmental effects

(5) Provide segregation for independent safety systems to provide protection from internal hazards and environmental effects

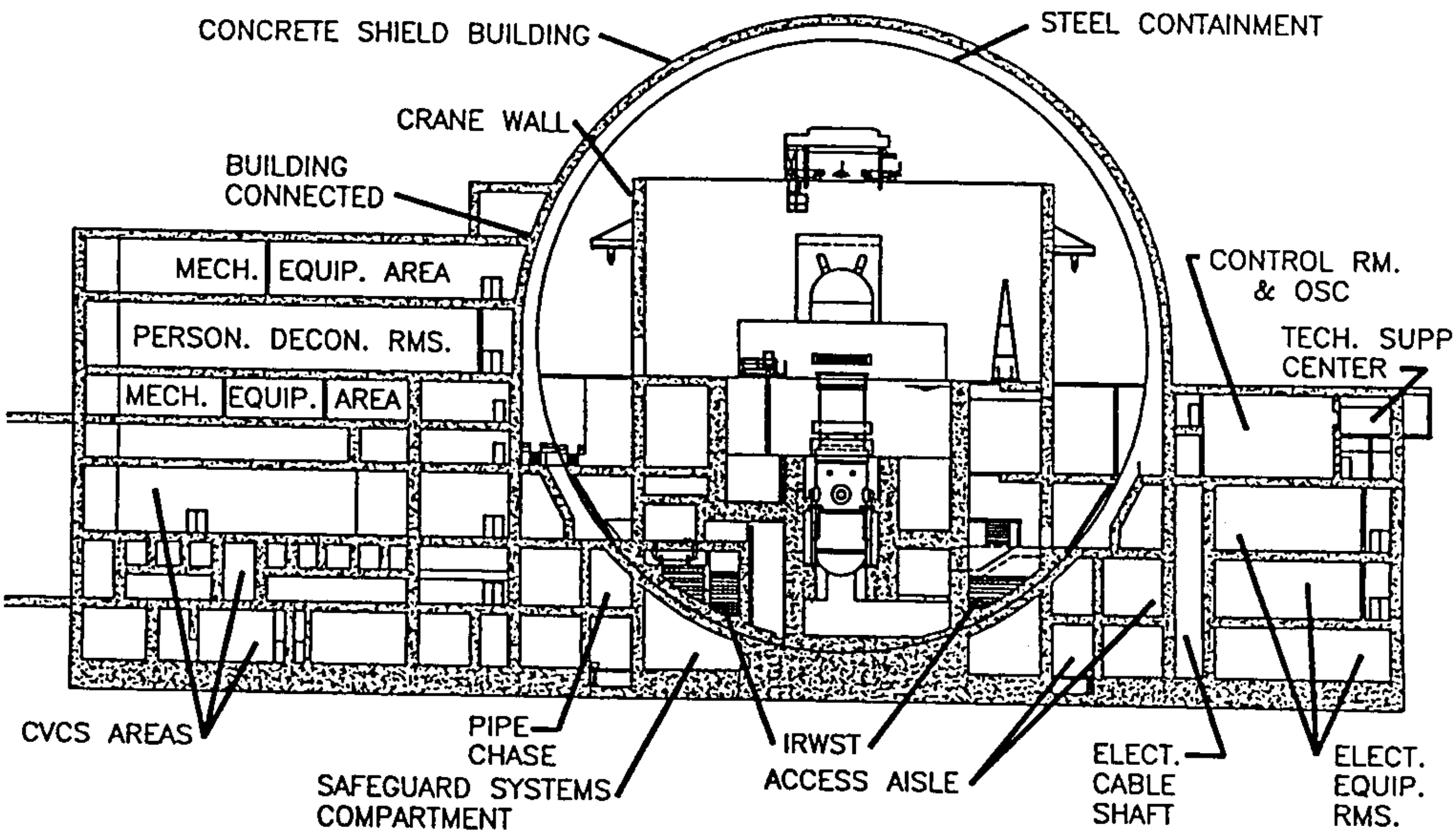
(6) Where appropriate, forms part of the engineered safety systems

ABB
ASEA BROWN BOVERI

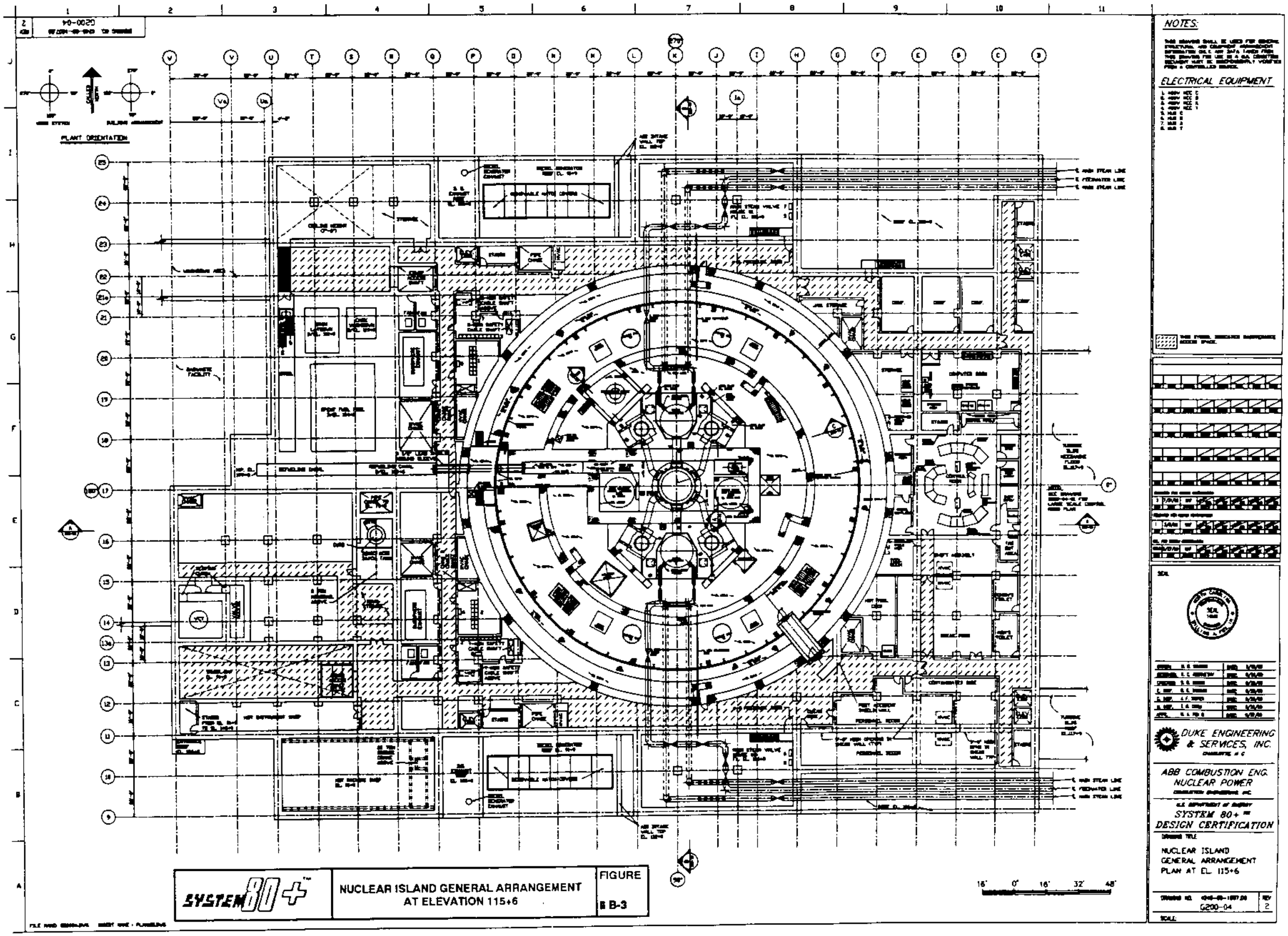


SYSTEM 80+™	SYSTEM 80+ NUCLEAR ISLAND ELEVATION VIEW	FIGURE II B-1

ABB
ASEA BROWN BOVERI



SYSTEM 80+™	SYSTEM 80+ NUCLEAR ISLAND ELEVATION VIEW ROTATED 90°	FIGURE II B-2



SYSTEM 80+

NUCLEAR ISLAND GENERAL ARRANGEMENT AT ELEVATION 115+6

FIGURE B-3

C. REACTOR AND REACTOR COOLANT SYSTEM

The Reactor and Reactor Coolant System (RCS) function together to provide the means for the generation and transfer of heat from the core and fuel to the steam generators. The steam generators subsequently transfer heat to the Main Steam System and the turbine.

The advanced System 80+ ALWR design has evolved from the System 80 NSSS reference design. The primary objective of this evolutionary design effort has been to increase operating margin in a manner which would enhance performance and reliability while preserving the well proven configuration.

For the Reactor, the core operating margin has been increased by reducing the normal operating hot leg temperature, revising core parameter monitoring methods, and by the use of an advanced burnable poison containing Erbrium. The ability to change operating power level (i.e., maneuver) using control rods only (without adjusting boron concentration in the Reactor Coolant System) has been provided, simplifying reactivity control during plant load changes and reducing liquid waste processing requirements.

Improvements have also been made to the RCS. For example, the reactor pressure vessel is ring-forged with material specifications that result in a sixty year end-of-life RT_{NDT} well below the current NRC screening criteria. The ring-forged design results in a significant reduction in the number of welds (with resulting reduction in inservice inspection) and eliminates concern for pressurized thermal shock. Also, the pressurizer volume has been increased relative to the System 80 design to enhance transient response and reduce unnecessary challenges to safety systems. Furthermore, the System 80+ steam generators include Inconel 690 tubes, improved steam dryer efficiency, and a seventeen percent increase in overall heat transfer area relative to System 80, and a ten percent margin for potential tube plugging. The steam generators have a twenty-six percent larger secondary feedwater inventory than System 80 to extend the "boil dry" time and improve system response to upset conditions. Steam generator improvements including larger and repositioned manways, a standby recirculation nozzle, and a redesigned flow distribution plate have been made to facilitate maintenance, and to maintain long term integrity.

The reactor and RCS include the reactor vessel, reactor internals, core, steam generators, pressurizer and associated coolant pumps, piping, and valves. There are two parallel heat transfer loops each containing one steam generator (SG) and two reactor coolant pumps (RCPs). The pressurizer is connected by a surge line to one of the reactor vessel outlet pipes.

The reactor is an advanced version of the light water-moderated and cooled pressurized water reactors (PWR) that are operating today. For use as a Plutonium burner, the fuel is mixed oxide (MOX) in the form of sintered pellets enclosed in standard Zircaloy tubes. Fuel alternatives or strategies for the selected mission objectives are described in Section III.

Control of the core power distribution and reactivity is achieved by a combination of fuel loading pattern, burnable absorber rod depletion, chemical shim reactivity control, and selected patterns of control element assembly (CEA) insertion and withdrawal.

The RCS arrangement is shown in Figure II-C-1 with a flow block diagram shown in Figure II-C-2. Reactor coolant enters the reactor vessel through inlet nozzles, flows downward between the reactor vessel shell and core barrel, into the lower plenum where flow distribution is equalized, and then upward through the core, removing heat generated by the nuclear reactor core. Coolant leaves the reactor vessel shell through outlet nozzles and enters the tube side of the vertical shell economizer steam generators, where heat is transferred to the secondary system. Steam generated in the shell side of the steam generator passes through moisture separators and dryers to ensure that moisture content at the steam generator outlet is minimized. After leaving the steam generators, reactor coolant is returned to the reactor vessel by limited leakage mechanical seal reactor coolant pumps.

The supports for each of the individual components of the Reactor Coolant System are designed to form an integrated support system which mitigates the effects of earthquakes, branch line breaks, and steam line breaks on the RCS. This capability has been achieved without sacrificing the ability of the Reactor Coolant System to expand thermally with minimum restraint. The NRC has approved elimination of design basis pipe breaks in the main loop piping.

Therefore, there are no pipe whip restraints or jet impingement shields required for the main loop piping, and dynamic loads due to postulated main loop pipe breaks are not imposed upon component support structures or other plant equipment.

The design pressure and temperature of the reactor coolant system are 2500 psia and 650°F. The system will operate at a pressure of 2250 psia and has a design lifetime of 60 years.

A description of the RCS major components including their function, design basis, major parameters and operation follows. Also, see Table II-1, for a listing of overall NSSS design characteristics.

1. Reactor Vessel

The reactor vessel is designed to contain and support the core and fuel. A major improvement in manufacturing and operation has been achieved through the use of ring-forgings. The use of a forging as opposed to rolled and welded plates used in previous vessel designs reduces the number of welds and the overall complexity of the vessel and thus reduces fabrication time. Furthermore, the remaining circumferential welds have been relocated to areas of lower neutron flux thus enhancing the vessels resistance to brittle fracture. To further ensure vessel integrity throughout its 60 year design life an analysis regarding vessel brittle fracture is performed.

The reactor vessel is designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Vessels.

The reactor vessel is comprised of the reactor vessel assembly, supports, level indication and surveillance specimens.

a. **Reactor Vessel Assembly**

The Reactor Vessel Assembly (Figure II-C-3) is a vertically mounted cylindrical vessel with an integral hemispherical lower head and a removable hemispherical upper closure head. The reactor vessel is fabricated from low alloy steel. The internal surfaces that are in contact with the reactor coolant are clad with austenitic stainless steel.

The reactor vessel consists of a vessel flange, three shell sections (upper, intermediate, and lower) and a bottom head. The vessel flange is a forged ring with a machined ledge on the inside surface to support the core support barrel, which in turn supports the reactor internals and the core. The flange is drilled and tapped to receive the closure studs and is machined to provide a mating surface for the reactor vessel closure seal rings. Each shell section consists of one 360 degree forged ring. The bottom head is constructed of a single hemispherical forging. The three shell sections, the bottom head forging and vessel flange forging are joined together by welding, along with four inlet nozzle forgings and two outlet nozzle forgings to form a complete vessel assembly.

The centerlines of the coolant flow nozzles are located on a common horizontal plane in the nozzle shell course. A boss is located around the two outlet nozzles on the inside diameter of the vessel wall to provide a mating surface for the internal structure and to guide outlet flow. This boss and the mating outlet sleeve on the core barrel are machined to a common contour to control reactor coolant bypass leakage. The transition section joining the nozzle and vessel shell courses is tapered externally.

The closure head is fabricated separately since it is joined to the reactor vessel by bolting. The closure head consists of a head flange and a dome.

The head flange is a forged ring. The flange is drilled to match the vessel flange stud hole locations, and the lower surface of the flange is machined to provide a mating surface for the vessel closure seal rings. The dome is constructed of a single hemispherical forging. The dome and flange are welded together to form the closure head, and the control element drive mechanism (CEDM) nozzles are welded into the head to complete the assembly.

The studs for the closure head are tensioned using hydraulic stud tensioners. Flange sealing is accomplished utilizing silver plated Ni-Cr-Fe alloy self-energizing O-Rings.

The four nozzles for direct vessel injection (DVI) of borated water from the Safety Injection System are provided near the top of the cylindrical vessel.

b. **Reactor Vessel Supports**

The reactor vessel is supported by four columns located under the vessel inlet nozzles (See Figure II-C-4). A pad on each inlet nozzle provides a surface to which the column is bolted. The sides of the pads are designed to mate with the reactor cavity embedment structure and allow radial movement of the vessel during thermal expansion while restraining it during earthquakes and branch line pipe breaks.

The columns are designed to support the vessel and restrain vertical motion during earthquakes and following branch line pipe breaks.

The lower end of each column terminates in a pad which acts as a keyway in conjunction with a key welded to the lower head of the vessel. The combination assists in restraining the vessel during earthquakes. Load transfer from the reactor vessel column to the building takes place through a bolted joint into an embedment in the concrete.

The analyses and criteria used in the design of the reactor vessel supports consider the forces associated with a Safe Shutdown Earthquake in combination with a branch line pipe break, including the asymmetric reactor internals hydraulic forces.

c. Reactor Vessel Level Indication

The water level within the reactor vessel is monitored using several methods. The first method is to monitor vessel water level inventory using the inadequate core cooling (ICC) monitoring instrumentation. The ICC instrumentation contains the reactor vessel level monitors which use heated junction thermocouples (HJTCs). The reactor vessel level monitors provide control room indication of the status of the liquid inventory in the reactor vessel.

Another method of monitoring the water level in the reactor vessel during refueling operations is the Shutdown Reactor Coolant Level Monitor (SRCLM). The SRCLM is designed to meet the EPRI ALWR reactor vessel level instrumentation requirement for measuring coolant level in the RPV during depressurized shutdown conditions, with or without the head in place. The SRCLM measures the coolant level with both wide- and narrow- range HJTCs and dP instruments.

The SRCLM consists of the piping, valves, transmitter, indicator and other equipment required for monitoring reactor coolant level during shutdown conditions. The SRCLM displays Reactor Coolant System (RCS) level in the control room via the Discrete Indication and Alarm System (DIAS) and the Data Processing System (DPS).

The SRCLM is permanently attached to the RCS hot leg piping, but is only valved-in during shutdown operation, thus eliminating the need for blind flanges (and the associated hookup problems). This configuration permits valving, testing and maintenance outside the secondary shield wall, and indication in the control room to improve ALARA considerations.

d. Reactor Vessel Design - Material Selection

The reactor vessel is fabricated from SA-508 steel forgings with controlled copper, nickel, sulfur, and phosphorous content in the beltline region of the vessel.

The cooldown time from operating temperature to refueling temperature of 27.5 hours is not limited by reactor vessel irradiation and the RT_{NDT} shift.

Likewise, there is no effect on the startup sequence. Plant heatup rate is limited by the heat input from the pumps. No additional limit needs to be imposed at the end of life as a result of reactor vessel irradiation and RT_{NDT} shift.

2. Reactor Vessel Internals

The Reactor Vessel Internals provide the structural support for the core and fuel located within the reactor vessel. The internals also provide for the alignment of the core with the CEA's, CEDMs, and instrumentation, and serve as a shield to protect the reactor vessel from radiation damage. In addition, they also accommodate the expansion and contraction of the core.

The Reactor Vessel Internals are composed of core support structures and internal structures. Core support structures are those that restrain the core. All other structures within the reactor vessel, exclusive of the fuel assemblies, in-core instrumentation, and control element assemblies, are considered internal structures. Core support structures are designed, fabricated and stamped in accordance with the ASME boiler and pressure vessel code Section III, Sub-section NG.

The Reactor Vessel Internals and their relationship to the rest of the reactor are shown in Figure II-C-5. The Reactor Vessel Internals guide the primary coolant through the core. The core support barrel assembly, in conjunction with the water in the downcomer annulus, provides a shield for the reactor vessel against radiation damage by the neutron flux. The core support structures also provides alignment of the core with the control drives, instrumentation, and the reactor vessel.

The reactor vessel internals are comprised of two basic components: (1) a core support barrel assembly and, (2) an upper guide structure assembly. Both assemblies are designed such that maximum operating stresses are less than the values specified in Section III, Sub-section NG of the ASME pressure vessel code, for normal operating and transient plant conditions. The reactor vessel internals are fabricated from austenitic stainless steel. A high strength stainless steel holddown ring provides an axial load on the internals to prevent harmful vibration.

Welded connections are used throughout the design. Full penetration welds, where required, are designed to develop full strength of joined members.

a. Core Support Barrel Assembly

The major structural member of the Reactor Internals is the core support barrel assembly. The core support barrel assembly consists of the core support barrel, the lower support structure, and core shroud. The material for the assembly is austenitic stainless steel.

The core support barrel assembly is supported at its upper end by the upper flange of the core support barrel, which rests on a ledge in the reactor vessel flange. Alignment is accomplished by means of four equally spaced keys in the flange, which fit into the keyways in the reactor vessel ledge and reactor vessel closure head. The lower flange of the core support barrel supports, secures, and positions the lower support structure, and is attached to this structure by means of a welded flexural connection. The lower support structure provides support for the core by means of support beams that transmit the load to the core support barrel lower flange. The locating pins in the beams provide orientation for the lower ends of the fuel assemblies. The core shroud, which provides a flow path for the coolant, is also supported and positioned by the lower support structure. The lower end of the core support barrel is restricted from excessive radial and torsional movement by six snubbers which interface with the pressure vessel wall.

b. Upper Guide Structure

The upper guide structure (UGS) assembly aligns and laterally supports the upper end of the fuel assemblies, maintains the control element spacing, holds down the fuel assemblies during operation, prevents fuel assemblies from being lifted out of position during severe accident condition, and protects the control elements from the effects of coolant flow in the upper plenum.

The upper guide structure consists of a flange forging, suspended from the reactor vessel ledge, and a cylindrical shell, welded to the flange supporting the upper end tube sheet structure that provides a guide path for individual control rods. The lower end of the tube sheet provides support and alignment of the upper end of the Fuel Assemblies. A shroud assembly within the cylindrical shell provides a guide path for the control element assemblies (CEAs). The arrangement of the upper guide structure is shown on Figure II-C-6.

The unique System 80+ tubesheet upper guide structure allows control assemblies to serve more than one fuel element and permits greater flexibility in selecting rod strength in relation to the functions performed. Two types of control assemblies are provided, utilizing four or twelve absorber rods. Four-rod assemblies are used for load maneuvering and shaping of radial power distribution. This design feature allows use of the same proven control element drive mechanism for all types of control assemblies.

Twelve rods or fingers are used for the reactivity shutdown control assemblies which bridges five fuel assemblies. This expanded spread of control fingers, permitted by the design of the tubesheet upper guide structure, provides more uniform and effective reactivity control. This improve method of control rod function is one of the key reasons for the plutonium capability of System reactors.

c. Flow Skirt

The flow skirt is a cylindrical structure having a large number of holes and located between the core support barrel and lower head of the reactor vessel. The flow skirt provides coolant flow distribution reduces the pressure drop in the lower plenum region. The flow skirt is designed to withstand the static, cyclic, and shock loads resulting from hydraulic pressure drop, vibration, and earthquake accelerations.

3. Reactor Core and Fuel

The Reactor Core is designed to generate the required thermal output of 3931 MWt. The core provides a flowpath for the forced circulation of coolant to remove heat generated by the core under all power operating conditions, and for the removal of decay heat by natural or forced circulation under shutdown conditions. The fuel contains the fuel rods which are designed to transfer the heat generated by fission reactions in the fuel pellets to the reactor coolant. The fuel also contains poison rods which provide negative reactivity in the fuel throughout core life in conjunction with soluble boron distributed throughout the reactor coolant. The heat transferred from the core and fuel is circulated by the reactor coolant to the steam generator where the heat is used to generate steam.

The fuel rod design provides two barriers to the escape of fission products. The ceramic pellets operate well below the melting point, and as a result, retain most of the fission products within the structure of the fuel. The small fraction of released fission products is contained within the hermetically sealed Zircaloy-4 cladding tube. In conjunction with the reactor coolant system envelope, the containment, and reactor building, these barriers provide five barriers against the release of fission products.

The combination of cladding thickness and fuel rod pressurization provides additional margin to the design criteria regarding internal pressure buildup within the fuel rod and the structural integrity of the fuel cladding. The positive spring loading feature of the leaf spring spacer design restricts lateral fuel rod motion and thus prevents fretting, yet allows free axial expansion of each element. These provisions, combined with the specified burnup and precise quality control during fabrication, reduce the probability of fuel rod failure.

The cylindrical control elements are provided with full length guidance in the core and in the area of radial flow above the core, and move downward under the influence of gravity when released. The unique design of the upper guide structure and control element assemblies provides exceptional mechanical simplicity, and ruggedness for withstanding and protecting all control element assembly fingers from the combined effects of seismic and blowdown loads resulting from a LOCA. This further adds to the reliability of operation of the reactor protection systems. The control element arrangement provides ample shutdown margin under all conditions and without restriction of the fuel management scheme, even with the control element of highest worth stuck out of the core.

a. Configuration

The Reactor Core and fuel for the design is made up of the fuel assemblies including MOX fuel and poison rods, control element assemblies (CEA) and drives (CEDMS).

Core

The reactor core consists of an array of 241 mechanically identical fuel assemblies in an arrangement that approximates a right circular cylinder. The core also includes control element assemblies (CEAs), in-core instrumentation assemblies, and neutron sources inserted into the fuel assemblies, together with the reactor vessel internals which support and position the core and guide the coolant flow. There are no internal shrouds, channels or poison curtains in the core. This simplifies the reactor design and permits flexibility in fuel arrangement. All of the structural materials in the active core zone, including the CEA guide tubes and the spacer grids, are Zircaloy, which eliminates concern over the formation of lower melting point authentic during a loss of coolant accident (LOCA) as a result of using dissimilar metals and which provides excellent neutron economy.

Figure II-C-7 shows the cross-sectional arrangement of the fuel assemblies as positioned in the reactor core, and the arrangement of the core within the core support structure. The individual components of the reactor core are described in the following paragraphs. Details of the Plutonium burner fuel design is included in Section III-A.

b. Fuel Rod

Each fuel rod consists of dish-ended and chamfered MOX pellets, upper and lower spacer discs, a cladding tube, upper and lower end caps, and a helical spring. The fuel is manufactured in the form of pellets having a material microstructure and configuration which minimizes the effects of fuel densification during irradiation. The pellet column rests on an aluminum oxide spacer disc at the bottom of the cladding tube, and is held in place by a stainless steel compression spring acting on a similar spacer disc at the top of the column. The space at the top of the column allows for fuel and gas expansion. A cold diametrical gap of sufficient size is allowed to provide for differential expansion between the cladding and fuel, which limits clad strain. Net unrecoverable circumferential clad strain will not exceed one (1) percent, as predicted by analysis considering clad creep and fuel-clad interaction effects.

The cladding tube is slightly cold-worked Zircaloy-4. The ratio of clad thickness to outer diameter is large enough to provide adequate margin in limiting the maximum short-term cladding stresses to less than the yield strength. The fuel rods are internally pressurized with helium to improve both their reliability and performance characteristics.

Pressurization results in reduced clad stresses and strains because of a smaller differential pressure across the cladding and a reduction in pellet-clad interactions. Thus, under internal pressure and with the relatively large wall thickness, the cladding tube is essentially free-standing. In addition, the high concentration of helium inside the tube improves gap thermal conductivity and thus lowers fuel pellet temperatures. Lower fuel temperatures are beneficial not only with respect to clad integrity at steady state conditions, but also in the case of transients where peak clad temperature is important.

The end caps are welded to the cladding tubes by an exclusive ABB-CE process involving magnetic force welding. This process has the advantage of producing a high degree of uniformity and consistency of weldment, thus providing a very high degree of leak tightness. There are no gas bubbles, as may be found in more conventional methods of welding, that could lead to weld porosity and leakage. All end cap welds are 100 percent helium leak tested.

c. Fuel Assembly

The fuel rods are arranged in a 16 x 16 square array to form a fuel assembly, as shown in Figure II-C-8. The lower end cap of each fuel rod engages the lower end fittings of the fuel assembly, which provides lateral support of the rods. Lateral support and positioning is maintained throughout the length of the rods by spacer grids. The upper end of the fuel rod is free to expand in the axial direction.

Five (5) Zircaloy guide tubes are welded to the spacer grids and mechanically attached to the upper and lower end fittings. These guide tubes are a key feature of the fuel assembly design. They provide channels which guide the CEAs over their entire length of travel within the core, as well as space for the in-core instrumentation and neutron sources. These guide tubes form the longitudinal structure of the assembly and offer the advantage of simple, rugged construction.

The guide tubes form a closely spaced repetitive array of water channels throughout the core, which permits the use of rugged, yet adequately flexible control elements. The use of rugged, large-size control elements simplifies the design and increases the reliability of associated reactor hardware.

The fuel spacer grids maintain spacing between the fuel rods and restrict their movement. These grids consist of a series of preformed strips joined in an eggcrate fashion and welded together. Each fuel rod is laterally positioned by two leaf springs which are integral with two of the adjacent strips. The springs press the fuel rod against two stiff arches, which are integral with the other two adjacent strips. The grid perimeter strips have punched-out sections adjacent to the fuel rods to provide local cooling. Although the spacer grids position the fuel rods laterally, the rods are still free to expand axially.

The fuel assembly lower end fittings contain coolant flow holes. Alignment of the lower end and support of the assembly is provided by the core support structure. Lateral location of the upper end of each fuel assembly is provided by the lower ends of the control element shroud tubes which extend downward from the alignment plate which is part of the guide structure above the core. Fuel assembly positioning is accomplished by the control element shroud tube extensions into which the fuel assembly extension is guided. The shroud tube extensions prevent the assemblies from moving so far upward as to disengage from the lower core support grid in the event of an accidental flow surge. Holddown springs on the upper end fitting react against the shroud tube extensions preventing fuel assembly uplift due to hydraulic forces and allowing differential expansion between the fuel assembly and the core internals. The upper end fitting also serves as the lifting fixture for the fuel assemblies.

d. <u>Control Element Assemblies (CEAs)</u>

Control Element Assemblies (CEA) can be either four (4) or twelve (12) fingered as shown in Figure II-C-9. The CEA fingers are fixed to a spider which serves as the central support structure for the 4- and 12-fingered CEA assemblies. The hub of the spider also couples the CEA to the drive mechanism through the Control Element Drive Mechanism (CEDM) extension shaft assembly.

Each control element in the CEA is guided and shrouded from the main coolant flow above the core by the individual control finger shroud tubes and by the CEA shroud. In-core guidance is provided by the guide tube, which is an integral part of the fuel assembly. The guide tubes are open near the bottom to allow entry of coolant. Inlet orifices and control element fingers limit bypass flow through the tubes.

All CEAs used in the core design contain neutron absorbing (poison) materials which extend over the active length when fully inserted.

e. Control Element Drive Mechanisms (CEDM)

The magnetic jack CEDM is a completely sealed, magnetically operated linear actuator. External coils produce a magnetic field, which operates driving and holding latches. Proper sequencing of these coils produces the linear motion of the driveshaft. Using external driving coils, the mechanism transfers loads with no sliding wear on latching members.

The general arrangement of the magnetic jack CEDM is indicated in Figure II-C-10, showing the location of the external coils and latches. The motor housing is of Type 403 heat treated stainless steel and the upper pressure housing is austenitic stainless steel. The pressure retaining components are designed in accordance with Section III of the ASME Boiler and Pressure Vessel Code for 2500 psi and 650°F.

Position indication is achieved by means of redundant reed switch position transmitter assemblies (two per CEDM) attached to the upper pressure housing, and also by counting power pulses which actuate the CEDM driving CEAs up or down. Each reed switch assembly is operated by a magnet in the upper end of the driveshaft. Travel is limited by reed switches which actuate upper and lower electrical limit relays. The travel is also limited by the length of the toothed section of the driveshaft which is designed to prevent the extension shaft assembly from being driven into the pressure housing.

All electrical connections are made at the top of the mechanism with pin type stainless steel electrical connectors. A shroud surrounds the upper pressure housing enclosing the reed switch position transmitter assemblies, providing additional structural support for the CEDM, and serving as a lifting device for removal of the coil stack assembly for inspection of the pressure housing.

The CEDM motor housing is completely seal welded to prevent leakage during operation. However, the design of the CEDM facilitates service of its internals. The top seal on the motor housing is designed to permit removal of the CEDM internals by remotely cutting (and rewelding) the pressure housing. The lower seal and attachment to the vessel nozzle will not be disturbed for any normal mechanism maintenance or inspection but is designed for housing replacement, should a housing become damaged.

f. Thermal and Hydraulic Design

The core is designed with sufficient margins to departure from nucleate boiling (DNB) and center fuel melting under normal and transient conditions to provide highly reliable reactor performance. The reactor control and protective systems provide for automatic trip or other corrective action before these design limits are reached.

g. Reactor Coolant Flow

The coolant flow path can be traced by referring to Figure II-C-5. Coolant enters inlet nozzles and flows in the annular region (downcomer) between the reactor vessel and core support barrel. It then flows downward and enters the lower plenum through the flow skirt.

The flow skirt acts on the coolant to provide a nearly uniform radial distribution of flow into the lower plenum. The flow then proceeds upwards through the bottom plate and core support structure, which further smooth out the flow and provide a nearly uniform flow distribution into the core. After passing through the core, the coolant enters the upper plenum region and flows outward through the upper guide structure tube bank. The tube sheet upper guide structure provides individual protection to each control finger from vibration and other hydraulically induced forces, yet permits adequate area between the shrouds for coolant flow.

h. Maximum Fuel Temperature

Fuel will be operated at all times at a temperature substantially below the melting point of ~2800°C (~5000°F). As a first approximation, the temperature rise in the fuel above the coolant temperature is proportional to the linear heat release rate, which is generally expressed in terms of kilowatts per foot.

i. Reactor Control Characteristics

Reactor power increases or decreases with reactivity insertion or removal. Reactivity insertion may be accomplished by Control Element Assembly (CEA) withdrawal, dilution of reactor coolant boron concentration or, to a lesser degree, by reduction of reactor coolant average temperature. Conversely, reactivity is removed (negative reactivity inserted) by CEA insertion, increasing the reactor coolant boron concentration, or increasing the reactor coolant average temperature.

Very slow but ultimately very large changes in reactivity may be made by adjusting the boron concentration. This means of changing reactivity is normally used to compensate for fuel burnup or to override fission product buildup and is manually controlled by the reactor operator.

Changes in reactor power in response to changes in turbine load are accomplished automatically by means of regulating CEA movement. No change in soluble boron concentration is necessary. The Reactor Regulating System (RRS) senses reactor power, reactor coolant temperatures, and a load reference (usually turbine power) and provides signals to the controlling CEA group demanding direction and speed of movement, if required.

After a load change, the RRS brings the reactor to a new steady-state power level. Simultaneously, reactor coolant average temperature is adjusted to a value programmed with reactor power to maintain design steam conditions. Average temperature is then allowed to "float" about the programmed value within a dead band to take advantage of the reactor coolant temperature reactivity effect, which provides a negative feedback loop. This effect stabilizes reactor power, eliminating the need for frequent CEA movement.

CEA position monitoring is performed by two diverse and independent indication systems. One system consists of reed switch assemblies (two independent stacks per CEDM) attached to the CEDM upper pressure housing. The reed switches are operated by a magnet in the upper end of the driveshaft and act to add or subtract resistance in a voltage divider network. A voltage signal proportional to CEA position is thereby generated and is displayed to the operator in bar-graph form on a video display unit. The other system utilizes the plant computer to count "RAISE-LOWER" pulses to the CEDMs. Position is displayed by means of digital indicators and by printouts.

Reactor Core Monitoring

The In-Core Neutron Monitoring System is used to determine the neutron behavior within the reactor core. The system is comprised of a Fixed In-Core Detector System (FICDS) which ABB-CE has provided for its pressurized water reactors over the last twenty years. The FICDS utilizes rhodium self-powered neutron detectors (SPNDs) to perform continuous on-line monitoring of the flux distribution in the reactor core which is used in the calculation of other important core parameters such as peak linear heat rates, axial power distributions, azimuthal tilts and limits to departure from nucleate boiling (DNB).

The FICDS has significant advantages for reactor core monitoring over a system which uses ex-core detectors in combination with a movable in-core detector flux mapping system. The basic advantage of using fixed instead of movable in-core detectors is that fixed detectors allow the core power distribution to be observed on-line and can be used continuously for core monitoring. Movable detectors permit only periodic observation of the core power distribution, and allowance must be made for changes between observations. Further, by using information from in-core rather than ex-core detectors to monitor the core, greater accuracy is achieved, which yields increased margins to thermal limits. These margin gains can be translated into one or more of the following: higher allowable power levels or peaking, additional operating flexibility, and additional flexibility in fuel management.

In addition to the FICDS, an optional Movable In-Core Detector System (MICDS) is available which provides redundant and diverse core power distribution information. The MICDS includes two movable miniature fission chambers, movable detector signal conditioning equipment, and the movable drive system hardware and software.

4. Steam Generators

The two vertical, U-tube steam generators in the NSSS provide the means of transferring heat from the primary system (i.e., Reactor and RCS) to the secondary system (i.e., main steam, turbine and feed and condensate systems.) One steam generator is located in each loop.

Each steam generator is a recirculating, vertical, U-tube heat exchanger with an integral axial flow economizer as shown in Figure II-C-11.

Both primary tube and secondary shell sides of the steam generator are designed and fabricated in accordance with ASME Boiler and Pressure Vessel Code Section III, Rules for Construction of Nuclear Vessels.

Reactor coolant enters the steam generator inlet plenum via the primary inlet nozzle, flows up through the tubesheet and U-bend heat transfer tubing, returns through the tubesheet to the steam generator outlet plenum and exits via two outlet nozzles. Vertical divider plates separate the primary inlet and outlet plenums of the steam generator. All surfaces in contact with reactor coolant are either Ni-Cr-Fe alloy or are clad with stainless steel or Ni-Cr-Fe alloy. Specifically, the heat transfer tubes are made with thermally treated Alloy 690. The tubesheet cladding is weld deposited to obtain the strongest type of metallurgical bond. The center support cylinder minimizes bending moments on the tubesheet, and provides a solid assembly for distribution of support loads. Large manways (21 in.) are provided in the two plenums.

The Alloy 690 heat transfer tubes are connected to the tubesheet by first being seal welded to the primary side cladding and then "explanded" (i.e., "explosively expanded") into the tube sheet by detonating an explosive contained in a plastic sheath inserted into the tube. The detonation applies a uniform force throughout the thickness of the tubesheet. In contrast to mechanical rolling processes, there is no scoring or local thinning and minimal springback of the tube material. "Explansion" through the full tubesheet thickness also minimizes the potential for solids collection and crevice corrosion by eliminating the gap between tube and tubesheet on the secondary side.

The integral axial flow economizer is shown in Figure II-C-12. Figure II-C-13 shows a cutaway view of the economizer region.

The economizer increases the cold leg side temperature difference for heat transfer by bringing relatively cold feedwater into contact with the primary outlet or cold leg side of the tube bundle. Feedwater enters the economizer region via the two main feedwater nozzles. This feedwater is introduced into a distribution box of rectangular cross section which forms a half ring around the cold leg portion of the tube bundle. Feedwater leaves the distribution box through uniformly distributed holes at the bottom and then flows radially inward across the tubesheet beneath a flow distribution baffle. The flow distribution baffle is designed to provide a uniform axial mass velocity of feedwater at the economizer inlet.

Above the flow distribution baffle, feedwater flows in axial counterflow through the tube bundle until heated to saturation. Tube supports in this region have the ABB-CE "eggcrate" design. A divider plate is mounted in the tube lane between the hot and cold leg sides to separate the economizer half (cold leg side) of the lower cylindrical section from the evaporator half (hot leg side). The divider plate is attached to the lower pressure shell and to a center support cylinder by tongue and groove joints so that there is no structural interaction with the secondary shell under pressure and thermal deflections. Localized, statically indeterminate stress problems with the pressure shell are therefore avoided. (A similar design is also used for the divider plate separating inlet and outlet plenums of the primary head).

At the top of the economizer section, the feedwater, having been heated to saturation conditions, mixes with recirculating water in the evaporator section of the tube bundle. In the evaporator, heat transfer by nucleate boiling occurs as the secondary fluid flows upward through the tube bundle, continually increasing in steam quality. Quality at the tube bundle exit is approximately 30 percent. Unitized steam-water separators (Figure II-C-14) mounted on a deck plate at the top of the tube bundle shroud separate the steam from the two-phase mixture, with the steam flowing upward through a secondary stage of steam-water separators or steam dryers (Figure II-C-15) and leaving the steam generator essentially dry (better than 99.75 percent quality).

Water removed by the steam separators recirculates into the evaporator section through an annular downcomer formed by the pressure shell and the tube bundle shroud. The downcomer and tube bundle shroud are designed so that approximately 60 percent of the recirculating water enters the bundle at the tubesheet on the hot leg side and approximately 40 percent enters through an opening in the shroud on the cold leg side at the top of the economizer section. In this manner, density differences between the hot leg (evaporator) side and the cold leg (economizer) side are reduced at the level of the economizer discharge, and the potential for lateral density wave instabilities is minimized. The net recirculation ratio (total evaporator mass flow to steam flow) is typically greater than 3.5 to 1.

Two feedwater inlet systems, one located in the economizer region and the other in the uppershell region, are provided for the steam generators. In the economizer region of the lower shell, two (2) 14-inch main feedwater nozzles supply feedwater to the economizer water box. Two nozzles are desirable for optimum flow distribution to the economizer, however no special control of feedwater split between these nozzles is required. In the upper shell, a downcomer feedwater nozzle which connects to an internal distribution ring is supplied. Feedwater discharges through flow holes on the top of the internal ring.

The downcomer feedwater nozzle and internal distribution ring have several functions. At very low power levels, when feedwater heaters are not yet effective due to lack of turbine extraction steam, main feedwater flow is admitted only through the downcomer feedwater inlet to preclude thermal shock to the pressure shell and economizer water box parts. In addition, cold emergency feedwater entering the steam generator during certain transient conditions is admitted through this path.

At full power approximately 10 percent of feedwater flow is admitted through the downcomer feedwater nozzle into the downcomer section. With partial feed flow to the downcomer, a degree of subcooling is obtained in the downcomer, promoting an increased recirculation ratio. Even without subcooling in the downcomer, recirculation ratios of 3.5 to 1 are expected in the steam generators. Introduction of feedwater into the downcomer will also retard boiling of the recirculating flow entering the evaporator region, further improving hydraulics in this region.

The flexibility provided by the downcomer feedwater inlet nozzle and internal distribution ring allows maximum protection for the steam generator during start-up and emergency conditions, and maximum thermal efficiency during power operation.

The tube support design provides maximum reliability in that protection from tube damage due to mechanical or flow-induced vibrations, or combined seismic and accident conditions, is provided, while offering minimum resistance to steam/water flow in the tube bundle. Tube supports in the economizer and straight tube portions of the evaporator employ the ABB-CE eggcrate design, which provides maximum open flow area. Particular attention is devoted to flow patterns and velocities at feedwater and recirculating water inlet regions which historically have been susceptible to vibration problems in heat exchangers.

In the U-bend region, the same process of designing with experimentally verified methods is applied to the two-phase steam and water flow region. Figures II-C-16 and II-C-17 show a detail of tube supports in the U-bend region.

Vertical support strips between each panel of tubes interlock with horizontal support strips to provide support against in-plane and out-of-plane flow vibrations, and to restrain the tubes from excessive bending during combined LOCA and seismic incidents. Some of the vertical strips are attached to support beams welded to the tube bundle shroud, providing restraint during steam line break accidents in combination with seismic loadings.

The grid type eggcrate tube support system offers considerable advantages from a hydraulic standpoint. Because of the relatively open eggcrate tube supports present in the design, localized crevices adjacent to tube surfaces are not formed. The tube support system for the steam generator provides a minimum of potential localized steam blanketed areas, such as might be present in the annular gaps between tubes and drilled support plate holes. The open flow area of each eggcrate support is approximately 69 percent. The large open flow area avoids the accumulation of boiler water deposits by eliminating local flow eddies and flat surfaces present in other commonly used tube bundle supports. Avoiding the accumulation of corrosion products helps to avoid the concentration of acid-producing chloride salts which in past steam generator designs has led to accelerated carbon steel support corrosion and subsequent tube denting.

The tube support material used for both horizontal and vertical grids is ferritic stainless steel. This material is employed due to its high resistance to general corrosion and thinning. Ferritic stainless is preferable to austenitic stainless because the coefficient of thermal expansion is more compatible with carbon steel and Inconel.

The steam generators use a zero solids (i.e., all volatile) feedwater treatment for chemistry control. Volatile chemistry control is specified to minimize the solids content of the water. Internal chemistry sampling systems are provided for the hot leg and cold leg side of each steam generator tube bundle via the blowdown lines. In addition, a final sample point is provided in the downcomer shell.

In addition to the use of volatile water treatment chemicals, continuous steam generator blowdown is specified to further minimize the solids content of the water. The steam generator design includes both hot and cold leg blowdown lines. The two lines are similar and consist of rectangular ducting which runs down the tube lane and around the center support cylinder at the tubesheet elevation (Figure II-C-18). Blowdown fluid enters the duct through holes located at the outside radius of the curved portion located adjacent to the center support cylinder. This is the region of lowest horizontal velocity across the tubesheet where settling of heavier solid particles is most likely to occur. The duct terminates before reaching the generator shell and flow is directed through drilled passages in the tubesheet exiting at the blowdown nozzles. The steam generator internal blowdown ducts and passages are sized to accommodate up to a maximum blowdown rate of approximately 10 percent of the individual steam generator's maximum steaming rate. The connecting steam generator blowdown system provides the means to quench, collect, and purify the blowdown fluid.

The steam generator design features dual 28-inch main steam outlet nozzles each containing integral flow elements for use in steam flow measurement and to restrict steam flow in the unlikely event of a steam line break.

A conical skirt welded to the bottom head of the steam generator provides a bolting surface for a heavy steel support plate (Figure II-C-19). Machined cutouts in the plate act as keyways for embedded keys which restrict horizontal motion of the steam generator during earthquakes. Low friction bearing plates at the interface between keys and keyways are utilized to minimize resistance to thermal expansion.

The plate rests on four horizontal spherical low friction bearings on which the steam generator slides during thermal expansion of the system. Spherical bearings are used to simplify erection procedures and to ensure a uniform bearing surface when the steam generator rotates from a true vertical position during thermal expansion. The design incorporates shims between the keys and keyways. The shim sizes are selected after welding of connecting piping to allow motion of the generator without causing the plate to contact the keys.

The upper steam generator supports provide horizontal restraint for the steam generator during earthquakes and following postulated branch line pipe breaks. Low friction bearing plates at the keys minimize resistance to thermal expansion.

The snubber assemblies are designed to take advantage of the low resistance of the snubbers to slow movement and, by means of the mechanical advantage of the lever, resist large forces and responses associated with earthquakes and branch line pipe breaks. Branch line pipe breaks considered are those postulated after application of the leak-before-break methodology.

5. Pressurizer

The pressurizer maintains RCS operating pressure and, in conjunction with the CVCS, compensates for changes in reactor coolant volume during load changes, heatup, and cooldown. During full-power operation, the pressurizer is about one-half full of saturated steam.

RCS pressure may be controlled automatically or manually by maintaining the temperature of the pressurizer fluid at the saturation temperature corresponding to the desired system pressure. A small continuous spray flow is maintained to the pressurizer to avoid stratification of pressurizer boron concentration and to maintain the temperature in the surge and spray lines, thereby reducing thermal shock as the spray control valves open. An auxiliary spray line is provided from the charging pumps to permit pressurizer spray during plant heatup, or to allow cooling if the reactor coolant pumps are shutdown.

During load changes, the pressurizer limits pressure variations caused by expansion or contraction of the reactor coolant. The average reactor coolant temperature is programmed to vary as a function of plant power level. Reduction in RCS load is followed by a decrease in the average reactor coolant temperature to the programmed value for the lower power level. The resulting contraction of the coolant lowers the pressurizer water level, causing the RCS pressure to decrease. This pressure reduction is partially compensated by flashing of pressurizer water into steam. All pressurizer heaters are automatically energized on low system pressure, generating steam and further limiting any pressure decrease. Should the water level in the pressurizer drop sufficiently below its setpoint, the letdown control valves close to a minimum value, and additional charging pumps in the chemical and volume control system are automatically started to add coolant to the system and restore pressurizer level.

When steam demand is increased, the average reactor coolant temperature is raised in accordance with the coolant temperature program. The expanding coolant from the reactor coolant piping hot leg enters the bottom of the pressurizer through the surge line, compressing the steam and raising system pressure. The increase in pressure is moderated by the condensation of steam during compression and by the decrease in bulk temperature in the liquid phase. Should the pressure increase be large enough, the pressurizer spray valves open, spraying coolant from the reactor coolant pump discharge (cold leg) into the pressurizer steam space. The relatively cold spray water condenses some of the steam in the steam space, limiting the system pressure increase. The programmed pressurizer water level is a temperature dependent function. A high level error signal, produced by an in-surge, causes the letdown control valves to open, releasing coolant to the chemical and volume control system and restoring the pressurizer to the programmed level. Small pressure and primary coolant volume variations are accommodated by the steam volume that absorbs flow into the pressurizer and by the water volume that allows flow out of the pressurizer.

A number of the heaters are connected to proportional controllers, which adjust the heat input to account for steady state losses and to maintain the desired steam pressure in the pressurizer. The remaining heaters are connected to on-off controllers. These heaters are normally deenergized but are automatically turned on by a low pressurizer pressure signal or a high level error signal. This latter feature is provided since load increases result in an in-surge of relatively cold coolant into the pressurizer, thereby decreasing the bulk water temperature. The CVCS acts to restore level, resulting in a transient pressure below normal operating pressure. To minimize the extent of this transient, the backup heaters are energized, contributing more heat to the water. Backup heaters are deenergized in the event of concurrent high-level error and high-pressurizer pressure signals. A low-low pressurizer water level signal deenergizes all heaters before they are uncovered to prevent heater damage.

The RCS pressurizer is a vertically mounted, bottom supported, cylindrical pressure vessel (Figure II-C-20). Replaceable direct immersion electric heaters are installed vertically in the bottom head. The pressurizer is furnished with nozzles for spray, surge, safety valves, and pressure and level instrumentation. The pressurizer surge line is connected to one of the reactor coolant hot legs and the spray lines are connected to two of the cold legs at the reactor coolant pump discharge.

Design and fabrication of the pressurizer conform to requirements of the ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Vessels Class 1. The pressurizer heaters are single unit, direct immersion heaters that protrude vertically into the pressurizer through Inconel 690 sleeves welded in the lower head. Each heater is internally restrained from high amplitude vibrations and can be individually removed for maintenance during plant shutdown. Protective devices for pressurizer heaters are provided to prevent damage by overloads and short circuits. These devices are usually located in the heater switchgear.

The pressurizer is supported by a cylindrical skirt welded to the bottom head of the pressurizer. The skirt ends in a flange which is drilled to accept anchor bolts. Four keys welded to the upper portion of the pressurizer shell mate with structural keyways to give additional support to the pressurizer during seismic, safety valve actuation, and following a branch line pipe break.

6. RCS Overpressure Protection

Overpressure protection of the RCS is provided in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III. Four (4) spring-loaded safety valves installed in piping connected to the top of the pressurizer are provided. The valves have an enclosed bonnet and a balanced bellows to compensate for backpressure. The valves are sized to pass sufficient pressurizer steam to limit the reactor coolant system pressure. The pressurizer safety valves discharge to the in-containment refueling water storage tank (IRWST) where the steam is released under water through spargers to be condensed and cooled. If the steam discharge exceeds the capacity of the IRWST, it is relieved to the containment atmosphere via vents installed in the tank.

The pressurizer safety valves are qualified for all fluid conditions expected during normal, transient, and accident operations. Operability of the valves is based on the results of test programs such as those conducted by EPRI for safety and relief valves. The set pressure of individual pressurizer safety valves is adjusted using steam, and seat leakage is checked using nitrogen or steam.

Indirect indication of pressurizer safety valve leakage is provided by a decrease of pressurizer pressure and pressurizer level, monitored by safety-grade instrumentation.

Positive indication of pressurizer safety valve position by acoustic monitoring of flow is provided in the control room. The sensing instrumentation is environmentally qualified to function in a post-LOCA environment in accordance with Regulatory Guide 1.89. A plant annunciator alarm is provided to indicate valve opening. The valve position instrumentation is powered from a reliable instrument bus with Class 1E backup power. The system is designed to meet the intent of the requirements in Regulatory Guide 1.97, Revision 3. In addition, temperature sensors are provided in piping downstream of the safety valves with displays in the control room. A temperature increase will indicate pressurizer safety valve leakage.

In sizing the pressurizer safety valves, it is assumed that loss of load does not trip the reactor, but that a delayed reactor trip does occur due to a high pressurizer pressure signal. The pressurizer safety valves are sized to protect the reactor coolant system against overpressure in this incident. No credit is taken for the action of pressurizer spray, letdown, heat transfer to the pressurizer, or steam dump.

The design basis conforms to the applicable sections for the ANS Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants prepared by the ANS-51

Subcommittee. These sections complement Section III of the ASME Boiler and Pressure Vessel Code.

7. Reactor Coolant Pumps and Motors

The Reactor Coolant Pumps and Motors provide the motive force for circulation of the heat transfer medium (i.e., reactor coolant) from the reactor to the steam generators during various modes of plant operation including startup, maneuvering and shutdown.

The Reactor Coolant Pumps and Motors provided for the design are highly reliable with respect to preserving seal integrity during various events including station blackout (SBO).

The reactor coolant is circulated by vertical, single stage, single bottom suction, single horizontal discharge, centrifugal pumps (Figure II-C-21). The pump shaft is sealed by controlled leakage mechanical shaft seals and driven by a vertical AC induction motor.

The pump rotating assembly is mounted in a diffuser type pump casing. The pump casing is a one-piece design in accordance with applicable sections of Section III, ASME Boiler & Pressure Vessel Code. The one piece casing reduces the ASME Section XI examination requirements to only the welds between the pump casing and the suction and discharge nozzle extension pieces and other minor items. The pump impeller is face spliced and locked to the pump shaft. The lower portion of the pump rotating assembly is supported by a self-aligning water lubricated bearing mounted in the pump cover and above the impeller. Additional shaft support is provided by oil lubricated tilting pad radial and thrust bearings mounted on the pump shaft and in the motor. The motor is coupled to the pump by a flexible diaphragm type coupling.

The shaft seal system consists of two mechanical face seals mounted in series with controlled bypass leakage to provide equal pressure differential across each seal. The seal system reduces reactor coolant system pressure to the volume control tank pressure.

Each mechanical seal is designed to withstand full system pressure should the other seal fail. A back-up vapor seal is located above the mechanical seals to prevent liquid or gaseous leakage from escaping to the containment. The back-up seal normally operates against volume control tank backpressure but is capable of sealing against full system pressure in the static condition and during coastdown, following failure of the main seals.

The temperature of the water in the seal assembly is maintained within acceptable limits by externally supplied seal injection water. Water cooled heat exchangers are also furnished to provide the necessary cooling should seal injection fail. The pump is capable of operating continuously without seal injection or without cooling water to the pump seal water coolers. If both normal seal injection and component cooling to the pump seal water coolers are lost, a dedicated positive displacement seal injection pump supplies cooling water to the seals. If this water supply is also lost, the pump must be shutdown. If component cooling water to the oil lubricated bearing cooler is lost, but seal cooling is available by either seal injection water or seal component cooling water, the pump can operate for up to ten (10) minutes without damage before shutdown is required. Once shutdown, the seals are capable of withstanding the effects of a loss of all cooling water without creating a small break LOCA.

The capability of ABB-CE RCP seals to withstand various loss of cooling events including loss of CCW and SI, and still maintain their integrity is demonstrated through factory testing. Seal integrity has also been confirmed through an actual operating event. During pump testing various loss of cooling and seal injection events are performed. Furthermore, an operating event occurred at a plant whereby the seals experienced a total loss of both CCW and SI and continued to maintain their integrity, while sustaining no damage. This particular event was potentially more severe than the coping criteria for station blackout because the RCP was operated without cooling for 10 minutes before the pump was shutdown.

In addition to demonstrated RCP seal integrity under both normal and off-normal operating conditions, the RCP seals can be installed or removed without draining the reactor coolant system or the pump casing, or removing the motor, during refueling operations. A seal removal crane is furnished for mounting in the motor support stand. The crane is used for removing the shaft coupling and the individual seal assemblies. Estimated time for changing the entire seal assembly is six (6) hours. Actual seal change time is dependent upon the degree of training of the maintenance personnel. The seals have a design life of 20,000 hours, and are the only major item requiring periodic maintenance. Performance of the seals is monitored by pressure and temperature sensing devices installed in the seal assembly. Controlled bypass leakage is also monitored by a flow measurement device.

The motors are sized for continuous operation over the design flow range with 1.0 to 0.74 specific gravity water. The motors are designed to start and accelerate to operating speed under full load with an initial drop to 80 percent of rated voltage at the motor terminals. Each motor is provided with an anti-reverse rotation device, designed to withstand each of the following conditions: motor starting torque resulting from reversed power leads; reactor coolant flow through the pump in the reverse direction of up to 52 percent of rated capacity with the motor deenergized; and reverse flow due to LOCA of sizes up to the largest remaining pipe break after application of leak before break methodology.

The motor is furnished with a flywheel to provide sufficient coastdown flow for core cooling following a loss of power to the pumps. The flywheel design and manufacture is based on applicable requirements including an overspeed test at 125 percent of rated speed. The pump rotating assembly, including the flywheel, is designed to withstand the overspeeds resulting from a LOCA up to the largest remaining pipe break after application of leak before break methodology without generating missiles.

An air-to-water heat exchanger is furnished on the motor to absorb the motor heat load and thereby reduce the cooling requirements to the containment ventilation system. Temperature sensing devices are furnished for monitoring stator temperatures and oil lubricated bearing temperatures.

The portion of the reactor coolant pump designed to ASME Section III is examined by radiography to the specified acceptance criteria during the manufacturing process. The design of the RCP permits access for the examinations required by Section XI. This design does not require the use of special tools, instrumentation, indicating devices, switches, etc. beyond those required to do the UT examination.

All rotating parts of the pump are statically and dynamically balanced. All pump assemblies are full scale performance tested in the vendor's shop in accordance with the Standards of the Hydraulic Institute to verify hydraulic performance as well as the ability of the pumps to function as required by the specifications. The vibration levels are monitored during this test.

The reactor coolant pump supports (Figure II-C-22) are designed to prevent significant motion of the pump during an earthquake and following branch line pipe breaks. The pump assembly, including the motor, is a relatively sensitive component which cannot tolerate large forces due to thermal expansion. The support design therefore accommodates the operating requirements of the pump assembly while providing restraint for abnormal conditions.

A skirt bolted to the pump casing provides attachment points for the four vertical columns which support the pumps. Each column is fitted with a ball and socket connection at both ends. This allows each column to act independently during thermal expansion. The result is an articulated support which does not impose significant forces during thermal expansion, but which provides vertical support for the pump during earthquakes and following branch line pipe breaks.

Horizontal support is provided by four horizontal columns, articulated in the same manner as the vertical columns, and by a system of horizontal snubbers. Two of the horizontal columns are located at the bottom of the skirt, the other two columns and snubber are located at the top of the motor support stand.

8. Reactor Coolant Piping and Valves

The Reactor Coolant Piping and Valves which provide the conduit for the flow of reactor coolant from the vessel and core to the Steam Generators is designed, fabricated, and installed in accordance with the applicable sections of the ASME Code. Furthermore, the pipe whip restraints required by the NRC on previous plant designs have been eliminated by the application of leak-before-break analyses.

The reactor coolant system consists of two heat transfer loops. The arrangement of the system components is shown in Figure II-C-1.

Each of the two heat transfer loops contains five sections of pipe: one (1) 42-inch internal diameter pipe between the reactor vessel outlet nozzle and steam generator inlet nozzle, two (2) 30-inch internal diameter pipes, one from each of the two (2) steam generator outlet nozzles to each reactor coolant pump suction nozzle, and two (2) 30-inch internal diameter pipes, one from each pump discharge nozzle to a reactor vessel inlet nozzle. These pipes are referred to as the hot leg, the pump suction legs, and the pump discharge legs, respectively. The pump suction and discharge legs taken together are referred to as the cold legs. The other major pieces of reactor coolant piping are the surge line and the spray line.

The 42-inch and 30-inch pipe diameters are selected to obtain coolant velocities which provide a reasonable balance between erosion-corrosion, pressure drop, and system volume. The surge line is sized to limit the frictional pressure loss through it during the maximum in-surge so that the pressure differential between the pressurizer and the heat transfer loops is no more than 5 percent of the system design pressure. Thermal stratification effects are also taken into account in the design of the surge line.

To reduce the amount of field welding during plant fabrication, the 42-inch and 30-inch pipes are supplied in major pieces, complete with shop-installed instrumentation nozzles and connecting nozzles to the auxiliary systems. Where necessary, nozzles incorporate thermal sleeves for protection of piping from thermal stresses. The amount of Safety Class 1 piping connected to the reactor coolant system is minimized by incorporating flow restrictions in all level sensing, pressure measurement, and sampling nozzles on the reactor coolant piping, reactor coolant pumps, pressurizer, and steam generators. The connecting lines are therefore classified as Safety Class 2.

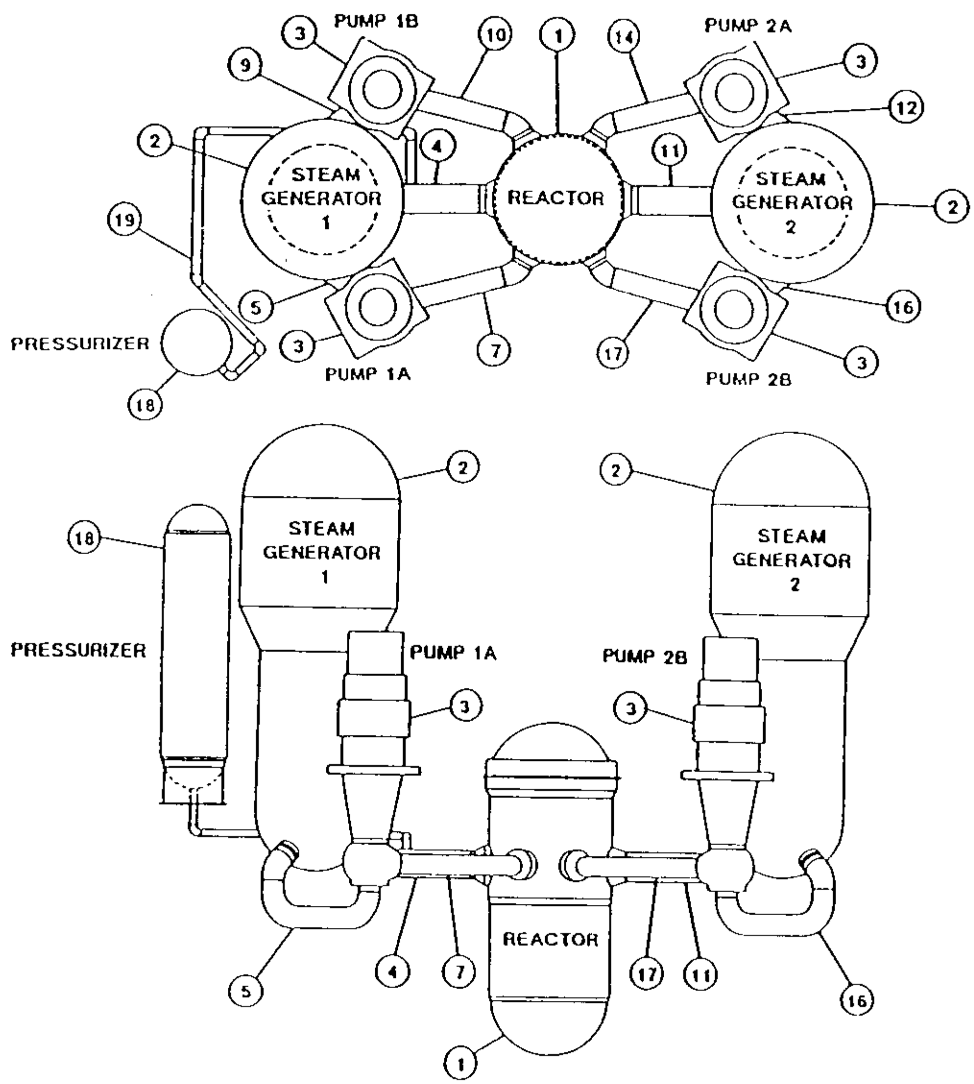
Reactor coolant system piping is allowed to expand essentially without restraint. There are no supports for the reactor coolant piping, other than the support provided by the major components to which the pipes are connected. The component supports are designed to allow essentially unrestrained thermal expansion. Piping flexibility stress analysis has determined that the stress levels are below the values allowed by ASME codes at the design conditions and during all postulated normal and abnormal modes of operation. Pipe material is carbon steel clad internally with stainless steel.

Reactor coolant piping is designed with the required instrumentation nozzles and connecting nozzles for piping to and from the auxiliary system equipment. Nozzles subjected to local thermal transients, caused by fluid entering the Reactor Coolant System from an auxiliary system, are analyzed to ensure the nozzles can accommodate these transients. Flow restricting orifices are provided in the nozzles for the RCS instrumentation to limit flow in the event of a break downstream of the nozzle.

Branch lines connected to the main loop pipes include the surge line, shutdown cooling lines, safety injection lines, charging lines, spray lines, and drain lines. Design of the branch lines considers requirements such as load limits, thermal and seismic movements, and pipe break criteria. The surge, shutdown cooling, and safety injection lines satisfy the NRC requirements for elimination of dynamic effects of pipe breaks, by application of the leak-before-break methodology. This minimizes the requirements for pipe whip restraints and jet impingement shields, and eliminates the need to consider other load effects.

ABB
ASEA BROWN BOVERI

BILL OF MATERIAL		
ITEM No.	QUAN.	NAME
1	1	REACTOR VESSEL ASSEMBLY
2	2	STEAM GENERATOR
3	4	REACTOR COOLANT PUMP
4	1	STEAM GENERATOR No. 1 INLET PIPING
5	1	PUMP 1A SUCTION PIPING
7	1	PUMP 1A DISCHARGE PIPING
9	1	PUMP 1B SUCTION PIPING
10	1	PUMP 1B DISCHARGE PIPING
11	1	STEAM GENERATOR No. 2 INLET PIPING
12	1	PUMP 2A SUCTION PIPING
14	1	PUMP 2A DISCHARGE PIPING
16	1	PUMP 2B SUCTION PIPING
17	1	PUMP 2B DISCHARGE PIPING
18	1	PRESSURIZER
19	1	PRESSURIZER SURGE LINE



SYSTEM 80+™	REACTOR COOLANT SYSTEM ARRANGEMENT	FIGURE II-C-1

ABB

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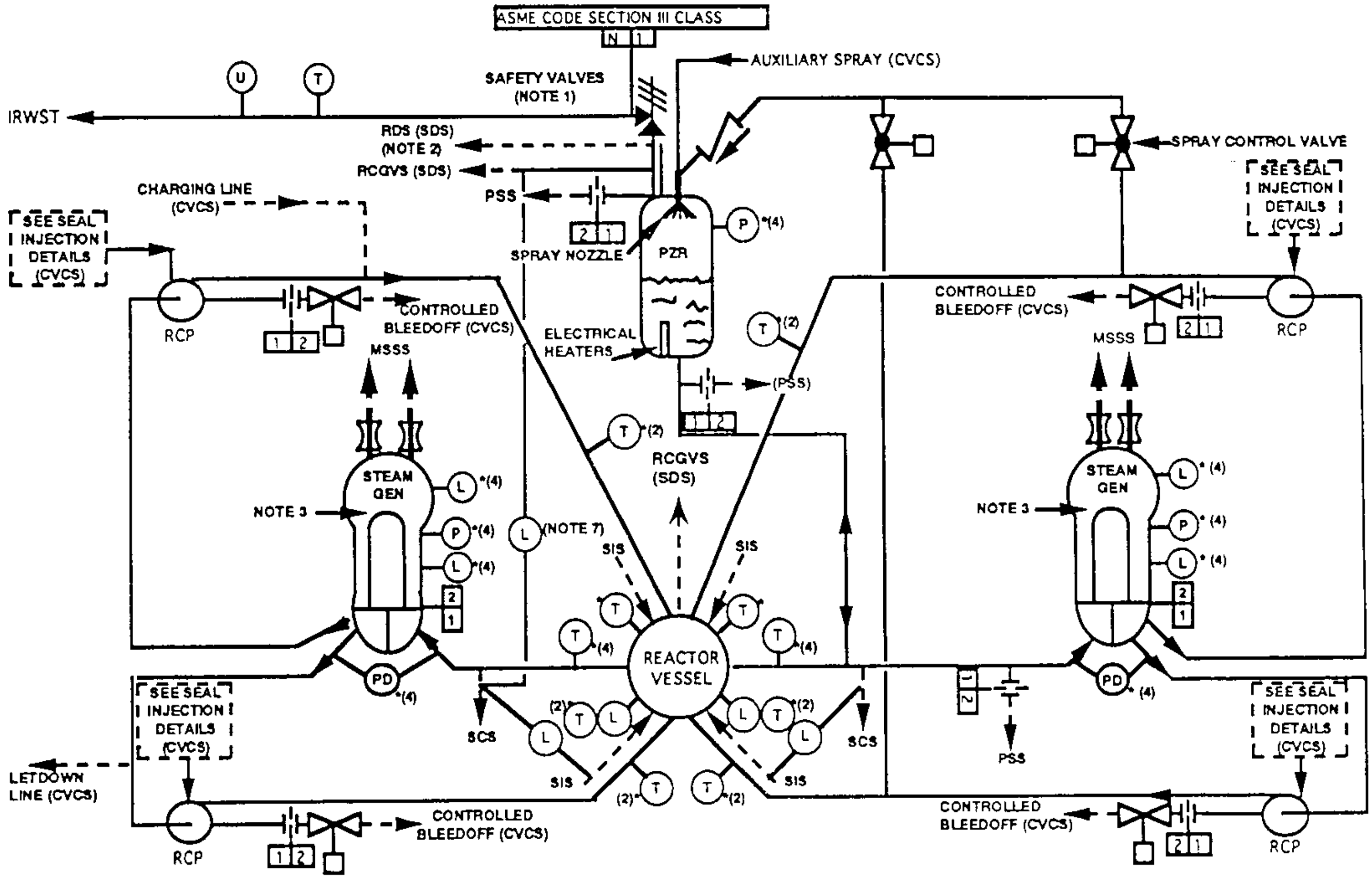
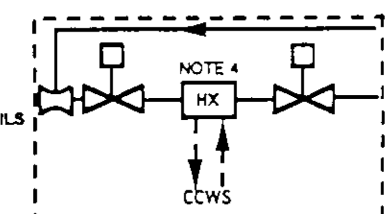




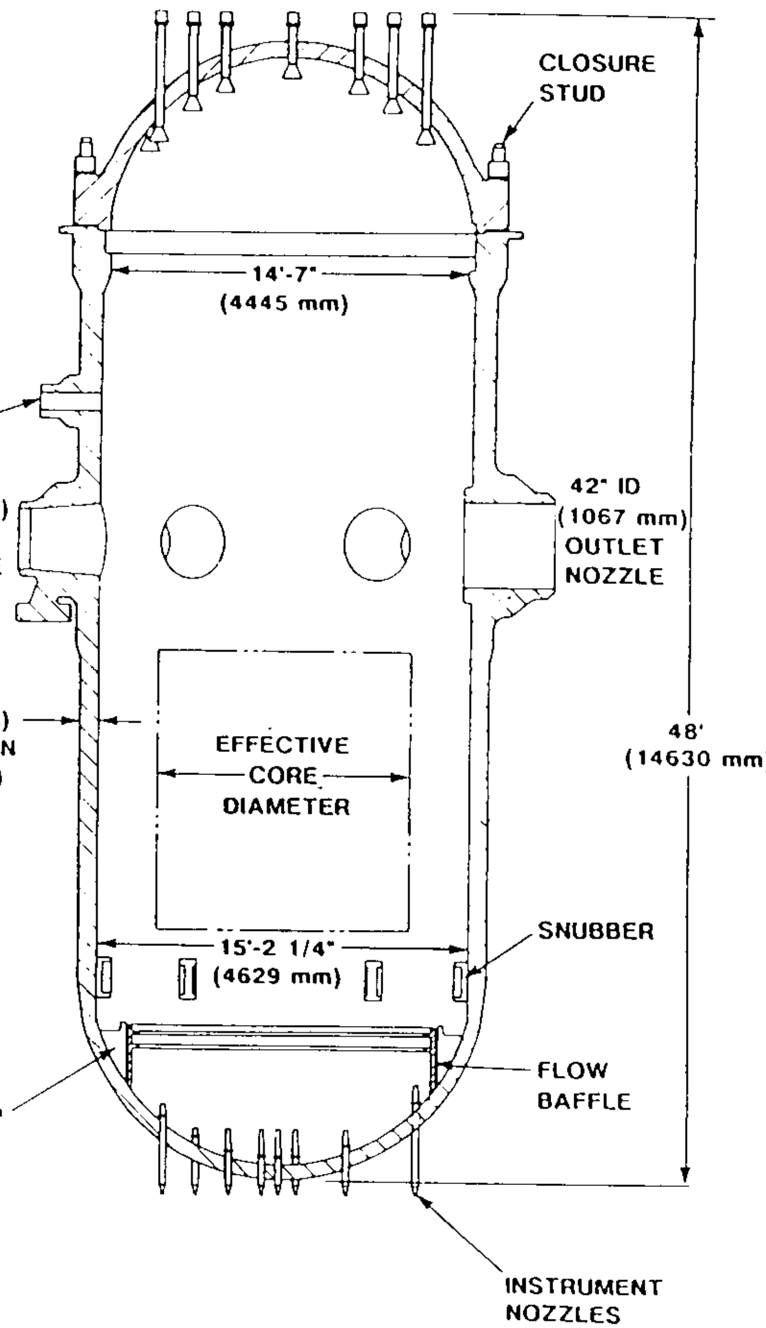
SEAL INJECTION DETAILS

NOTES:
1. ONE OF FOUR SAFETY VALVES IS SHOWN.
2. ONE OF TWO RDS LINES IS SHOWN.
3. SG TUBES ARE ASME CODE SECTION III CLASS 1.
4. TUBE SIDE IS CLASS 1 AND SHELL SIDE IS NON-ASME SECTION III.
5. *: EQUIPMENT FOR WHICH PARAGRAPH NUMBER 3 OF THE BASIC CONFIGURATION DEFINITION APPLIES.
6. FOR INSTRUMENTS, THE NUMBER OF REDUNDANT DETECTORS AND CHANNELS IS LISTED IN PARATHESES.
7. ANOTHER LEVEL INSTRUMENT (NOT SHOWN) IS PROVIDED BETWEEN THE OTHER SCS SUCTION LINE AND THE OTHER RCGVS LINE FROM A DIFFERENT SAFETY VALVE (NOT SHOWN).

SYSTEM 80+™	SYSTEM 80+ ™ REACTOR COOLANT SYSTEM	FIGURE II-C-2

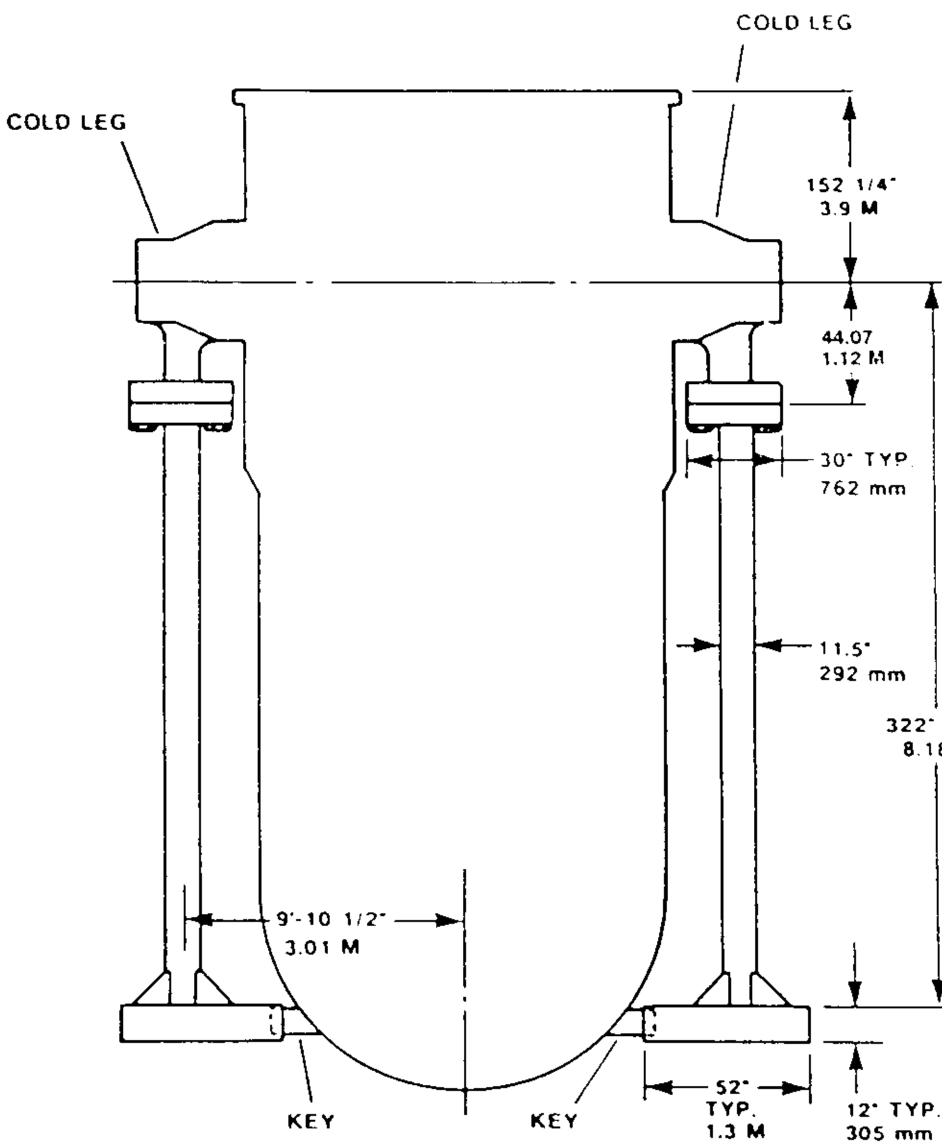
ABB
ASEA BROWN BOVERI

NOZZLE SCHEDULE	
SERVICE	
COOLANT INLET	4
COOLANT OUTLET	2
CEDM NOZZLES	97
INSTRUMENTATION	61
VENT	1
SEAL LEAK MONITOR	1
DVI NOZZLES	4



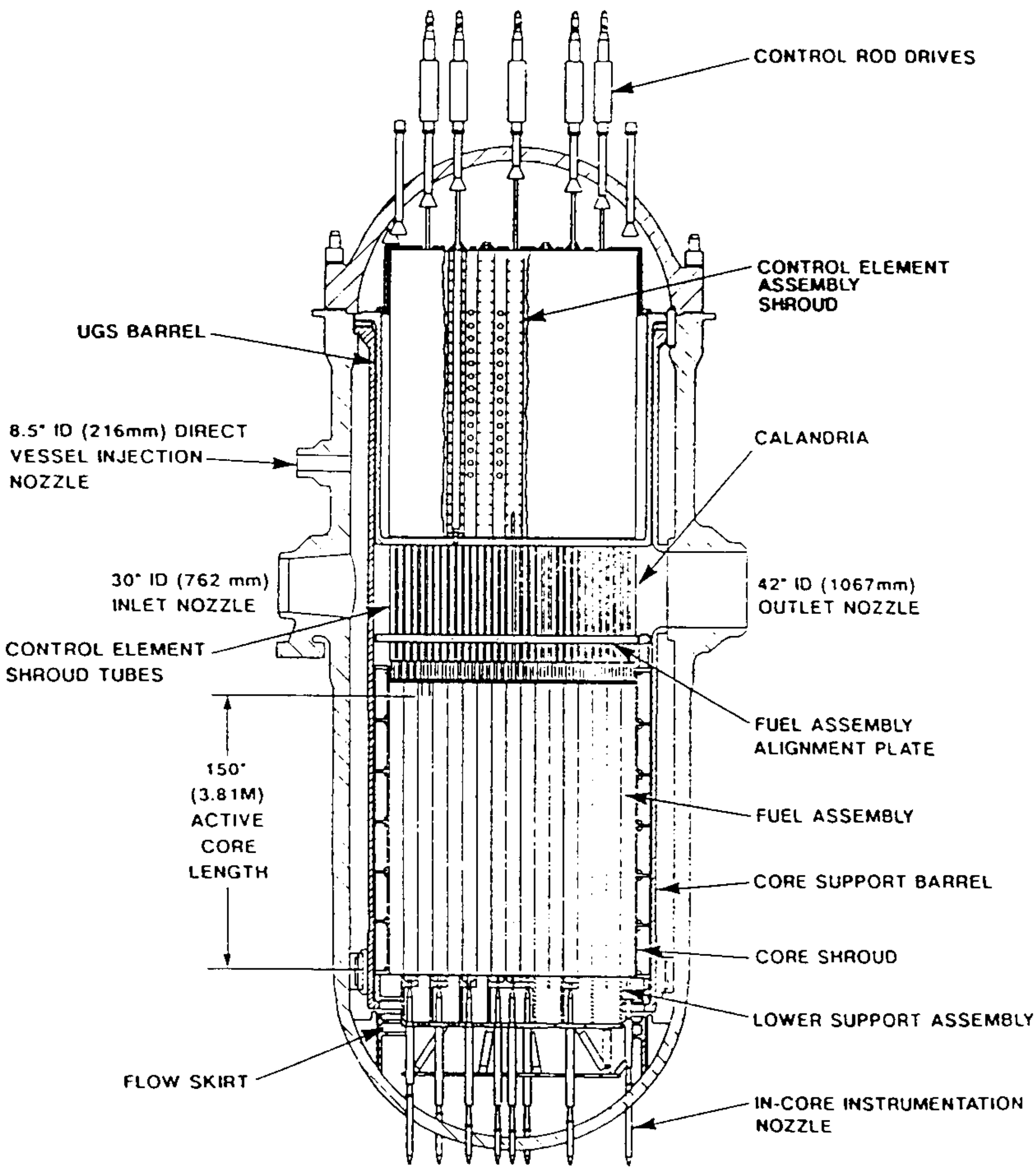
SYSTEM 80+™	REACTOR VESSEL	FIGURE II-C-3

ABB
ASEA BROWN BOVERI



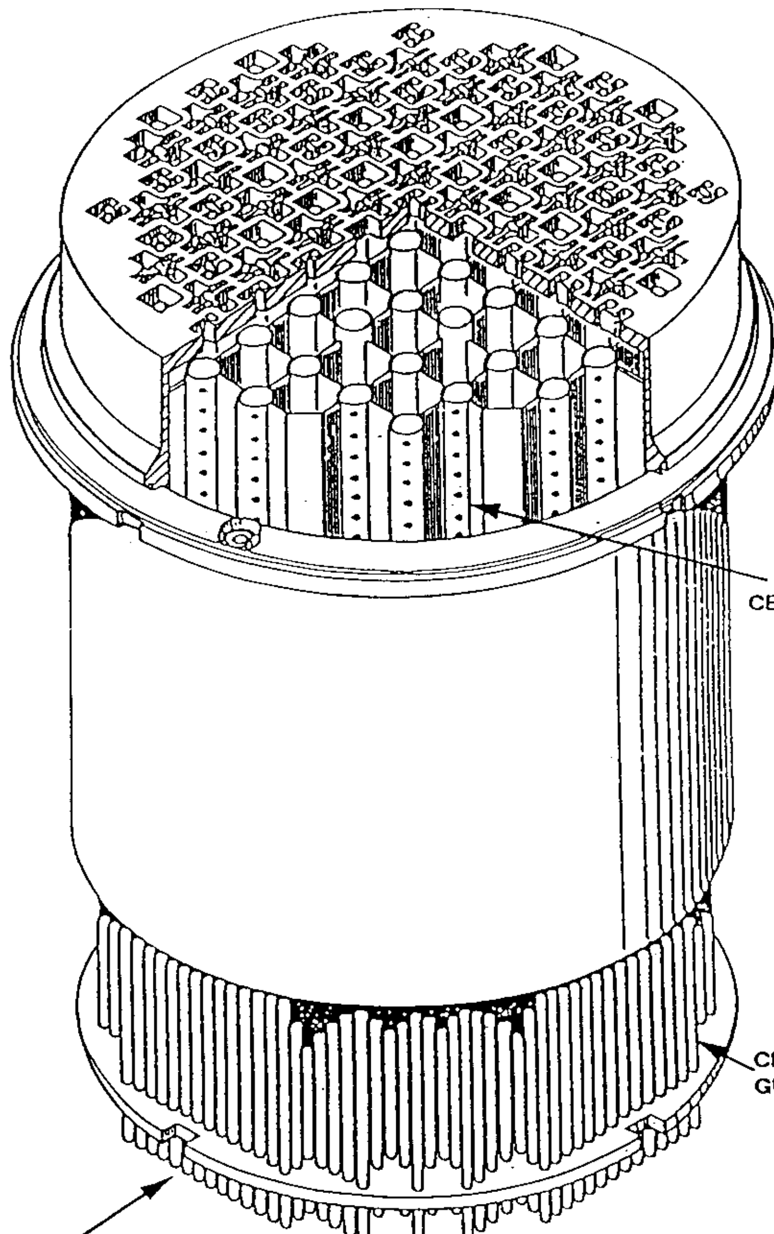
SYSTEM 80+™	REACTOR VESSEL SUPPORTS	FIGURE II-C-4

ABB
ASEA BROWN BOVERI



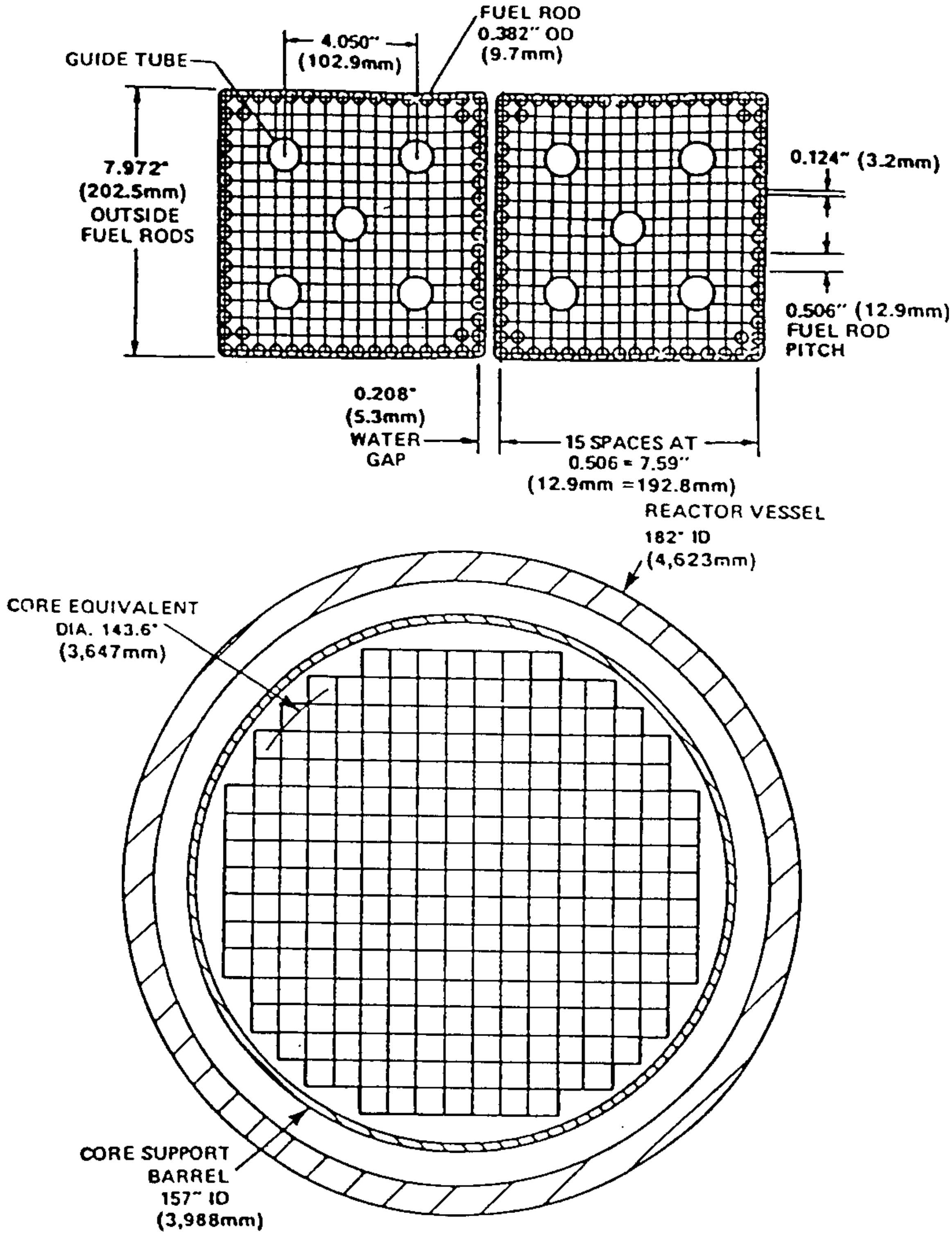
SYSTEM 80+™	SYSTEM 80 +™ REACTOR ARRANGEMENT	FIGURE II-C-5

ABB
ASEA BROWN BOVERI

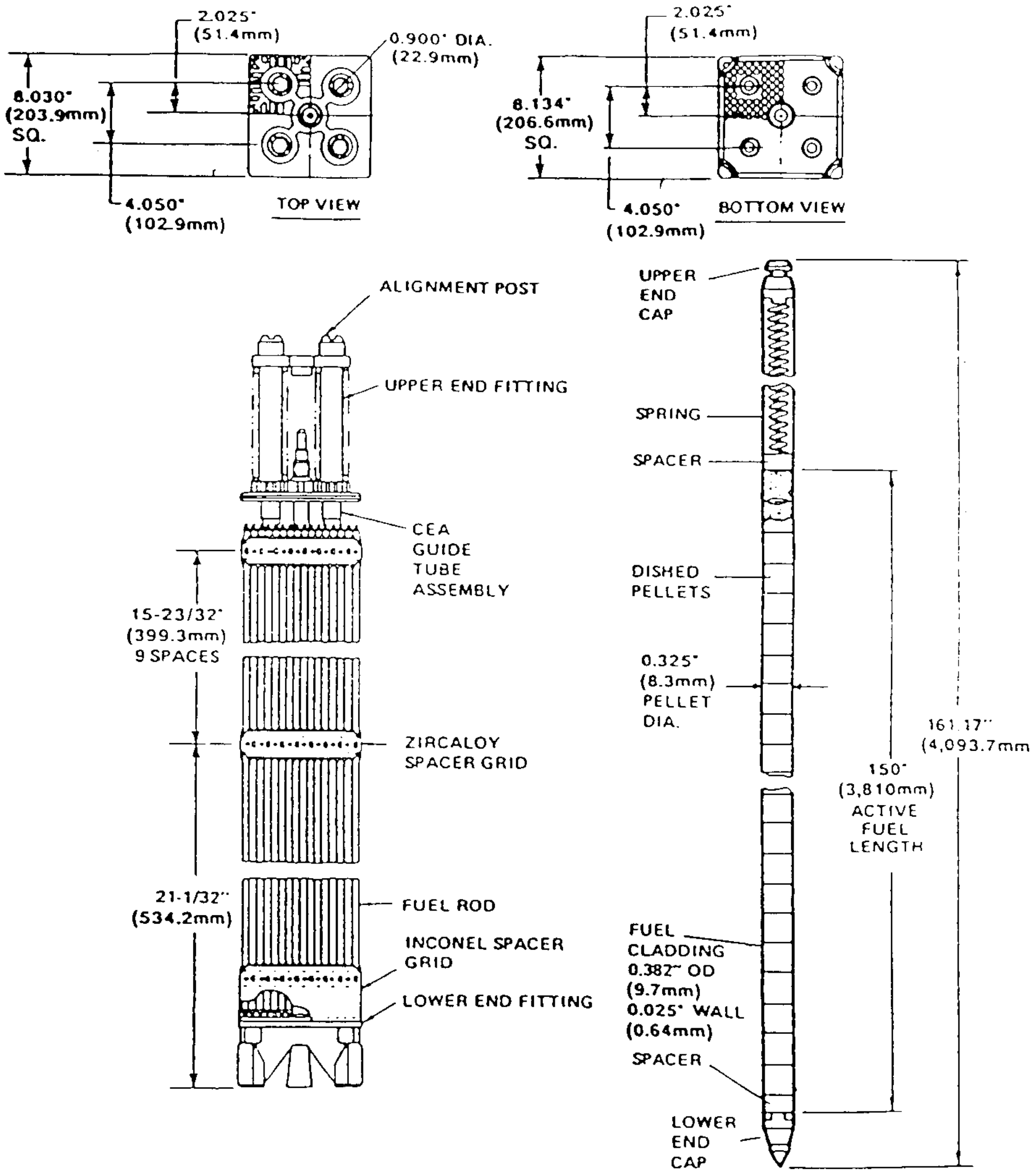


SYSTEM 80+™	UPPER GUIDE STRUCTURE ASSEMBLY	FIGURE II-C-6

ABB
ASEA BROWN BOVERI

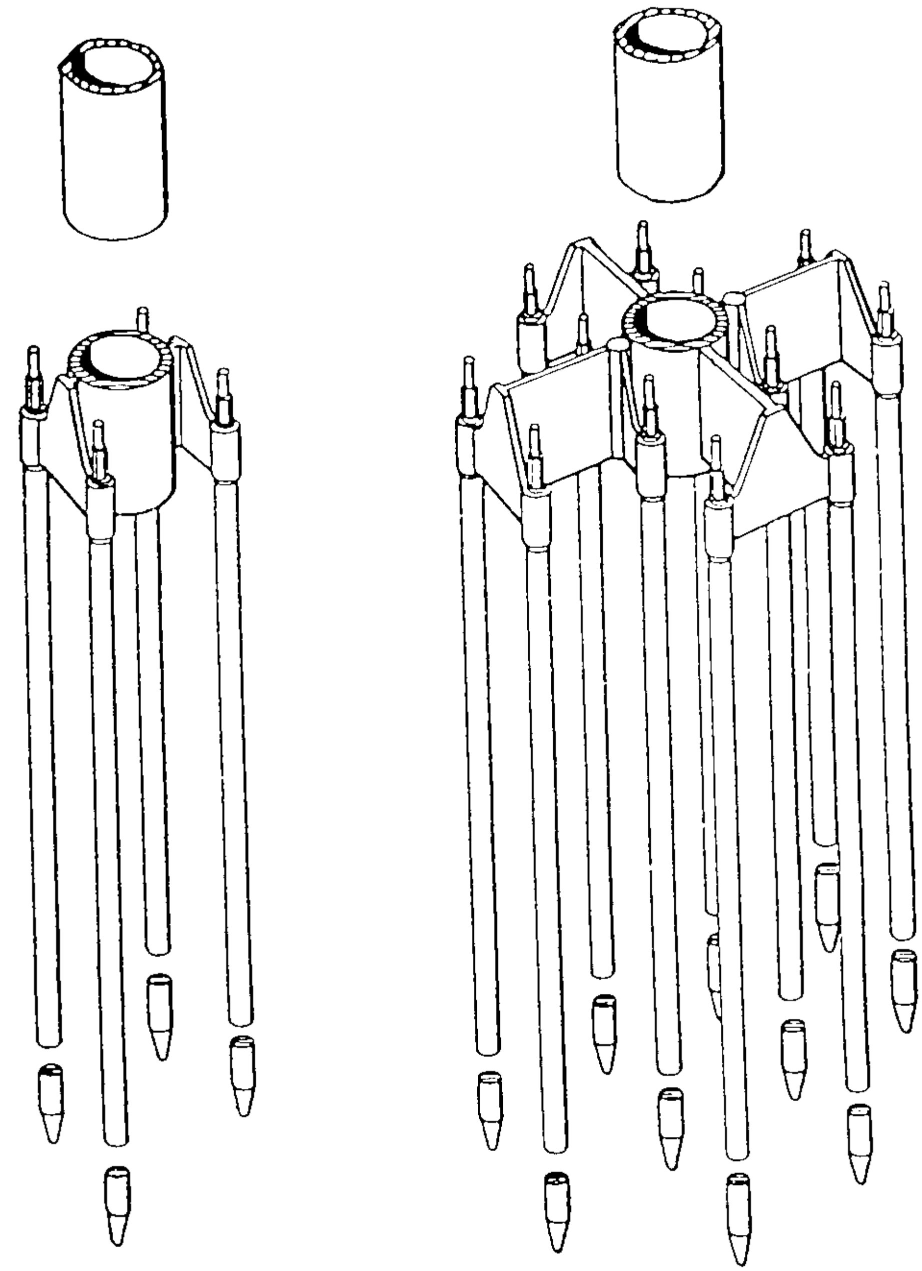


SYSTEM 80+™	REACTOR CORE CROSS SECTION	FIGURE II-C-7



SYSTEM 80+™	FUEL ASSEMBLY	FIGURE II-C-8

ABB
ASEA BROWN BOVERI

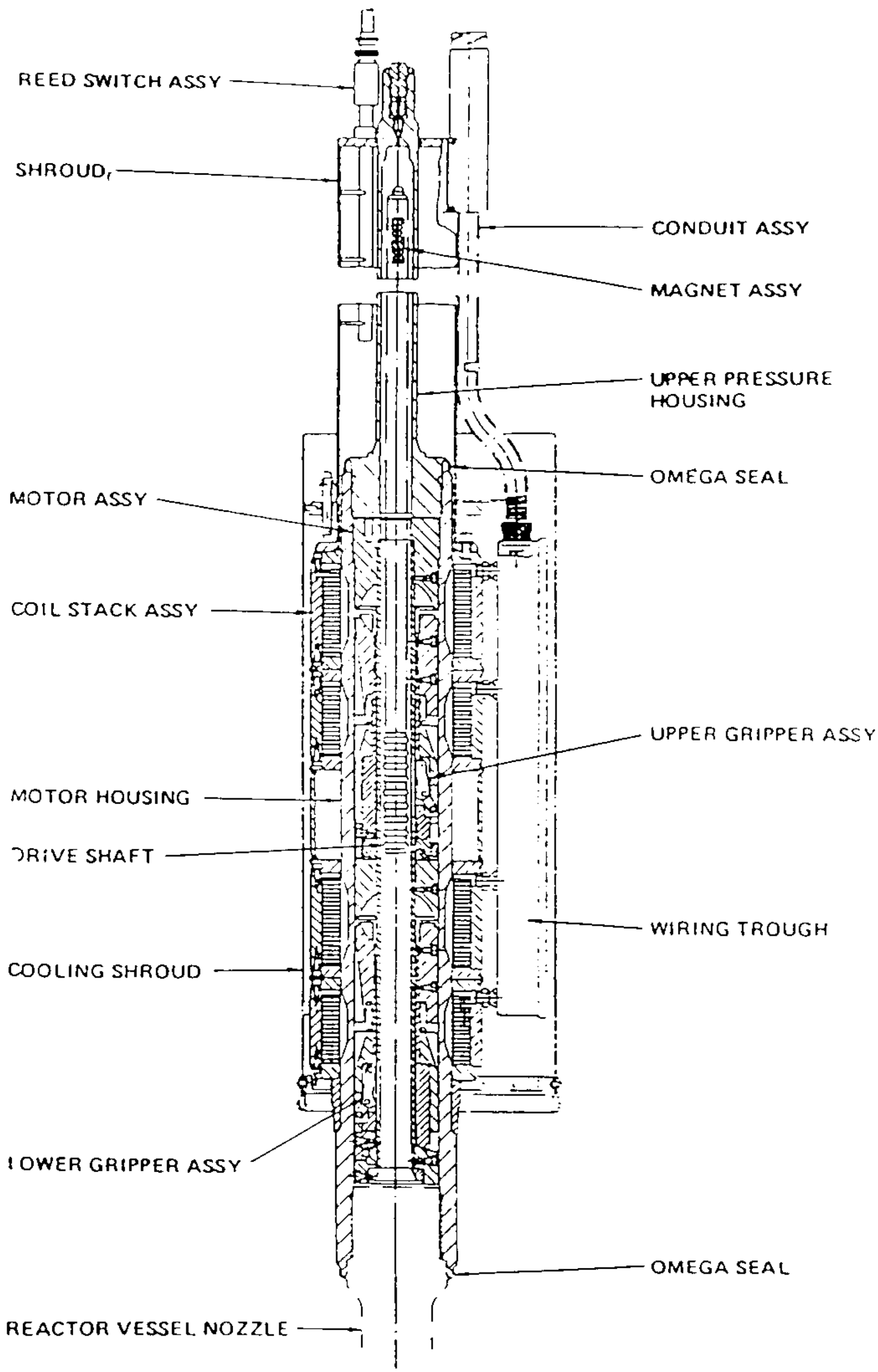


4 FINGER CEA

12 FINGER CEA

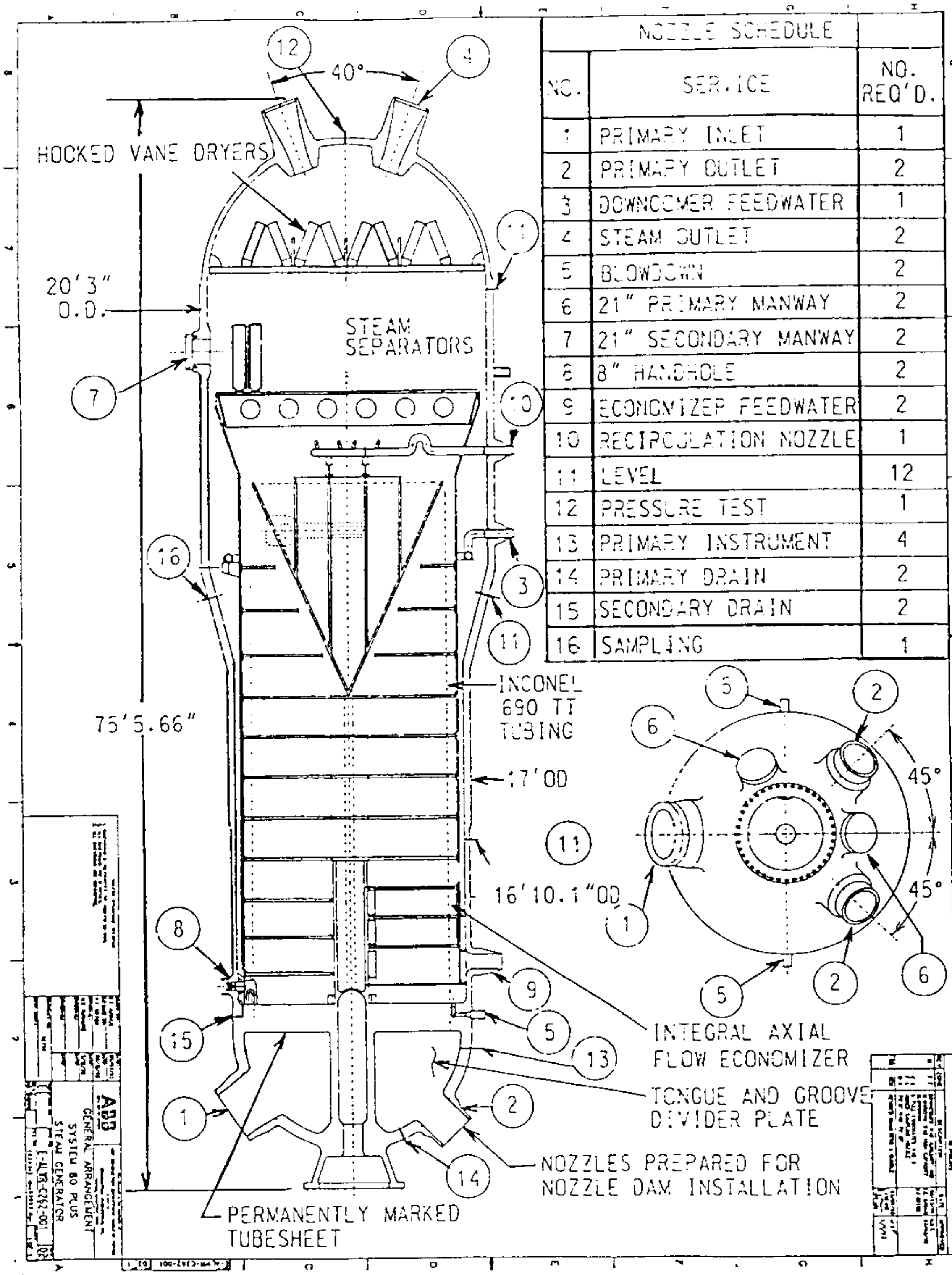
SYSTEM 80+™	4 AND 12 FINGER CONTROL ELEMENT ASSEMBLIES	FIGURE II-C-9

ABB
ASEA BROWN BOVERI



SYSTEM 80+™	CONTROL ELEMENT DRIVE MECHANISM	FIGURE II-C-10

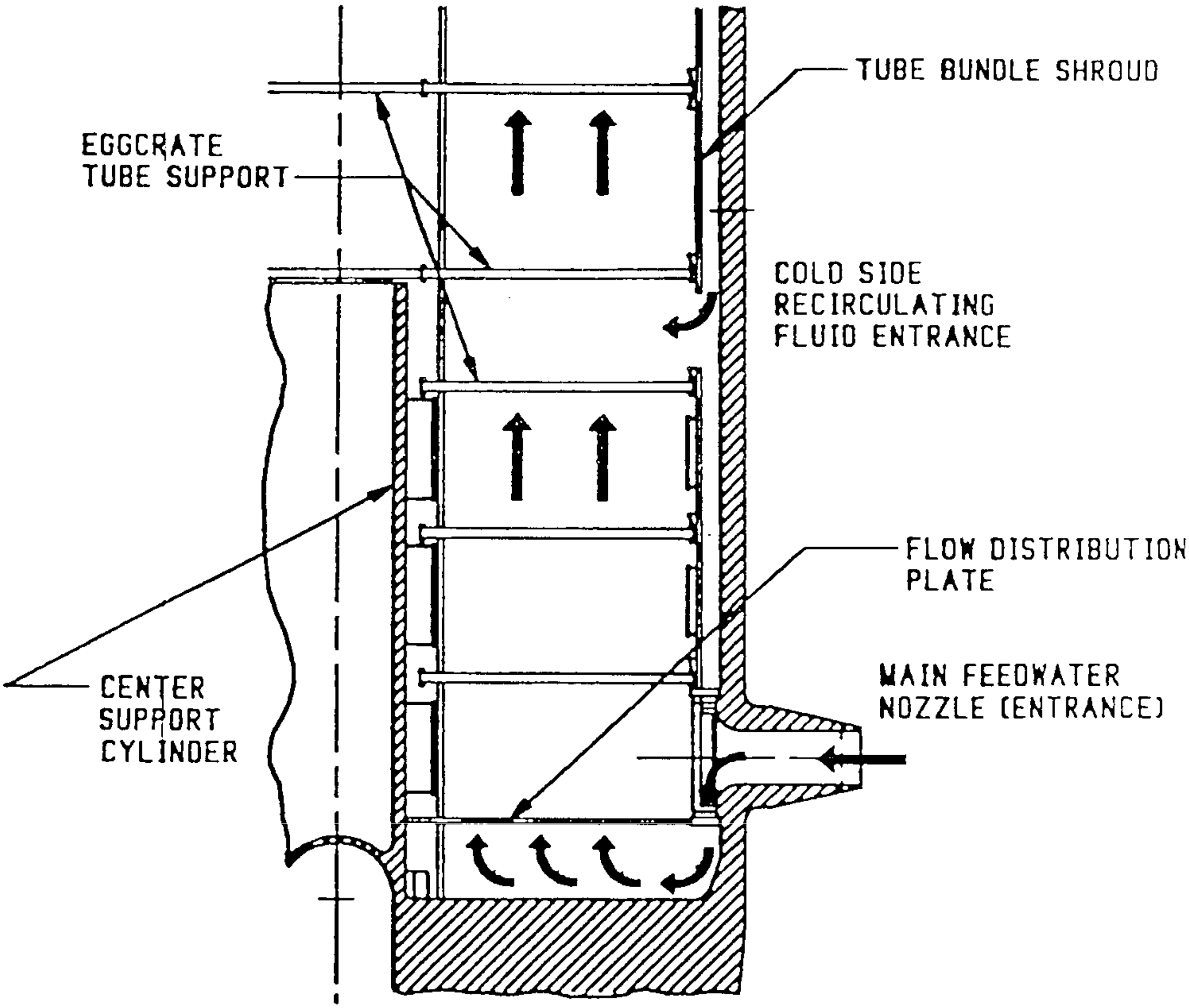
ABB
ASEA BROWN BOVERI



NOZZLE SCHEDULE		
NO.	SERVICE	NO. REQ'D.
1	PRIMARY INLET	1
2	PRIMARY OUTLET	2
3	DOWNCOMER FEEDWATER	1
4	STEAM OUTLET	2
5	BLOWDOWN	2
6	21" PRIMARY MANWAY	2
7	21" SECONDARY MANWAY	2
8	8" HANDHOLE	2
9	ECONOMIZER FEEDWATER	2
10	RECIRCULATION NOZZLE	1
11	LEVEL	12
12	PRESSURE TEST	1
13	PRIMARY INSTRUMENT	4
14	PRIMARY DRAIN	2
15	SECONDARY DRAIN	2
16	SAMPLING	1

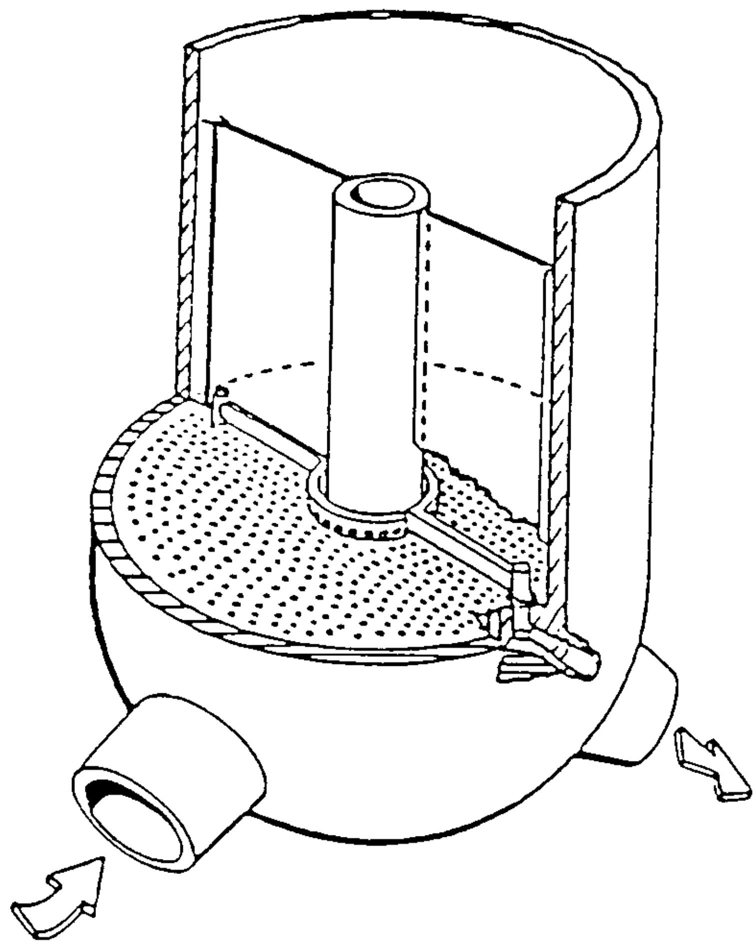
SYSTEM 80+™	STEAM GENERATOR	FIGURE II-C-11

ABB
ASEA BROWN BOVERI



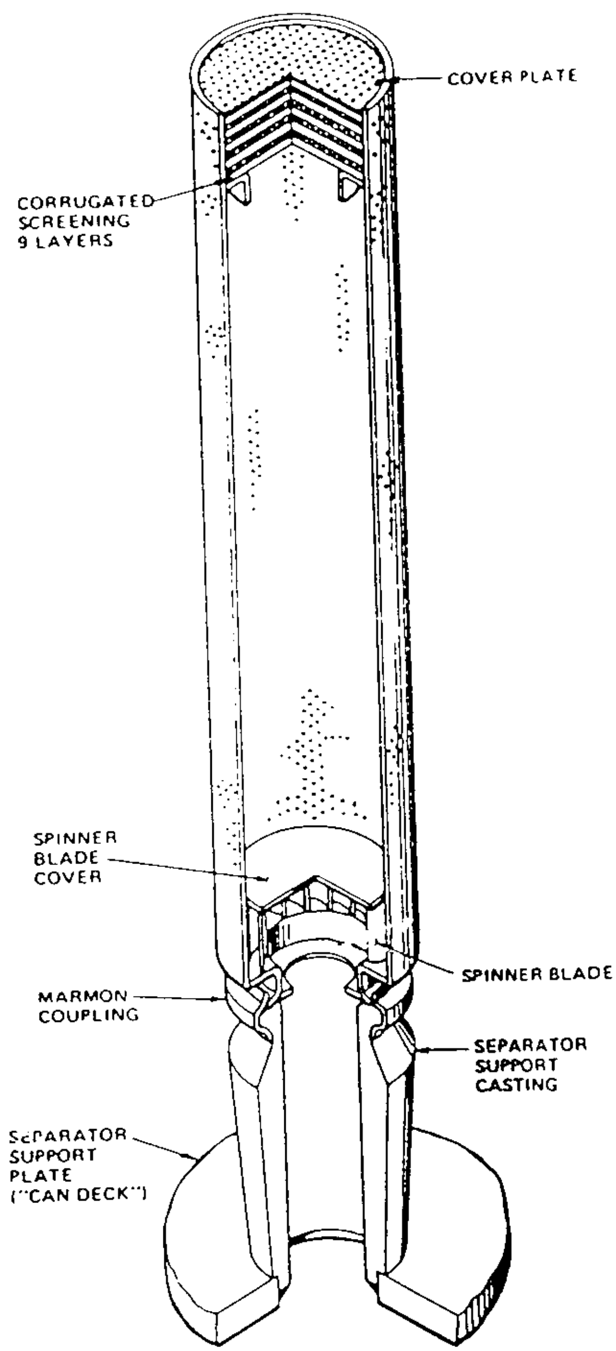
SYSTEM 80+™	STEAM GENERATOR ECONOMIZER ARRANGEMENT (ELEVATION VIEW)	FIGURE II-C-12

ASEA BROWN BOVERI



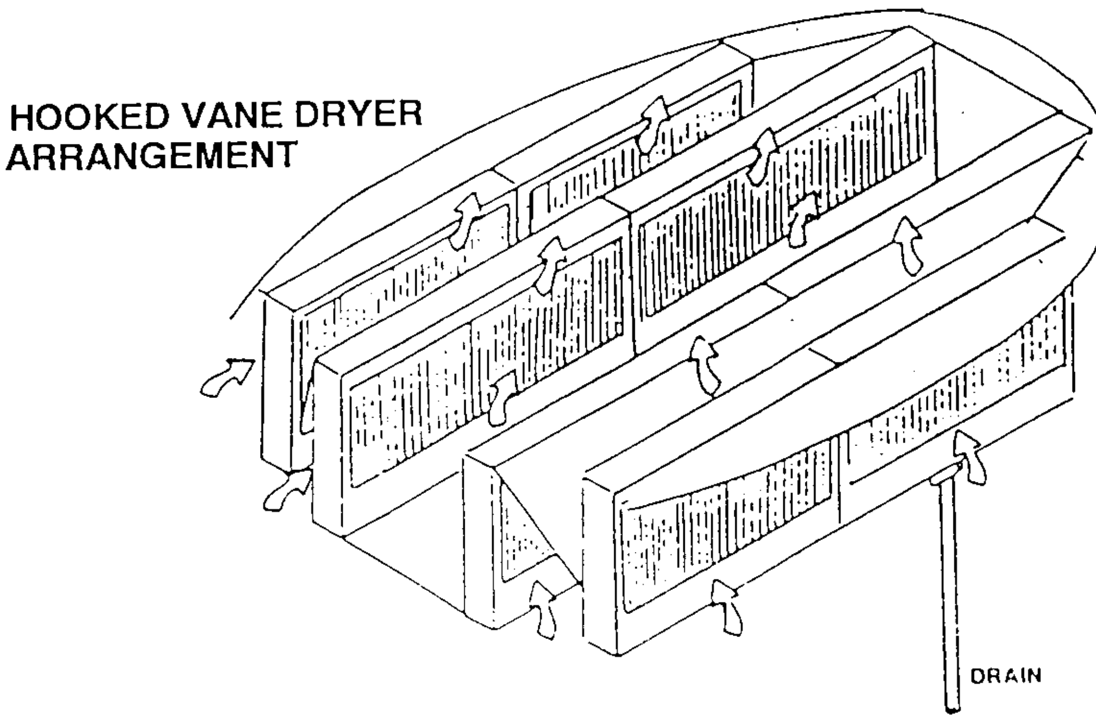
SYSTEM 80+™	ECONOMIZER REGION	FIGURE II-C-13

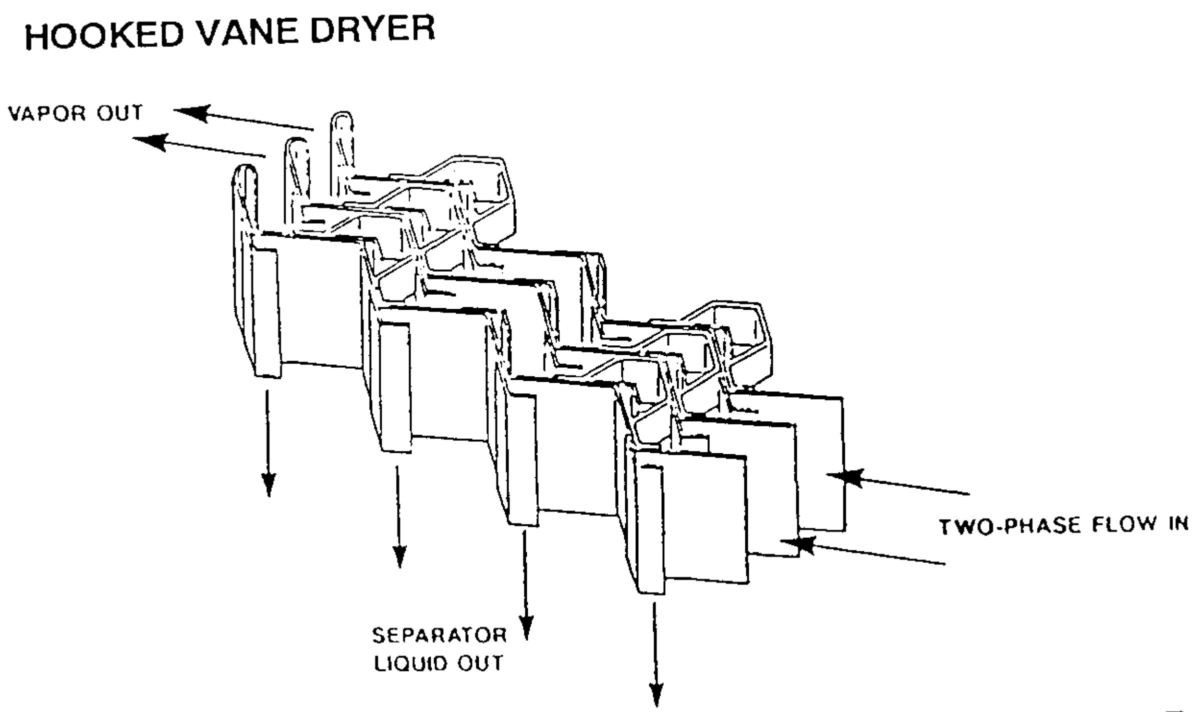
ABB
ASEA BROWN BOVERI



SYSTEM 80+™	CENTRIFUGAL SEPARATOR	FIGURE II-C-14

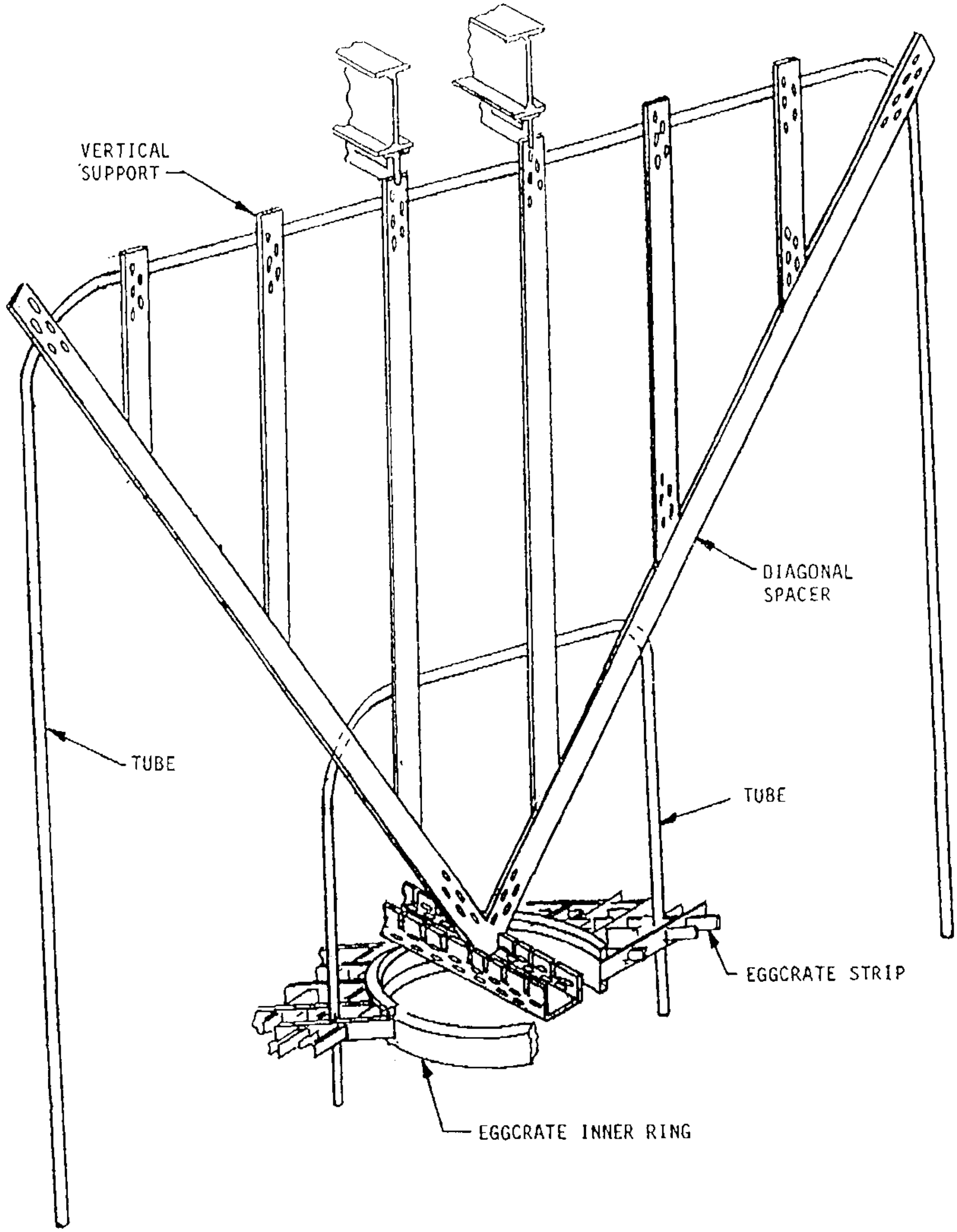
ABB
ASEA BROWN BOVERI





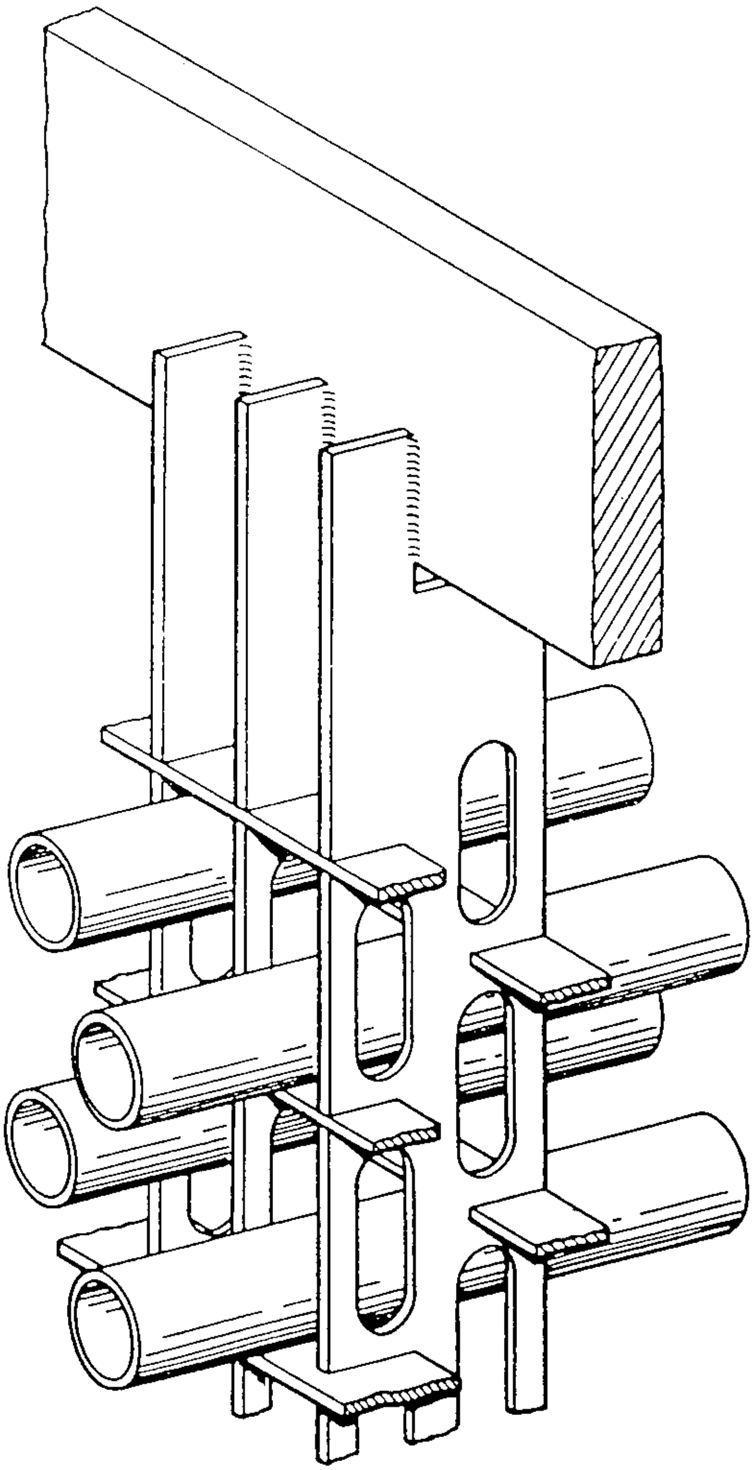
SYSTEM 80+™	HOOKED VANE DRYER ARRANGEMENT & HOOKED VANE DRYERS	FIGURE II-C-15

ABB
ASEA BROWN BOVERI



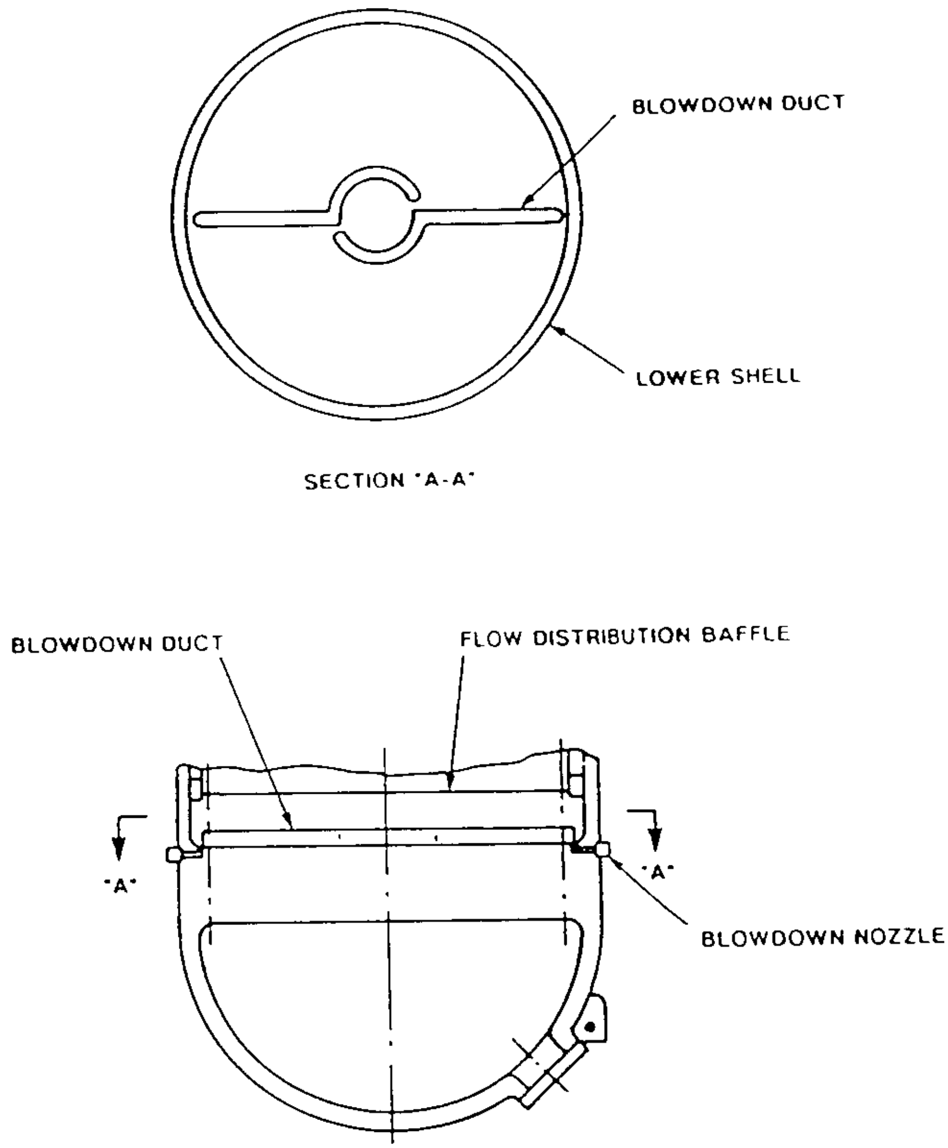
SYSTEM 80+™	STEAM GENERATOR BEND REGION TUBE SUPPORTS	FIGURE II-C-16

ABB
ASEA BROWN BOVERI



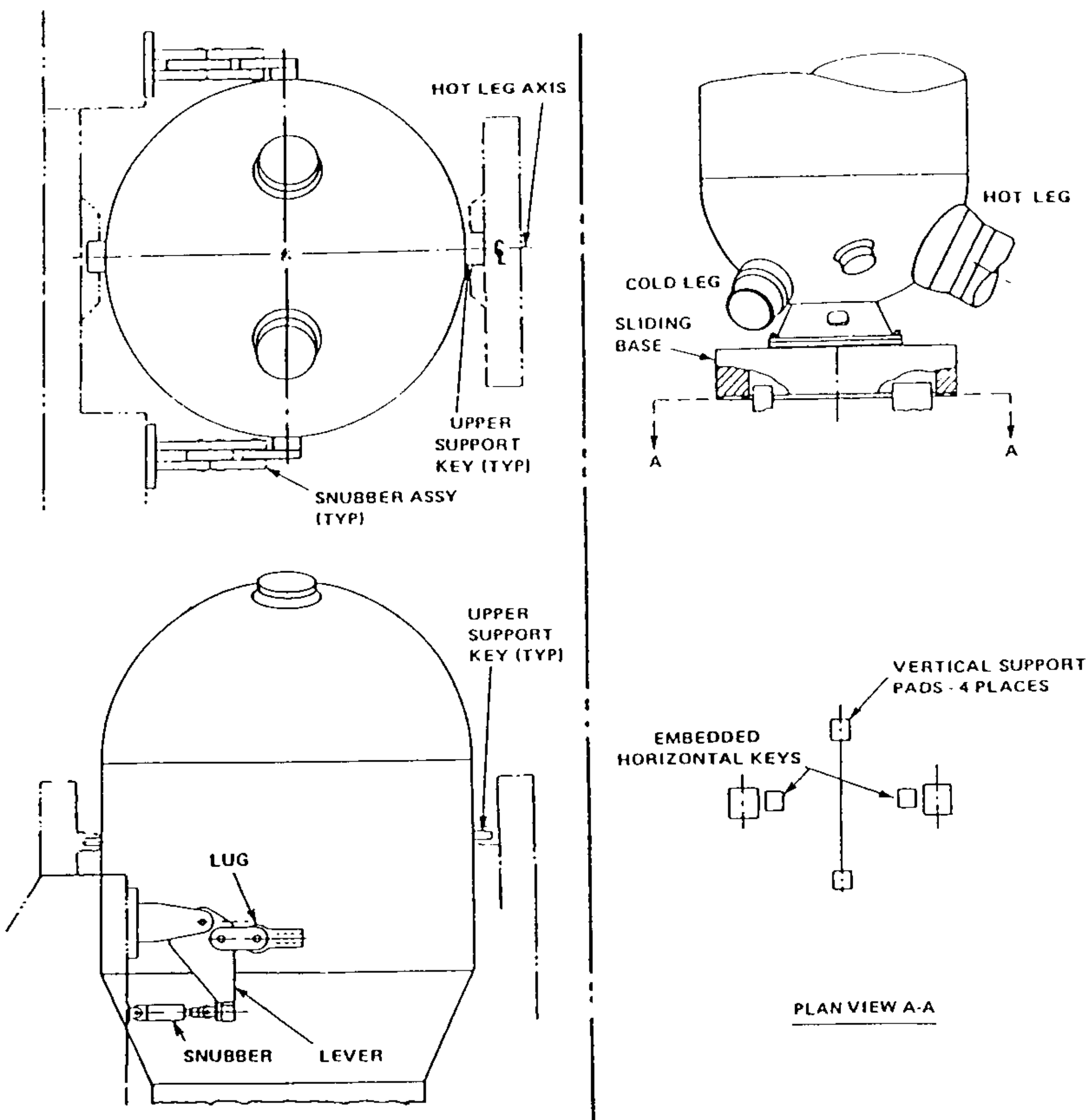
SYSTEM 80+™	STEAM GENERATOR HORIZONTAL TUBE SUPPORT DETAILS	FIGURE II-C-17

ABB
ASEA BROWN BOVERI



SYSTEM 80+™	STEAM GENERATOR BLOWDOWN DUCT ARRANGEMENT	FIGURE II-C-18

ABB
ASEA BROWN BOVERI



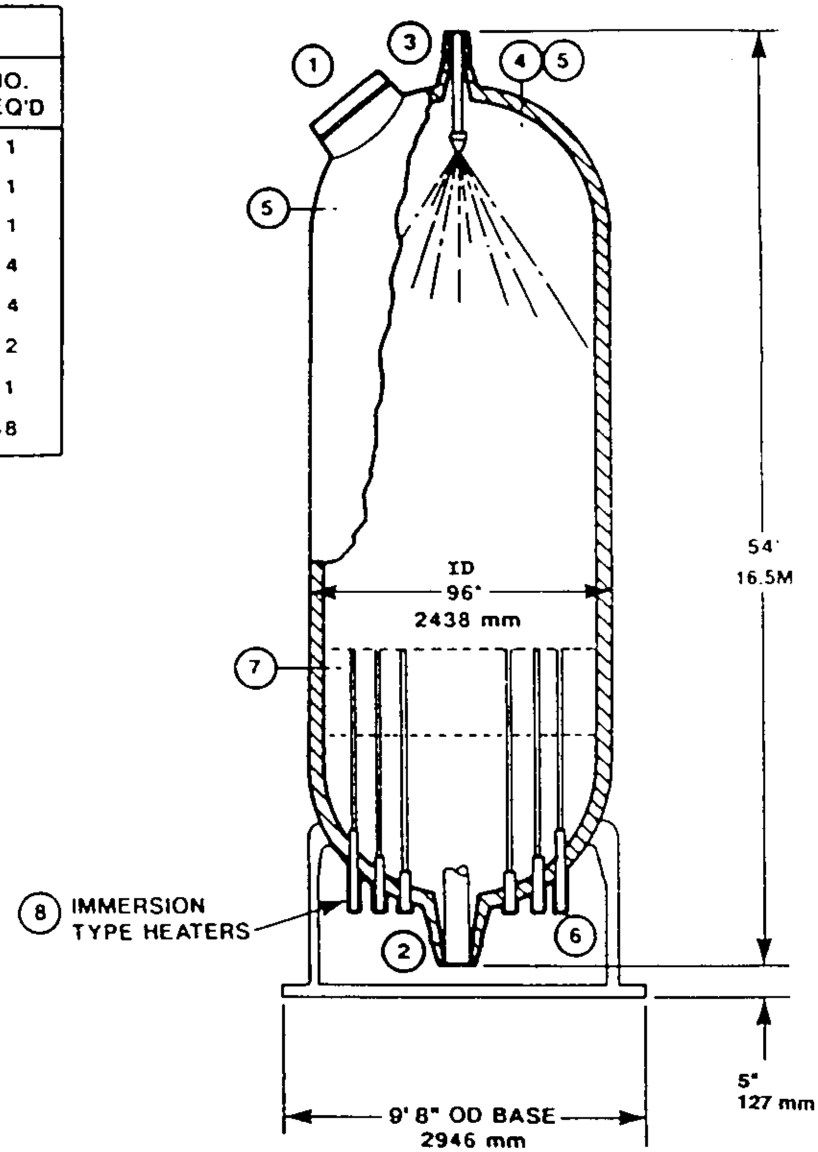
UPPER SUPPORTS

LOWER SUPPORTS

SYSTEM 80+™	STEAM GENERATOR SUPPORTS	FIGURE II-C-19

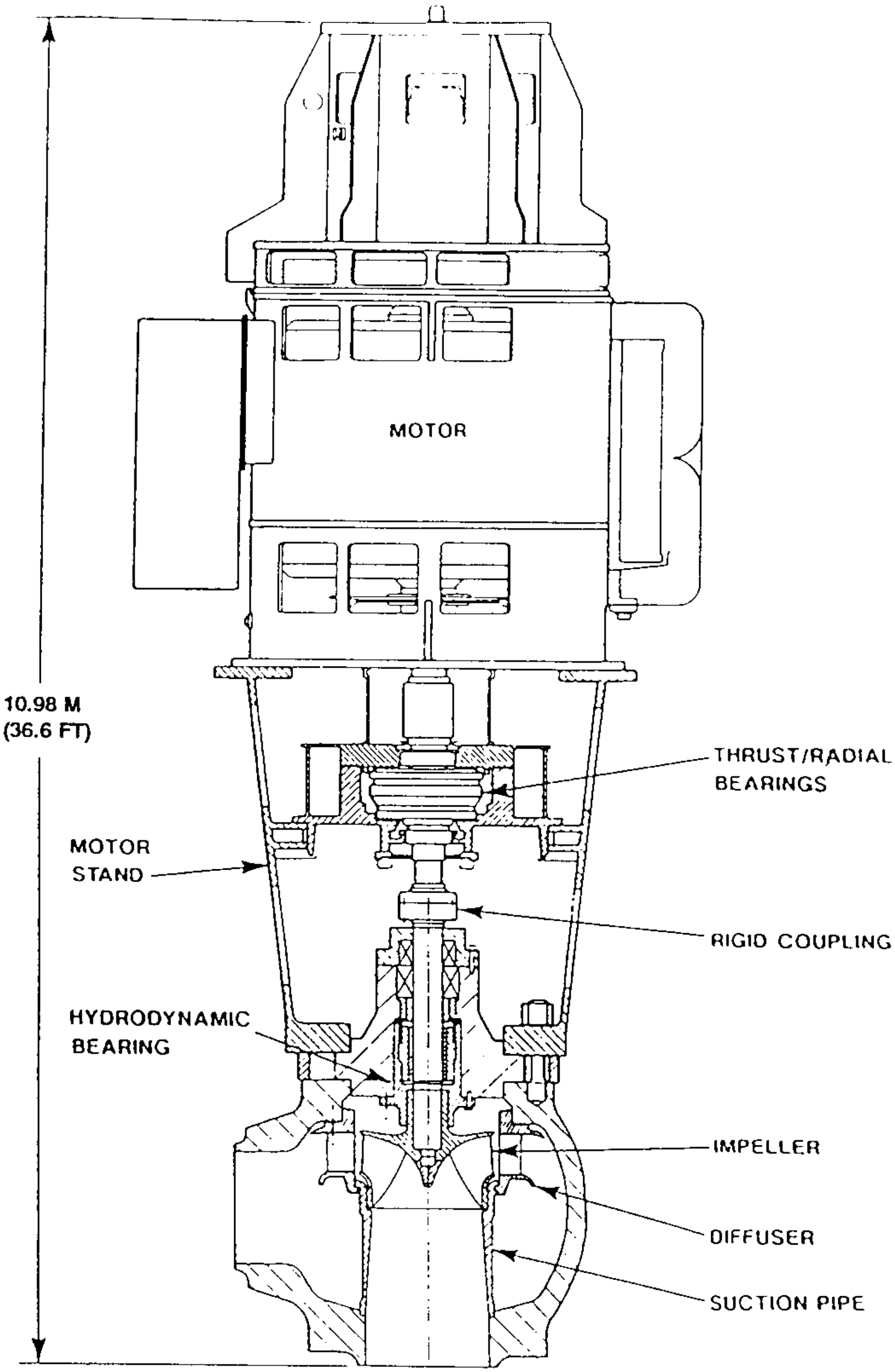
ABB
ASEA BROWN BOVERI

NOZZLE SCHEDULE		
NO.	SERVICE	NO. REQ'D
1	MANWAY	1
2	SURGE	1
3	SPRAY	1
4	SAFETY VALVE	4
5	INSTRUMENT	4
6	INSTRUMENT	2
7	TEMPERATURE	1
8	HEATER	48



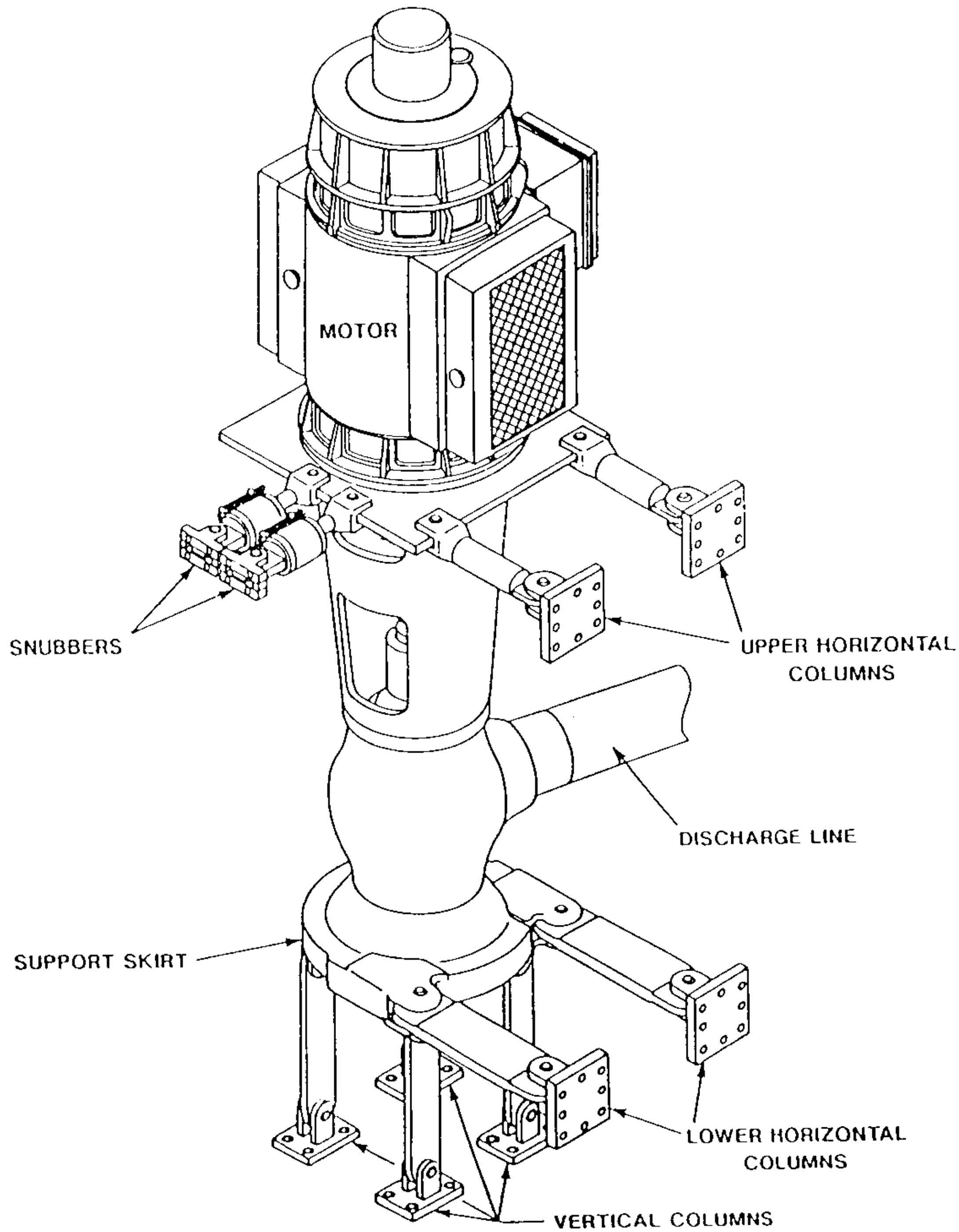
SYSTEM 80+™	PRESSURIZER	FIGURE II-C-20

ABB
ASEA BROWN BOVERI



SYSTEM 80+™	REACTOR COOLANT PUMP	FIGURE II-C-21

ABB
ASEA BROWN BOVERI



SYSTEM 80+™	REACTOR COOLANT PUMP SUPPORTS	FIGURE II-C-22

D. SAFETY SYSTEMS

Probabilistic Risk Assessment (PRA) has proven to be a valuable tool in evaluating the various safety systems. Although these systems in the original System 80 design were found to be very reliable, further improvements have been made to significantly lower the probability of core damage by more than two orders of magnitude.

1. Safety Depressurization System

The Safety Depressurization System (SDS) is designed to perform the following functions: venting of the Reactor Coolant System (RCS), and safety depressurization (i.e., feed and bleed) of the RCS (Figure II-D-1).

a. Reactor Coolant Gas Vent (RCGV) Function

The Reactor Coolant Gas Vent (RCGV) function provides a safety grade means of venting non-condensible gases from the pressurizer and the reactor vessel upper head to the Reactor Drain Tank (RDT) during post-accident conditions. In addition, the RCGV provides a safety grade means to cooldown the RCS in the event that pressurizer Main Spray and Auxiliary Spray systems are unavailable for plant cooldown.

Non-condensible gases are removed from the pressurizer through a vent line to the RDT, through one or both of the parallel isolation valves, and from the reactor vessel upper head through a vent line to the RDT, through a flow restricting orifice and one or both of the parallel isolation valves. Venting under accident conditions would be accomplished using only one source (reactor vessel or pressurizer) at a time.

If the operator determines that non-condensible gases have collected in the reactor vessel upper head or in the pressurizer steam space, the operator manually opens the RCGV valves to vent the reactor vessel upper head or the pressurizer steam space from the Main Control Room. The system is operable following all design basis events. The vent path from either the pressurizer or the reactor vessel upper head can accommodate a single active failure with the active components powered from a normal AC power source and an emergency AC back-up power source. Parallel valves powered from alternate power sources are provided at both vent sources to assure a vent path exists in the event of a single failure of either of the valves or of the power source.

The reactor drain tank (RDT) quenches any steam relieved from either the pressurizer or the reactor vessel. The RDT stores small quantities of non-condensible gas from the RCS: (1) without the need for releasing the highly radioactive gas directly into containment; and, (2) without influencing containment hydrogen concentration levels. The Gaseous Waste Management System is used to process the non-condensible gases collected in the RDT.

The operator may use the Reactor Coolant Gas Vent (RCGV) function to cooldown and depressurize the pressurizer in the event the Main Spray and Auxiliary Spray systems are not operable. The operator manually opens the RCGV valves on the top of the pressurizer, releasing steam to the RDT. If voids form in the reactor vessel upper head, the operator may open the RCGV valves on the top of the reactor vessel to vent steam from the vessel head, allowing the reactor vessel to be refilled. The RCGV flow, and therefore the depressurization rate, are controlled by modulating valves in the vent lines from the top of the pressurizer and by opening and closing two series RCGV valves from the top of the reactor vessel head.

b. Safety Depressurization Function

The Safety Depressurization function (i.e., feed and bleed) provides a safety grade means of quickly depressurizing the RCS when normal and emergency feedwater (EFW) are unavailable to remove core decay heat through the steam generators. This function is achieved via remote manual operator control. Whenever any event (e.g., a total loss of feedwater) results in high RCS pressure with a gradual loss of RCS liquid inventory, the SDS valves may be opened by the operator, resulting in a controlled depressurization of the RCS. As the RCS pressure decreases, the Safety Injection pumps start, initiating feed flow to the RCS and restoring the RCS liquid inventory.

Core decay heat removal, using the SDS, is accomplished by a once-through cooling process in which water is injected directly into the reactor vessel downcomer via the normal Safety Injection System. Once in the reactor vessel, the cooling fluid passes through the vessel downcomer to the lower plenum, up through the core (where decay heat is removed) and out to the hot leg, through the surge line to the pressurizer and out through the dedicated rapid depressurization bleed valves through piping to the sparger in the IRWST where quenching and cooling of the bleed flow is accomplished.

Bleed and feed and, therefore, core cooling can continue even without the initiation of flow through the Shutdown Cooling heat exchanger. Without IRWST cooling, the IRWST's vent system will relieve the steam formed in the tank to the containment. The discharged steam will be condensed by the containment cooling system and eventually returned to the IRWST via gravity drains. Cooling of the IRWST can be accomplished by either the Containment Spray or Shutdown Cooling heat exchangers.

2. Shutdown Cooling System

The Shutdown Cooling System (SCS) is a forced circulation heat removal loop designed to transfer heat from the Reactor Coolant System to the Component Cooling Water System at temperatures where the steam generators are ineffective. The Shutdown Cooling System consists of two independent subsystems, each utilizing a shutdown cooling pump to circulate coolant through a shutdown cooling heat exchanger. The Shutdown Cooling System is used for normal shutdown, emergency shutdowns, refueling and maintenance operations. In a normal cooldown to refueling temperatures both shutdown cooling trains are used to minimize cooldown times. All safety functions are provided by a single train. A flow diagram of the Shutdown Cooling System is shown in Figure II-D-2.

The initial cooldown of the Reactor Coolant System is accomplished by heat rejection to the secondary side of the steam generators and then releasing steam via the Steam Bypass System or atmospheric dump valves. The components required for the Safety Injection System remain aligned for emergency use until the Reactor Coolant System pressure and temperature are reduced. After the reactor coolant temperature and pressure have been decreased to approximately 350°F and 400 psia, the shutdown cooling system can be placed in operation.

During shutdown cooling, the reactor coolant flows out of the RCS through the shutdown cooling nozzles located on each hot leg. Reactor coolant is circulated by the shutdown cooling pumps through the shutdown cooling heat exchangers and then returned to the RCS through the four safety injection lines to the reactor vessel. The cooldown rate is controlled by adjusting flow through the heat exchangers with a throttle valve on the discharge of each heat exchanger. Flow indicators provide indication so that the operator can maintain a constant total shutdown cooling flow to the core. The operator then adjusts the heat exchanger bypass flow to compensate for changes in flow through the heat exchangers. Operator control of shutdown cooling is only required until full flow is established through the shutdown cooling heat exchangers.

The SCS is designed to permit a portion of the flow to be diverted at the outlet of the shutdown cooling heat exchangers to the purification portion of the Chemical and Volume Control System. The bypass stream passes through a purification ion exchanger and boronometer to provide for continued cleanup of the RCS during plant cooldown, and to provide a means of monitoring the boron concentration. The capability to continue reactor coolant purification during cooldown and while at refueling conditions is desirable since crud release can result from temperature changes and fuel handling.

The SCS can be used to provide additional cooling for the spent fuel pool. Connections are provided on the discharge lines from each shutdown cooling heat exchanger and on the shutdown cooling suction lines.

The SCS is protected from overpressurization by pressure interlocked isolation valves and by pressure relief valves. The shutdown cooling suction line isolation valves receive electrical power in a manner such that no fault to a single power supply can open the valves to connect the RCS and SCS inadvertently, nor can a fault to a single power supply prevent opening the valves of at least one suction line for initiation of shutdown cooling. The RCS can be brought to refueling temperature using one shutdown cooling pump and one shutdown cooling heat exchanger.

Overpressure protection of the RCS during low temperature conditions is provided by the relief valves located in the shutdown cooling system (SCS) suction lines.

Alignment of the SCS relief valve to the RCS is provided via plant procedures to ensure RCS overpressure protection for all temperatures below the pressure-temperature (P-T) operating curve limits corresponding to the pressurizer safety valve set pressure of 2500 psia. For temperatures above the temperature limit which corresponds to the pressurizer safety valve setpoint, overpressure protection is provided by the pressurizer safety valves.

During heatup, RCS pressure is maintained below the maximum pressure for SCS operation until RCS cold leg temperature exceeds the applicable P-T operating curve temperature corresponding to 2500 psia. If the SCS suction isolation valves are open and RCS pressure exceeds the maximum pressure for SCS operation, an alarm will notify the operator that a pressurization transient is occurring during low temperature conditions. Either SCS relief valve will terminate inadvertent pressure transients which occur when the RCS temperature is below the aforementioned temperature limit. Above the maximum Low Temperature Over Pressurization (LTOP) temperature, overpressure protection is provided by the pressurizer safety valves when the SCS relief valve is isolated from the RCS.

During cooldown whenever RCS cold leg temperature is below the applicable temperature for LTOP, the SCS relief valves provide the necessary protection. If the SCS is not aligned to the RCS before cold leg temperature decreases below the critical LTOP value, an alarm will notify the operator to open the SCS suction isolation valves. The maximum temperature requiring LTOP is based upon an evaluation of the applicable P-T curves. However, the SCS cannot be aligned to the RCS until the pressure is below the maximum pressure allowing SCS operation.

These LTOP conditions are within the SCS operating range. Technical Specification requires the SCS suction line isolation valves to be open when operating in the LTOP mode. Also, this Technical Specification ensures that appropriate action is taken if one or more SCS relief valves are out of service during the LTOP mode of operation.

The SCS relief valves are spring loaded liquid relief valves with sufficient capacity to mitigate the most limiting overpressurization event. Either SCS relief valve will provide sufficient relief capacity to prevent any pressure transient from exceeding the isolation interlock setpoint. Since each SCS relief valve is a self actuating spring loaded liquid relief valve, control circuitry is not required. The valve will open when pressure exceeds its setpoint. This method of providing low temperature overpressure protection has been approved by the U.S. Nuclear Regulatory Commission and is in use on System 80 at the Palo Verde Nuclear Generating Station.

3. Safety Injection System

The Safety Injection System (SIS) provides core cooling in the event of a loss of coolant accident. The SIS is designed to supply sufficient cooling to prevent significant alteration of core geometry, to preclude fuel melting, to limit the cladding metal-water reaction, and to remove the energy generated in the core for an extended period of time following a loss of coolant accident.

Figures II-D-3 and II-D-4 depict the SIS in the two modes of operation for emergency core cooling: the Short-Term Mode and the Long-Term Mode.

The principal components of the SIS are the four safety injection pumps, the four safety injection tanks, and the In-containment Refueling Water Storage Tank (IRWST). These components are utilized in redundant active and passive injection subsystems to provide core protection for the complete spectrum of reactor coolant pipe breaks, satisfying both short-term and long-term cooling requirements.

a. <u>Short-Term Mode</u>

The short-term cooling requirements following a loss of coolant accident are met by a passive injection sub-system consisting of the pressurized safety injection tanks and four active sub-systems consisting of the safety injection pumps, and associated valves and piping.

The four safety injection tanks are used to flood the core with borated water upon depressurization of the Reactor Coolant System. Each tank is piped into a safety injection nozzle located near the top of the reactor vessel cylindrical shell. During normal plant operation each safety injection tank is isolated from the Reactor Coolant System by two check valves in series. However, the safety injection tanks will automatically discharge into the Reactor Coolant System if the Reactor Coolant System pressure decreases below safety injection tank pressure during reactor operation. Adequate fluid is contained in the tanks to assure that the required volume reaches the core, assuming that the content of one tank is lost, i.e., spills through the postulated break in the Reactor Coolant System.

In the active injection system, four (4) full capacity pumps inject borated water from the in-containment refueling water storage tank into the four (4) nozzles on the reactor vessel cylindrical shell. Injection is automatically initiated upon a low pressurizer pressure signal or a high containment pressure signal. A subsystem of two (2) pumps is capable of performing the short term cooling function with one of its injection flow paths discharging through the break. The Safety Injection System pumps and valves are connected to the normal power source, and the emergency diesel generators. The connections are through two (2) independent buses so that in the event of a loss of coolant accident in conjunction with a single active failure in the emergency electrical supply system, the flow from two (2) safety injection pumps is available.

b. <u>Long-Term Mode</u>

Long-term cooling following a loss of coolant accident is accomplished by manually opening the hot leg injection valves. Simultaneous hot leg and direct vessel injection results in a circulation flow through the reactor core. For small pipe breaks, the SIS pumps provide makeup for spillage, while the RCS is cooled down and depressurized to shutdown cooling initiation conditions utilizing the steam generator Atmospheric Dump Valves and Emergency Feedwater System.

4. Containment Spray System

The Containment Spray System (CSS) is a safety grade system designed to reduce containment pressure and temperature following a main steam line break or loss-of-coolant-accident and to remove fission products from the containment atmosphere following a loss of coolant accident. Fission product removal is required so that in the event of containment leakage, activity at the site boundary due to radioactive iodine will be reduced.

The CSS uses the in-containment refueling water storage tank (IRWST) and has two independent trains (two containment spray pumps, two containment spray heat exchangers, two independent spray headers, and associated piping, valves, and instrumentation). The system is shown in Figures II-D-3 and II-D-4.

The CSS provides borated water spray to the containment atmosphere from the upper regions of the containment. The spray flow is provided by the containment spray pumps which take suction from the in-containment refueling water storage tank. Upon receipt of a Containment Spray Actuation Signal (CSAS), the containment spray header isolation valve opens and the containment spray pump starts in each of the two redundant trains.

The pumps discharge through the containment spray heat exchangers and the spray header isolation valves to their respective spray nozzle headers, and then into the containment atmosphere. The spray headers are located in the upper part of the containment building to allow the falling spray droplets time to approach thermal equilibrium with the steam-air atmosphere. Condensation of the steam by the falling spray results in reduction in containment pressure and temperature. When the water reaches the containment floor it drains to the holdup volume and subsequently back to the IRWST.

A CSAS occurs on a two out of four high-high containment pressure signal. The CSAS may also be initiated manually in the control room. The specific sequence of CSS pump and valve actuation depends on which power source is available. If offsite power is not available, the safeguards loads are divided between the two plant emergency diesel generators and are sequentially started after the diesel generators are running.

In the event that one or both of the shutdown cooling pumps is unable to perform its function of reducing RCS temperature to refueling temperature, one or both of the containment spray pumps can perform this function. Cross-connect valves to the suction and discharge lines of the CS pumps must be manually opened for shutdown cooling.

5. Emergency Feedwater System

The Emergency Feedwater (EFW) System provides an independent safety-related means of supplying secondary-side, quality feedwater to the steam generators for removal of heat and prevention of reactor core uncovery during emergency phases of plant operation. The EFW System is a dedicated safety system which has no operating functions during normal plant operation.

The EFW System is designed to be automatically or manually initiated, supplying feedwater to the steam generators for any event that results in the loss of normal and startup feedwater and requires heat removal through the steam generators, including the loss of normal onsite and normal offsite AC power.

Following the event, the EFW System maintains adequate feedwater inventory in the steam generators for residual heat removal. It is capable of maintaining hot standby, and facilitating a plant cooldown at the maximum administratively controlled rate of 75°F/hr, from hot standby to Shutdown Cooling System initiation. The Shutdown Cooling System becomes available for plant cooldown when the RCS temperature and pressure are reduced to 350°F and 400 psia.

The EFW System is designed to be initiated by the operator following a major loss of coolant accident, to keep the steam generator tubes covered. Covering the steam generator tubes following a LOCA minimizes potential containment bypass leakage, should pre-existing primary-to-secondary leakage be present.

The EFW System is configured into two separate subsystems as shown in Figure II-D-5. Each subsystem is aligned to feed its respective steam generator, and consists of one Emergency Feedwater Storage Tank (EFWST), one 100% capacity motor-driven pump, one 100% capacity steam-driven pump, associated valves, one cavitating venturi, and specified instrumentation. Each pump takes suction from its respective EFWST, and discharges through a check valve, flow regulating valve, steam generator isolation valve and steam generator isolation check valve. The motor-driven and steam-driven trains are joined together inside containment to feed their respective steam generator through a common EFW header which connects to the steam generator downcomer feedwater line. Each common EFW header contains a cavitating venturi to restrict the maximum EFW flow rate to each steam generator. The cavitating venturi restricts the magnitude of the two pump flow as well as the magnitude of individual pump runout flow to the steam generator.

A cross-connection is provided between each EFWST so that either tank can supply either train of EFW. The two EFWSTs are safety grade tanks of seismic design. Each tank contains 100% of the total volume required to meet the EFW system design bases. A local, normally locked closed, manually operated isolation valve is provided for each EFWST to provide separation. A line connected to a non-safety source of condensate is also provided with local manual isolation so that it can be manually aligned for gravity feed to either of the EFWSTs, should the EFWSTs reach low level before Shutdown Cooling System entry conditions are reached. A check valve and a local, normally locked closed, manually operated isolation valve are provided for separation of the non-safety source of condensate from the safety-related sources.

Pump discharge crossover piping is provided to enhance system versatility during long-term emergency modes, such that a single pump can feed both steam generators. Two local, normally locked closed, manually operated isolation valves are provided for subtrain separation.

The EFW System can either be manually actuated or automatically actuated by an Emergency Feedwater Actuation signal (EFAS) or by the Alternate Protection System (APS). At the low steam generator water level setpoint, the EFAS or the APS will actuate the EFW System as follows:

- Starts the associated motor-driven pump and opens the associated turbine steam supply bypass valve
- Starts associated turbine driven pump by opening the associated turbine steam supply valve and opens the associated steam generator isolation valves
- Assures that the associated EFW flow control valves are in their full open position
- Assures that turbine governor speed control is at full rated speed.

After the EFW System has been actuated, the plant operator will control the flow to the steam generators in order to control the steam generator water level. The operator has at least 30 minutes after EFW actuation before operator action is essential. The operator can control steam generator water level by either positioning the associated EFW pump flow control valves, by opening and shutting the associated EFW steam generator isolation valves, by adjusting the turbine governor control speed, or by using on/off operation of the motor driven EFW pumps.

The EFAS provides automatic protection functions, should the operator fail to control flow. The EFAS will automatically shut the steam generator isolation valves at a steam generator level setpoint higher than normal water level, to prevent steam generator overfill. If the steam generator water level falls to the low steam generator water level setpoint for EFW actuation, the EFW system is reactuated, as described above. An EFAS override is provided for each steam generator so that EFW flow can be terminated in the event of a steam generator fault, such as a main feedwater or main steam line break.

6. Component Cooling Water System

The component cooling water system (CCWS) is a closed loop cooling water system, which cools components and heat exchangers located in the Nuclear Systems Annex, Radwaste, and Containment Buildings. Heat transferred by these components to the CCWS is rejected to the essential service water system (ESWS) via the CCWS heat exchangers.

Safety design bases applicable to the CCWS are as follows:

- The CCWS, in conjunction with the ESWS (including the reserve ultimate heat sink), is capable of removing sufficient heat to ensure a safe reactor shutdown coincident with a loss of offsite power.

- A single failure of any component in the CCWS will not impair the ability of the CCWS to meet its functional requirements.

- Adverse environmental occurrences will not impair the ability of the CCWS to meet its functional requirements.

- The CCWS is designed to withstand the effects of a safe shutdown earthquake (SSE).

The CCWS consists of two separate, independent, redundant, closed loop, safety-related divisions (Figure II-D-6). Either division of the CCWS or a single pump in each division is capable of supporting 100 percent of the cooling functions required for a safe reactor shutdown. Post-LOCA, all pumps in both divisions automatically start, however, only one of the pumps is required to operate to support the cooling function after the cooling supply to the non-safety-related loads and fuel pool cooling heat exchangers are isolated.

The CCWS operates at a higher pressure than the ESWS, as protection against leakage into the CCWS from the ESWS in case of tube leakage in the CCWS heat exchanger.

Each division of the CCWS includes two heat exchangers, a surge tank, two 100 percent pumps, a chemical addition tank, piping, valves, controls, and instrumentation.

Makeup water to the CCWS is supplied by the demineralized water system. Should the demineralized water system be unavailable, during an accident, makeup can be supplied from the assured makeup source.

Each division of the CCWS provides cooling for redundant safety-related components. These include:

- Shutdown cooling heat exchangers
- Safety injection pump motor coolers
- Containment spray heat exchangers
- Shutdown cooling pump motor coolers
- Containment spray pump motor coolers
- Motor driven emergency feedwater pump motor coolers
- Essential chillers

Each division can also provide cooling for the following non-safety-related components:

- Reactor coolant pump (RCP) motor air coolers
- RCP motor upper bearing oiler coolers
- RCP motor lower bearing coolers
- RCP oil coolers
- RCP seal coolers
- RCP high pressure cooler
- Letdown heat exchanger
- Fuel pool heat exchanger
- Sample heat exchangers
- Gas stripper
- Boric acid concentrator
- Normal chillers
- CEDM air coolers
- Containment spray pump mini-flow heat exchangers
- Shutdown cooling pump mini-flow heat exchangers
- Charging pumps
- Instrument air compressors
- Other miscellaneous components

Cross-connections are provided with locked, closed valves to be utilized during shutdown, if one division is out for maintenance.

7. AC Power Systems

There are three AC power systems which service station electrical loads: the class 1E AC power system, the non-1E AC power system, and the non-1E alternate AC power system.

a. Class 1E AC Power System

The class 1E AC power system consists of a 4160 volt power system, and a 480 volt auxiliary power system. The 4160 volt system is a redundant system, divided into safety divisions I and II. It normally receives power from the 4160 volt normal power system. Upon loss of normal power, emergency power is provided either by separate and independent emergency diesel generators, or by an onsite standby gas turbine generator. All safety related equipment requiring power during a loss of offsite power event, or during other accident conditions, is fed by this system.

The 480 volt auxiliary power system includes two load centers tied to each division of the 4160 volt system. These four redundant load centers furnish power for large heater loads, large 480 volt motors, and 480 volt class 1E motor control centers.

b. **Non-1E Uninterruptible AC Power System**

The Non-1E uninterruptible AC power system is divided into four subsystems: the unit main power system, the 13,800 volt normal auxiliary power system, the 4160 volt normal auxiliary power system, and the 480 volt normal auxiliary power system.

The unit main power system consists of the main generator, isolated phase bus, generator circuit breaker, four unit step-up transformers, and two half-sized unit auxiliary transformers. This system generates and transmits power to the transmission system, while simultaneously supplying power to the unit auxiliaries. In the event that the main generator is not in service, this system can be used to supply power from the transmission system to the unit auxiliaries.

The 13,800 volt normal auxiliary power system consists of four non-safety buses which are connected to the unit auxiliary transformers. This system furnishes power to large motors, such as the reactor coolant pump motors and condensate water pump motors. Protective relays are provided for both the motor loads and the buses.

The 4160 volt normal auxiliary power system consists of four switchgear groups and a non-class 1E source (onsite gas turbine). The first switchgear group is connected to a unit auxiliary transformer, to power large non-safety related loads such as service water and component cooling water pumps. The second switchgear group is connected to the remaining unit auxiliary transformer to power the remaining large, non-safety related loads.

The third switch gear group, designated "permanent non-safety", provides power to auxiliary and service loads which must typically remain operational, independent of the plant operating condition, or during plant outages. The normal power source is the 4160 volt unit auxiliary transformer. If necessary, it can be switched to either the station auxiliary transformer, or the alternate AC source (gas turbine). The fourth switchgear group is also designated "permanent non-safety", and is configured in a manner similar to the third switchgear group.

The four non-class 1E switchgear groups, with four diverse sources of power, and the ability to energize the Division I and II safety loads, reduces the likelihood of a total station blackout occurrence.

c. **Non-Class 1E Alternate AC Power Source**

A non-class 1E onsite alternate AC power source is provided for motors and other electrical loads which have a special regulatory or operational significance, but which are not classified as safety related (e.g., CVCS charging pumps, CEDM cooling fans, instrument air compressors). The power source is a gas turbine, and it is connected to the two 4160 volt buses designated as "permanent non-safety". These buses are normally supplied by the unit auxiliary transformer, and if necessary, can be switched to the station transformer.

This diverse alternate AC power source has been incorporated as a backup to the diesel generators. This gas turbine generator is capable of handling all necessary loads in the event of a station blackout.

8. Dual Steel Containment

The Steel Containment Vessel (SCV), penetration assemblies, equipment hatch, personnel air locks and isolation valves for systems penetrating the SCV form the Containment System. The SCV is a free standing, low leakage (essentially leak tight) welded steel pressure vessel designed in accordance with the requirements of the ASME Code for pressure vessels.

The SCV is housed in a reinforced concrete shield building with an annular area between the concrete and the free standing SCV (Figure II-D-8). The shield building is designed to protect the SCV from the outside environment, i.e., normal winds, tornado winds, tornado generated missiles, extreme temperature fluctuations, snow loads, etc. The SCV is similarly protected from internal missiles, jet impingements and transient pressure differentials by the reinforced concrete crane wall.

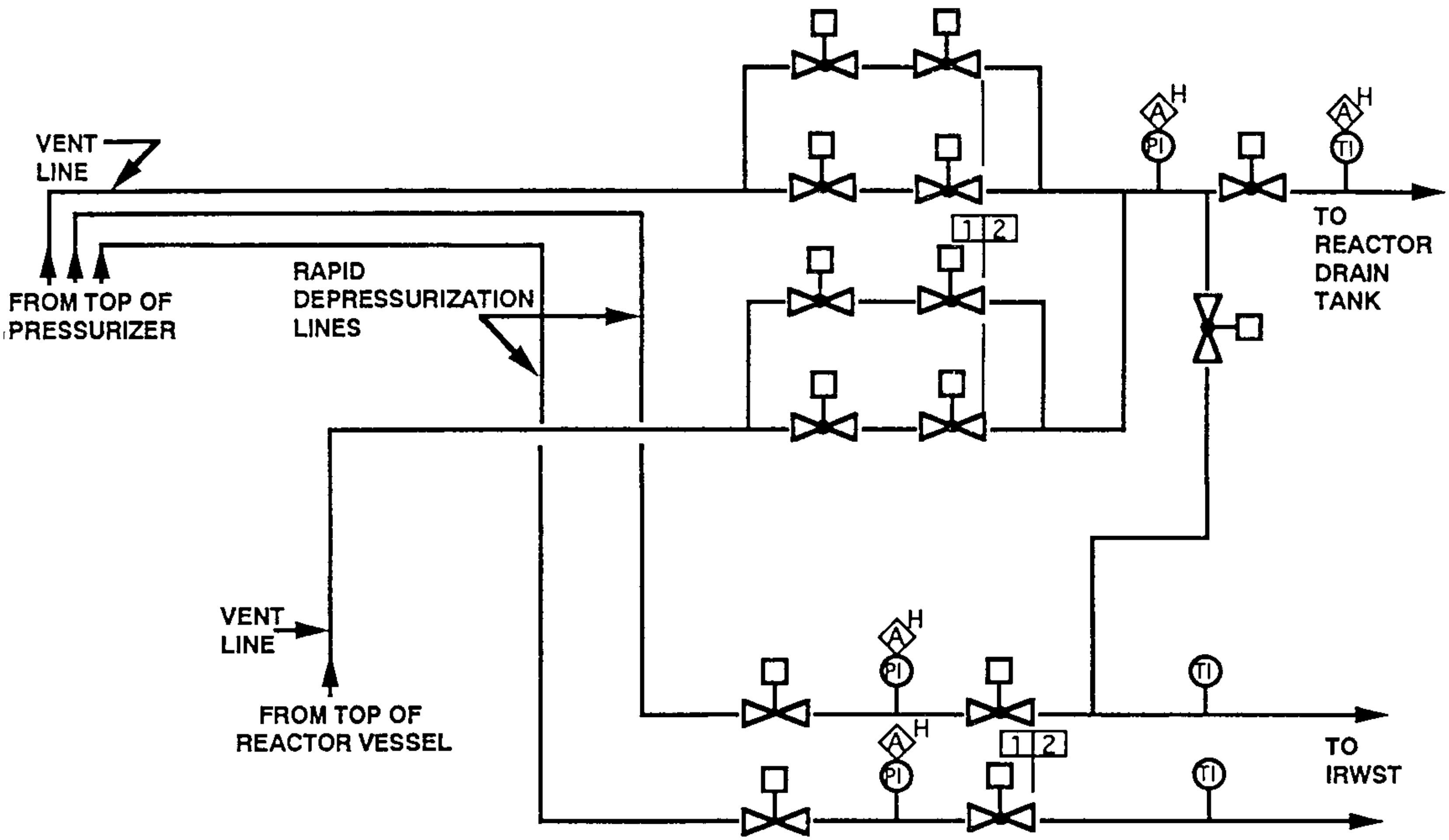
The Containment System is required to provide an essentially leak tight barrier enclosing the Reactor Coolant System. This barrier serves to control and limit the consequence of releases of energy and radioactivity in the event of a breach of the Reactor Coolant System, so that public health and safety will not be impaired. The Containment System is also required to withstand all internal and external environmental conditions that may reasonably be expected to occur during the design life of the plant. This includes both the short and long term effects following a loss-of-coolant accident or main steam line break, in accordance with the General Design Criterion 16, "Containment Design", of Appendix A to 10 CFR Part 50 "General Design Criteria for Nuclear Power Plants". Those requirements state that a Reactor Containment and associated systems be provided for the following purposes:

- to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment
- to assure that the containment design conditions important to safety shall not be exceeded for as long as postulated accidents require.

The SCV pressure boundary includes all penetration sleeves that are attached to the SCV including the equipment hatch and the two personnel air locks.

The Containment Vessel is classified as an MC component. Subsection NE of the ASME Code establishes the rules for material, design, fabrication, examination, inspection, testing, and certification of the metal Containment System.

All materials used in the fabrication and erection of the SCV meet the requirements of Article NE-2000 of the ASME code. The material for the SCV shell plate is SA 537 Class 2. The material for the penetration assemblies is SA 671 or SA 537 Class 2.



NOTES:

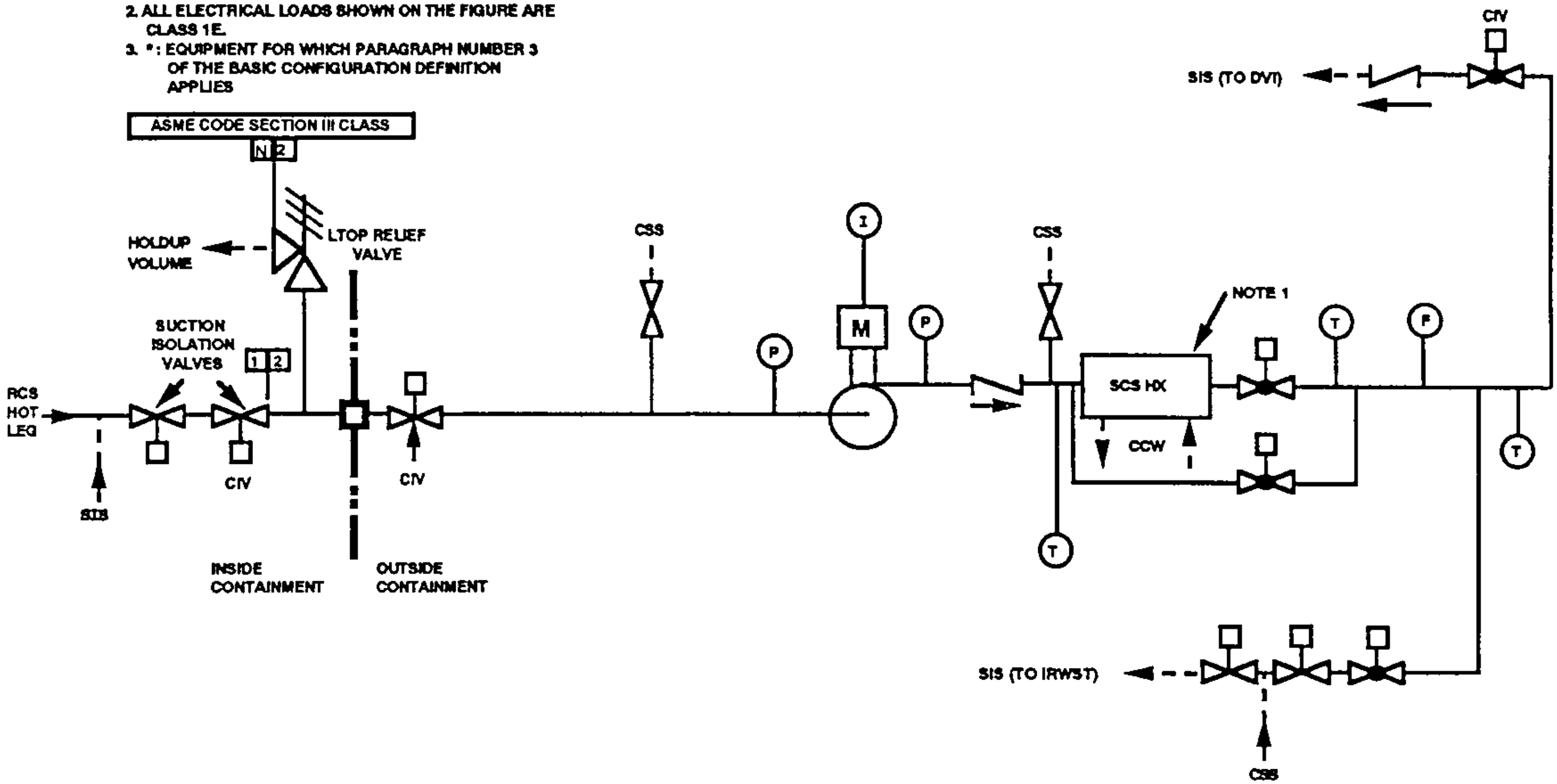
ALL ITEMS SHOWN EXCEPT INDICATIONS & ALARMS ARE INSIDE THE CONTAINMENT BUILDING

IRWST: IN CONTAINMENT REFUELING WATER STORAGE TANK

SYSTEM 80+™	SAFETY DEPRESSURIZATION SYSTEM	FIGURE II-D-1

NOTE:

1. TUBESIDE IS ASME CODE SECTION III CLASS 2 AND SHELLSIDE IS ASME CODE SECTION III CLASS 3.
2. ALL ELECTRICAL LOADS SHOWN ON THE FIGURE ARE CLASS 1E.
3. *: EQUIPMENT FOR WHICH PARAGRAPH NUMBER 3 OF THE BASIC CONFIGURATION DEFINITION APPLIES

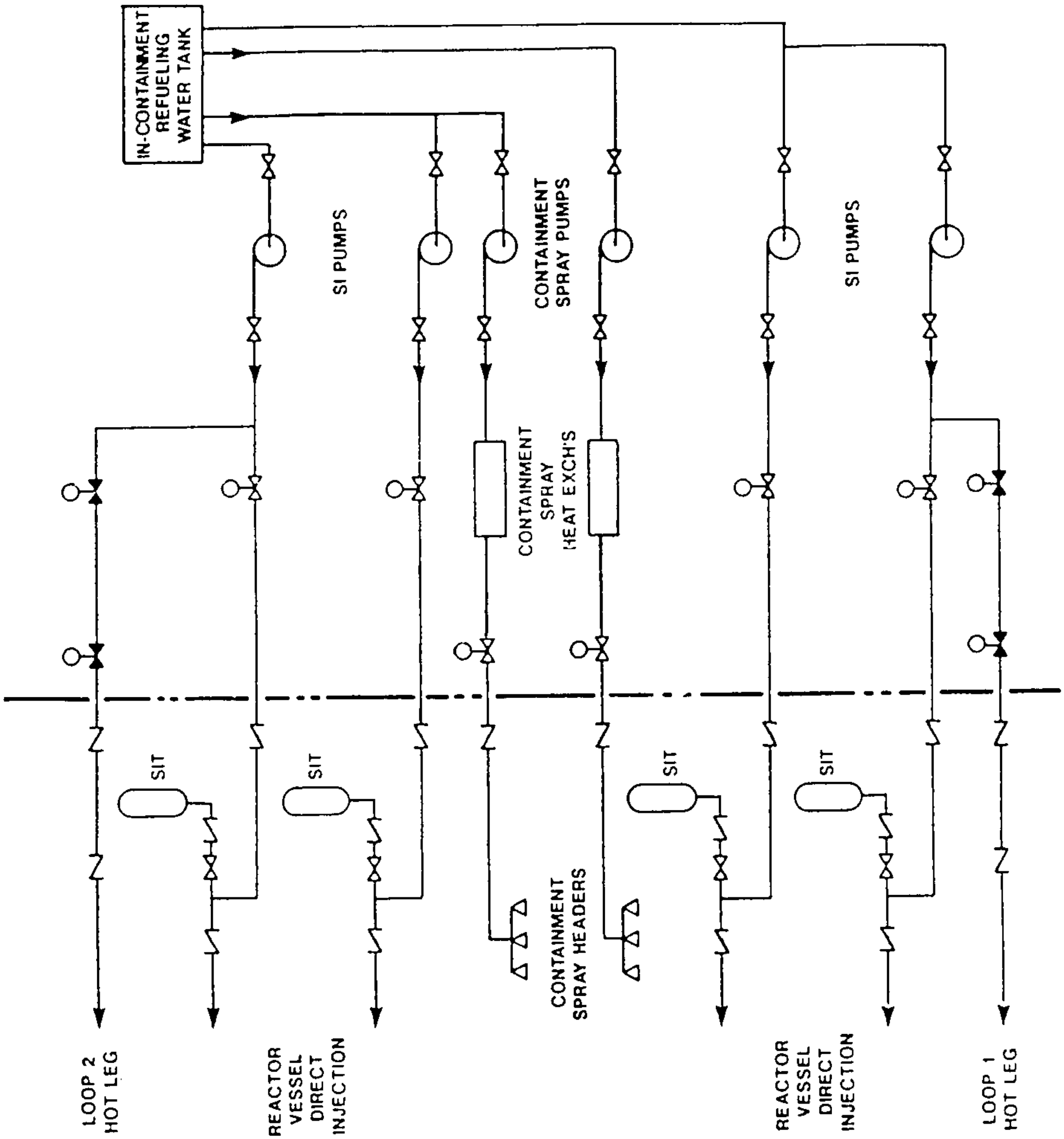


SYSTEM 80+™

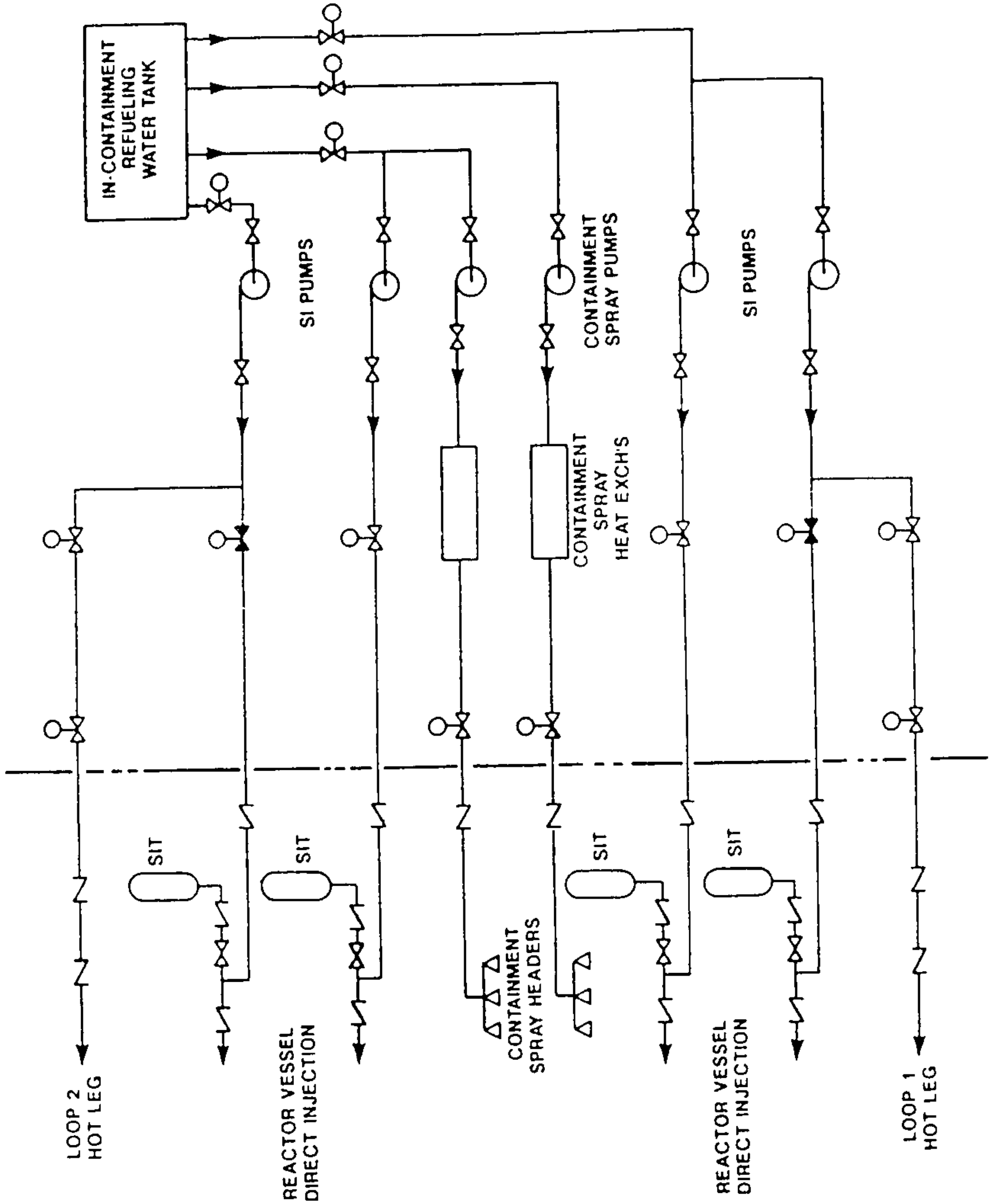
SHUTDOWN COOLING SYSTEM

FIGURE II-D-2

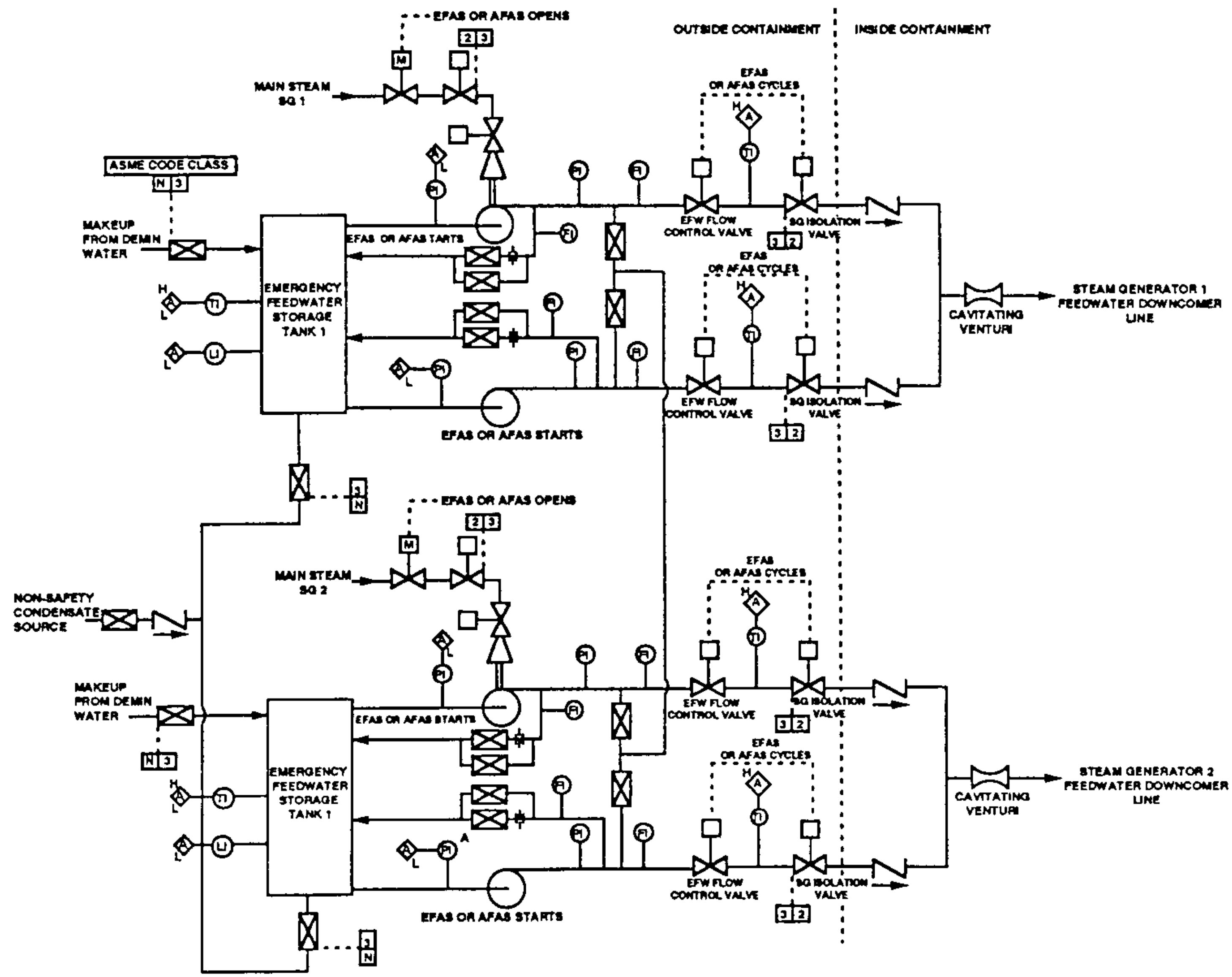
ABB
ASEA BROWN BOVERI



SYSTEM 80+™	SAFETY INJECTION SYSTEM /CONTAINMENT SPRAY SYSTEM SHORT-TERM MODE	FIGURE II-D-3



SYSTEM 80+™	SAFETY INJECTION SYSTEM /CONTAINMENT SPRAY SYSTEM LONG-TERM MODE	FIGURE II-D-4

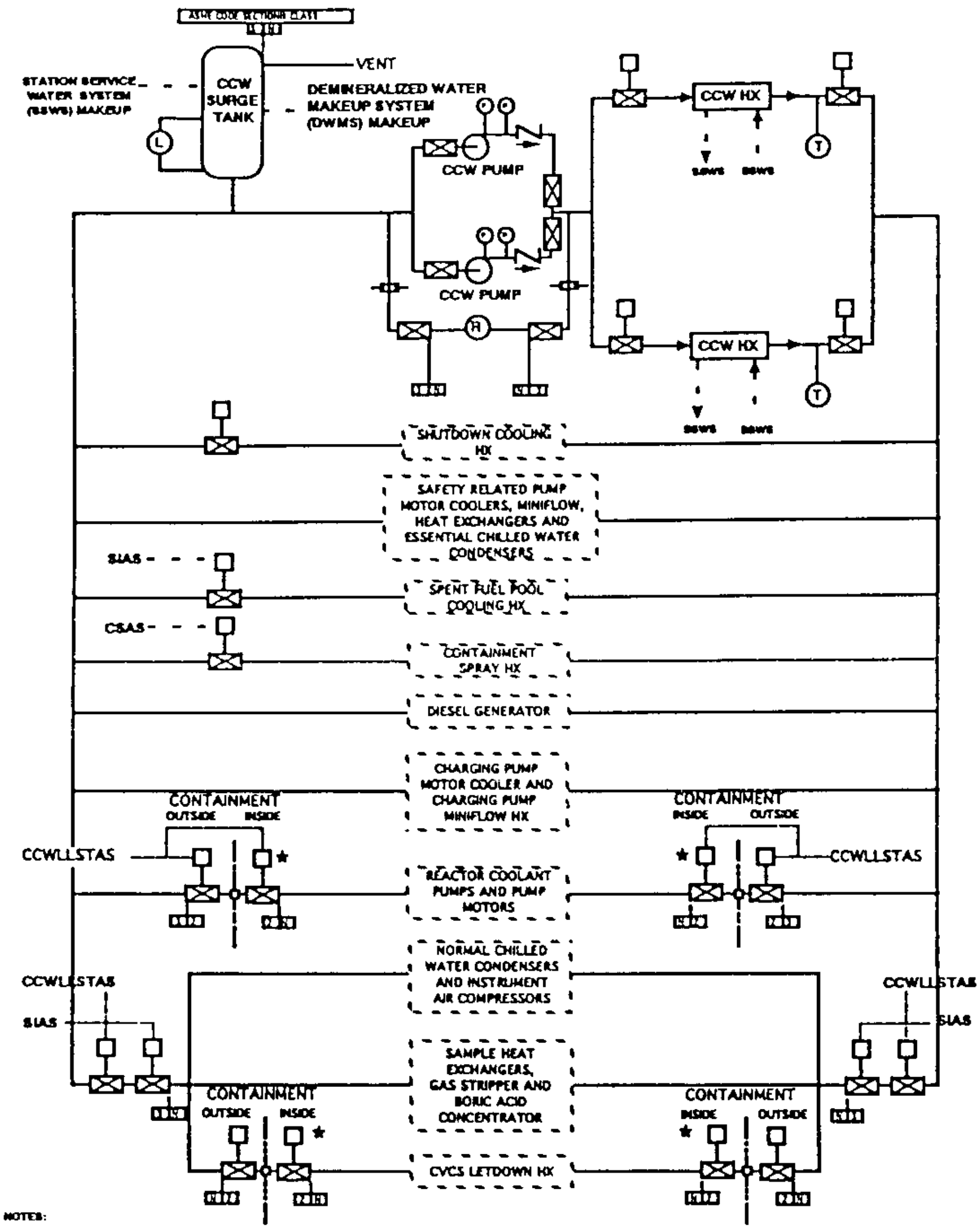


SYSTEM 80+™

EMERGENCY FEEDWATER SYSTEM

FIGURE II-D-5

ABB
ASEA BROWN BOVERI



NOTES:

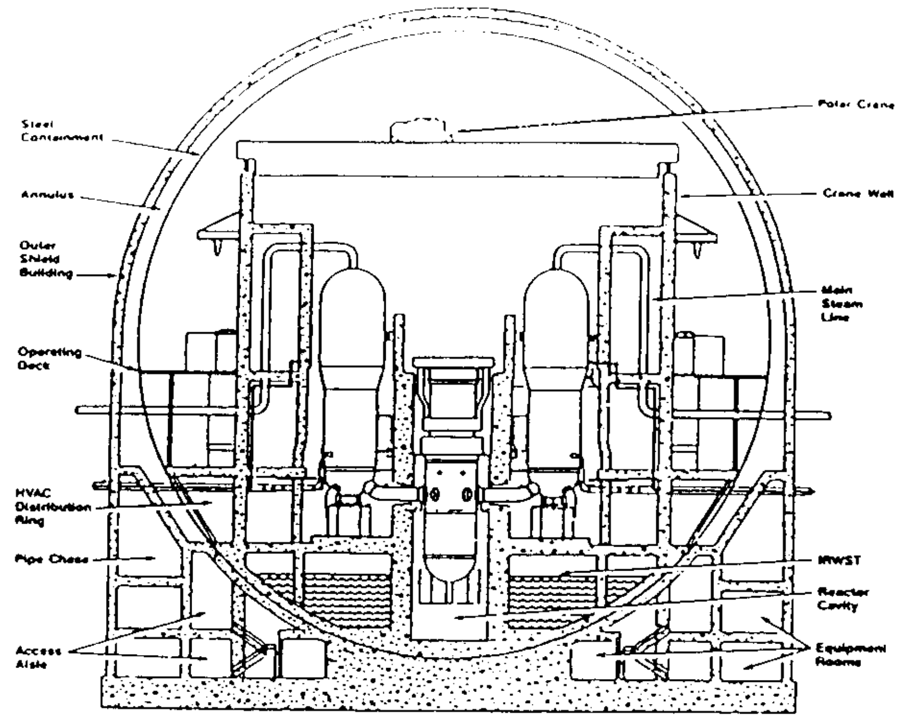
A. ALL ASME CODE CLASS 2 AND 3 COMPONENTS SHOWN ON THE FIGURE ARE SAFETY RELATED.

B. ASSIGNMENT OF THE NON-SAFETY RELATED CCWS HEAT REMOVAL LOADS TO THEIR RESPECTIVE CCWS DIVISION IS DEPENDENT UPON THE LOCATION OF COMPONENTS ASSOCIATED WITH THOSE LOADS

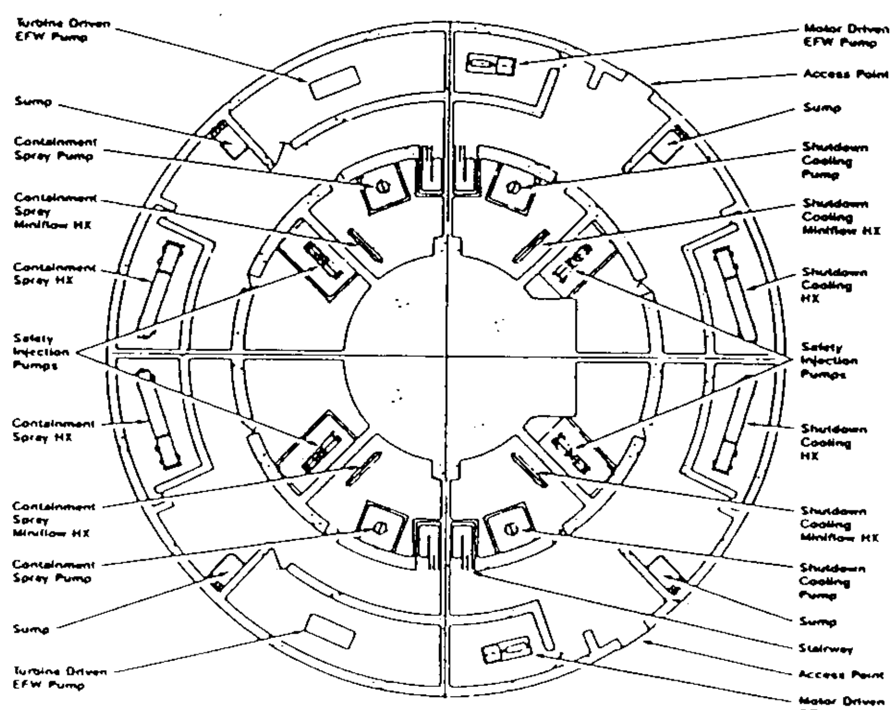
C. SAFETY RELATED COMPONENTS AND EQUIPMENT SHOWN ON THE FIGURE ARE POWERED FROM THEIR RESPECTIVE CLASS 1E BUS.

* EQUIPMENT FOR WHICH PARAGRAPH NUMBER 3 OF THE BASIC CONFIGURATION DEFINITION APPLIES.

SYSTEM 80+™	COMPONENT COOLING WATER SYSTEM (ONE OF TWO DIVISIONS)	FIGURE II-D-6



Elevation View of System 80+ Containment



Plan View of System 80+ Containment

SYSTEM 80+ ™	SYSTEM 80 + ™ CONTAINMENT GENERAL ARRANGEMENT	FIGURE II-D-7

E. AUXILIARY SYSTEMS INCLUDING I&C

Feedback based on operational experience has permitted incremental improvements in the auxiliary systems.

The following auxiliary systems are of particular interest:

- Chemical and Volume Control System
- Process Sampling System
- Spent Fuel Pool Cooling and Cleanup System

1. Chemical and Volume Control System

Although not required to perform any accident mitigation or safe shutdown functions, the CVCS is essential for the normal day-to-day operation of the plant. The CVCS has therefore been provided with a high degree of reliability and redundancy and has been designed in accordance with accepted industry standards and quality assurance commensurate with its importance to plant operations.

The chemical and volume control system is designed to perform the following functions:

- Maintain the chemistry and purity of the reactor coolant during normal operation and during shutdowns.
- Maintain the required volume of water in the reactor coolant system, compensating for reactor coolant contraction or expansion resulting from changes in reactor coolant temperature and for other coolant losses or additions.
- Receive, store,and separate borated waste for reuse and discharge to the liquid waste management system (LWMS).
- Control the boron concentration in the RCS to obtain optimum Control Element Assembly positioning, to compensate for reactivity changes associated with major changes in reactor coolant temperature, core burnup, and xenon variations, and to provide shutdown margin for maintenance and refueling operations.
- Provide auxiliary pressurizer spray for operator control of pressurizer pressure during the final stages of shutdown, and to allow for pressurizer cooling.
- Provide a means for functionally testing the check valves which isolate the safety injection system from the RCS.
- Provide injection water at the proper temperature, pressure, and purity for the reactor coolant pump seals, and collect the controlled bleedoff from the reactor coolant pump seals.

- Leak test the RCS.
- Provide a reactor makeup water supply to various auxiliary equipment.
- Provide a means for sluicing ion exchanger resin to the solid waste management system.
- Provide a means for continuous removal of noble gases from the RCS.
- Provide makeup to the spent fuel pool.
- Provide purification of shutdown cooling flow.
- Provide makeup for losses from small leaks in RCS piping.
- Provide a means to purify contents of the IRWST.
- Provide a means to add makeup and adjust chemistry of IRWST.

The normal reactor coolant flowpath through the CVCS is indicated by heavy lines on Figures II-E-1 (Sheets 1 and 2).

Letdown flow from the reactor coolant system passes through the tube side of the regenerative heat exchanger, where an initial temperature reduction takes place via heat transfer to cooler charging fluid on the shell side of the heat exchanger. The regenerative heat exchanger is designed to cool letdown fluid to a maximum of 450°F for all normal operations, and to heat the charging flow by a minimum of 100°F. A final temperature reduction to the purification subsystem operating temperature is made by the letdown heat exchanger. The letdown heat exchanger is sized to cool inlet water from the maximum regenerative heat exchanger outlet temperature to 120°F (or lower) for most operating conditions. Both the letdown and the regenerative heat exchangers are designed for full RCS pressure and both are located inside containment.

Letdown fluid pressure is reduced from full system pressure to the operating pressure of the purification subsystem in two stages. The first pressure reduction occurs at the letdown orifice and the second occurs at the letdown control valve downstream of the letdown orifice. The letdown orifice is sized to pass the maximum letdown flow at full RCS pressure with the control valve fully open. A bypass orifice around the normal letdown orifice is provided for low pressure operation.

Following temperature and pressure reduction, the coolant flows through a filter where filterable impurities are removed to reduce crud buildup on ion exchanger resin. The entire flow passes through a purification ion exchanger where ionic corrosion and fission products are removed.

Two full capacity purification ion exchangers are provided. The unit not in continuous service is used to periodically remove lithium from the coolant to control the lithium concentration within the desired range. This unit becomes exhausted for lithium capacity during one core cycle and is used in the next cycle for continuous purification. A deborating ion exchanger is provided for use near the end of core life for removal of boron, since the normal feed and bleed process becomes much less efficient at low boron concentrations. All ion exchanger resins are protected from high temperature coolant by a diversion valve, which automatically bypasses the ion exchangers if the temperature of the coolant entering the ion exchangers exceeds 140°F. Coolant flows from the ion exchangers through a strainer which provides a backup for the ion exchanger retention elements. The coolant is then sprayed into the volume control tank.

For continuous degasification capability, the letdown fluid, after passing through the filter, ion exchanger(s) and strainer, can be diverted from the volume control tank, and processed through the gas stripper before being returned to the volume control tank. Use of the gas stripper on a continuous basis significantly reduces the gaseous activity in the coolant, thereby mitigating the radiation hazards involved in primary to secondary leakage or a loss of coolant accident.

The volume control tank is designed to accumulate letdown water from the RCS, to provide for control of hydrogen concentration in the reactor coolant, and to provide a reservoir of reactor coolant for the charging pumps. The charging pump normally takes suction from the volume control tank and discharges to the RCS. One letdown and one charging pump flow control valve are selected for use. The charging pump mini-flow heat exchanger uses component cooling water to cool the recirculation flow from the operating charging pump.

Seal injection water is supplied to the reactor coolant pumps by diverting a portion of the charging flow at a point in the system just downstream of the charging pumps. This seal flow is then heated in a steam heater to approximately 125°F before filtering. Once the flow has been filtered, the seal injection fluid is distributed to the reactor coolant pumps. The undiverted charging fluid is sent to the regenerative heat exchanger where it is heated before injection into the RCS.

If necessary, a chemical addition tank and a chemical addition metering pump are used to transfer chemical additives to the charging line downstream of the seal injection takeoff connection.

a. Chemistry and Purity Control

The chemistry and purity of the reactor coolant are controlled to provide the following:

- minimize the corrosion of primary side hardware, which includes minimizing the fouling of heat transfer surfaces,
- adjust the chemical shim properly to control core reactivity throughout the life of the core, and
- ensure that the quality of the reactor coolant is being maintained.

Chemistry control of the reactor coolant consists of preoperational removal of oxygen by hydrazine addition (hydrazine scavenging), degasification (via the gas stripper) of RCS fluid (when necessary), control of oxygen concentration by maintaining an excess hydrogen concentration, and pH control by maintaining lithium within a specific control band during normal operation.

Preoperational oxygen scavenging is accomplished by the addition of hydrazine. During hot functional testing, 30 to 50 ppm of hydrazine is maintained in the reactor coolant whenever the reactor coolant temperature is below 150°F. This prevents halide-induced attack, which could occur if significant quantities of fluorides or chlorides and significant amounts of dissolved oxygen are present. During heatup, any dissolved oxygen is scavenged by the hydrazine, eliminating the potential for general corrosion. At higher temperatures, the hydrazine decomposes, forming ammonia. The resultant increase in pH aids in the development of passive oxide films on reactor coolant system surfaces. It has been well established that the corrosion rates of Ni-Cr-Fe Alloy-690 and 300 series stainless steels decrease with time when exposed to normal reactor coolant chemistry conditions, approaching low steady state values within approximately 200 days. The high pH condition produced by a high ammonia concentration (to 50 ppm) minimizes corrosion product release and assists in the rapid development of the passive oxide film. Most of the film is established within seven days at hot, high pH conditions. To aid in maintaining the pH during this passivation period, lithium in the form of lithium hydroxide, is added to the coolant and maintained within a 1-2 ppm lithium-7 range.

At power, oxygen concentration is limited by maintaining excess dissolved hydrogen gas in the coolant. The excess hydrogen forces the water decomposition/synthesis reaction in the reactor core toward water synthesis rather than hydrogen and oxygen decomposition.

In order to minimize the effect of crud deposition on the reactor core heat transfer surfaces, lithium-7 hydroxide additions to the reactor coolant are made. The lithium-7 hydroxide produces pH conditions within the reactor coolant (at normal operating temperatures) that reduce the corrosion product solubility and, hence, the dissolved crud inventory in the circulating reactor coolant. The elevated pH promotes conditions within the coolant for selective deposition of corrosion products on cooler surfaces (steam generators) rather than hotter surfaces (core). An additional advantage is the formation of a more stable and tenacious passive oxide layer on out-of-core system surfaces. The lithium concentration is maintained with a 0.2-2.3 ppm lithium-7 range during operation.

A chemical addition tank and pump is used to transfer hydrazine and/or lithium hydroxide to the discharge side of the charging pumps for injection into the RCS. The hydrogen concentration in the reactor coolant is controlled by a hydrogen overpressure in the volume control tank.

The control of other impurities is accomplished by the continuous operation of a purification ion exchanger which contains resin which has been converted to the lithium or ammonia lithium form. The resin bed removes soluble nuclides by an ion exchange mechanism and insoluble particles by the impingement of these particles on the surface of the resin beads.

The normal method of adjusting RCS boron concentration is by "feed and bleed". To change concentration, the makeup portion of the CVCS supplies either reactor makeup water or boric acid to the volume control tank. Meanwhile, the letdown stream is diverted to the pre-holdup ion exchanger. Toward the end of the core cycle, however, the quantities of waste produced due to this "feed-and-bleed" operation become excessive and the deborating ion exchanger is used to further reduce the reactor coolant system boron concentration. An anion resin, initially in the hydroxyl form, is converted to a borate form as boron is removed.

b. Reactor Coolant Inventory

The volume of water in the RCS is automatically controlled using level instrumentation located on the pressurizer. The pressurizer level setpoint is programmed to vary as a function of RCS temperature in order to minimize the transfer of fluid between the RCS and CVCS during power changes. Reactor power is determined for this situation using the average reactor coolant temperature derived from hot and cold leg temperature measurements. A level error signal is obtained by comparing the programmed setpoint with the measured pressurizer water level. Volume control is achieved by automatic control of the charging and letdown flow control valves in accordance with the pressurizer level control program.

Two parallel charging pump flow control valves, a letdown orifice, letdown flow control valves, and two parallel charging pumps are provided. During normal operation, one charging pump is running with the other as an installed spare. In addition, one of the letdown and one of the charging pump flow control valves are selected for use. The selected charging pump flow control valve is maintained by the pressurizer level control program at a preset position which gives a constant desired flow rate at normal operating pressures. The position of the selected charging pump flow control valve is maintained constant by the pressurizer level control program, except in response to a high or low pressurizer level condition. Fine control of pressurizer level is accomplished via letdown control. The position of the selected letdown flow control valve is varied by the pressurizer level control program in response to the level error in order to compensate for small changes in pressurizer level and to keep it within the programmed level band.

As stated previously, the volume control tank provides a reservoir of reactor coolant for the charging pump. The level in the volume control tank is controlled automatically. The letdown flow is diverted to the holdup tank via the pre-holdup ion exchanger and gas stripper when the control band high volume control tank level is reached. Conversely, the volume control tank low level signal automatically initiates makeup flow to the volume control tank. The makeup system is normally set up for the automatic mode of operation, in which flow at a preset blend of boric acid from the boric acid storage tank (BAST) and demineralized water from the reactor makeup water tank (RMWT) is automatically initiated. A low-low level signal automatically closes the outlet valve on the volume control tank, opens the boric acid flow valves, and starts the boric acid makeup pumps to provide flow directly from the BAST through the boric acid makeup pumps to the suction of the charging pumps.

c. Boron Recovery

The boron recovery portion of the CVCS accepts the letdown flow diverted from the volume control tank as a result of feed and bleed operations for shutdowns, startups, and boron dilution over core life. Additionally, reactor coolant quality water from valve and equipment leakoffs, drains, and reliefs within the containment is collected in the reactor drain tank and scheduled for batch processing. Recoverable reactor coolant quality water from various equipment and valve leakoffs, reliefs, and drains outside containment is collected in the equipment drain tank and scheduled for batch processing.

Reactor coolant collected in the reactor and equipment drain tanks is periodically discharged by the reactor drain pumps through the reactor drain filter and pre-holdup ion exchanger. The diverted letdown flow, which has been previously passed through a purification filter and ion exchanger, also passes through the pre-holdup ion exchanger. The pre-holdup ion exchanger retains cesium, lithium, and other ionic radionuclides with high efficiency. All process flow then passes through the gas stripper where hydrogen and fission gases are removed with high efficiency, thus precluding the buildup of explosive gas mixtures in the holdup tank, and minimizing the release of radioactive fission product gases from other aerated vents or via any liquid discharge. The degassed liquid is automatically pumped from the gas stripper to the holdup tank. When a sufficient volume accumulates in the holdup tank, it is pumped by the holdup pump to the boric acid concentrator where the bottoms are concentrated.

The boric acid concentrator bottoms are continuously monitored for proper boron concentration. Normally, the concentrator bottoms are pumped directly to the boric acid storage tank. In the event that abnormal quantities of radionuclides are present, the bottoms are concentrated to 12 weight percent boric acid and are discharged to the solid waste management system (SWMS).

The concentrator distillate passes through a boric acid condensate ion exchanger, where boric acid carryover is removed. The distillate is collected in the reactor makeup water tank for reuse in the plant. If recycle is not desired, the condensate is diverted to the liquid waste management system (LWMS).

d. Auxiliary Pressurizer Spray

Auxiliary pressurizer spray flow is provided by the charging pumps for use during the latter stages of cooldown when the reactor coolant pumps are secured due to insufficient net positive suction head (NPSH).

e. Reactor Coolant Pump Seal Injection and Controlled Bleedoff

Reactor coolant pump seal injection water is supplied by the chemical and volume control system to cool RCP seal cavities and to provide a flushing flow to minimize deposition of radioactive crud in any seal cavity.

During normal operation, the flow to the reactor coolant pumps is supplied by the charging pumps. The seal injection flow is diverted from the charging line just downstream of the charging pumps. The seal injection flow is sent to a heat exchanger supplied with steam, where a temperature of approximately 120°F is maintained. The seal injection flow then mixes with CCW fluid as it passes through the pump.

The controlled bleedoff flow returns to the chemical and volume control system volume control tank. If controlled bleedoff increases due to seal system wear or failure, the seal water supply can be automatically increased to maintain the desired in-leakage flow. High controlled bleed-off flow is limited by an orifice in the control bleedoff discharge. When bleedoff reaches the highest acceptable value, the seals are considered failed, and the reactor coolant pump must be secured. The seal water supply and bleedoff for a given pump can be secured without affecting the other pumps.

A diverse and highly reliable dedicated seal injection system (DSIS) is provided to increase the reliability of the reactor coolant pump seals by further reducing the probability of seal failure; in particular, to ensure seal integrity during Station Blackout (SBO).

The DSIS is a diverse design with respect to normal seal cooling. It features a small capacity positive displacement pump which is placed in parallel with the chemical and volume control system centrifugal charging pumps. This positive displacement pump is provided with both suction stabilizers and discharge dampeners and uses the same CVCS piping as for normal seal injection. During normal operation, the DSIS does not operate. The DSIS would be operated during off-normal plant conditions involving losses of all other reactor coolant pump seal cooling methods. The pump is supplied with 1E power can also be powered by the alternate AC (AAC) power source.

f. Removal of Noble Gases

When continuous degasification of the RCS is desired, the letdown flow is diverted from the inlet line of the volume control tank to the gas stripper, bypassing the pre-holdup ion exchanger. The letdown flow is processed in the gas stripper and is then returned to the volume control tank via the normal spray nozzle.

The hydrogenated water is degassed at the gas stripper so that explosive gas concentrations do not occur in subsequent process equipment. Stripped gases are piped to a hydrogenated gas system in the gaseous waste management system where they are processed and later released under controlled conditions.

g. Shutdown Cooling System Flow Purification

When the shutdown cooling system is operational, a flow path through the CVCS can be established for purification. This is accomplished by diverting a portion of the flow from the shutdown cooling heat exchanger to the letdown line upstream of the letdown heat exchanger. The flow then passes through the purification filter, purification ion exchanger, and letdown strainer, and is returned to the suction of the shutdown cooling pumps.

h. Purification of the IRWST

Sufficient connections exist between the CVCS and the in-containment refueling water storage tank (IRWST) to allow for purification, inventory adjustments, and boron adjustments of the contents of that tank.

2. Process Sampling System

The process sampling system is designed to collect and deliver representative samples of liquids and gases in various process systems to sample stations for chemical and radiological analysis. The system permits sampling during reactor operation, cooldown, and post-accident modes without requiring access to the containment. Remote samples can be taken of fluids in high radiation areas without requiring access to these areas.

The process sampling system includes sampling lines, heat exchangers, sample vessels, sample sinks or racks, analysis equipment, and instrumentation. Sampling points are selected to provide the required chemical and radiological information while keeping the system simple for reliability and ease of maintenance. Figures II-E-2 (Sheets 1 and 2) show a flow diagram for the process sampling system, including primary, secondary, gas, and post-accident sampling. Sample locations, types of samples, and analysis requirements are shown. Chemical and radiochemical analyses are performed to determine boron concentration, fission and corrosion product activity, crud concentration, dissolved gas and corrosion product concentrations, chloride concentration, coolant pH, conductivity of the reactor coolant, and noncondensible gas concentration in the pressurizer. The results of the analyses are used to regulate the boron concentration, monitor the fuel cladding integrity, evaluate ion exchanger and filter performance, specify chemical additions to the various systems, and maintain the proper hydrogen concentration in the reactor coolant systems.

A data management and surveillance system gives daily evaluation of plant chemistry, and tracks and plots chemistry trends.

The system configuration is such that, under normal operation, samples are transported from a number of different locations to central sample racks, where samples are cooled and depressurized as necessary. For samples containing dissolved gases, sample bombs are located at the exit of each sample cooler to facilitate the collection of samples. In addition, the system configuration is such that, under post-accident conditions, samples of containment atmosphere and containment liquids are transported to an accessible location for grab sampling.

3. Spent Fuel Pool Cooling and Cleanup System

The safety-related spent fuel pool cooling system consists of two independent cooling trains. The system is located in the Nuclear Annex, a Seismic Category I building, which provides protection from effects of natural phenomena and missiles. The spent fuel pool cooling system (piping, pumps, valves, and heat exchangers) is safety-related, Quality Group C. The spent fuel pool (SFP) receives normal borated water makeup from a water source which is Seismic Category I, Quality Group C. In addition, the backup to the normal makeup system consists of piping and/or hoses from an alternate water source. Non-safety-related, Seismic Category I sources provide normal nonborated demineralized water backup and boric acid backup to the spent fuel pool.

Each cooling train is designed to service the spent fuel pool, with designed spent fuel assembly loading, and to maintain the bulk fluid temperature of the spent fuel pools below 140.0°F during normal operation.

The spent fuel pool cooling system removes decay heat from fuel stored in the spent fuel pool. Heat is transferred from the spent fuel pool cooling system, through the heat exchanger to the component cooling water system.

When either cooling train is in operation, water flows from the spent fuel pool to the spent fuel pool cooling pump suction, is pumped through the tube side of the heat exchanger, and is returned to the pool. The suction line is provided with a strainer and is located at an elevation well above the normal fuel level, while the return line contains an antisiphon device to prevent gravity drainage of the pool.

Each of the two spent fuel pool cleanup trains consists of a pump, a strainer, a demineralizer and a filter to maintain spent fuel pool water clarity and purity. Transfer canal water may also be circulated through the same demineralizer and filter. This purification loop is sufficient for removing fission products and other contaminants which may be introduced if a leaking fuel assembly is transferred to the spent fuel pool.

The demineralizer and filter of either cleanup train may be used to clean and purify the refueling water while spent fuel pool heat removal operations proceed. Refueling water can be pumped from either the in-containment refueling water storage tank (IRWST) or the RCS through a filter and demineralizer, and discharged to the reactor vessel or the IRWST. To assist in maintaining spent fuel pool water clarity, the SFP surface is cleaned by a skimmer.

The pool water may be separated from the water in the transfer canal by a gate. The gate is installed so that the transfer canal may be drained to allow maintenance of the fuel transfer equipment.

Two full-size fuel pool cooling pumps and two full-size fuel pool coolers will be provided to ensure 100-percent redundant cooling capacity. This portion of the system is Seismic Category I and Safety Class 3. The Seismic Category I cooling portion of the fuel pool cooling and cleanup system is independent of the nonseismic purification portion. Failure of the purification portion in an earthquake will not affect the operation of the cooling trains.

For all normal plant operations and normal spent fuel pool heat load conditions, the maximum spent fuel pool bulk water temperature is 120.0°F. Given a single active failure, the maximum temperatures are 140.0°F. Design heat loads are evaluated utilizing ANSI/ANS 5.1 correlations.

Following a design basis accident with loss of power, the reactor plant component cooling water system is not available to cool the spent fuel pool coolers. Power from the emergency diesel generators is not immediately available due to loading considerations. A loss of cooling evaluation has be performed to which shows that the spent fuel pool temperature reaches a temperature 200.0°F in approximately 12.5 hours. This provides sufficient time to manually initiate pool cooling. Once the cooling is restarted, the temperature decreases to 150.0°F in less than 6 hours. Redundant safety grade fuel pool temperature indication is provided in the Main Control Room. Redundant safety Class 3 level instruments are located in the fuel pool which indicate both locally and in the Main Control Room.

An emergency make-up water provision is provided via a capped connection outside the fuel building with a line that runs into the building and terminates pointing down at the spent fuel pool. This enables cooling to be provided via this hose connection from a fire engine or other fire equipment, without requiring access to the fuel building.

4. Plant Control (I&C)

Advanced Control Complex

Nuplex 80+ breaks down the historical boundaries between the NSSS and BOP through a total integration of plant-wide I&C systems. The Advanced Control Complex (Figure II-E-3) is comprised of the Main Control Room, the Technical Support Center, the Remote Shutdown Control Room, and the I&C Equipment Rooms (which house control, protection and monitoring systems plant-wide). Each of these are described in the following paragraphs.

a. Main Control Room

The Control Room (shown in Figures II-E-4 and II-E-5) consists of the Controlling Workspace (where the plant's control panels are located) and offices for the plant operating staff. The design accommodates six people continuously. Due to advanced automation and man-machine interface features, however, only one operator is required at the controls during the hot standby and power operating modes. Within the controlling workspace, the Master Control Console provides the instrumentation and controls for the primary steam and electrical generation systems, as well as plant-wide-monitoring capabilities. The safety and auxiliary consoles provide the instrumentation and controls for the Engineered Safety Features and the Auxiliary Systems, respectively. The Control Room Supervisor's Console provides plant-wide communications and monitoring functions; the desk area provides working space for control room personnel as well as a natural boundary for interaction with other plant staff.

b. Technical Support Center (TSC)

The TSC located above the operators' offices, in Figure II-E-5 has been designed for emergency, as well as normal, operations. It includes communications and plant-wide monitoring equipment. Its direct visibility into the Main Control Room allows plant staff, emergency officials and plant visitors to view control room activities without interference with plant operations. The TSC is entered independently and is secured from the Main Control Room to allow easier access for all users.

c. Remote Shutdown Control Room (RSCR)

The RSCR includes a single sit-down/stand-up control console which accommodates one or two operators. This console provides dedicated hot shutdown control and access to all plant controls through touch sensitive video displays. These displays facilitate maintenance operations in the Main Control Room during power operation, as well as achieving cold shutdown during events involving evacuation of the Main Control Room.

d. Equipment Rooms

To enhance fire, flood and sabotage protection, Nuplex 80+ includes four equipment rooms for redundant Class 1E systems, two equipment rooms for non-safety systems, and one computer room. Fiber optic communication cables between each of the equipment rooms, and to the Main Control Room, ensure electrical independence of all areas.

5. Man-Machine Interface

The man-machine interface was developed by a multi-disciplined design team including experienced reactor operators and specialists in Human Factors Engineering. Redundancy and diversity in all information processing and display ensures the correctness of information presentation and allows continued plant operation with equipment failures (e.g., loss of CRTs). The integration of information from the Safety Parameter Display System and the Post Accident Monitoring Instrumentation (PAMI) into normal operating displays allows the same display to be used during all plant conditions. The equipment and techniques for processing information to the operator are described in the following paragraphs.

a. Alarm Reduction Processing

Alarms are based on validated signal inputs with logic and setpoints that account for plant and equipment operating modes. Alarms are presented in functional groups via a minimum set of spatially dedicated alarm tiles. In addition, CRT displays present alarms within the context of dynamic color graphic plant mimic displays and through various alarm list formats. To correctly direct operator attention, four levels of alarm prioritization are employed with predefined as well as "operator established" alarms. Momentary audible cues with periodic reminder tones require no silencing to reduce alarm annoyance. Individual alarm acknowledgement with global "stop flash" and "resume" features ensure that all alarms are recognized without operator task overload. Alarm acknowledgements are common to both the tile and CRT display media, and in either case provide direct access to alarm details and supporting displays.

b. Display Processing

Dedicated (discrete indicator) and selectable (CRT) displays reduce the amount of data to be presented by using cross-channel signal validation and correlation of related parameters. CRT graphic presentations are arranged in a hierarchical format with each level selectable through touch sensitive access. This enhances rapid information retrieval.

c. Advanced Operator Aids

Information and alarm data is prioritized using algorithms that monitor critical functions and success paths for both safety and power production. Post-trip critical function monitoring is operator selectable for uncomplicated trips, or functional or optimal accident recovery. These features ensure that the alarms and displays directly support normal, alarm response, and emergency procedures. Another operator aid directly supports periodic surveillance testing by monitoring test activities to ensure correct test procedures and restoration of equipment to pre-test conditions.

d. Man-Machine Interface (MMI) Equipment

A combination of dedicated and selectable information and controls is used to optimize the Man-Machine Interface. The apex of the information hierarchy is the Integrated Process Status Overview (IPSO) which resides at the front of the control room, as shown in Figure II-E-4. The IPSO is a 4.5' x 6' wall mounted color graphic display. It provides a fixed plant-wide overview of functions and success paths critical to safety and power production. The IPSO uses simple graphic symbols to allow quick comprehension of plant conditions from all locations in the Control Room and by the entire operating staff.

Lower in the hierarchy of Nuplex 80+ information are the dedicated system level, subgroup level and component level displays that are located at each control panel. As an example, the Reactor Coolant System panel is shown in Figure II-E-6. Specific features are described below.

(1) Alarm Tiles

Electro-luminescent displays duplicate conventional alarm tiles to facilitate pattern recognition through spatial dedication. High level tile descriptions for all Priority 1 and 2 alarm conditions provide touch sensitive acknowledgement and automatic display of alarm details.

(2) Discrete Indicators

Spatially dedicated indicators on the vertical section of each panel replace conventional analog meters with continuous digital, bar graph and trend displays for key system-level parameters. Displays use data reduction techniques such as cross-channel signal validation, auto-range selection and like-parameter correlation. Touch sensitive screens allow access to all parameter details and related (but less frequently monitored) data.

(3) Process Controllers

Spatially dedicated controls on the desk section of each panel provide continuous display of functionally related controlled parameters. Touch sensitive screens allow access to controls for mode selection, selection of controlling sensors, setpoint change and manual output for each control loop.

(4) Component Controls

Shape-coded control pushbuttons with backlit indicators provide normal and discrepancy status for pumps, valves, heaters, fans, etc.

(5) Operators Modules

For I&C systems such as the Plant Protection System and the Power Control System, operator modules provide selectable displays for operating status, bypass conditions and auto test diagnostics.

Supporting all levels of dedicated display information are the CRT display pages. Touch sensitive CRTs are located at each control panel, in all offices and at the TSC. All display pages are accessible at any location to allow maximum transportability of plant information. Display pages support all levels of operator information needs from high level critical function monitoring to detailed component performance diagnostics.

6. Instrumentation and Control Systems

The standardization of components, while maintaining a level of hardware and software diversity among systems, to protect against common mode failures. Within systems, multiple redundant processors with automatic failure detection are used to distribute functions in a highly reliable fault tolerant configuration. This defense-in-depth approach ensures high availability and promotes regulatory acceptance of advanced computer technology.

Fiber optic data communications are used throughout Nuplex 80+ to maximize electrical independence of systems. These multiplexed interfaces along with geographically distributed processing and remote I/O multiplexing minimize plant cabling. The end result is that costs less than a conventional control complex; installation schedules are shortened; and, future system changes are accommodated, without pulling additional cables. The various systems that comprise Nuplex 80+ are shown in Figure II-E-3 and described in the following paragraphs.

a. Discrete Indication and Alarm Systems (DIAS)

The DIAS (Figure II-E-7) processes data for the alarm tile and discrete indicator displays as well as the process parameters on the IPSO display. The system consists of multiple processors in a functionally distributed architecture. DIAS is seismically qualified for PAMI display.

b. Data Processing System (DPS)

The DPS (Figure II-E-8) processes data for the CRT displays and for the IPSO display. It is diverse from DIAS to enhance information integrity and availability. It utilizes super mini-computers with internal parallel processors. Multiple distributed display processors provide user-friendly operator workstations. Optical disks provide long-term historical data storage and retrieval.

c. Plant Protection System (PPS)

The PPS (Figure II-E-9) provides integrated reactor trip and engineered safety features actuation in a geographically distributed four channel configuration. Within each channel, separate processors provide core protection calculations, bistable functions, and two-out-of four coincidence logic. The system automatically initiates pre-trip control actions for rapid power reduction without scram. Continuous automatic software testing eliminates most manual periodic surveillance tests.

d. Engineered Safety Features-Component Control System (ESF-CCS)

The ESF-CCS (Figure II-E-10) provides normal ESF actuation, and emergency sequencing controls for pumps, valves, fans, etc. in the plant's safety systems. The system consists of four independent trains, each with geographically distributed processors located in close proximity to the controlled equipment.

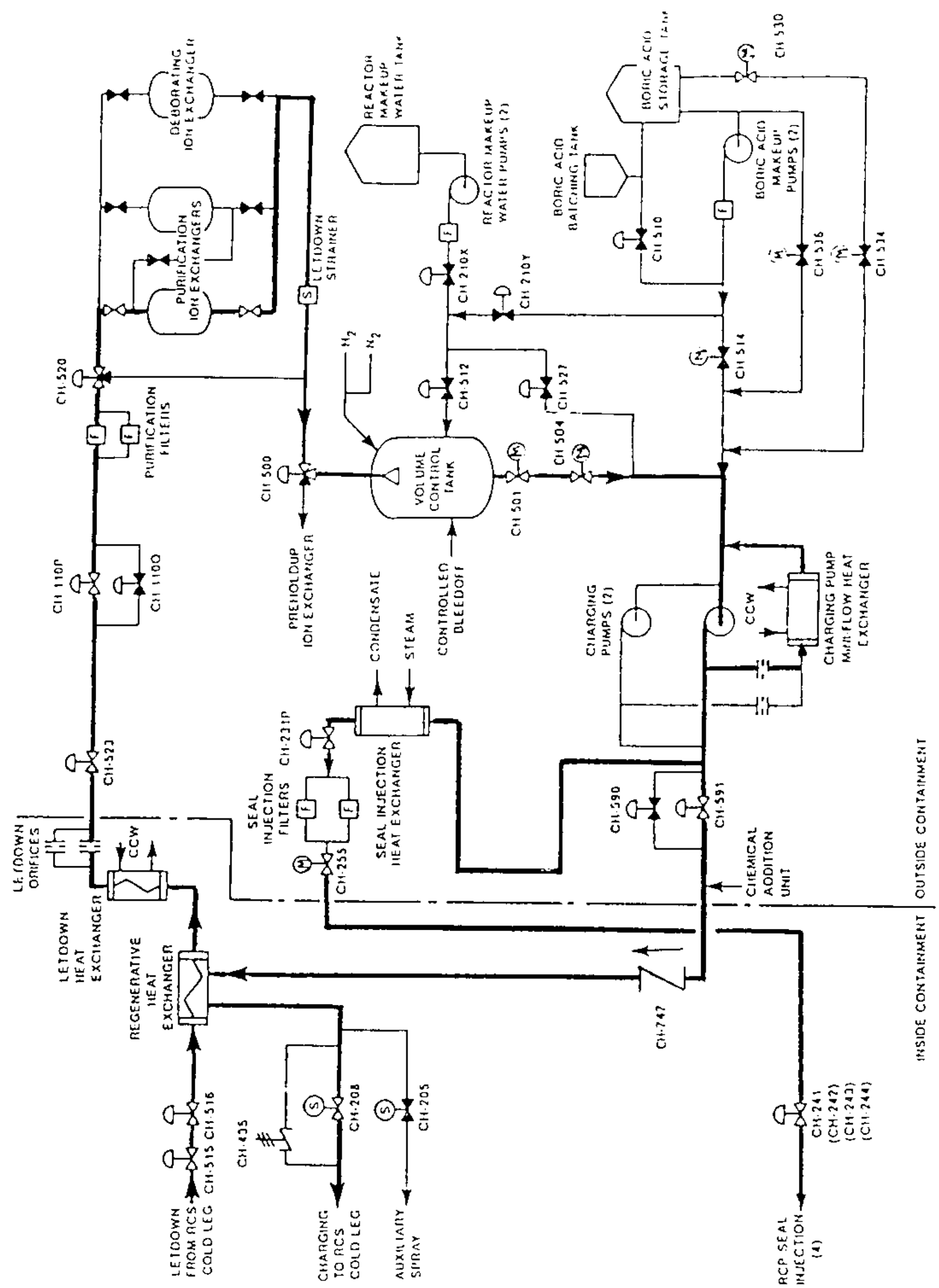
e. Process-Component Control System

The Process-CCS provides similar functions to the ESF-CCS but for non-safety plant systems. It is diverse from the ESF-CCS to ensure that normal and emergency plant systems are not subject to common mode failures.

f. Power Control System (PCS)

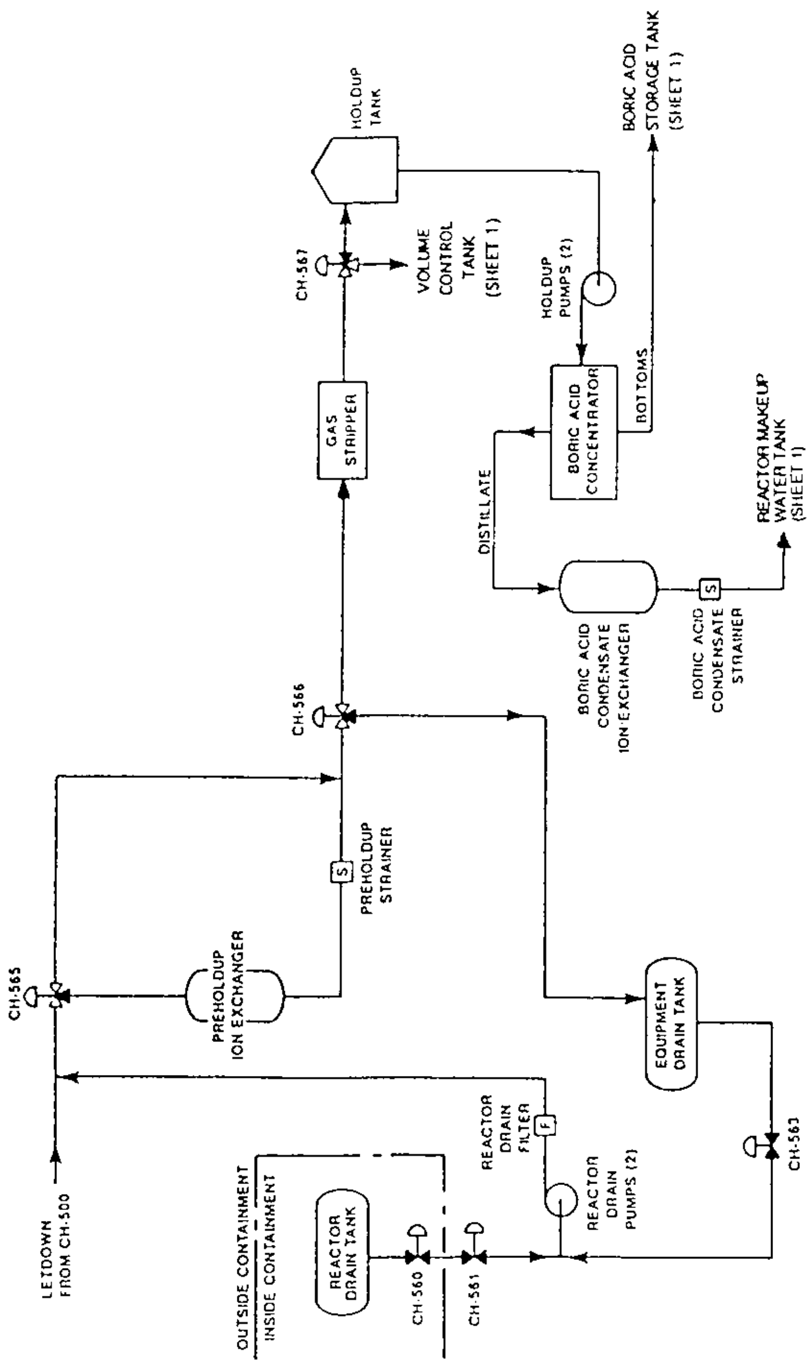
The PCS (Figure II-E-11) consists of several processors, each performing distinct functions related to plant power control. The Megawatt Demand Setter (MDS) screens load change demands from the operator or remote load dispatcher to ensure consistency with plant operating limits. The MDS also initiates automatic power reductions when plant conditions may result in unacceptable trip margins. The Control Element Drive Mechanism (CEDM) controller adjusts control rod position using automatic, closed-loop control logic. This controller also controls rapid reactor power cutback for pretrip conditions. The Reactor Regulating Controller adjusts reactor power and reactor coolant temperature to automatically follow turbine load changes. All three controllers share redundant internal and external communications networks.

ABB
ASEA BROWN BOVERI



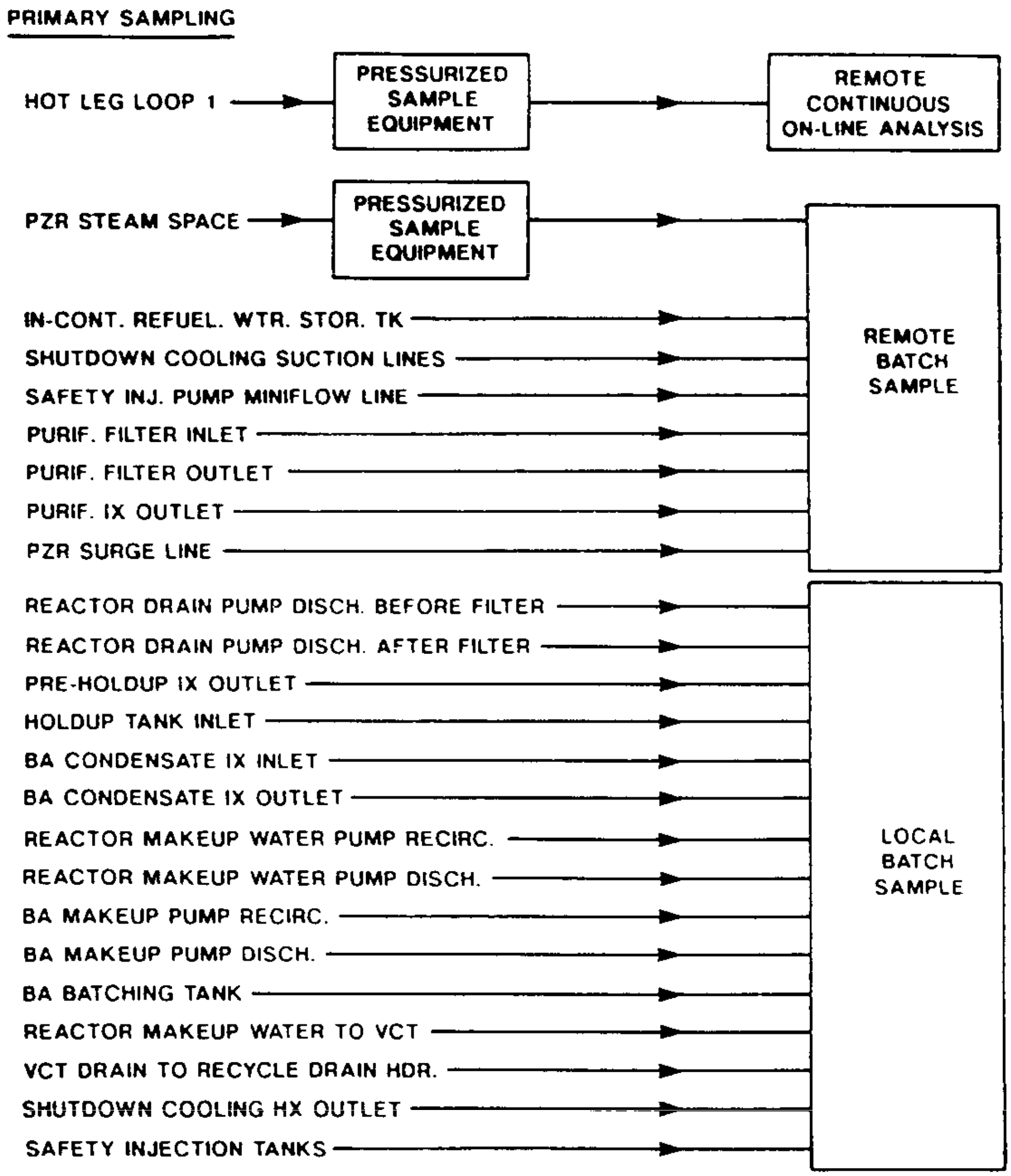
SYSTEM 80+™	CHEMICAL AND VOLUME CONTROL SYSTEM BLOCK DIAGRAM (SHEET 1 OF 2)	FIGURE II-E-1

ABB
ASEA BROWN BOVERI



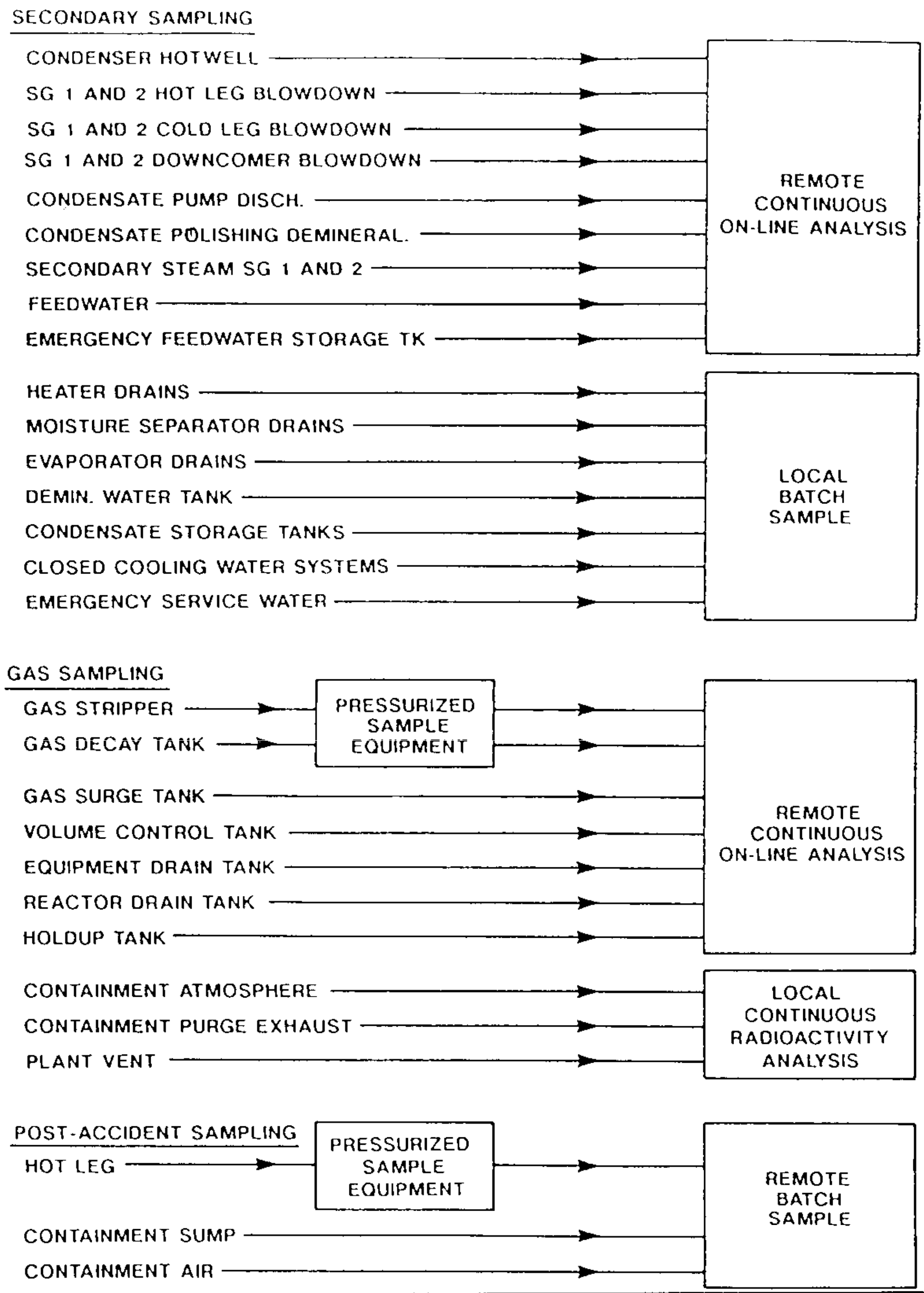
SYSTEM 80+™	CHEMICAL AND VOLUME CONTROL SYSTEM BLOCK DIAGRAM (SHEET 2 OF 2)	FIGURE II-E-1

ABB
ASEA BROWN BOVERI



SYSTEM 80+™	PROCESS SAMPLING SYSTEM FLOW DIAGRAM (SHEET 1 OF 2)	FIGURE II-E-2

ABB
ASEA BROWN BOVERI



SYSTEM 80+™	PROCESS SAMPLING SYSTEM FLOW DIAGRAM (SHEET 2 OF 2)	FIGURE II-E-2

SYSTEM 80+™

NUPLEX 80+ ADVANCED CONTROL COMPLEX

FIGURE II-E-3

ABB
ASEA BROWN BOVERI

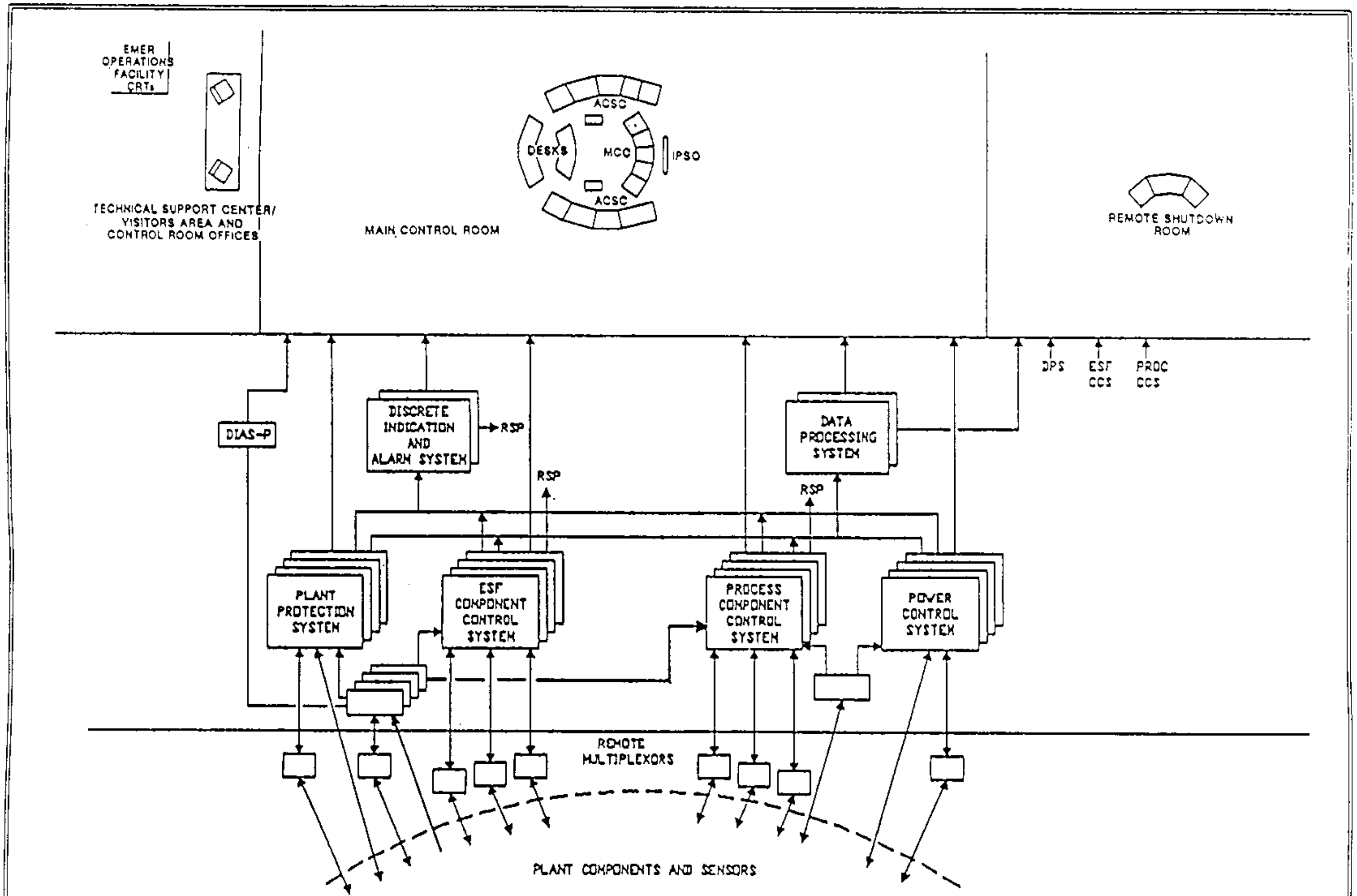
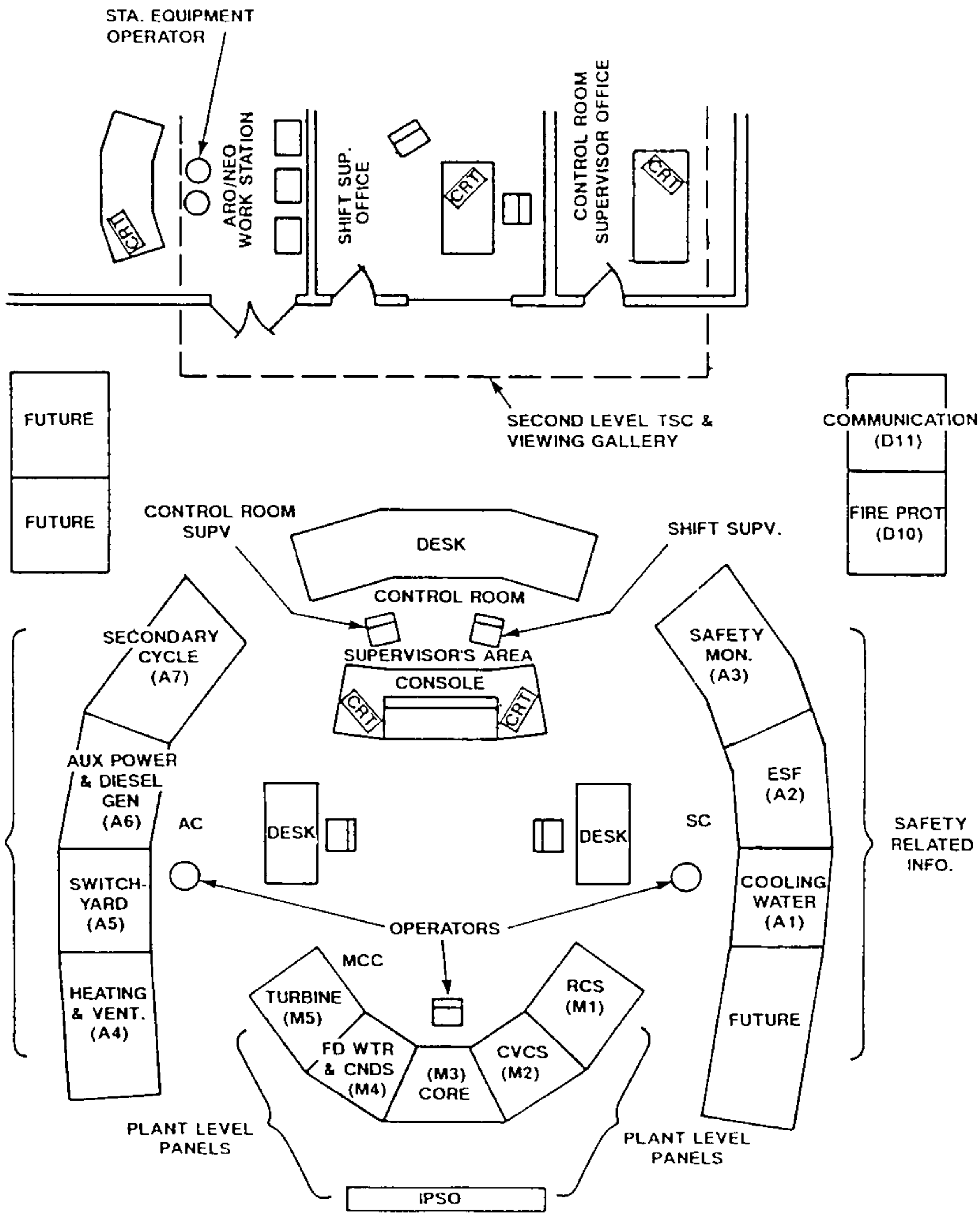
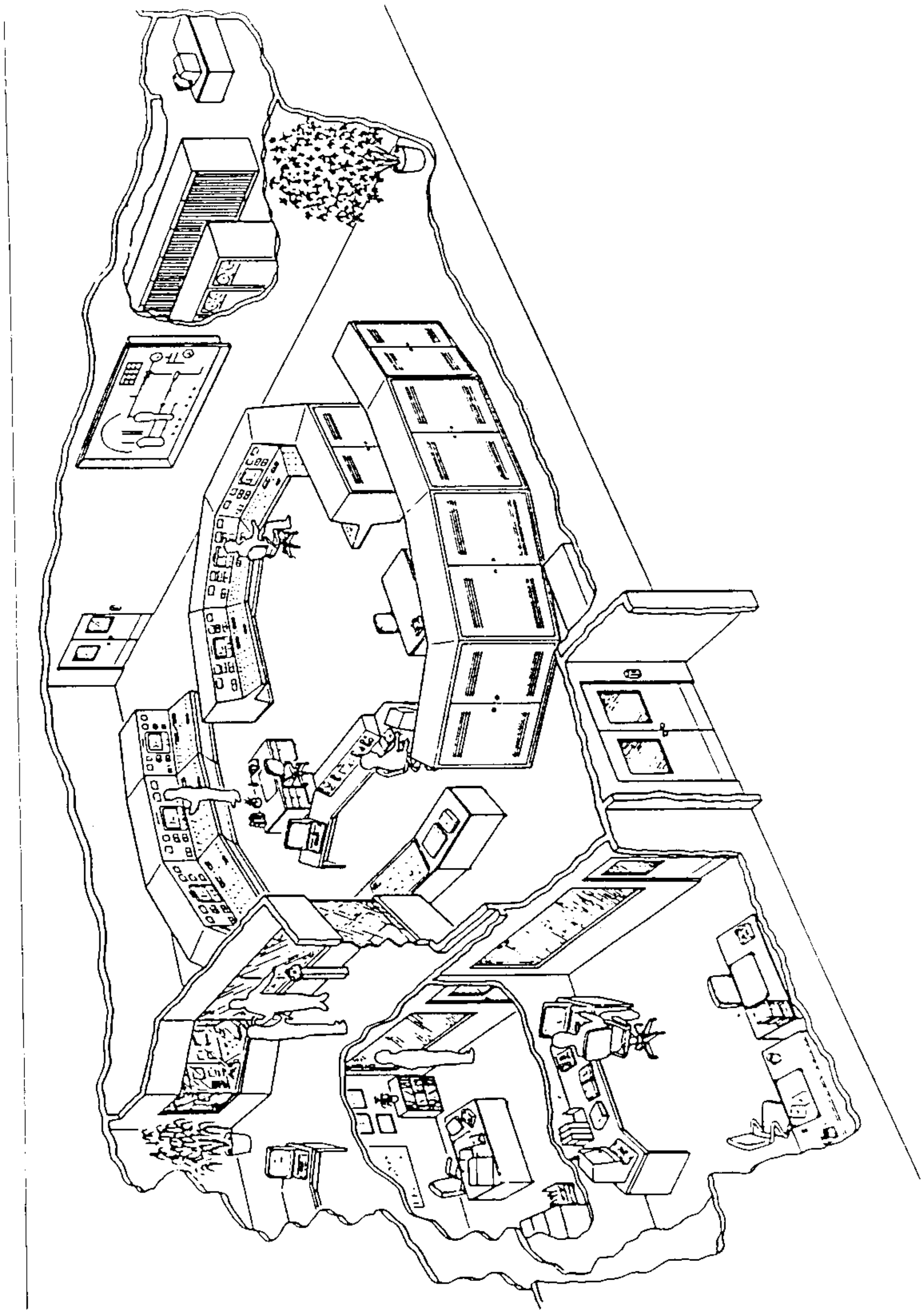


ABB
ASEA BROWN BOVERI



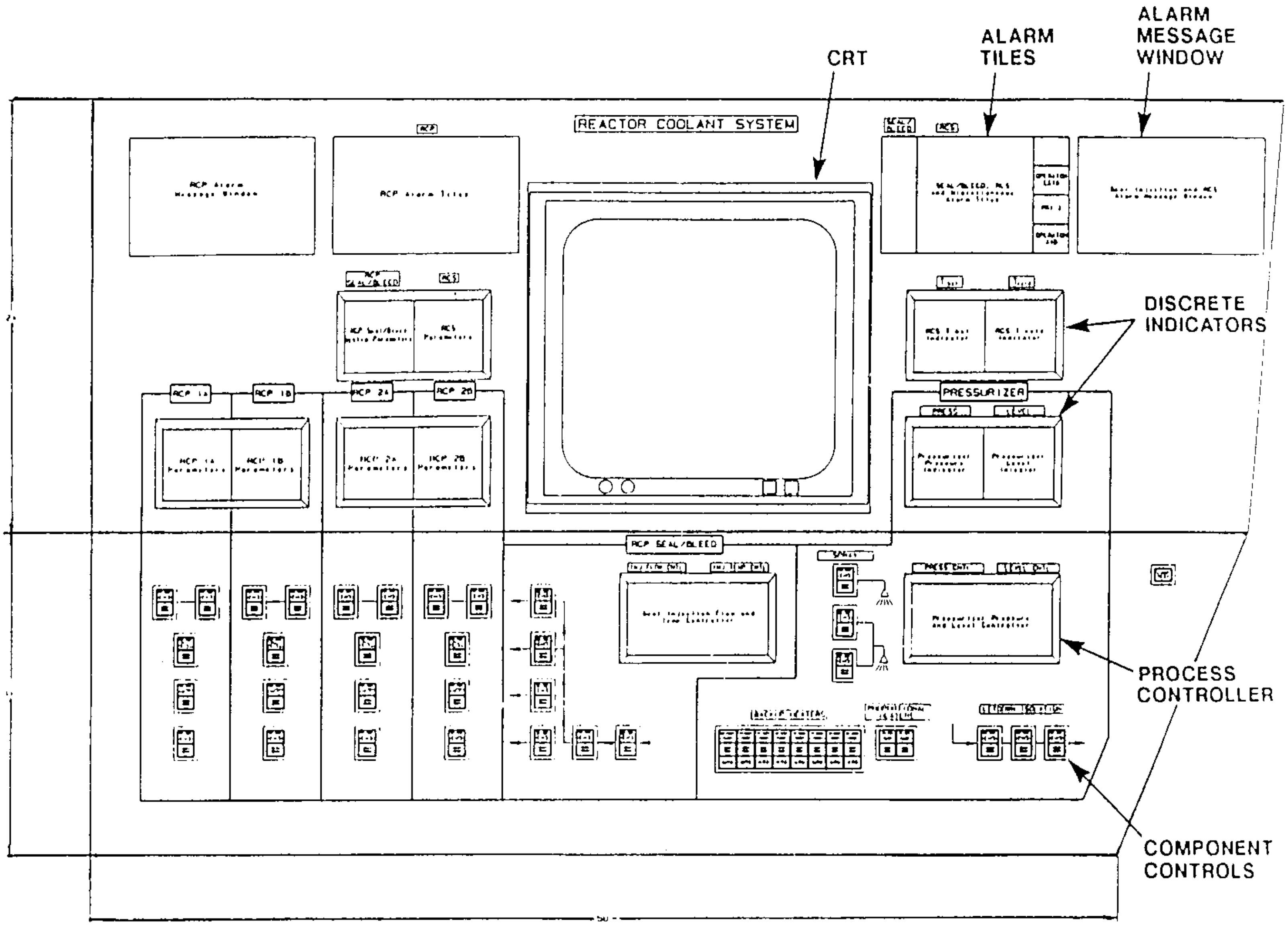
SYSTEM 80+™	THE NUPLEX 80+ CONTROL ROOM SUPPORTS A VARIETY OF STAFF ARRANGEMENTS	FIGURE II-E-4

ABB
ASEA BROWN BOVERI



SYSTEM 80+™	TECHNICAL SUPPORT CENTER	FIGURE II-E-5

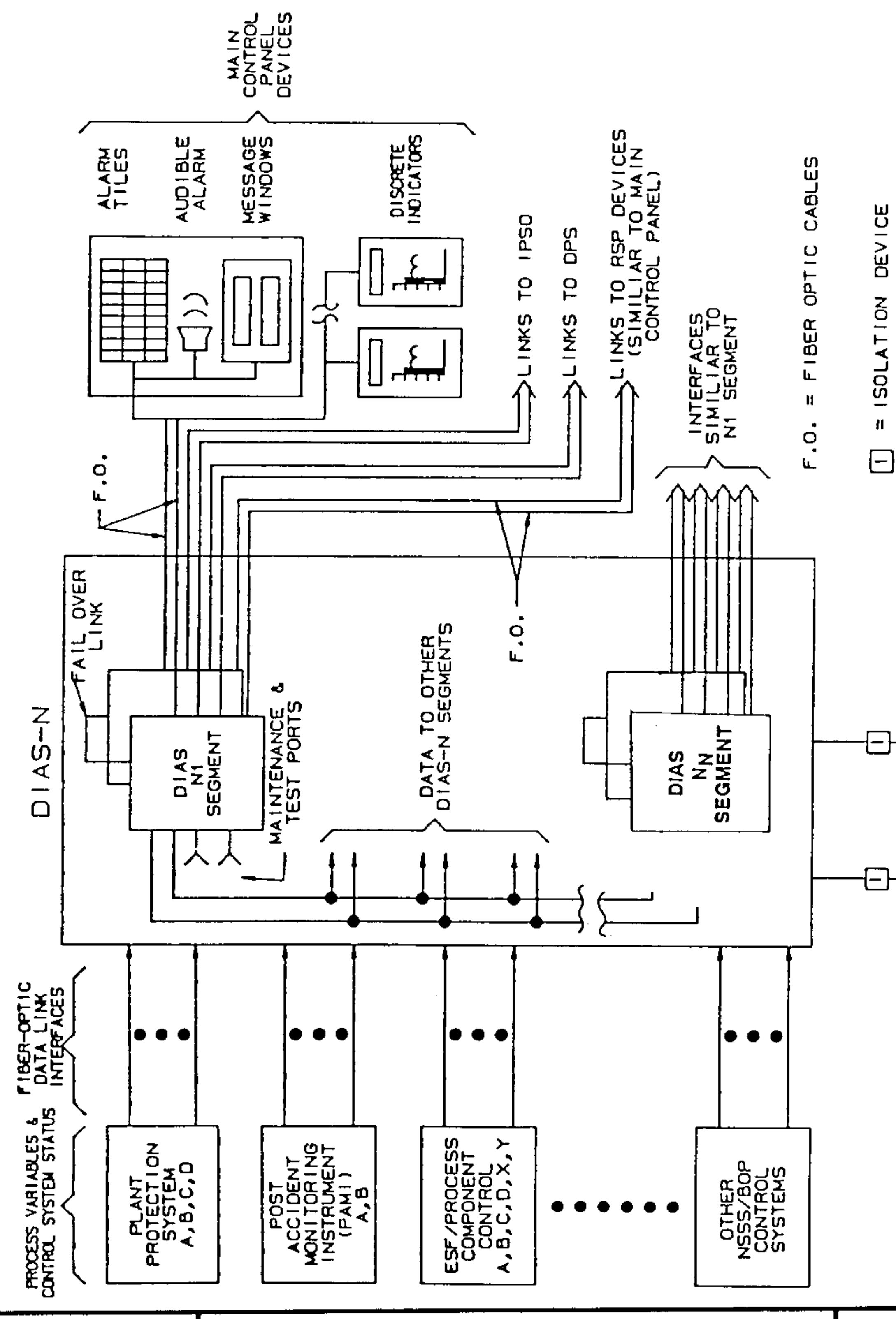
SYSTEM 80+™



TYPICAL NUPLEX 80+ CONTROL PANEL

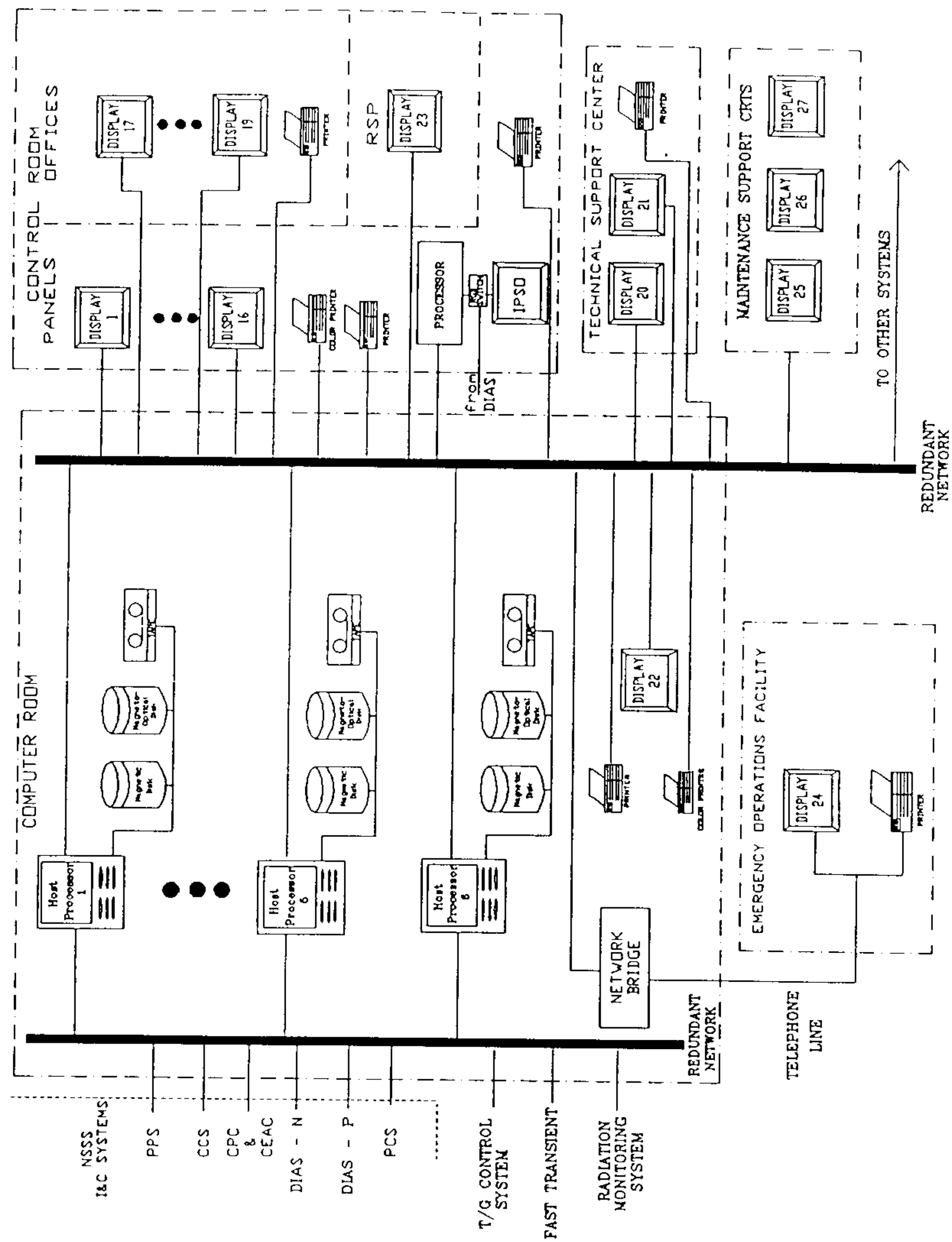
FIGURE II-E-6

ASEA BROWN BOVERI

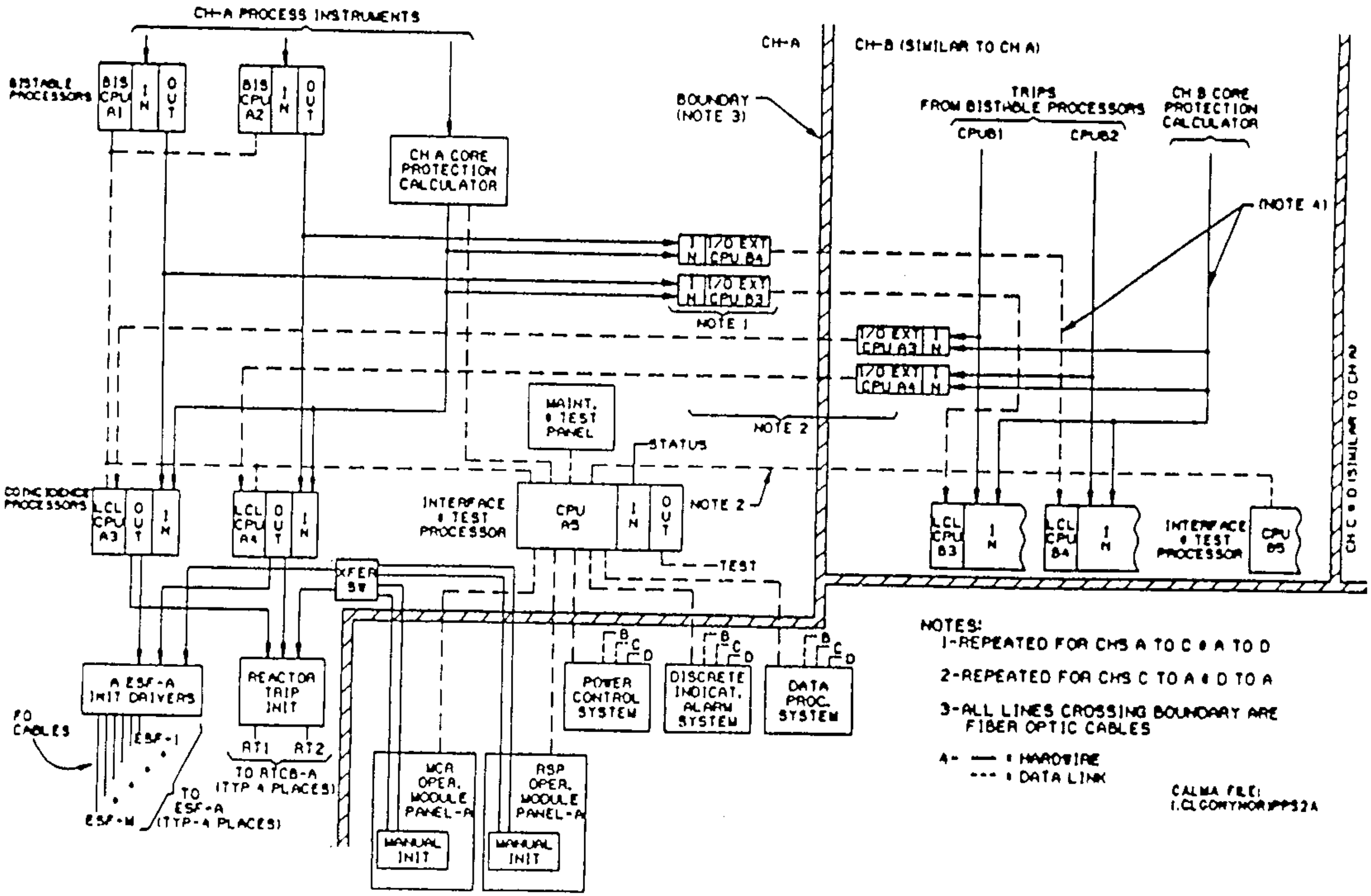


SYSTEM 80+™	DIAS-N SEGMENT BLOCK DIAGRAM	FIGURE II-E-7

ASEA BROWN BOVERI



SYSTEM 80+™	DATA PROCESSING SYSTEM CONFIGURATION	FIGURE II-E-8

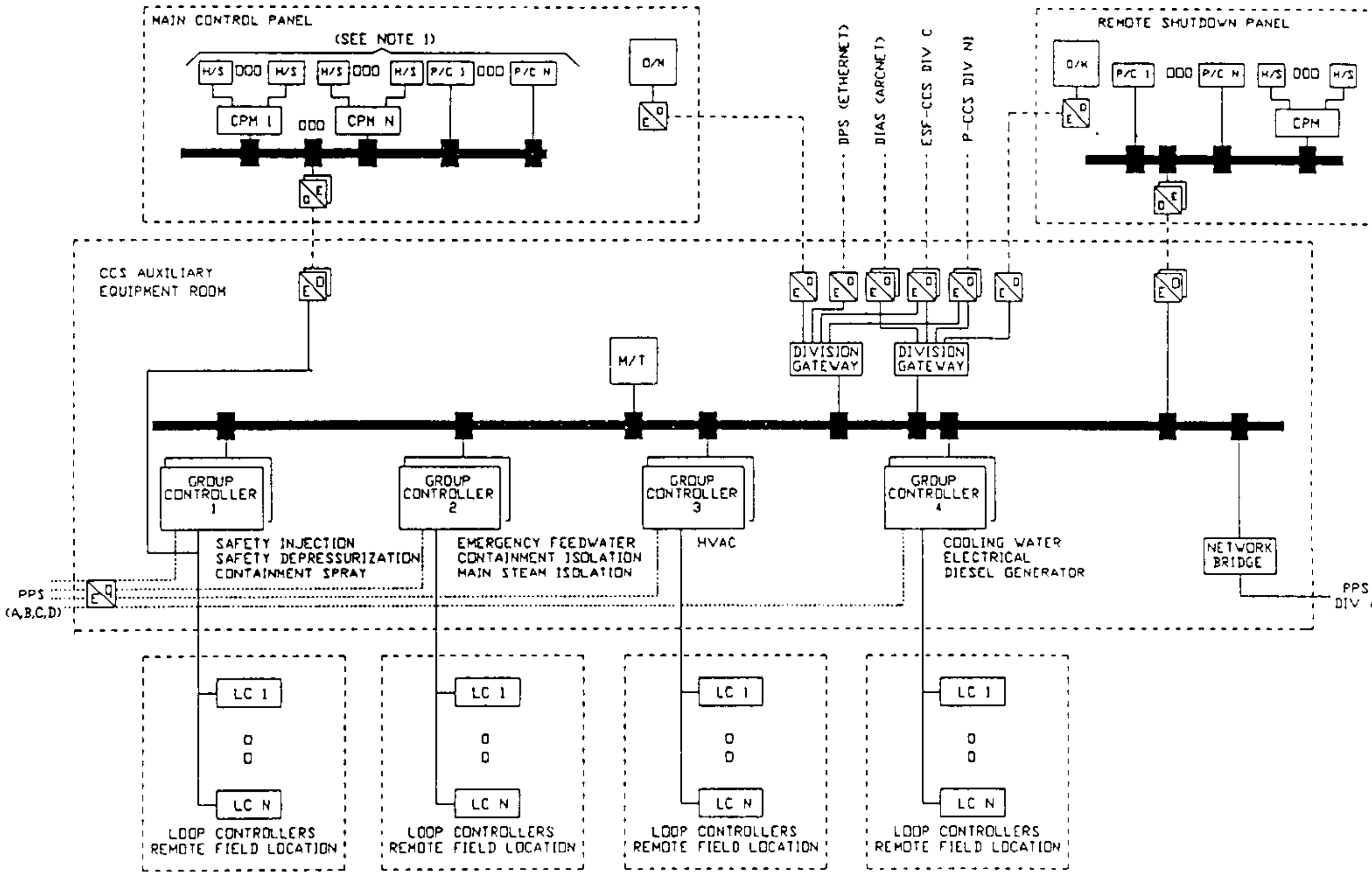


System 80+™

NUPLEX 80+™ PPS CH-A
SIMPLIFIED BLOCK DIAGRAM

FIGURE II-E-9

ABB
ASEA BROWN BOVERI



H/S	HAND SWITCH
P/C	PROCESS CONTROLLER
O/M	OPERATOR'S MODULE
DPS	DATA PROCESSING SYSTEM
DIAS	DISCRETE INDICATION AND ALARM SYSTEM
CPM	CONTROL PANEL MULTIPLEXER
M/T	MAINTENANCE AND TEST PANEL
SR	SAFETY RELATED (ESF-CCS)
PPS	PLANT PROTECTION SYSTEM
PCS	POWER CONTROL SYSTEM

FIBER-OPTIC MODEM
- - - - OPTICAL DATA LINK
—— WIRE DATA LINK
.... DISCRETE FIBER-OPTIC

NOTE: 1. TYPICAL FOR CONTROL PANEL DEVICES IN EACH GROUP.
2. AVERAGE 4 COMPONENTS PER LOOP CONTROLLER
3. TOTAL NUMBER OF LC/DIV = $\frac{\text{NUMBER OF COMPONENTS}}{\text{4 COMPONENTS/LOOP}}$

NUPLEX 80+ ESF-CCS DIVISION A

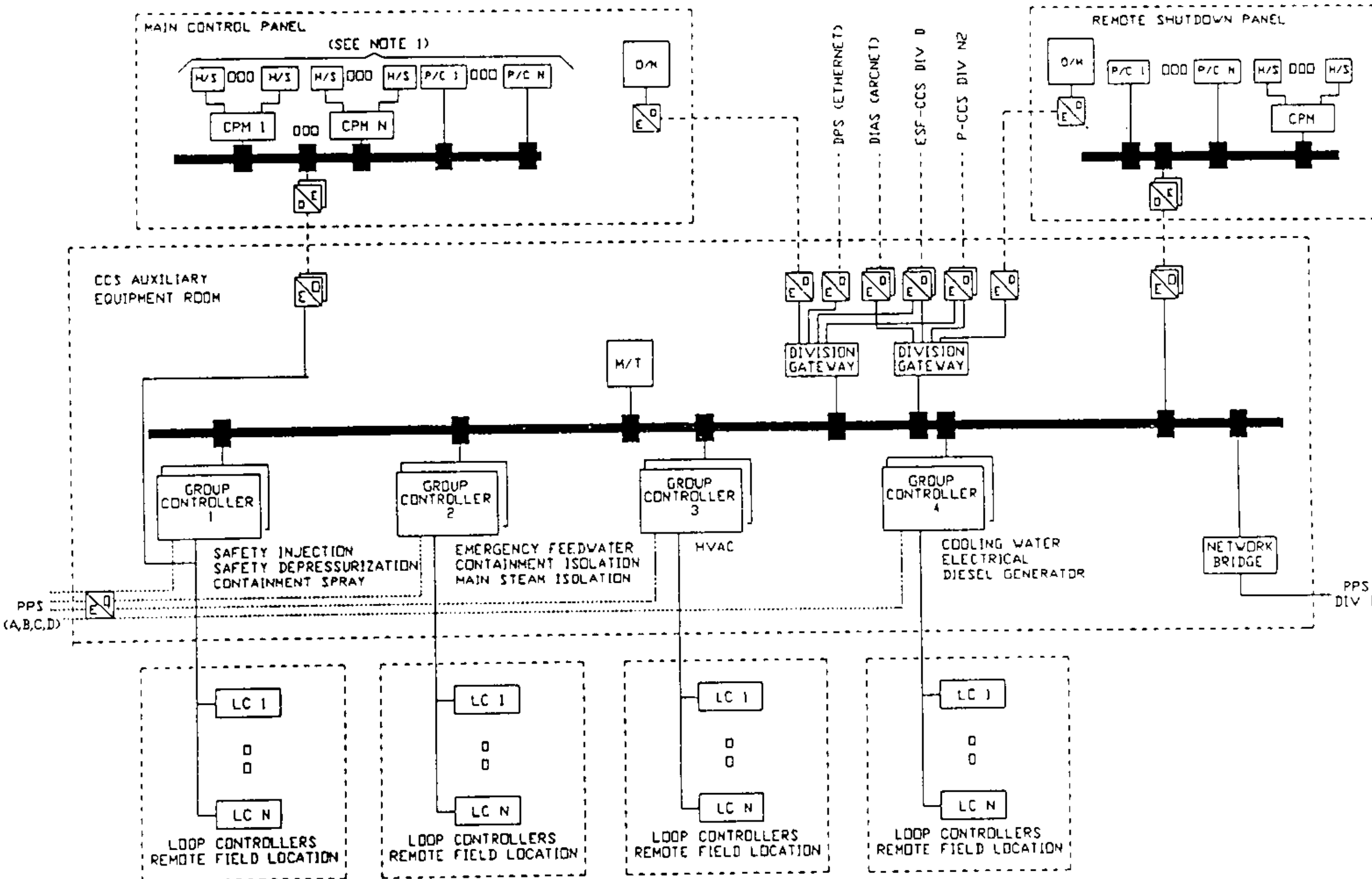
ASEA BROWN BOVERI

SYSTEM 80+™

SYSTEM 80+™ ESF-CCS SIZING CONFIGURATION
(SHEET 1 OF 4)

FIGURE II-E-10

ABB
ASEA BROWN BOVERI



H/S HAND SWITCH
P/C PROCESS CONTROLLER
O/M OPERATOR'S MODULE
DPS DATA PROCESSING SYSTEM
DIAS DISCRETE INDICATION AND ALARM SYSTEM
CPM CONTROL PANEL MULTIPLEXER
M/T MAINTENANCE AND TEST PANEL
SR SAFETY RELATED (ESF-CCS)
PPS PLANT PROTECTION SYSTEM
PCS POWER CONTROL SYSTEM

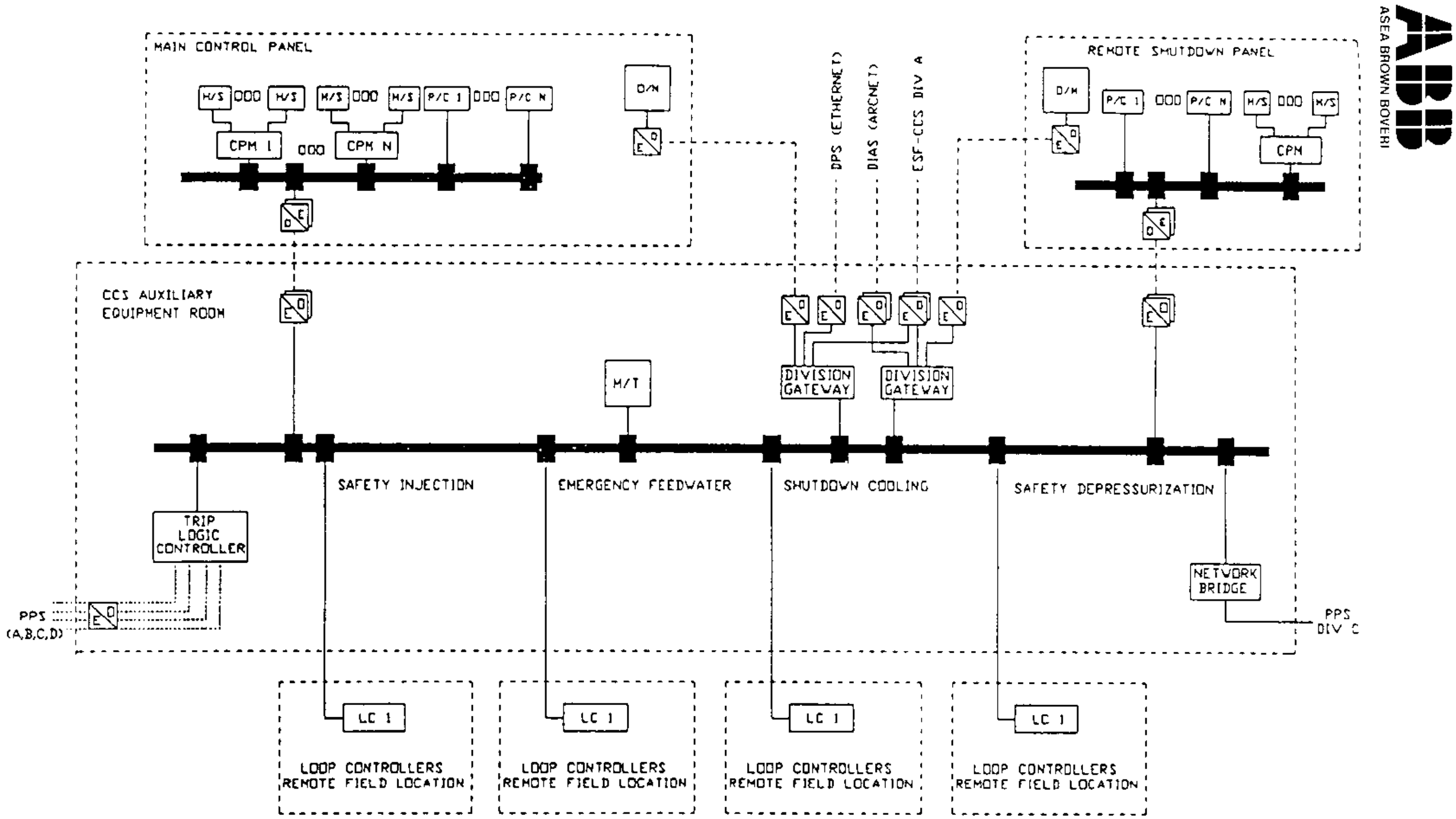
FIBER-OPTIC MODEM
---- OPTICAL DATA LINK
—— WIRE DATA LINK
...... DISCRETE FIBER-OPTIC

NOTE: 1. TYPICAL FOR CONTROL PANEL DEVICES IN EACH GROUP.
2. AVERAGE 4 COMPONENTS PER LOOP CONTROLLER
3. TOTAL NUMBER OF LC/DIV = $\frac{\text{NUMBER OF COMPONENTS}}{\text{4 COMPONENTS/LOOP}}$

DATE 1/19/93
FILE T1ACC1

NUPLEX 80+ ESF-CCS DIVISION B

SYSTEM 80+™	SYSTEM 80+™ ESF-CCS SIZING CONFIGURATION (SHEET 2 OF 4)	FIGURE II-E-10



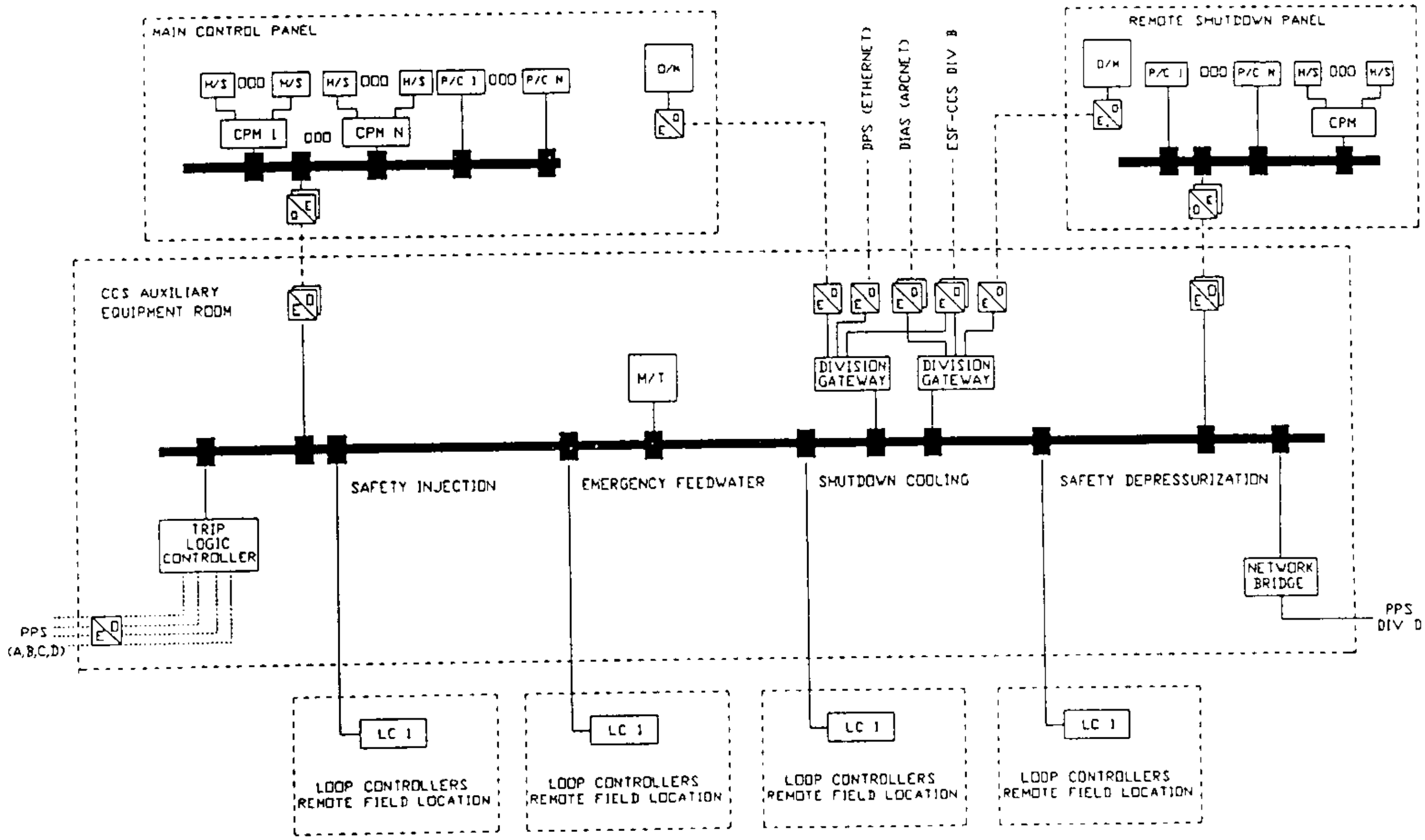
H/S HAND SWITCH
P/C PROCESS CONTROLLER
O/M OPERATOR'S MODULE
DPS DATA PROCESSING SYSTEM
DIAS DISCRETE INDICATION AND ALARM SYSTEM
CPM CONTROL PANEL MULTIPLEXER
M/T MAINTENANCE AND TEST PANEL
SR SAFETY RELATED (ESF-CCS)
PPS PLANT PROTECTION SYSTEM
PCS POWER CONTROL SYSTEM

FIBER-OPTIC MODEM
- - - - OPTICAL DATA LINK
——— WIRE DATA LINK
.... DISCRETE FIBER-OPTIC

NOTE 1. AVERAGE 4 COMPONENTS PER LOOP CONTROLLER
2. TOTAL NUMBER OF LC/DIV = $\frac{\text{NUMBER OF COMPONENTS}}{\text{4 COMPONENTS/LOOP}}$

NUPLEX 80+ ESF-CCS DIVISION C

SYSTEM 80+™	SYSTEM 80+™ ESF-CCS SIZING CONFIGURATION (SHEET 3 OF 4)	FIGURE II-E-10



H/S HAND SWITCH
P/C PROCESS CONTROLLER
O/M OPERATOR'S MODULE
DPS DATA PROCESSING SYSTEM
DIAS DISCRETE INDICATION AND ALARM SYSTEM
CPM CONTROL PANEL MULTIPLEXER
M/T MAINTENANCE AND TEST PANEL
SR SAFETY RELATED (ESF-CCS)
PPS PLANT PROTECTION SYSTEM
PCS POWER CONTROL SYSTEM

FIBER-OPTIC MODEM
OPTICAL DATA LINK
WIRE DATA LINK
DISCRETE FIBER-OPTIC

NOTE 1. AVERAGE 4 COMPONENTS PER LOOP CONTROLLER
2. TOTAL NUMBER OF LC/DIV = $\frac{\text{NUMBER OF COMPONENTS}}{\text{4 COMPONENTS/LOOP}}$

NUPLEX 80+ ESF-CCS DIVISION D

SYSTEM 80+™

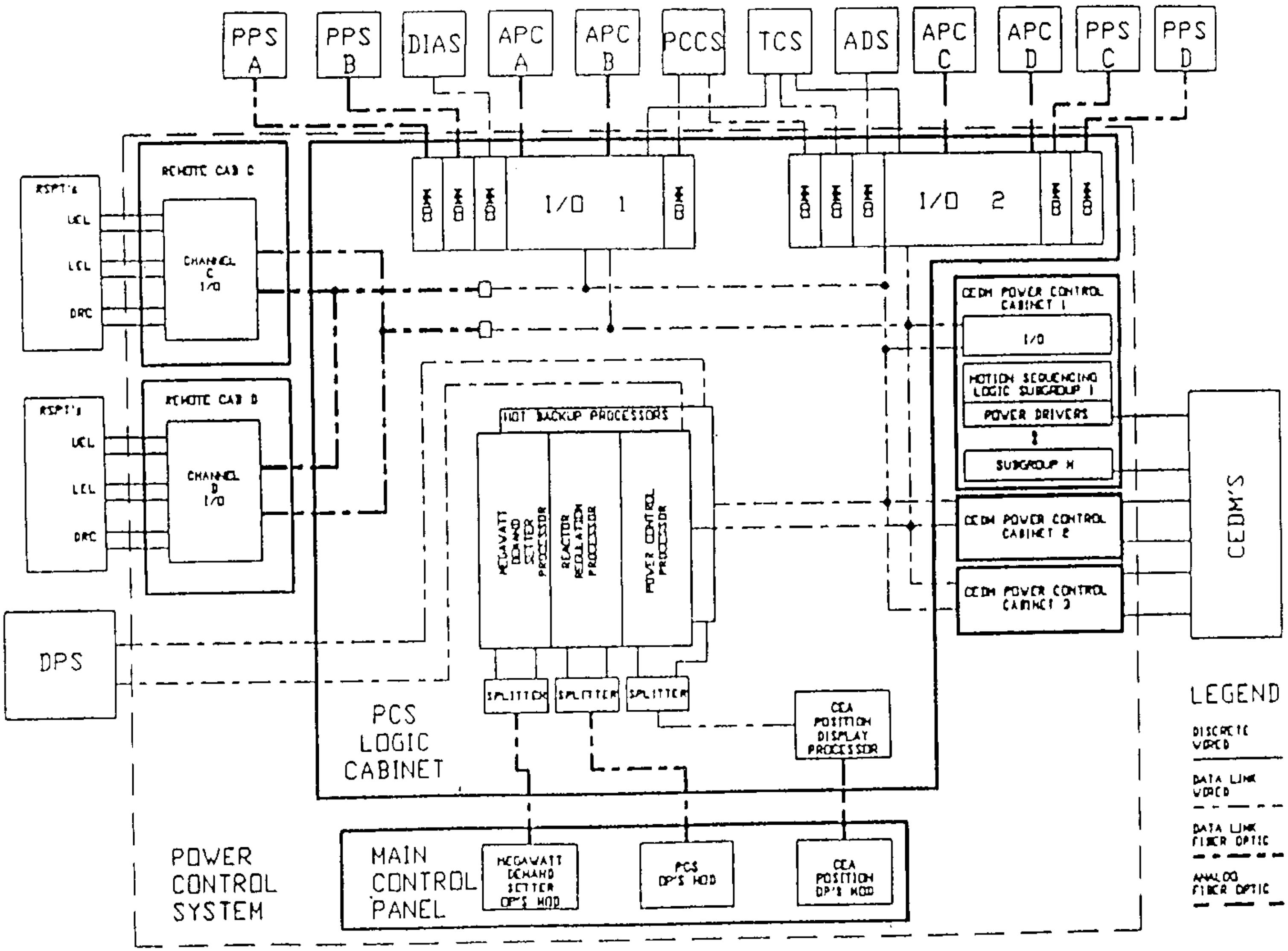
SYSTEM 80+™ ESF-CCS SIZING CONFIGURATION

(SHEET 4 OF 4)

FIGURE II-E-10

SYSTEM 80+™

ABB ASEA BROWN BOVERI



POWER CONTROL SYSTEM SIMPLIFIED BLOCK DIAGRAM

FIGURE II-E-11

F. FUEL HANDLING STORAGE

Introduction

The System 80+ Refueling System consists of the equipment, tools and procedures for refueling the reactor in a safe, efficient and rapid manner. The Refueling System tools and equipment result from the evolution of the design during the past twenty five years. Experience from operating reactors is continuously factored into the design to enhance the operation and reliability of the Refueling System. Major improvements have been factored into the current System 80+ design. The foremost of these improvements are summarized below:

- The building arrangements provide improved personnel and equipment flow and staging for activities in the reactor building. Examples of these improvements are a large closed staging area outside the equipment hatch to improve material flow to support reactor building activities. Access to the reactor building operating floor can be gained through either the personnel lock or the equipment hatch. Both access points allow personnel traffic to the 360 degree hallway outside of the crane wall. Jib cranes and hoists are located at strategic points in the reactor building to support maintenance activities and lessen the work load previously assigned to the polar crane.

- A permanent pool seal is included in the design eliminating the handling of the removable pool seal previously required to accommodate the pressure increases due to the blowdown loads resulting from double ended pipe breaks.

- The multiple stud tensioner design has advanced to permit detensioning and removal of 100 percent of the reactor vessel studs at one time.

- The head area missile shield is incorporated in the head area cable tray structure eliminating the handling of a separate missile shield.

- The addition of a computer system to the refueling equipment improves the locating accuracy of the equipment and provides for increased operating speeds without jeopardizing the safety of the fuel assembly handling operations.

- A quick opening transfer tube penetration sleeve closure is incorporated to significantly reduce personnel radiation exposure and preparation time for the fuel transfer system.

- A refueling machine simulator is incorporated in the design to allow operator training prior to the refueling outage.

- The spent fuel machine duplicates the refueling machine in control and operation to ease operator training requirements.

Additional features that are also incorporated in previous System 80+ plants to minimize outage time include the following:

- The bottom mounted ICI assembly design allows the withdrawal and insertion of the instrumentation at the same time as the reactor vessel closure head disassembly/reassembly operations. Top mounted ICI assembly design requires that these activities be performed in sequence.

- The control element assemblies are removed from the fuel assemblies in the reactor core as part of the removal of the upper guide structure assembly removal operations. This combined removal eliminates the time consuming handling and exchange of control element assemblies during the fuel handling operations. The control element assemblies are installed back into the reactor core when the upper guide structure assembly is installed in the reactor vessel.

The system includes the equipment for the removal and replacement of the following: the reactor vessel closure head assembly, the head area cable tray system (HACTS), control rod drive mechanisms (CEDMs), in-core instrument (ICI) assemblies, reactor vessel internals, control element assemblies (CEAs), and fuel assemblies. The system also includes equipment for new and spent fuel assembly storage, inspection, and shipment.

Water is used to provide radiation shielding for refueling operations. System 80+ does not have an external refueling water storage tank. Water required for refueling operations is stored inside the reactor building in the In-containment Refueling Water Storage Tank. The efficient transfer of the required water into and out of the refueling cavity is an important consideration during refueling operations. This tank is also used in the mitigation of postulated design basis events.

The Refueling System tools and equipment can be conveniently divided between those required in the reactor building and those required in the nuclear annex. Major refueling equipment located within the reactor building includes: the refueling machine with its computer control system and underwater closed circuit television system, fuel transfer system upender and operating console, CEA change platform, CEA elevator, reactor vessel closure head lift rig, head area cable tray system with integral missile shield, multiple stud tensioner, upper guide structure (UGS) lift rig assembly, and the core support barrel (CSB) lift rig assembly. Major refueling system equipment located within the nuclear annex includes: spent fuel handling machine with auxiliary hoist and computer control system; fuel transfer system upender, console and fuel carrier; new and spent fuel storage racks; and new and spent fuel inspection stations. The proposed arrangement of equipment in the reactor building is shown in Figures II-F-1 through II-F-3, respectively.

The refueling system is designed such that the refueling can be accomplished in a minimum amount of time with (1) maximum safety, (2) a minimum probability of mishandling, (3) efficiency of equipment operation and procedures to minimize the length of the refueling outage, and (4) as low as reasonably achievable (ALARA) personnel radiation exposures. These requirements are accomplished by the following features. Positive grappling is provided for all fuel assembly and component handling devices to preclude inadvertent release of the components being handled. Electrical and mechanical interlocks are included to ensure correct sequential operations of system components. Travel limits and handling load limits are used to ensure that component design parameters are not exceeded. Sufficient water coverage during fuel assembly and core component handling and storage is maintained to minimize the radiation levels in the work area.

Crevice free designs of wetted components facilitate decontamination. Utilization of parallel disassembly and reassembly operations minimize the outage time. Manual backup features for all electrical controlled hoisting and travel functions are provided. Computer controlled fuel handling equipment provides optimized travel paths, minimize referral to written operating instructions in technical manuals (help menus and displays are available on touch screen computer displays) and automatically record fuel assembly movements for core verification. The equipment contains, where practical, devices which will function to bring the operation to a safe condition in the event of a failure. Maximum visibility is maintained and visual displays provide the most efficient operation of the equipment.

Refueling Operations

A brief description of the refueling procedure follows as an aid to understanding the descriptions of the tools and equipment.

1. Refueling Procedure

Prior to plant shutdown, operators are trained in the use of the fuel handling system equipment. The training can be performed with either the refueling machine simulator or the spent fuel handling machine which duplicates the controls and operations of the refueling machine. The new fuel assemblies are moved to the spent fuel storage racks where they will be readily accessible to the spent fuel handling machine. At the start of the refueling outage the generators are disconnected and the plant is shutdown and cooled to the required refueling temperature. After containment access is gained, the control rod drive mechanisms are disengaged from their extension shaft assemblies by de-energizing the coil stack assemblies, and the control element drive mechanisms (CEDMs) and heated junction thermocouple (HJTCs) cabling connectors are disconnected. The head area cable tray structure, with integral missile shield, is removed from over the reactor vessel head. In parallel with these operations, the reactor coolant level is lowered to at least 12 inches below the reactor vessel flange.

The CEDM cooling manifold is disconnected from its ductwork, and the reactor vessel head vent line, as well as sections of the reactor vessel head insulation are removed. The multiple stud tensioner is employed to remove the preload on the reactor vessel studs and to remove the studs, nuts, and washers to the storage area. After two reactor vessel closure head alignment pins and the full complement of stud hole plugs are installed, the closure head lifting frame is installed on the head lift rig assembly, and the closure head removed to its storage location. As the head is removed, the containment pool is flooded with water.

The bottom mounted ICI assemblies are withdrawn in parallel with the disconnection of the CEDM cables and removal of the reactor vessel studs. This is accomplished by disconnecting mechanical and electrical connections at the ICI seal table and the pool wall and removing the ICI cable tray frame assembly. An ICI holding frame is installed at the top of the pool wall directly over the seal table and each ICI is withdrawn from the seal table, with the CEA change platform hoist, and secured to the holding frame.

The transfer tube penetration sleeve quick opening closure is removed as soon as practical to permit preoperational testing of the fuel transfer system. The transfer tube valve is then closed during all heavy lifts within the flooded containment pool to preclude water level changes in the spent fuel pool should a heavy load be inadvertently dropped.

The upper guide structure lift rig is removed from the upper guide structure laydown area and installed on the upper guide structure assembly. The lift rig work platform is lowered, the extension shaft assemblies latched to the work platform, and the extension shaft assemblies and control element assemblies are then withdrawn into the upper guide structure assembly. The upper guide structure assembly is lifted from the reactor vessel by the polar crane and transferred to the upper guide structure laydown area where it remains completely submerged during the refueling outage. At this point, the fuel transfer valve is opened to permit the fuel handling sequence to begin.

The refueling machine, controlled by a state of the art computer system, positions the refueling machine at the desired core location in accordance with preprogrammed software instructions. Once the refueling machine is located over the proper fuel assembly, the fuel hoist box is lowered and the operator actuates a fuel spreading device to minimize contact of adjacent fuel assemblies with the fuel assembly being handled. The hoist is lowered further until the grapple engages the fuel assembly and automatically stops by an underload interlock. After the computer verifies that the grapple is at the correct grappling elevation, a grapple permit interlock is energized and the operator positively locks the grapple to the fuel assembly. The hoist motor is energized and the fuel assembly is raised into the fuel hoist box which supports and protects the fuel assembly. The hoist box and fuel assembly are raised into the refueling machine mast. As the fuel assembly is lifted, the grapple is mechanically keyed to the hoist box and to the fuel assembly to preclude disengagement of the fuel assembly, even if the grapple is inadvertently actuated. An on-board closed circuit television system monitors fuel grappling and movement, and allows remote verification of proper fuel location and orientation.

After removal from the core, the spent fuel assembly is moved underwater to either the intermediate fuel storage rack or the fuel transfer system depending on the fuel handling sequence. At the fuel transfer system upender, it is installed into the empty fuel carrier cavity, then the refueling machine is indexed over the other cavity where the new fuel assembly is located. The new fuel assembly is removed from the fuel carrier and installed into the core while the upender moves the fuel carrier to the horizontal position and transfers it through the transfer tube to the fuel building.

When the fuel carrier reaches the fuel building, a second upender returns the carrier to the vertical position. The spent fuel handling machine installs a new fuel assembly into the empty fuel carrier cavity then indexes over the other fuel cavity and removes the spent fuel assembly. The spent fuel handling machine moves to the assigned fuel rack cavity location while the new fuel assembly is moved to the reactor building . In this manner, both the refueling machine and the spent fuel handling machine always travel with a fuel assembly which minimizes the refueling time. The removed spent fuel assemblies are inspected at the spent fuel inspection station as required by the fuel handling procedure.

The equipment and procedures are designed to permit handling of the control element assemblies and ICI assemblies to proceed in parallel with the fuel handling operations. Complete CEA replacement operations are only expected to occur at approximately 15 year intervals. Routine ICI assembly changeout is performed as part of the normal refueling outage. During control element assembly replacement operations, the control element assemblies are removed from the stored upper guide structure assembly and placed into the CEA elevator with long handling tools. Once in the elevator, the CEA is disassembled, the CEA rod is cut into two pieces and placed into a transport container for shipment to the nuclear annex via the fuel transfer system. New control element assemblies are assembled in the CEA elevator and installed into the stored UGS.

Replacement of an ICI assembly is accomplished by winding up all but the highly radioactive end portion onto a reel located on the CEA change platform. The end portion is cut into pieces by an ICI cutter. During the cutting operation, the pieces are placed into a transport container which is transported to the nuclear annex by the fuel handling equipment for eventual disposal.

At the completion of the fuel handling operation, the fuel transfer tube valve is closed and the upper guide structure assembly is reinserted into the reactor vessel. The control element assemblies and extension shaft assemblies are lowered into position and the UGS lift rig is uncoupled from the extension shaft assemblies and the upper guide structure assembly is moved to the upper guide structure laydown area. The water in the pool is lowered and the reactor vessel closure head assembly is moved from its storage location until it is positioned over the reactor vessel and slowly lowered until the reactor vessel alignment pins are engaged. After engagement, the closure head is slowly lowered until seated. The alignment pins and stud hole plugs are removed, the head bolted down, the transfer tube closure installed, and the ICI assemblies inserted into the seal table. Following this, the ICI cable tray support assembly is installed, the CEDM cooling manifold installed, and the head area cable tray system replaced.

After pool wall and refueling equipment decontamination, containment leak testing, and physics testing are completed and the generator is synchronized, completing the refueling outage.

2. Reactor Vessel Level Indication for Refueling and Cold Drain-Down

Instrumentation is provided to monitor the water level in the refueling cavity during refueling operations to ensure adequate radiation shielding when assemblies are being handled or the reactor internal components are being transferred between the laydown areas and the reactor vessel.

Additional instrumentation is provided to monitor the water level during reduced inventory operations with shutdown conditions. The instrumentation provided for water level monitoring during refueling, particularly mid-loop operation, represents a significant evolutionary design improvement. The water level monitoring instrumentation substantially reduces the risk of loss of shutdown cooling capability. The instrumentation provides data that will enable the operator to take earlier action to correct the water level and mitigate the potential consequences of an unplanned reduction in the water level.

a. Normal Refueling Water Level

During normal fuel assembly handling operations the refueling cavity water level is maintained at an elevation 7.3 m (24 ft) above the reactor vessel flange. This level provides sufficient shielding to plant operations personnel during fuel handling. When the reactor internals components are being transferred in the reactor cavity the water level is varied as necessary to support component handling. Whenever the water is being changed or heavy components, such as the reactor internals, are being handled in the refueling pool the transfer tube valve is closed to maintain the water level in the spent fuel pool.

During normal refueling operations, the water level in the refueling cavity is monitored with two (2) wide range differential pressure based sensors. These sensors alarm when the water level reaches either the high or low level setpoint. These wide range sensors are discussed in the following section. Additionally, a Refueling pool level switch is activated to directly monitor the normal refueling cavity water level. Protection for overfilling the refueling cavity is provided by overflow skimmers in the cavity wall.

b. Reduced Inventory Operations Water Level

Instrumentation is provided to determine the water level and water temperature in the reactor vessel when the water level is lowered during reduced inventory operations. Reduced inventory operations occur whenever the water level is lowered to more than three (3) feet below the reactor vessel flange, to a minimum level at the centerline of the reactor vessel outlet nozzles (referred to as mid-loop level). This lowering occurs for activities such as installation of the steam generator nozzle dams. In order to minimize the radiation dose rates in the reactor building, the reactor vessel head is installed when operating at the mid-loop condition. The water level is closely controlled at mid-loop to ensure shutdown cooling is maintained. The instrumentation provides the operator with diverse and redundant indications that the level is being properly maintained.

Four (4) instrument pairs are provided for the measurement of the water level during RCS draindown and reduced inventory operations. These instruments make up the refueling water level indication system. Two (2) wide, range, differential pressure based level sensors are provided which function to redundantly measure the water level anywhere between the pressurizer and the junction of the shutdown cooling suction line on the reactor coolant system hot leg.

Two (2) narrow range differential pressure based level sensors are utilized to determine reactor coolant system water level within the reactor vessel. The narrow range level sensors function to measure the water level in the reactor vessel between the direct vessel injection nozzles and the junction of the shutdown cooling suction line with the reactor coolant system hot leg.

One wide range and one narrow range differential pressure instrument are connected to each shutdown cooling suction line. Separate lower level taps are provided for each instrument. The wide range and narrow range instruments operate with or without the reactor vessel head in place. These instruments can be used in all plant operating modes with or without the reactor vessel closure head installed.

In addition to the differential pressure sensor instruments, a mid-loop monitoring system is provided to measure the water level during plant shutdown conditions with the reactor vessel closure head installed. This system consists of two (2) probe assemblies containing of a series of heated junction thermocouples and unheated reference thermocouples. The temperature readings of the heated junction thermocouple are compared to the reference thermocouple. When any heated junction thermocouple is not immersed, the temperature rises in comparison to the reference thermocouple due poor heat transfer in air as compared to in water thereby indicating the water level.

The heated junction thermocouples are spaced vertically from the above the fuel alignment plate in the upper guide structure assembly to above the top of the reactor vessel outlet nozzle. Closely spaced thermocouples indicate the water level during reduced inventory operations to ensure shutdown cooling suction is maintained.

Several instruments are available for continuous temperature measurements during reduced inventory operations with the reactor vessel closure head installed. These include:

- core exit thermocouples
- shutdown cooling heat exchanger inlet and return line temperature sensors
- hot leg resistance temperature detectors
- mid-loop monitor system thermocouple

All of these instruments provide representative indications of the core exit temperature when the shutdown cooling system is operational. If the shutdown cooling system fails, the core exit thermocouples, hot leg RTDs, and mid-loop monitor system thermocouples are available to track the response to the loss of shutdown cooling or the approach to boiling in the core.

Refueling System Tools and Equipment

The refueling system includes the tools and equipment that are required to prepare the reactor for refueling and to store and handle the reactor core components prior to and during the refueling outage. The reactor core components consist of the fuel assemblies, control element assemblies, neutron sources and ICI assemblies. As servicing of the reactor vessel and steam generator are closely allied with the refueling evolutions, the tools and equipment associated with servicing these components are also described in this section.

The refueling system tools and equipment are located in the reactor building and nuclear annex. The following descriptions are organized to divide the tools and equipment by the buildings that they are primarily used in.

1. Reactor Building Tools and Equipment

The following tools and equipment support the refueling activities in the reactor building. These items are primarily used only during the refueling outage and are not required during plant operation. The arrangement of the reactor cavity pool and the major equipment is shown in Figure II-F-1. Tools and equipment in the reactor building are presented below in groups of items having related functions.

a. Fuel Handling Equipment

The reactor building fuel handling equipment transfers the fuel assemblies between the reactor vessel and the transfer system upender. Movements of fuel assemblies within the reactor vessel are also done with this equipment.

- **Refueling Machine (Figure II-F-4)**

The refueling machine is used to relocate fuel assemblies within the core, and move fuel assemblies between the core, the fuel transfer system, and temporary storage in the intermediate fuel storage rack located in the reactor building. In addition, the machine is used to move the ICI/CEA transport container between the transport container rack and the fuel transfer system. The refueling machine also serves as a movable work platform within the pool area for other related tool handling operations.

The machine consists of a bridge assembly and trolley assembly that permits precise positioning of the mast assembly containing the fuel grapple over each fuel assembly to be handled. The hoist system, in the mast assembly, incorporates speed and load control interlocks to permit the safe withdrawal of fuel assemblies from and insertion into the core. The machine is also provided with an articulated grapple and fuel spreading device to accommodate deviations of the fuel assembly from the theoretical position to ensure the fuel assembly being handled does not hang up on the adjacent assemblies in the reactor core.

Positive mechanical grappling between the fuel assembly and refueling machine grapple is provided that permits the grapple to opened only at elevations corresponding to the core seating surface and the transfer system upender. Positive mechanical stops are used to control the maximum height that the fuel assembly can be lifted relative to the reactor core seating surface to ensure sufficient water coverage for radiation shielding.

The refueling machine rails are used to support the refueling machine and the CEA change platform during their movement within the reactor building . The rails are machined and installed with controlled tolerances to maintain the required positioning alignment of the refueling machine with respect to the fuel assemblies within the reactor vessel. Precise positioning allows the fuel handling process to be accelerated and minimizes potential fuel handling damage. The rails incorporate seismic holddowns to preclude the equipment from falling into the pool during a seismic event.

The refueling machine set-up and alignment fixtures are used to initially align the fuel handling equipment during initial equipment installation to ensure that it will operate in accordance with the required precision.

- **Fuel Transfer System (Figures II-F-5 and II-F-6)**

The fuel transfer system is used to move fuel assemblies, the ICI/CEA transport container, and the dummy fuel assembly through the fuel transfer tube assembly connecting the nuclear annex and the reactor building. The transfer system consists of a two cavity fuel carrier, upenders, devices for the remote actuation of the upenders, control consoles, and a winch mechanism to propel the carriage between the upenders. The two cavity fuel carrier permits the refueling machine and the spent fuel handling machine to travel fully loaded at all times, thus significantly reducing the fuel handling time. As an example, the spent fuel handling machine places a new fuel assembly in one carrier cavity and translates to the second carrier cavity to remove the spent fuel assembly being transferred from the reactor building.

Load interlocks preclude damage to the fuel carrier in the event of contact with other equipment or objects. The upenders provide precise positioning of the upended fuel carrier for access to the fuel assembly by the refueling machine and the spent fuel handling machine. A dual cable drive system ensures access to the fuel carrier in the event that one cable fails. Whenever the fuel carrier is retracted into the transfer canal in the nuclear annex, the transfer tube valve can be closed. Therefore personnel entry to the transfer canal is not required to set up the transfer system. The upenders are designed to be remotely removable from the pool without requiring the water level in the refueling cavity or spent fuel storage pool to be lowered.

The hook and wrench tool is used to remotely unlock the transfer system upender from the floor embedments to permit the upender to be removed for servicing from the flooded pool without requiring the pool water level to be lowered.

- **Fuel Handling System Computer System**

The control system is based upon modern, state-of-the-art equipment and techniques, with each machine having either a programmable logic controller or a computer as the heart of the control system. Signals are transmitted between the fuel handling machines to preclude actions that are detrimental to the fuel assemblies or the machines.

The new fuel elevator, fuel transfer system, CEA change platform, and the CEA elevator are controlled by programmable logic controllers. The refueling machine and the spent fuel handling machine and the simulator are computer controlled. The computers are AT-compatible and use software configured for the specific application. All machines in each refueling system are networked together via a shielded pair of conductors and/or a coax cable (the computer network) to provide a simple, powerful method for communicating machine status and other pertinent information from one machine to another. The facility wiring using this networking is greatly reduced compared to previous control systems and provides greater flexibility for future modifications or enhancements.

The refueling machine computers perform the following functions: hoist motion interlocks; hoist speed control, including acceleration/ deceleration parameters, maximum speed setting and speed restrictions; bridge and trolley motion interlocks; bridge and trolley, including acceleration/deceleration parameters, maximum speed setting, and speed restrictions; automatic fuel spreader control; automatic hoist latch control; operating zone boundary monitoring; load weighing interlocks; grapple zone control; computer and underwater tv camera display; security access to data files; data logging of movements and fuel serial numbers; resolver verification and trending; core mapping; automatic compensation; report generation; manual/automatic control; help/diagnostics menus; and graphic displays.

- **Intermediate Fuel Storage Rack**

The intermediate fuel storage rack is used for the temporary underwater storage of up to six fuel assemblies in the refueling pool core support barrel laydown area. They may be either new fuel assemblies or spent fuel assemblies depending on the operations being performed. The intermediate fuel storage rack is also used to provide temporary storage of the ICI/CEA transport container during its movement from the transport container rack to the fuel transfer system. The polar crane is used to move the ICI/CEA transport container from the transport container rack to the intermediate fuel storage rack, and the refueling machine moves the transport container from the intermediate fuel storage rack to the fuel transfer system.

b. Closure Head Area Service Equipment

The closure head area service equipment is used to prepare the reactor for refueling. This equipment provides additional functions during the operation of the plant. However, the design of this equipment is an integral part of the refueling outage.

- **Closure Head Lift Rig Assembly (Figure II-F-7)**

The closure head lift rig assembly is used to lift and transport the closure head assembly and includes the following components: reactor vessel closure head; heated junction thermocouple pressure housings; mid loop monitoring system pressure housings; control element drive mechanisms; closure head insulation; and cabling from the heated junction thermocouple system, the mid loop monitoring system,the acoustic leak monitoring system, and the loose parts monitoring system. The closure head rig assembly, in conjunction with the control element drive mechanism cooling manifold, provides a cooling air flow path to route cooling air around the control element drive mechanisms.

The closure head lift rig assembly is permanently bolted to pads on the reactor vessel head and includes a skirt, plenum plate, work platform and lifting frame assembly. The work platform provides access to the top of the control element drive mechanisms for connecting and disconnecting the various cabling that runs from the closure head assembly to the head area cable tray system. To minimize the number of heavy lifting equipment that is stored in the reactor building, the lifting frame assembly is also used to lift and transport the multiple stud tensioner with the full complement of reactor vessel studs, nuts and washers.

The closure head lift assembly is assembled from painted carbon steel components.

- **CEDM Cooling Manifold (Figure II-F-8)**

The CEDM cooling manifold is used to provide a uniform distribution of the CEDM cooling air flow from the cooling ductwork to the closure head lift rig assembly flow skirt. The manifold is fabricated in four segments, with quick acting fasteners, to permit the rapid attachment and removal of the ductwork from the closure head lift rig assembly. The reactor vessel flange insulation is an integral part of the cooling manifold. This feature reduces the number of lifts and the time required to prepare the reactor vessel studs for removal.

The CEDM cooling manifold is assembled from stainless steel components.

- **Head Area Cable Tray System (HACTS) (Figure II-F-9)**

The head area cable tray system is used to support a network of trays, walkways and termination panels for the reed switch position transmitters (RSPTs), the control element drive mechanism (CEDM) power cables, the heated junction thermocouple (HJTC) cables, and other cables routed in the head area. These cables emanate from the top of the head lift rig assembly and end in junction panels mounted on the HACTS adjacent to the steam generator walls. The HACTS simplifies the cable interfaces by segregating the CEDM and HJTC cables by control channel and function at the easily accessible junction panels. The cable tray system also provides a missile shield to prevent damage to safety related equipment from potential missiles generated in the head area during plant operation. Once the cables have been disconnected at the junction panels and at the interface connections between the head area and the cable tray, the entire system is removed to storage by the polar crane thereby providing complete access to the reactor vessel head area.

The main support structure consists of two beams that span the refueling pool and bolt to corbels on the steam generator walls. The end connections incorporate lead-in features to accurately locate the HACTS over the reactor vessel closure head assembly and to accommodate the thermal growth of the HACTS during plant operation. The corbels are situated such that they do not interfere with movement of the refueling machine and CEA change platform.

- **Reactor Cavity Pool Seal**

The refueling pool seal connects the reactor pressure vessel upper flange to the floor of the refueling cavity to permit filling of the refueling cavity for fuel handling activities.

During normal refueling operations, the pool seal is designed to withstand the pressure resulting from a water head that is the full depth of the refueling cavity from the elevation of the operating floor, and overpressure in a full cavity caused by seismic events. The pool seal will also withstand the impact of a fuel assembly dropped from the maximum height that it is raised above the pool seal by the refueling machine during transit.

Pool seal welds required for structural integrity or sealing integrity are inspectable. Access ports are provided for ex-core detector servicing and inspections, and for cavity ventilation.

c. <u>Reactor Vessel Internals Service Equipment</u>

The reactor vessel internals service equipment is used to remove, install and store the upper guide structure assembly and core support barrel assembly in the refueling cavity. The upper guide structure assembly is removed to gain access to the fuel assemblies at each refueling outage. The core support barrel assembly is removed only for ASME Boiler and Pressure Vessel Code inspections at approximately ten (10) year intervals.

- **Upper Guide Structure Lift Rig Assembly (Figure II-F-10, Sheets 1 and 2)**

The upper guide structure lift rig assembly is used to withdraw the control element assemblies from the core and to remove the upper guide structure assembly and control element assemblies from the reactor vessel to the upper guide structure laydown area. All extension shaft assemblies with their attached control element assemblies are latched to a movable plate in the lift rig, and are raised and lowered simultaneously with the plate. The UGS lift rig assembly is attached to the upper guide structure assembly with three (3) bolts that are operated from the top of the lift rig. The UGS lift rig assembly is lifted with the polar crane.

At the conclusion of the outage the UGS lift rig is used to replace the upper guide structure in the reactor vessel and reinsert the control element assemblies in the reactor core. During CEA handling operations, the upper guide structure lift rig is used to temporarily store the CEA extension shaft assemblies when it is placed on the core support barrel storage stand.

- **Core Support Barrel Lift Rig Assembly**

The core support barrel (CSB) lift rig assembly is used to couple the core support barrel to the polar crane as the core support barrel assembly is removed from and installed in the reactor vessel. The spreader weldment and tie rod assembly is shared with the UGS lift rig assembly to reduce the amount of lifting equipment stored in the reactor building. Three (3) column adapters and bolt assemblies connect the weldment to the core support barrel assembly for lifting with the polar crane.

d. <u>CEA Servicing Equipment</u>

The CEA servicing equipment is used to assemble, transport and dispose of control element assemblies in the reactor building. The CEA change platform and ICI/CEA transport container are also used for ICI assembly handling activities.

- **CEA Change Platform (Figure II-F-11)**

The CEA change platform is used for new and spent control element assembly (CEA) handling operations, spent CEA disposal operations, ICI assembly withdrawal operations from the ICI seal table, new ICI assembly insertion operations, and the cutting and disposal of spent ICI assemblies . Most of these operations are performed in parallel with the fuel handling.

The CEA Change Platform consists of a motorized work platform and gantry hoist that spans the refueling pool and travels on the same rails as the refueling machine. The controls for operation of the CEA handling tools, ICI cutter assembly and CEA cutter assembly are incorporated in the CEA change platform control consoles.

An ICI retrieval cart is mounted on rails located on the platform bridge. The retrieval cart is positioned over the ICI assembly scheduled for removal. The minimally radioactive upper portion of the ICI assembly is coiled on a disposable reel for shipment off site. The retrieval reel contains interlocks to ensure the highly radioactive portion of the ICI assembly is not withdrawn above the safe water depth.

- **CEA Elevator (Figure II-F-12)**

The CEA elevator is used to assemble new control element assemblies and to disassemble and dispose of spent control element assemblies . The elevator consists of a carriage that holds either a 12 rod or a 4 rod CEA and the ICI/CEA transport container. The carriage is mounted on rails attached to the pool wall and is powered by an electric hoist mechanism. By positioning the carriage at the designated elevation, and using the appropriate tooling, the various CEA assembly and disassembly operations can be performed. Load and mechanical interlocks ensure adequate water coverage over the irradiated components at all times.

New CEA Assembly Tools

A group of specialized tools are provided to assemble new control element assemblies in the refueling cavity. These tools are used in conjunction with the CEA elevator. The control element assembly rods are received at the site in a special shipping container and must be removed from the container and placed in the CEA elevator for assembly. Fixtures are provided to lift the rods from the container and position the rods in the configuration of either a four finger or twelve finger control element assembly. After the rods are positioned, additional tools are used to position a CEA spider on the rods. Nuts are placed on the rods and torqued to the prescribed value before crimping the nuts to lock them to the CEA rods.

CEA Handling and Disposal Tools

Special handling tools are used to lift the control element assemblies between the CEA elevator and upper guide structure. The control element assemblies are kept under water during handling and disposal to minimize personnel radiation exposure. These tools provide support and guidance for the control element assemblies as well as maintain the rod pattern during handling. These tools are connected to the hoist on the CEA change platform when in use. The CEA handling tools contain interlocks to prevent mishandling.

When the control element assemblies are to be replaced, the assemblies are transferred from the upper guide structure to the CEA elevator. The CEA disassembly tool is used to remove the crimp nuts from the CEA rods so that the CEA spider can be removed. After the spider is removed, the rods are lifted from the elevator to the transport container and cut to lengths shorter than a fuel assembly using the CEA cutter. When the ICI/CEA transport container is full, it is moved to the spent fuel pool using the fuel handling equipment.

e. <u>Reactor Vessel Head Disassembly Equipment</u>

The reactor vessel head disassembly equipment is required for the removal of the reactor vessel studs.

- **Multiple Stud Tensioner**

The multiple stud tensioner accomplishes simultaneous controlled tensioning/detensioning of the reactor vessel studs, rotating of the stud nuts, screwing in and out of the studs and transport of the studs and nuts to a storage stand on the operating floor. The multiple stud tensioner is fully automatic and completes each of these individual tasks on all reactor vessel studs simultaneously.

Operation of the multiple stud tensioner is performed from the control station located on the operating floor. The tensioner includes the tensioning units, rotation devices for stud and nuts as well as elongation indicators to measure the stud tension. The tensioner is powered by a hydraulic pumping system.

f. <u>ICI Handling Tools and Equipment</u>

The ICI handling tools and equipment are used to install new ICI assemblies in the reactor, withdraw the ICI assemblies for refueling, insert the ICI assemblies following refueling and to dispose of spent ICI assemblies. The ICI assemblies are inserted into the fuel assemblies during plant operation and must withdrawn from the fuel assemblies before the fuel is moved.

The CEA change platform and ICI/CEA transport container are used in conjunction with these tools and equipment.

- **ICI Holding Frame**

The ICI holding frame is used to hold the ICI assemblies after they have been withdrawn from the seal table during refueling.

The holding frame is mounted at the top of the refueling pool wall directly above the ICI seal table during the refueling outage. It is removed and stored on the operating floor during plant operation. The holding frame maintains the upper end of the ICI assemblies out of water but below the operating floor level to permit passage of the refueling machine bridge over the assemblies while the refueling machine is enroute to the transport container rack. The frame support the weight of the ICI assemblies and also a work platform which permits operator access to the ICI assembly electrical connectors.

The ICI assemblies are withdrawn and latched to the holding frame using the hoist on the CEA change platform.

The CEA change platform can be driven over the holding frame for these operations.

- **ICI Cable Tray Assembly**

The ICI cable tray assembly supports and separates the ICI cabling from the ICI seal table to the cable connector panels located on the pool wall above the seal table. The cable tray assembly includes a work platform to provide easy operator access to the cable junction panels at the side of the refueling cavity.

The ICI cable tray is removed from the refueling cavity at the beginning of the refueling outage before the ICI holding frame is installed.

ICI Tools

The ICI assemblies must be withdrawn from the reactor core prior to refueling the reactor as the assemblies are inserted into the fuel assemblies. Withdrawal is performed by attaching an ICI retrieval tool to each ICI assembly. The hoist on the CEA change platform is connected to each retrieval tool in turn to raise the end of the ICI assemblies to the ICI holding frame where the retrieval tools are used to fasten the assemblies to the frame.

Following refueling or when installing new ICI assemblies, insertion tools are supplied for seating the ICI assemblies in the fuel assemblies. The primary insertion tool straightens the ICI assembly as it is pushed into the guide tube. When the assembly is within approximately one foot of its seated position, the secondary tool is attached to the seal plug on the assembly to push the assembly into its seated position.

The ICI assemblies require periodic replacement due to depletion of the rhodium detectors which affects the instrument sensitivity. Special tools are provided to perform this activity. The CEA change platform includes a windup reel for ICI assembly disposal. The slightly contaminated portion of the ICI assembly that is in the guide tube is wound onto the reel and the lower 30 feet that is normally in the reactor is cut and placed in the transport container using the ICI cutter assembly. The transport container is moved to the spent fuel pool using the fuel handling equipment.

g. <u>CEDM and Extension Shaft Assembly Servicing Equipment</u>

The CEDM and extension shaft assembly servicing equipment is provided to perform the initial installation of the control element drive mechanisms (CEDMs), disassemble and reinstall CEDMs, and to handle and install extension shaft assemblies on the control element assemblies. With the exception of the venting tools, this equipment is used infrequently.

h. <u>HJTC Servicing Tools</u>

The HJTC servicing tools are used to service the heated junction thermocouple (HJTC) pressure housings and mid-loop monitoring system pressure housings on the reactor closure head during disassembly/assembly of the pressure boundary.

i. <u>Training and Test Equipment</u>

Equipment is provided to allow operator training in fuel assembly and control element assembly handling without the need for use of actual reactor core components. This equipment is also used to test the refueling system tools and equipment prior to the refueling outage.

- **Refueling Simulator**

The refueling simulator is a free standing, portable console which mounts and houses the equipment to perform diagnostic checks and validation of hardware and software functions in the refueling computers and to train personnel in the operation of the refueling machine and the spent fuel handling machine.

The simulator includes motors coupled to resolvers and electrical loads to simulate brakes, audible alarms, and other devices mounted on the machine. The simulator also mounts the necessary pushbuttons, toggle switches, indicators, and other control operators to duplicate the refueling machine operation. The simulator connects to the existing cable on the refueling computer systems.

The simulator utilizes a computer system similar to that used in the refueling machine and the spent fuel handling machine.

- **Dummy Fuel Assembly (Figure II-F-13)**

The dummy fuel assembly is used for training operating personnel in the handling of fuel assemblies, load testing the refueling machine and the spent fuel handling machine hoist and grapple system to 125 percent of the weight of the fuel assembly, load testing the various other fuel handling tools, transporting the source assembly through the fuel transfer tube, and carrying the surveillance capsule from the reactor building to the nuclear annex after it has been removed from the reactor vessel.

- **Four Rod Dummy CEA**

The four rod dummy CEA is used for operator training in CEA assembly, handling, and disassembly operations and for checkout of the CEA handling tooling during fabrication and acceptance testing.

- **Twelve Rod Dummy CEA (Figure II-F-14)**

The twelve rod dummy CEA is used for operator training in the CEA assembly, handling, and disassembly operations and for checkout of the CEA handling tooling during fabrication and acceptance testing.

2. Nuclear Annex Tools and Equipment

The following tools and equipment support the refueling activities in the nuclear annex. These tools and equipment also support preparation of the fuel assemblies for installation in the reactor and shipment of spent fuel assemblies from the site.

a. <u>Fuel Handling Equipment</u>

The fuel handling equipment in the nuclear annex handles fuel assemblies going to or returning from the reactor core. This equipment is also required for handling core components that are removed from the reactor building via the transfer system.

- **Spent Fuel Handling Machine (Figure II-F-15)**

The spent fuel handling machine is used to move fuel assemblies between the new fuel elevator, the spent fuel storage racks, the fuel transfer system upender, the spent fuel inspection station, the fuel consolidation station, and the spent fuel shipping cask. In addition, since the nuclear annex crane is locked out from passage over the spent fuel racks when they contain fuel assemblies, the spent fuel handling machine is provided with an auxiliary one ton hoist for other handling tasks within the pool area such as source relocation within the fuel assemblies, fuel reconstitution, and other light loads.

The spent fuel handling machine replicates the refueling machine to the maximum extent possible to permit the operators to train under simulated refueling machine operating conditions prior to entering the reactor building for the refueling operation. Lights, nomenclature, and operating instructions are identical to develop proper operating skills and avoid potential operating problems due to console layout and component nomenclature.

The spent fuel handling machine operation is controlled by a state-of the-art computer system.

The spent fuel handling machine rails are used to support the spent fuel handling machine during its movement within the fuel building. The rails are machined and installed with controlled tolerances to maintain the required positioning alignment of the spent fuel handling machine with respect to the fuel rack cavities. Precise positioning allows the fuel handling process to be accelerated and minimizes fuel handling damage. The rails incorporate seismic holddowns to preclude the equipment from falling into the pool during a seismic event.

b. <u>Fuel Storage Racks</u>

The requirements for receiving and storing the plutonium fuel assemblies necessitate that the new fuel assemblies be handled in a different manner than is normally used in the System 80+ plant design. It is expected that the entire complement of new fuel assemblies will be received on site early in the operating life of the plant. Furthermore, there is a potential for all of the fuel assemblies for the operating life to be irradiated and stored for additional burnup in a later fuel cycle. Therefore all fuel assemblies will be stored in racks under water in the spent fuel pool and there will be no racks or building area dedicated to new fuel storage. Storage capacity will be provided for the maximum number of fuel assemblies that will be used during the life of the plant.

The spent fuel storage racks are used to store the new fuel assemblies and spent fuel assemblies. Twenty (20) of the fuel rack cavities are designed specifically for the storage of failed fuel assemblies. Transport containers filled with expended control element assemblies and ICI assemblies are also stored in the spent fuel racks in ten other locations.

The basic rack module consists of a monolithic, honeycomb structure, the elements of which are the individual storage cells. Each rack module is fabricated from type 304 stainless steel and is formed by joining the individual fuel storage cavities into a load bearing honeycomb structure. Each rack module is welded construction and each weld is a structural weld visually inspected consistent with this type of construction for ASME code Class 3 components. The modules are a free standing design and is seismically qualified without mechanical dependence on pool walls. This feature facilitates remote installation and permits removal for pool maintenance if required. Each rack module is elevated and leveled to avoid floor obstructions and to provide the space needed so that pool water can flow freely under the rack and adequately cool every storage cell.

c. Fuel Transfer Tube Assembly (Figure II-F-16)

The fuel transfer tube assembly provides a passage for the transfer of fuel assemblies between the reactor building and nuclear annex.

- **Fuel Transfer System Rails**

The fuel transfer system rails are welded within the fuel transfer tube and support the fuel transfer system fuel carrier during its passage through the transfer tube. The rails incorporate replaceable wear surfaces for ease of rail repair in the event of damage and holddowns to maintain the fuel carrier on the rails during a seismic excursion.

The rails are not located within the fuel transfer tube valve as this would interfere with valve closing. The fuel carrier has been designed to span the opening between the fuel transfer system upender and the transfer tube rails.

- **Fuel Transfer Tube**

The fuel transfer tube connects the spent fuel pool with the refueling pool to allow the underwater transfer of fuel assemblies, expended ICI assemblies, expended control element assemblies, the ICI/CEA transport container, and the dummy fuel assembly between the nuclear annex and the reactor building .

The transfer tube is located within the penetration sleeve and quick opening penetration sleeve closure which forms the reactor building boundary. Therefore the transfer tube is not a safety related component.

- **Quick Opening Transfer Tube Penetration Sleeve Closure**

The quick opening transfer tube penetration sleeve closure is used during plant operation to close and seal the end of the penetration sleeve within the reactor building. It is designed for compatibility with the fuel transfer system and can be easily and rapidly removed to reduce personnel radiation exposure and minimize refueling down time. An auxiliary hoist provided can be located on brackets directly above the closure and used for closure removal in the event the polar crane is not readily available. When the polar crane becomes available, the load can be transferred to the polar crane and the closure removed to the operating floor for inspection and preparation for reinstallation.

- **Transfer Tube Support Stand**

The transfer tube support stand supports the end of the transfer tube that is located in the nuclear annex.

All the transfer tube loads i.e., seismic, transfer system emergency withdrawal, transfer tube valve, etc., are reacted by the stand.

- **Transfer Tube Valve Operator Standoffs**

The transfer tube valve operator standoffs provide lateral restraint for the transfer tube valve operator during a seismic event and normal valve operation and handling.

- **Transfer Tube Valve**

The transfer tube valve is used to isolate the water in the transfer canal of the spent fuel pool from the refueling cavity. The valve is closed whenever the refueling cavity is partially filled or during heavy lifts over the reactor vessel pool seal assembly when the refueling cavity is filled.

d. Fuel Handling Tools and Support Equipment

The fuel handling tools and support equipment are used to position fuel assemblies in the nuclear annex. The items are primarily used when handling new fuel assemblies for inspection and movements into and out of the new fuel storage racks.

- **Spent Fuel Handling Tool (Figure II-F-17)**

The spent fuel handling tool is used to handle spent and new fuel assemblies within the spent fuel pool. It is a backup to the spent fuel handling machine grapple which is the primary means for handling fuel assemblies. The tool is supported from the spent fuel handling machine auxiliary hoist and is operated from its walkway.

- **Spent Fuel Handling Tool Storage Rack**

The spent fuel handling tool storage rack is used to store the spent fuel handling tool in the spent fuel pool area for access by the spent fuel handling machine.

- **New Fuel Handling Tool (Figure II-F-18)**

The new fuel handling tool is used to transfer the new fuel assembly from the new fuel shipping container to the new fuel inspection device or the new fuel racks. It is also used to transfer the new fuel assemblies from the new fuel racks to the new fuel elevator. It may also be used to handle the dummy fuel assembly prior to its placement in the pool.

ABB
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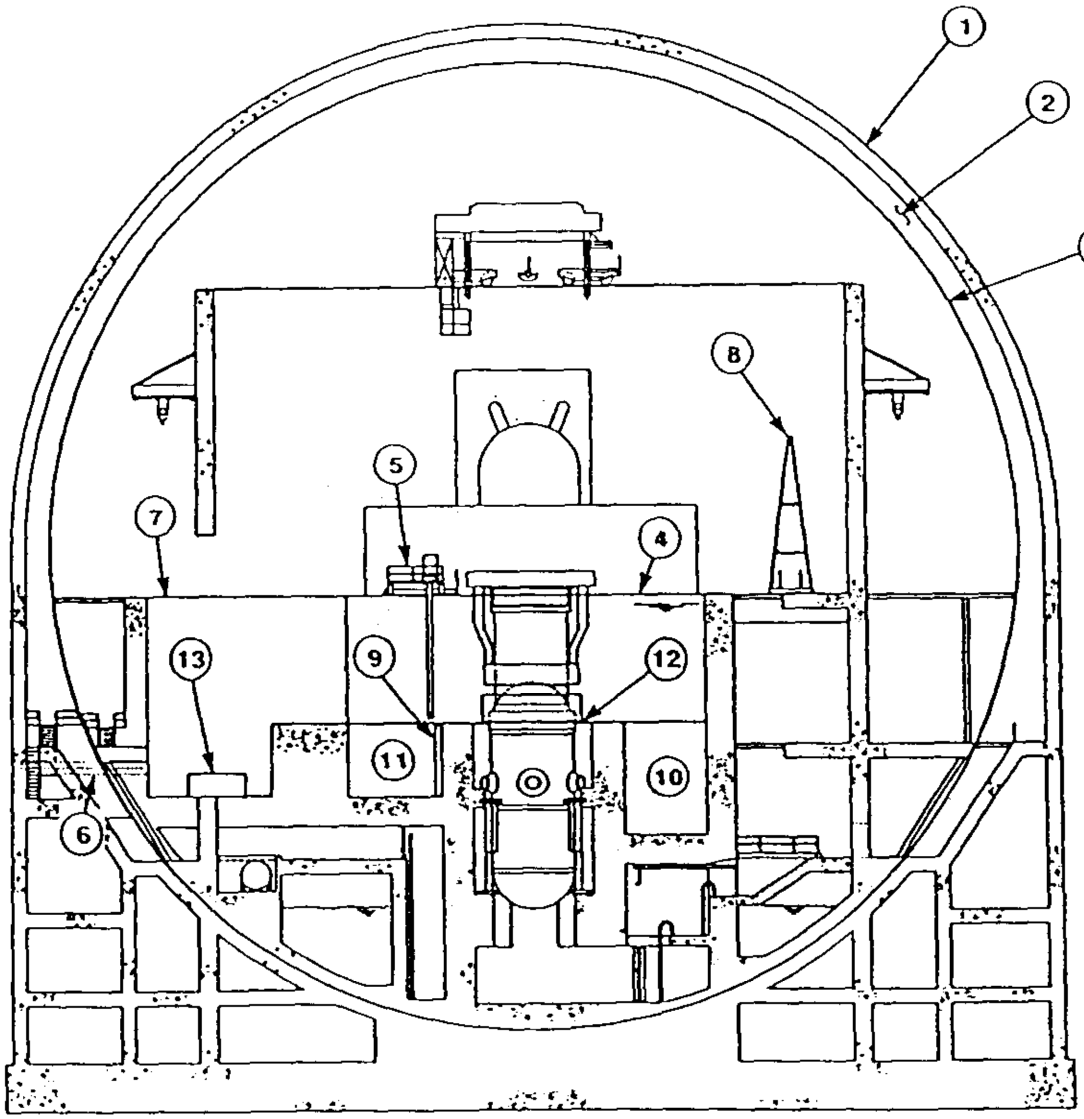
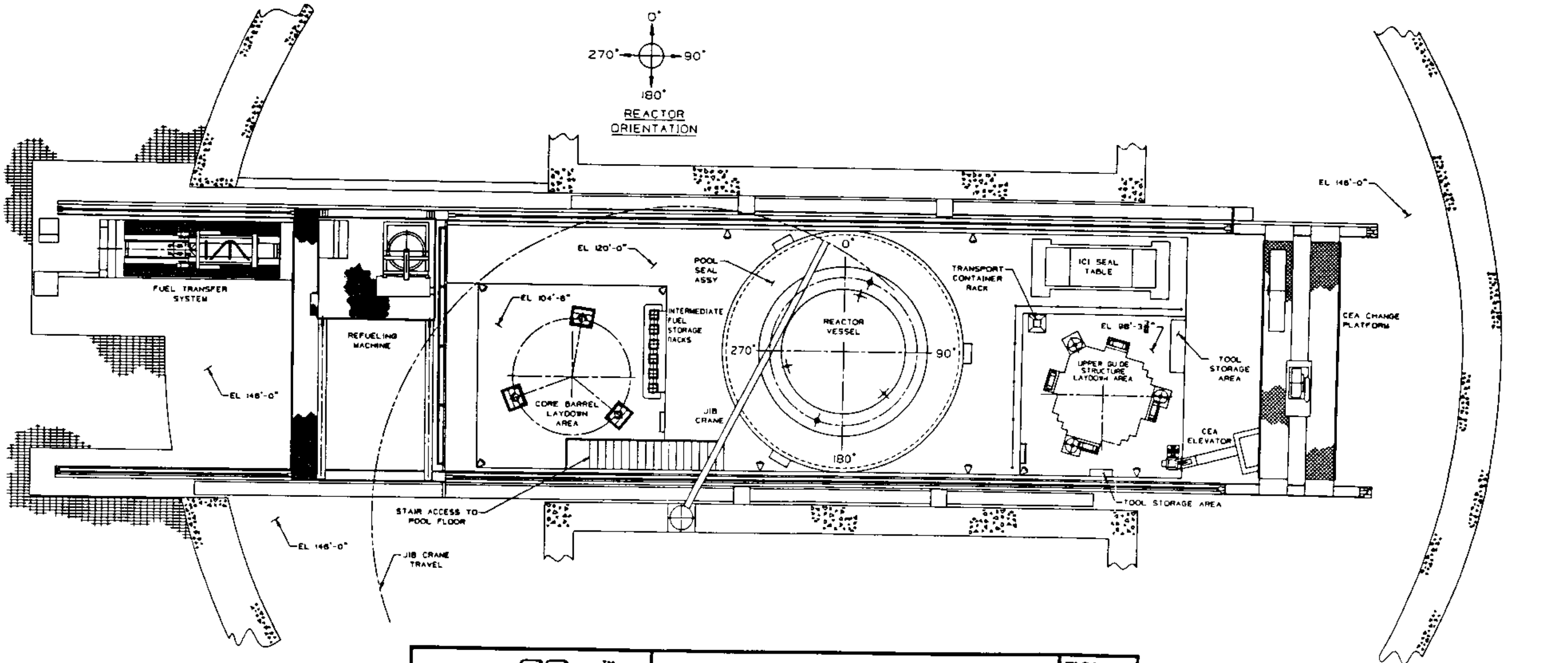
1. REACTOR BUILDING
2. ANNULUS
3. CONTAINMENT
4. OPERATING FLOOR
5. REFUELING MACHINE
6. FUEL TRANSFER TUBE
7. REFUELING CANAL
8. CEA CHANGE PLATFORM
9. TEMPORARY FUEL ASSEMBLY STORAGE RACK
10. UPPER GUIDE STRUCTURE LAYDOWN AREA
11. CORE SUPPORT BARREL LAYDOWN AREA
12. POOL SEAL
13. FUEL TRANSFER SYSTEM UPENDER



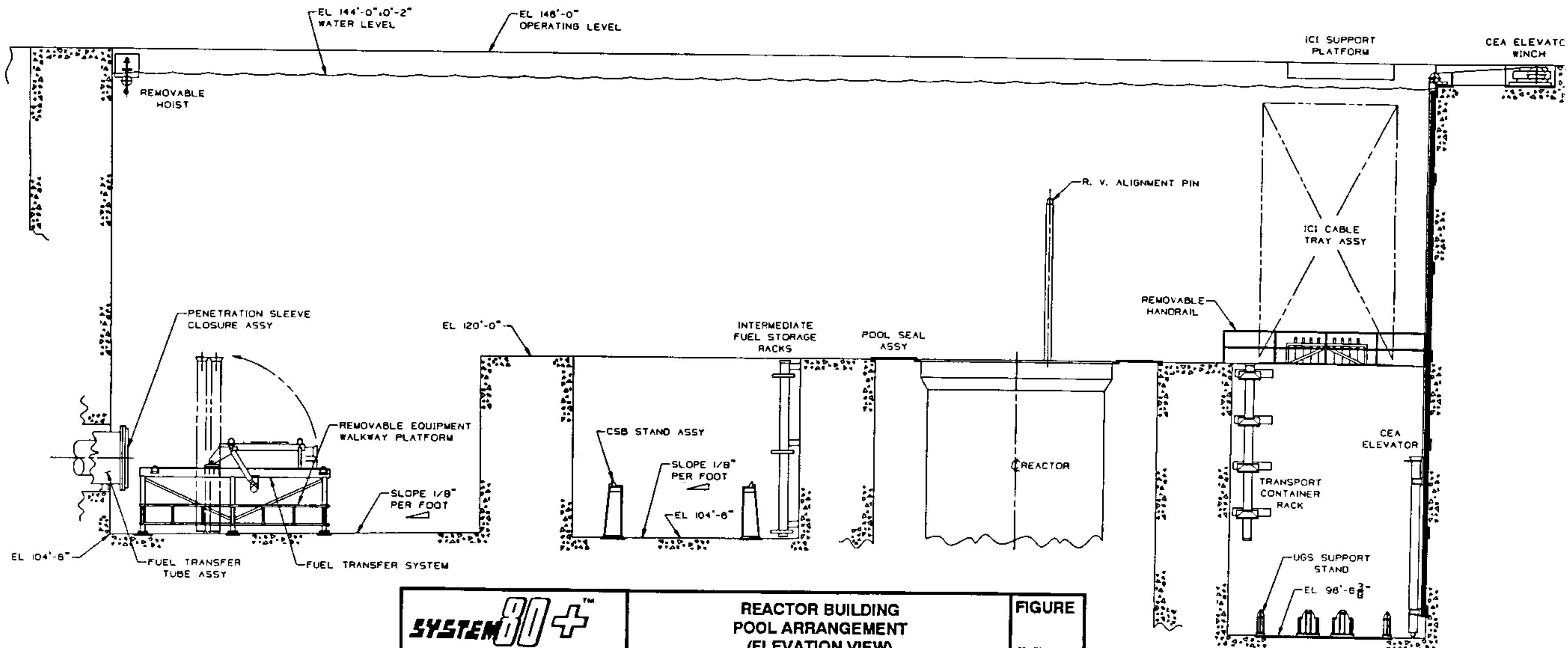
SYSTEM 80+™	REACTOR BUILDING REFUELING EQUIPMENT ARRANGEMENT	FIGURE II-F-1



SYSTEM 80+™

REACTOR BUILDING POOL ARRANGEMENT (PLAN VIEW)

FIGURE II-F-2



SYSTEM 80+™	REACTOR BUILDING POOL ARRANGEMENT (ELEVATION VIEW)	FIGURE II-F-3

ABB
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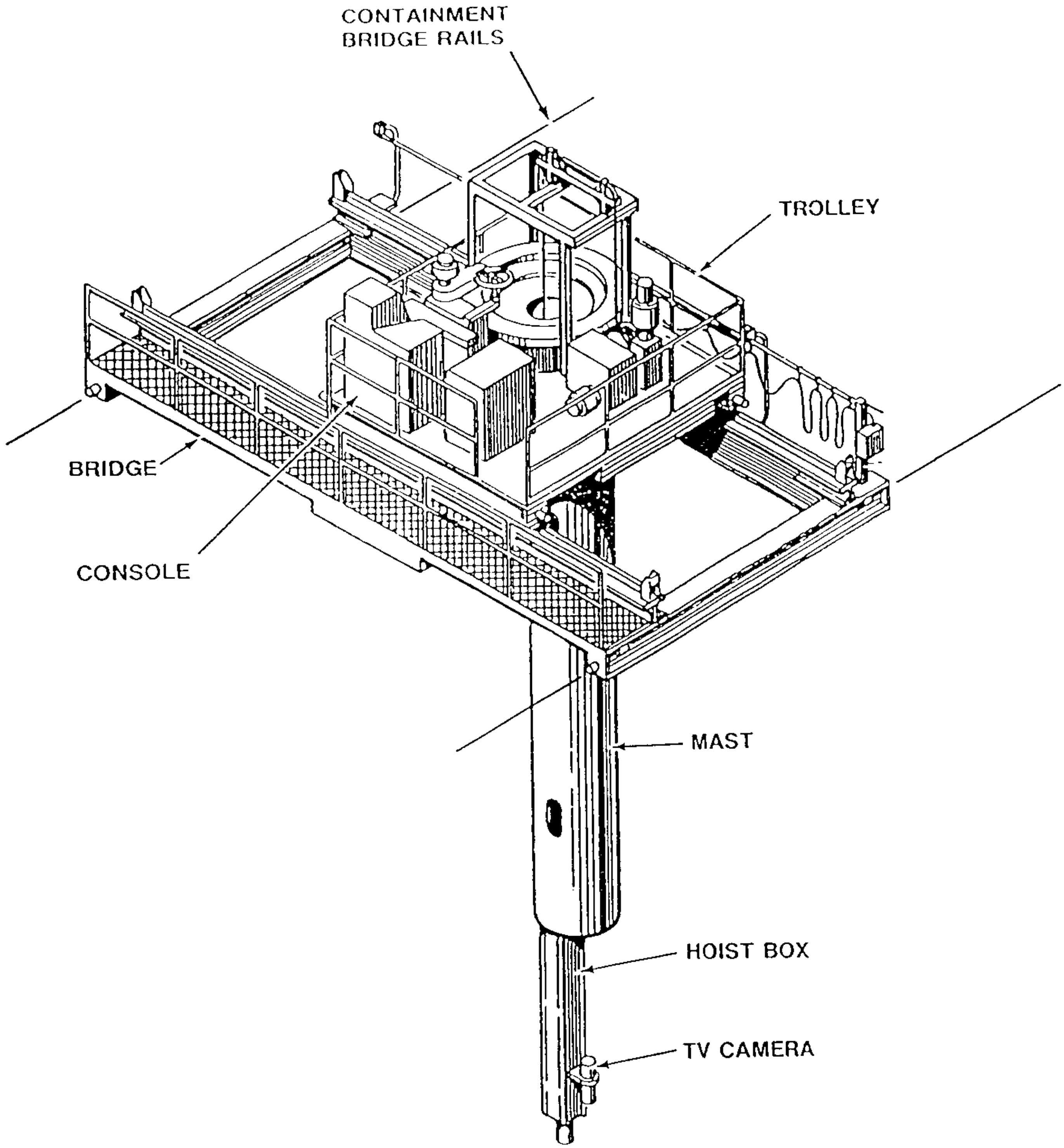

SYSTEM 80+™	REFUELING MACHINE	FIGURE II-F-4

ABB
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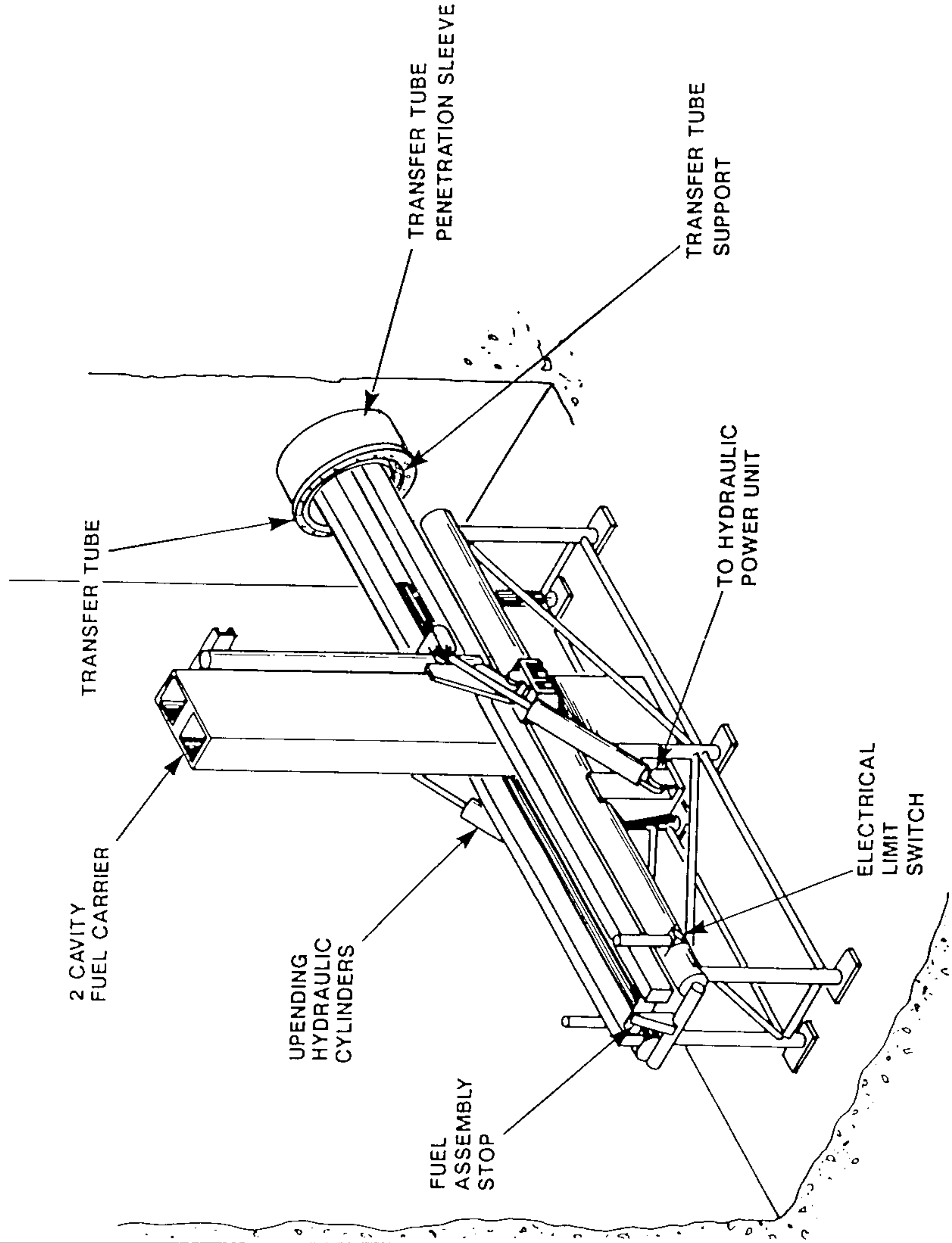

SYSTEM 80+™	FUEL TRANSFER SYSTEM CARRIAGE AND UPENDER (REACTOR BUILDING)	FIGURE II-F-5

ABB
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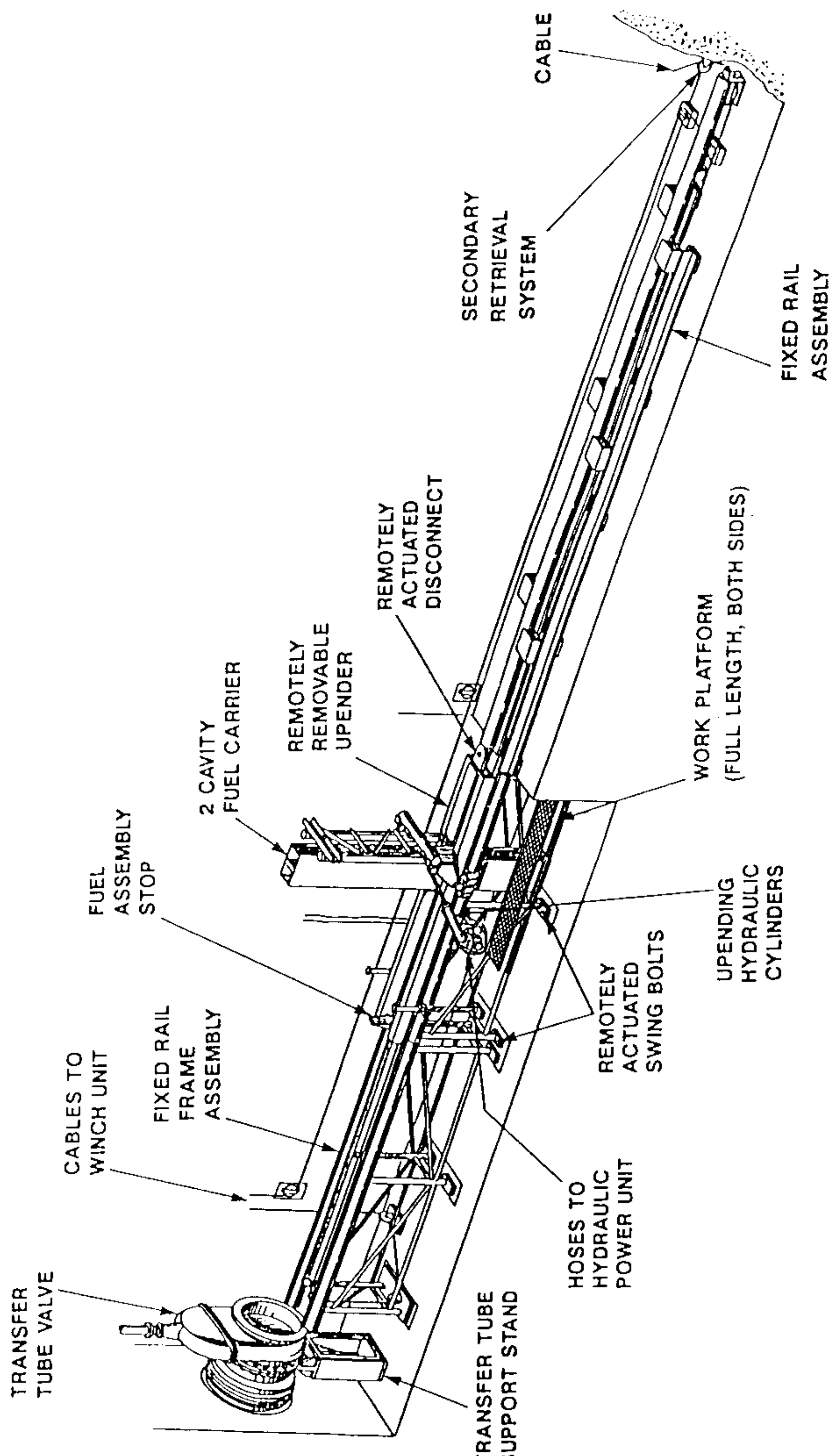

SYSTEM 80+™	FUEL TRANSFER SYSTEM CARRIAGE AND UPENDER (NUCLEAR ANNEX SYSTEM)	FIGURE II-F-6

ABB
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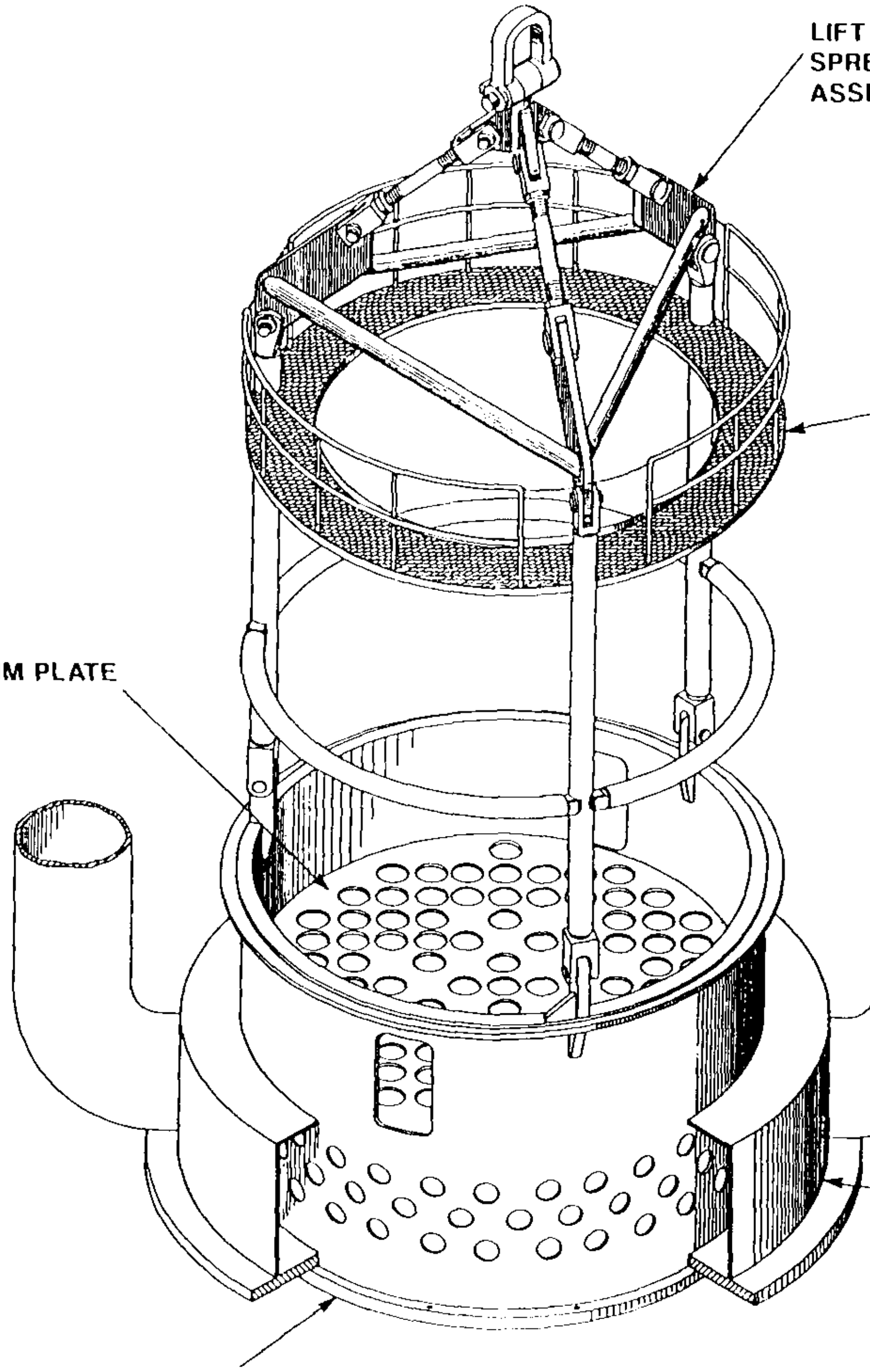
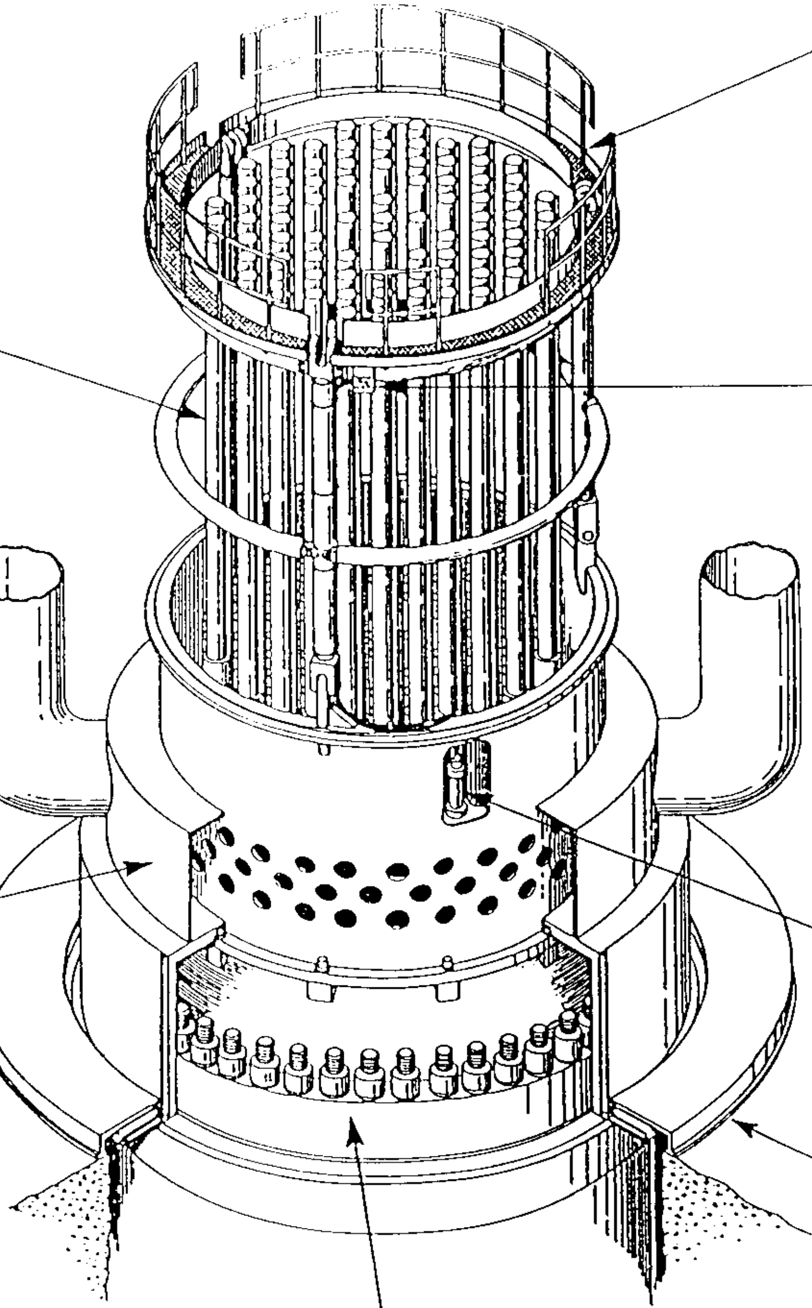
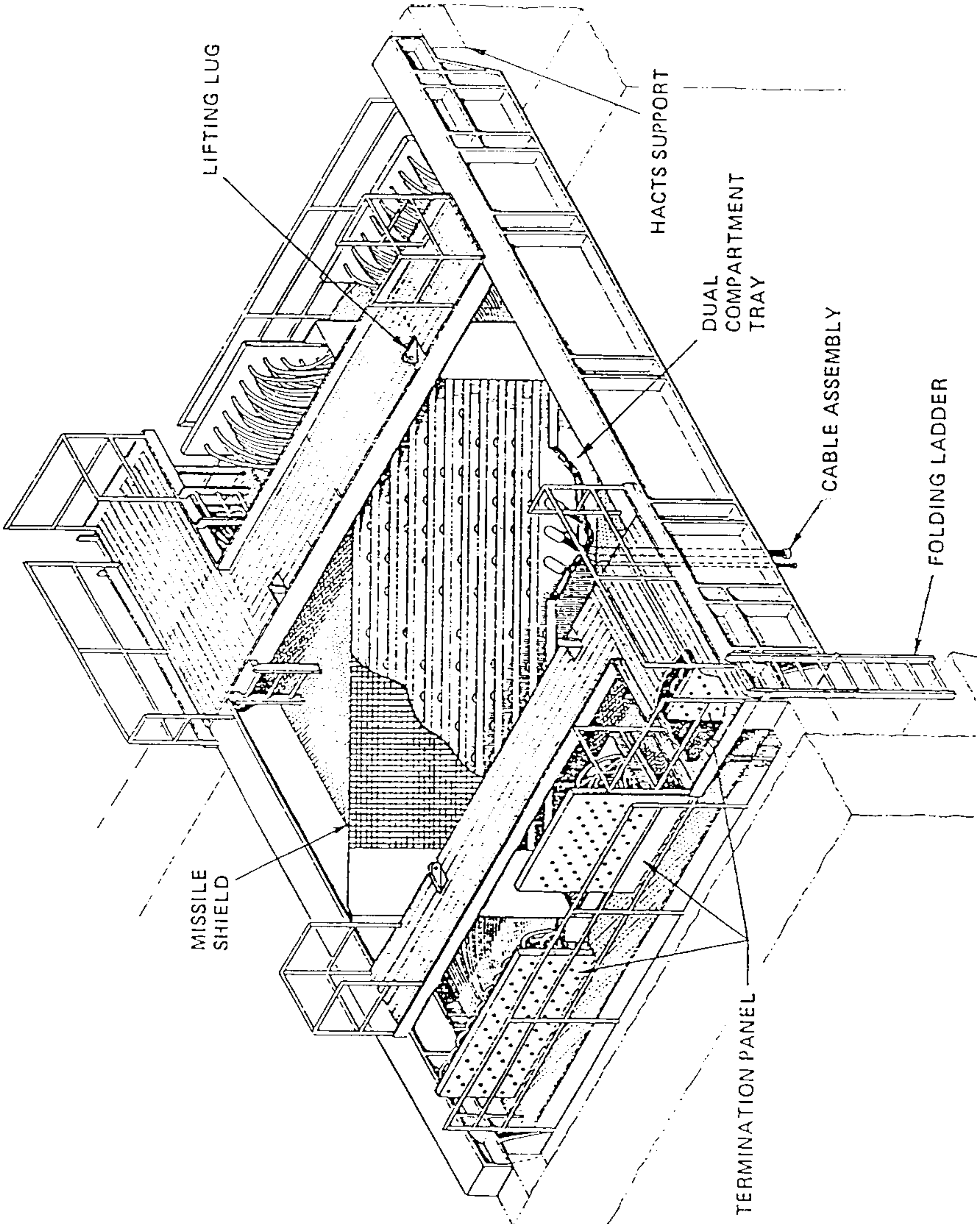


SYSTEM 80+™	CLOSURE HEAD LIFTING RIG ASSEMBLY	FIGURE II-F-7



SYSTEM 80+™	CEDM COOLING MANIFOLD	FIGURE II-F-8

ABB
ASEA BROWN BOVERI



SYSTEM 80+™	HEAD AREA CABLE TRAY SYSTEM	FIGURE II-F-9

ABB
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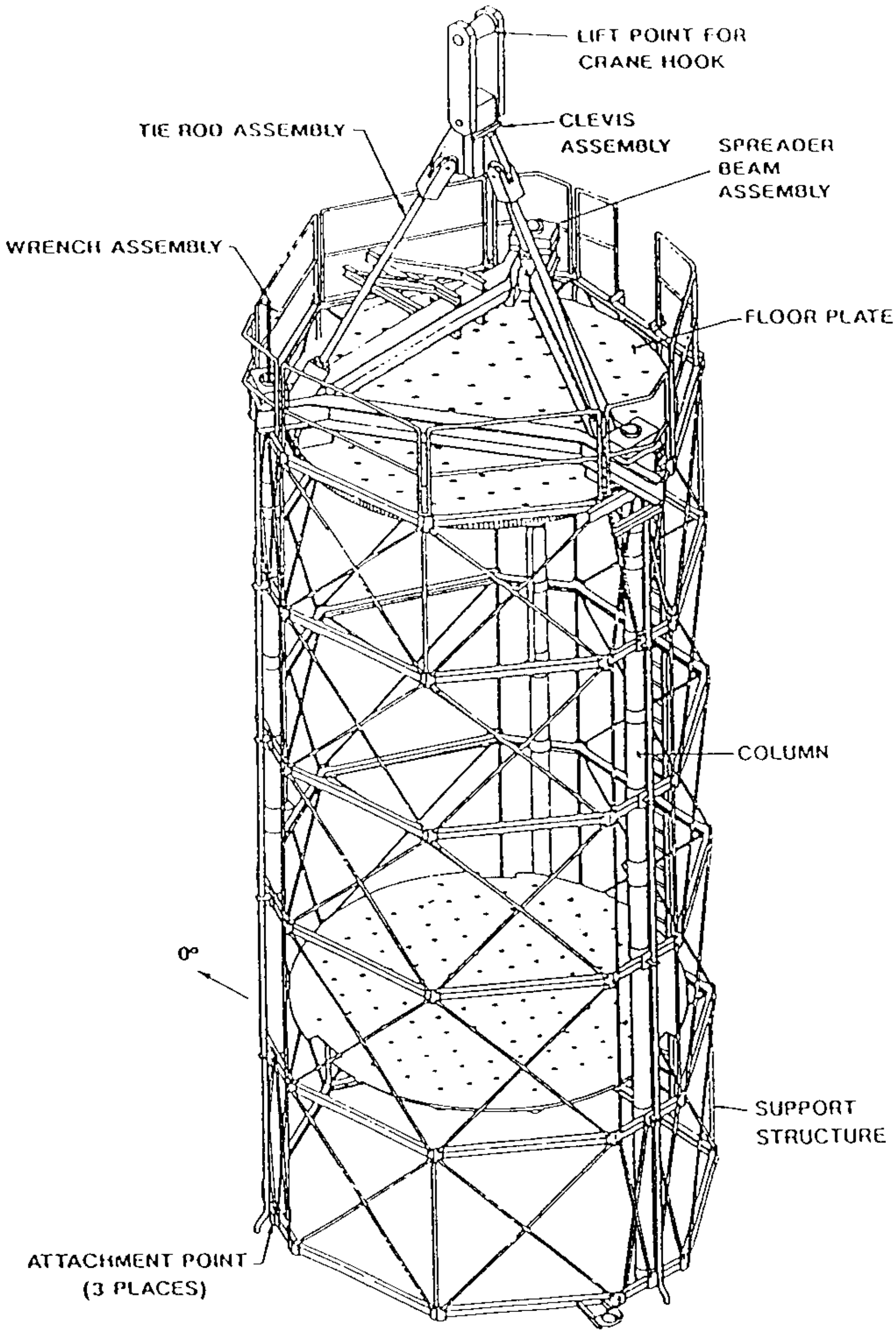
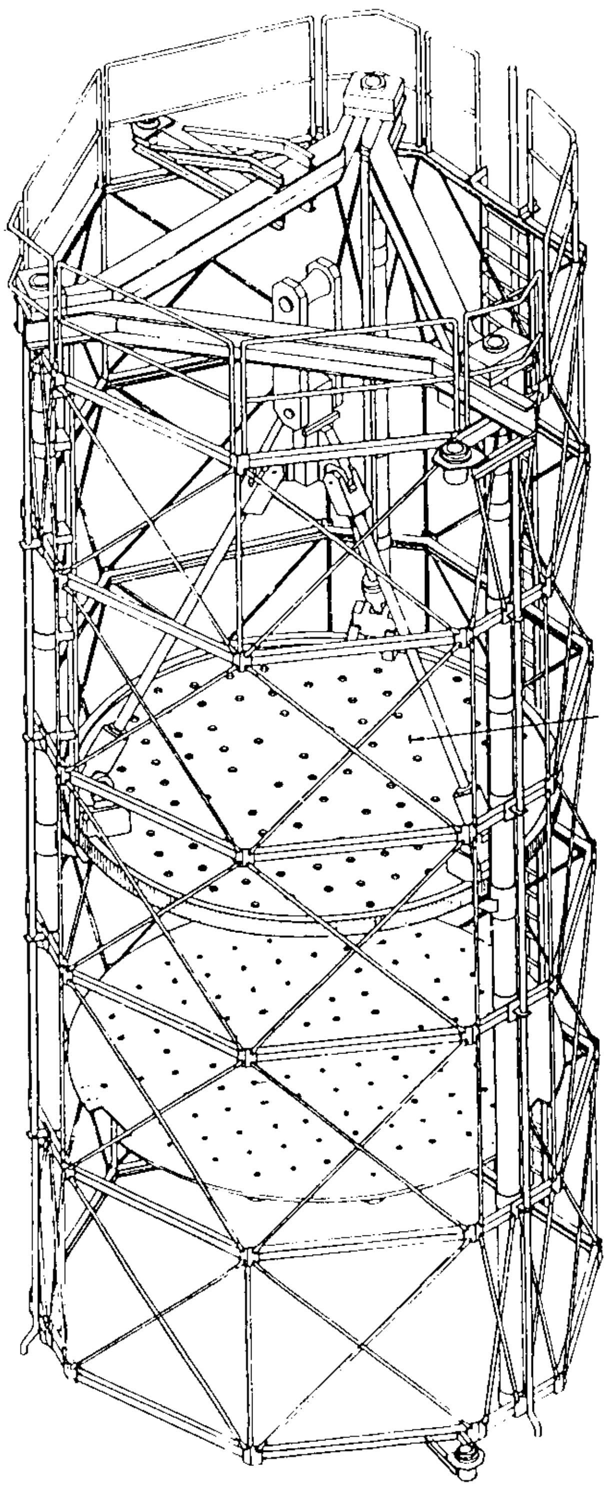
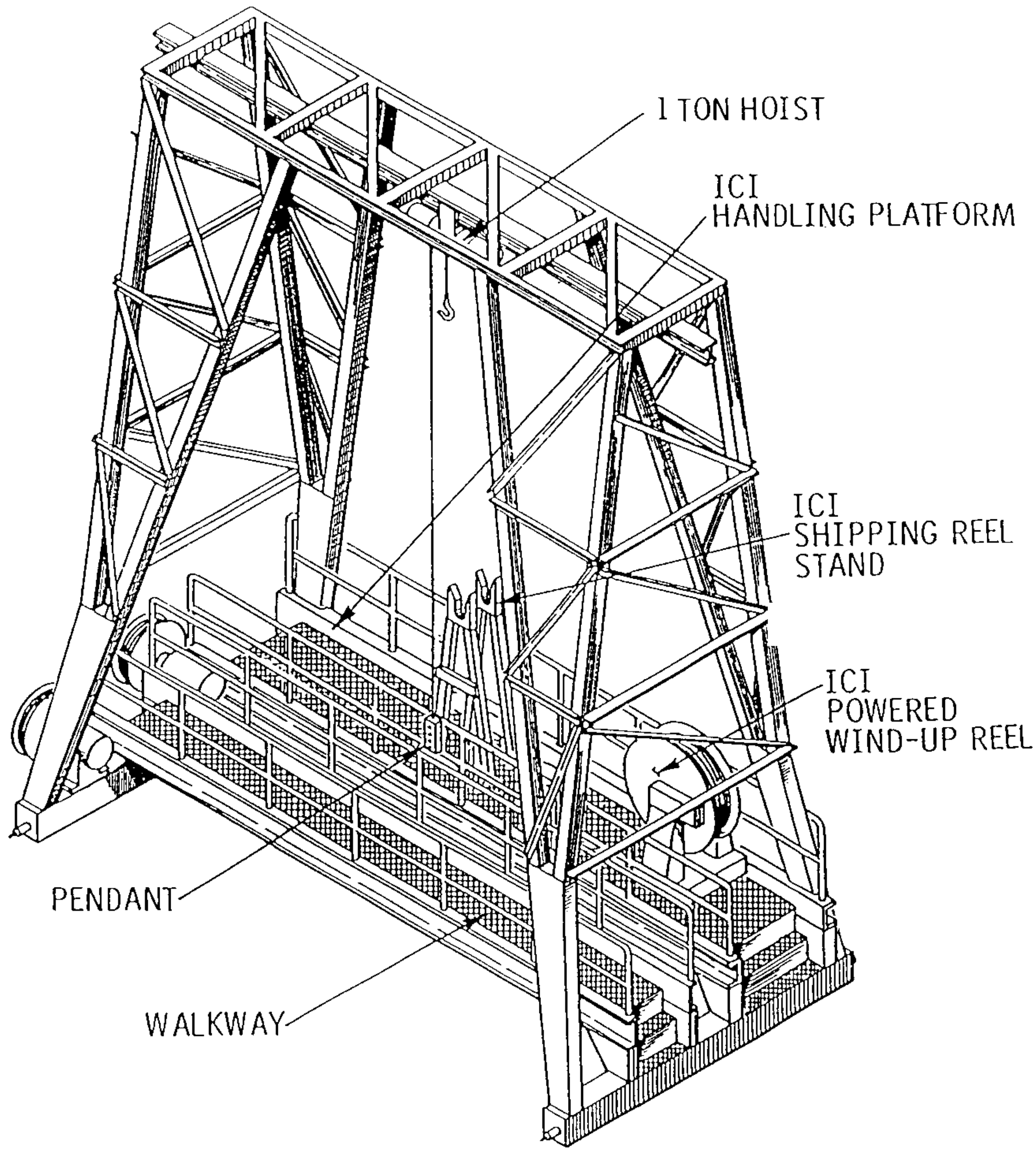


SYSTEM 80+™	UPPER GUIDE STRUCTURE LIFT RIG ASSEMBLY (SHEET 1 OF 2)	FIGURE II-F-10

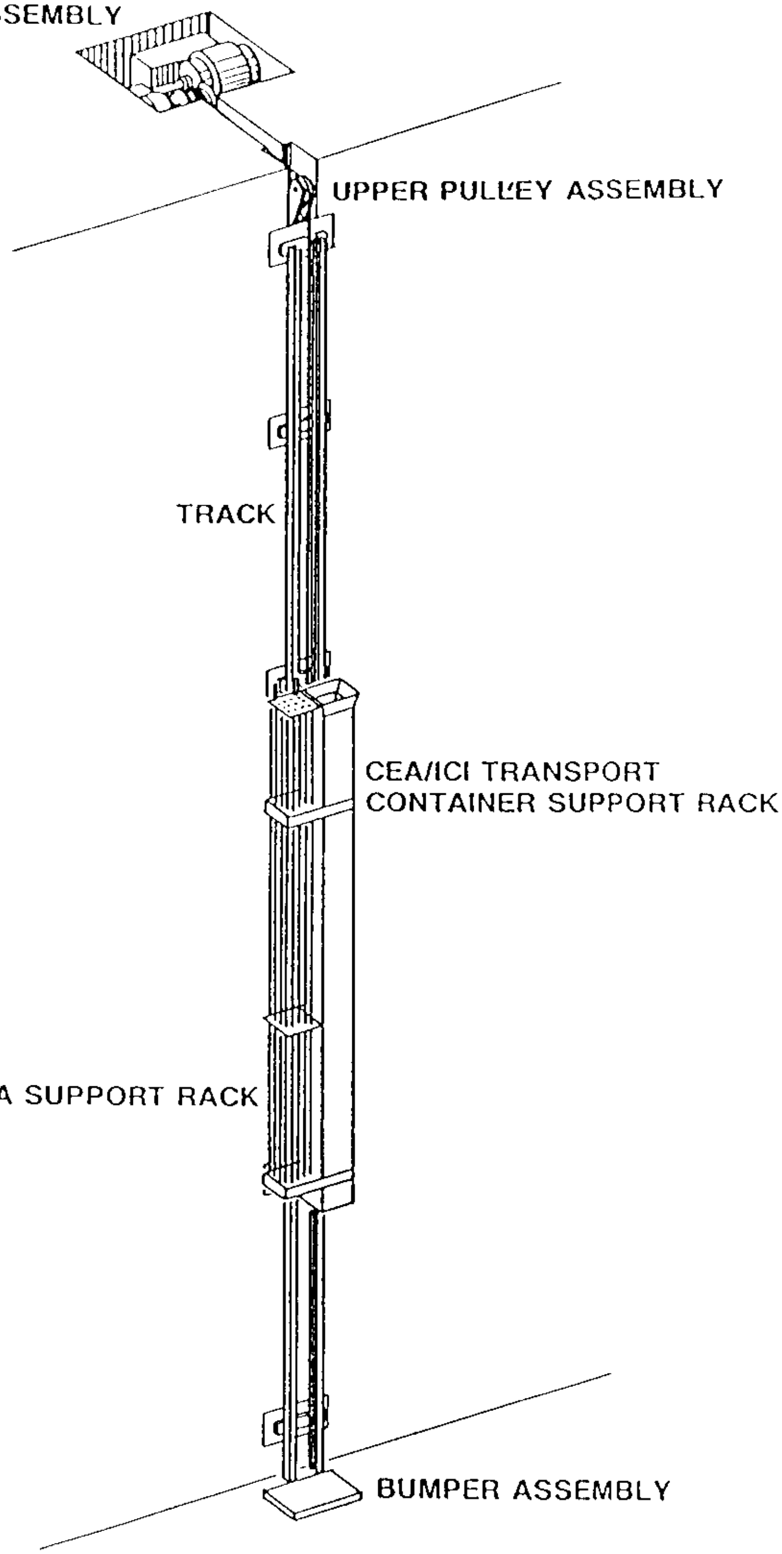


SYSTEM 80+™	UPPER GUIDE STRUCTURE LIFT RIG ASSEMBLY (SHEET 2 OF 2)	FIGURE II-F-10

ABB
ASEA BROWN BOVERI

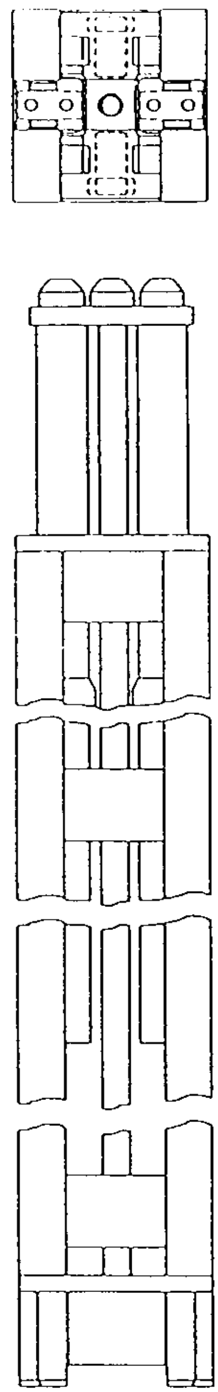


SYSTEM 80+™	CEA CHANGE PLATFORM	FIGURE II-F-11



SYSTEM 80+™	CEA ELEVATOR	FIGURE II-F-12

ASEA BROWN BOVERI



SYSTEM 80+™	DUMMY FUEL ASSEMBLY	FIGURE II-F-13

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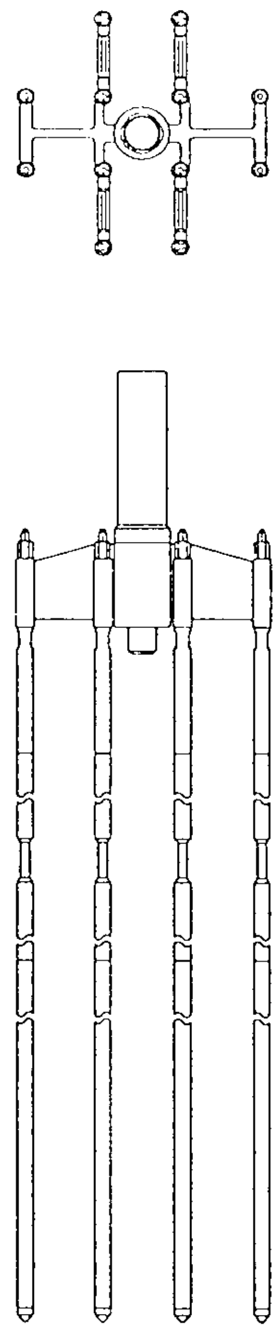

SYSTEM 80+™	12 ROD DUMMY CEA	FIGURE II-F-14

ABB
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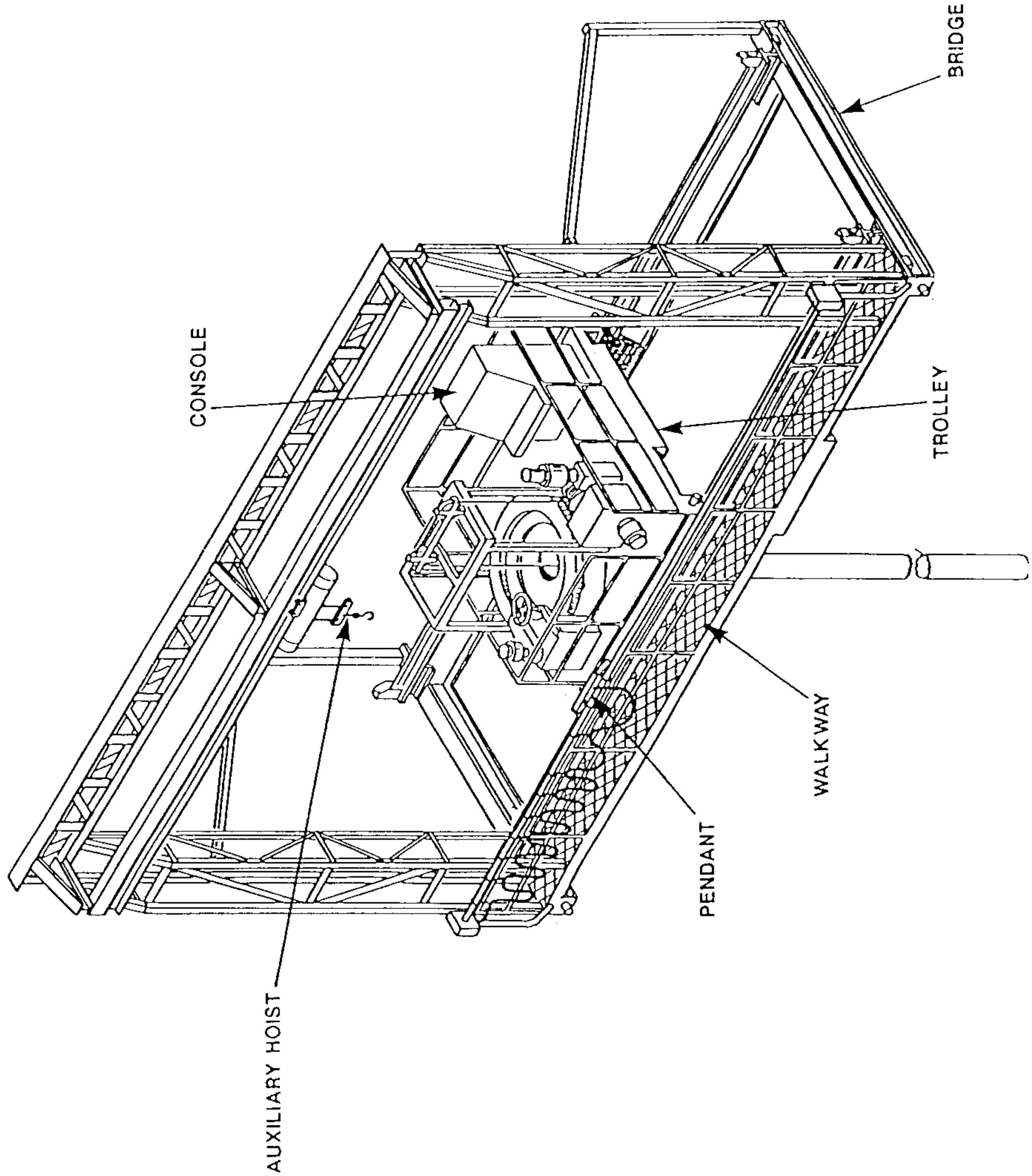
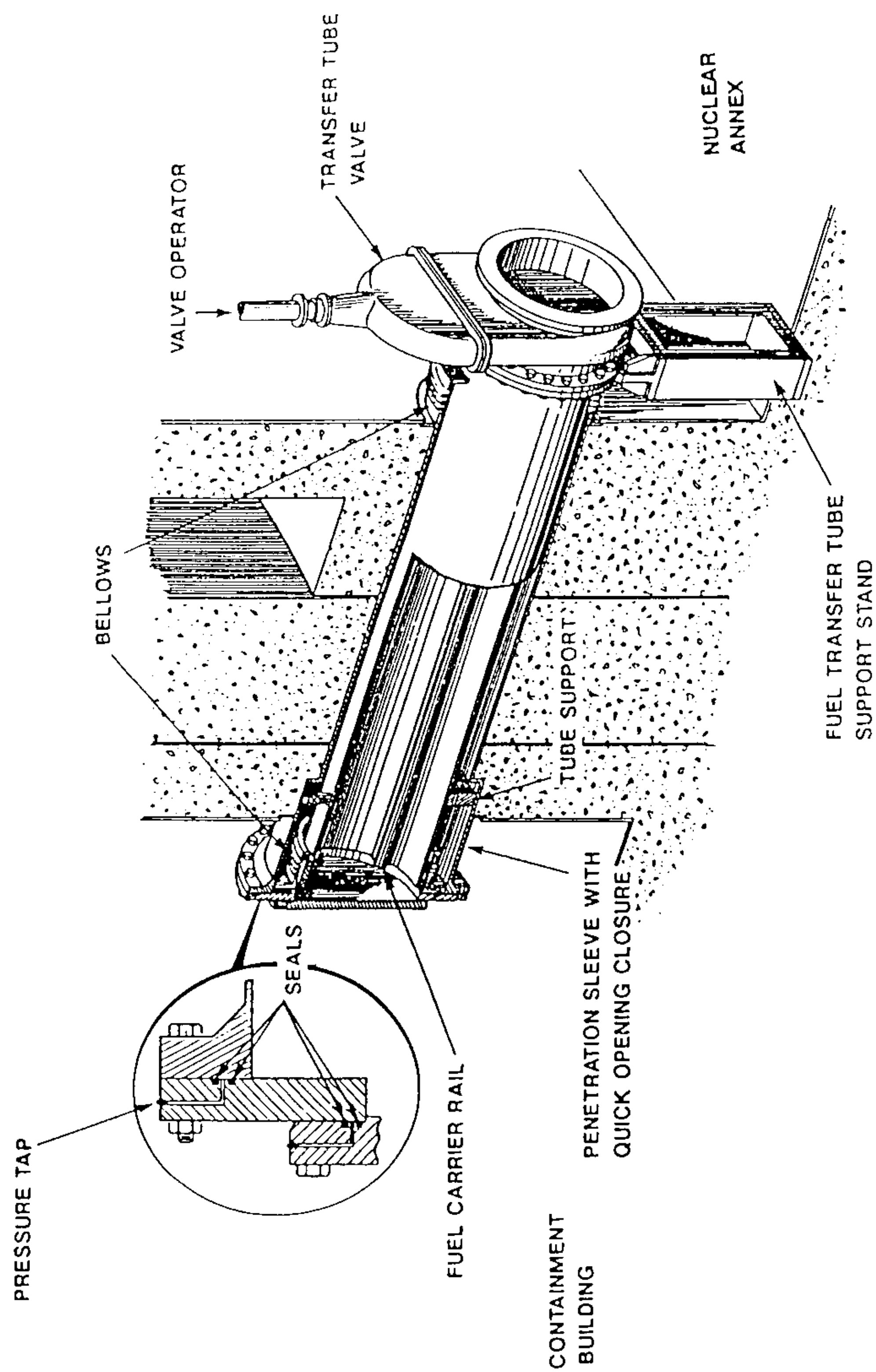
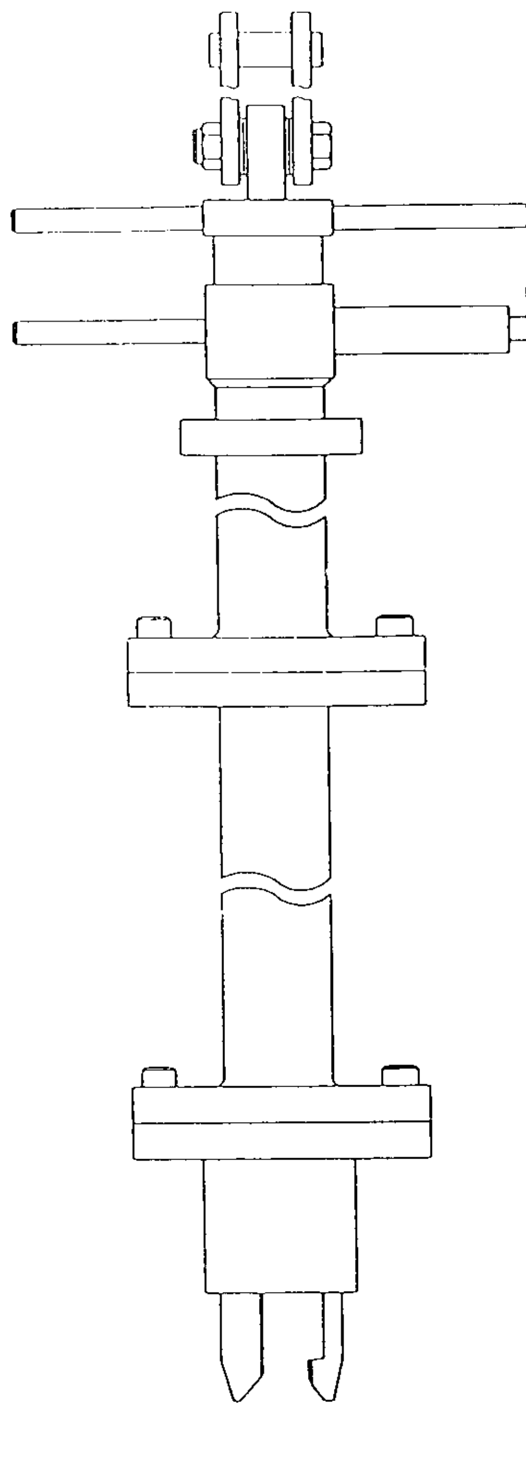


SYSTEM 80+™	SPENT FUEL HANDLING MACHINE	FIGURE II-F-15

ASEA BROWN BOVERI

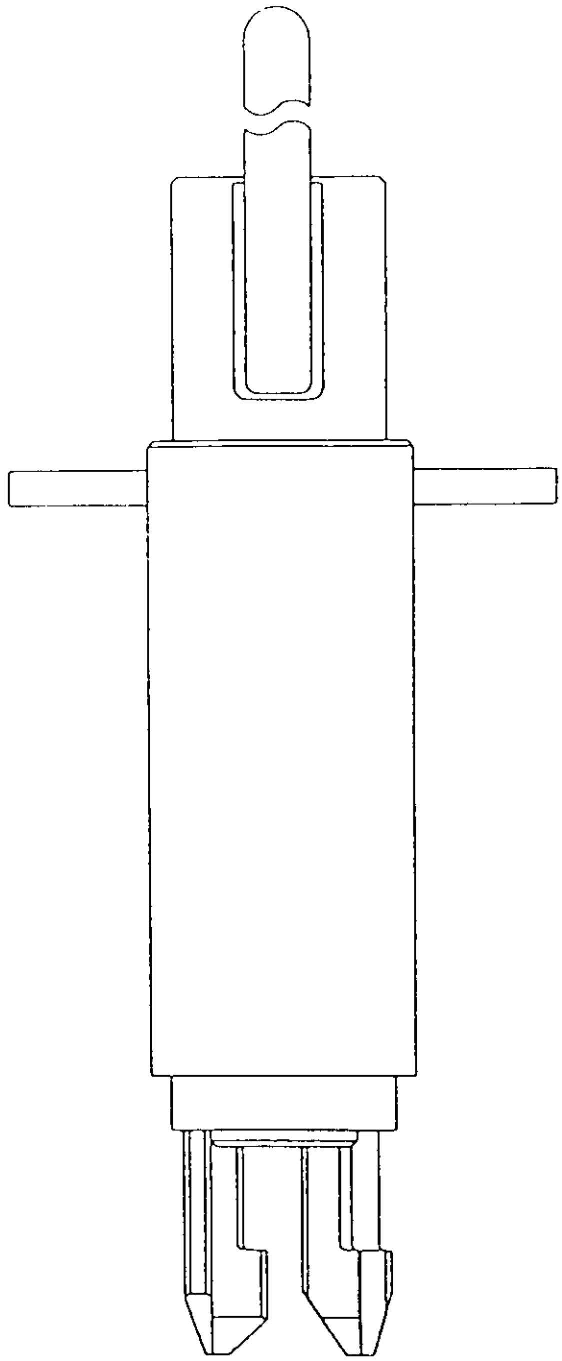


SYSTEM 80+™	FUEL TRANSFER TUBE ASSEMBLY	FIGURE II-F-16



SYSTEM 80+™	SPENT FUEL HANDLING TOOL	FIGURE II-F-17

ABB
ASEA BROWN BOVERI



SYSTEM 80+™	NEW FUEL HANDLING TOOL	FIGURE II-F-18

G. BALANCE OF PLANT SYSTEMS

1. General

This section provides a description of the turbine generator and equipment.

The layout of the turbine island integrates the NSSS, the turbine and auxiliary systems into a highly functional nuclear power station with the turbine island performing no safety functions.

The components of the steam power plant, including turbine generator, condenser system, and preheater system are housed in the Turbine Building. The main feedwater pumps, the deaerator/feedwater tank, and the condensate cleaning plant are arranged in an annex to the main turbine building.

The ABB turbine plant consists of a double-flow high-pressure turbine and 3 double-flow low-pressure turbines, the latter with a total exhaust area of 113 m^2 (1,216 ft^2). Of the nominal output of the turbine, about 40 percent is developed in the high-pressure turbine and the remainder in the low-pressure turbines.

The high-pressure turbine basically consists of a cast outer casing, a cast steel inner casing carrying the guide vanes, a forged and welded rotor fitted with the rotating blades and one shaft gland at each end of the rotor. The outer and inner casings are both split in the horizontal plane, and the two halves are bolted together.

The LP turbine consists of 3 double-flow cylinders. Each cylinder is essentially of double casing construction with the inner and outer casings horizontally divided and bolted together with flanged joints.

The high-pressure rotor and each of the low-pressure rotors consist of several forgings which are welded together. The advantage of this design is that the relatively small forgings making up a rotor are easier to manufacture at a high level of quality. The scrapping of a forging would have a minimal impact on delivery intervals compared to a fault in a one-piece rotor forging.

The turbine rotors are supported in journal bearings mounted in bearing pedestals. One bearing is provided for each shaft between two rotors, and this provides a well-defined location of the shaft and well-defined bearing loadings. The thrust bearing is located after the high-pressure rotor and is independent of other bearings.

Steam from the reactor is supplied through four main steam pipes connected to four identical steam inlet valves upstream of the high-pressure turbine. Part of the steam is extracted upstream of the steam inlet valve and is used for reheating the steam after the high-pressure turbine. This arrangement increases the efficiency of the turbine plant and reduces the moisture content in the low-pressure turbines. The exhaust steam from the HP turbine passes through in line high velocity moisture separations and reheaters before entering the three parallel LP turbine cylinders.

In order to protect the turbine from overspeed in the event of emergency tripping, intercept control and stop valves are fitted in the transfer pipes between the reheater and the low-pressure turbines. In addition, the turbine is capable of full load rejection utilizing approximately 55 percent steam dump to the condenser in conjunction with the reactor cutback system, with no reliance on the main steam safety valves unless the condenser is unavailable. The control, shut-off and dump valves of the turbine are controlled entirely electro-hydraulically and incorporate an electronic governor for controlling the turbine speed and the reactor pressure by means of the control and dump valves of the turbine. An electric signal from the governor controls the steam valve through electro-hydraulic converters.

The generator is of 4-pole design of ABB manufacture. Direct water cooling is employed for the stator windings. The rotor windings and the stator core are cooled with hydrogen. The excitation equipment is of the static type. The system consists of an excitation transformer, a thyristor rectifier bridge and a field circuit breaker and voltage regulator.

Electrically, the turbine plant is equipped with busbars with single-phase enclosure between the generator and the main transformer, with tappings to plant transformers.

The turbine plant is completely automatic and is supervised from the Main Control Room. The automatic control equipment carries out automatic run-up and shut-down of the turbine plant, the change-over to stand-by units, and reversion to a safe operating status in the event of disturbances in the process.

The turbine plant also includes a complete range of service systems, such as ventilation, fire-fighting, make-up water, tap water, general service and auxiliary cooling water, sampling, etc.

2. Steam Turbine

a. <u>General</u>

There are four reaction axial-flow turbines,

- one double-flow high pressure (HP)
- three double-flow low pressure (LP 1-3)

Running speed is 1800 rpm with clockwise rotation seen from HP turbine end towards the generator. The turbine rotors and the generator rotor are connected by integral forged coupling flanges to form the rotor train.

A preliminary heat balance diagram is included as Figure II-G-1.

b. <u>Turbine Governor, Turbine Protection and Turbine Supervisory Equipment</u>

The turbine controller TURBOTROL has the following features:

- Saving of time and costs due to automatic run-up and loading

- High plant availability due to automatic limiters
- Coordinates with the turbine bypass system
- Coordinates with DCS and overall control systems
- Modern control structure which reacts quickly and efficiently to disturbances in the grid but prevents unnecessary control operations during normal operation.

(1) Principle of Operation

The main parts of the TURBOTROL are the speed governer with automatic run-up and the load adjuster with load and gradient preselection. The actuating variable of the speed controller and the load set value of the automatic loading device are added together and the sum becomes the actuating variable for the main control valves.

(2) Reheater Pressure Control

The purpose of the reheater pressure control is to control the pressure in the heating steam system to a given set value.

The pressure set value is derived from the actual load set value by the function generator in a way that the reheater control valves are closed for low load and are fully opened at 50% load.

In order to prevent high stresses in the reheater a temperature limiter is provided to reduce the pressure set value over the minimum value selector. From the differential temperature measured at the tubesheets, which is equivalent to the thermal stresses, the reduced pressure set value is derived by the function generator.

c. Main Steam and Gland Sealing System

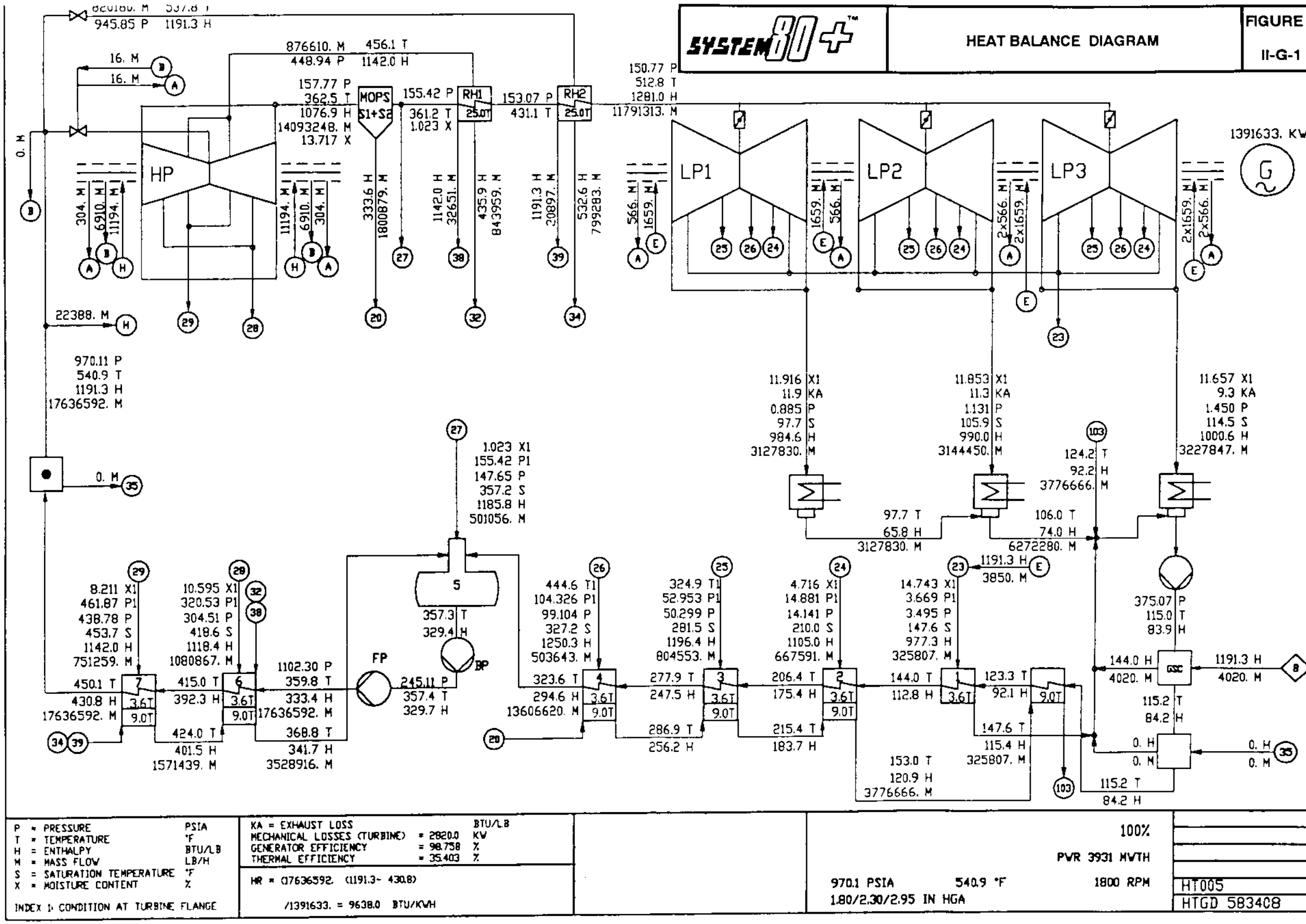
(1) Main Steam System

The Main Steam System is shown on Flow Diagrams FSK 3-1 A, B, C & D (Figures II-G-2-4 sheets).

The water/steam circuit incorporates those systems of the secondary section of the nuclear power plant directly involved in power generation. The main systems concerned are those for main steam, main condensate, LP and HP feedwater heating, feedwater, extraction steam, and heating condensate.

The main steam lines from each of the four main steam isolation valves are led to a header inside the turbine building which is connected by four pipes to the four main admission valves of the HP turbine.

The main steam isolation valves are located inside the Main Steam Valve House.

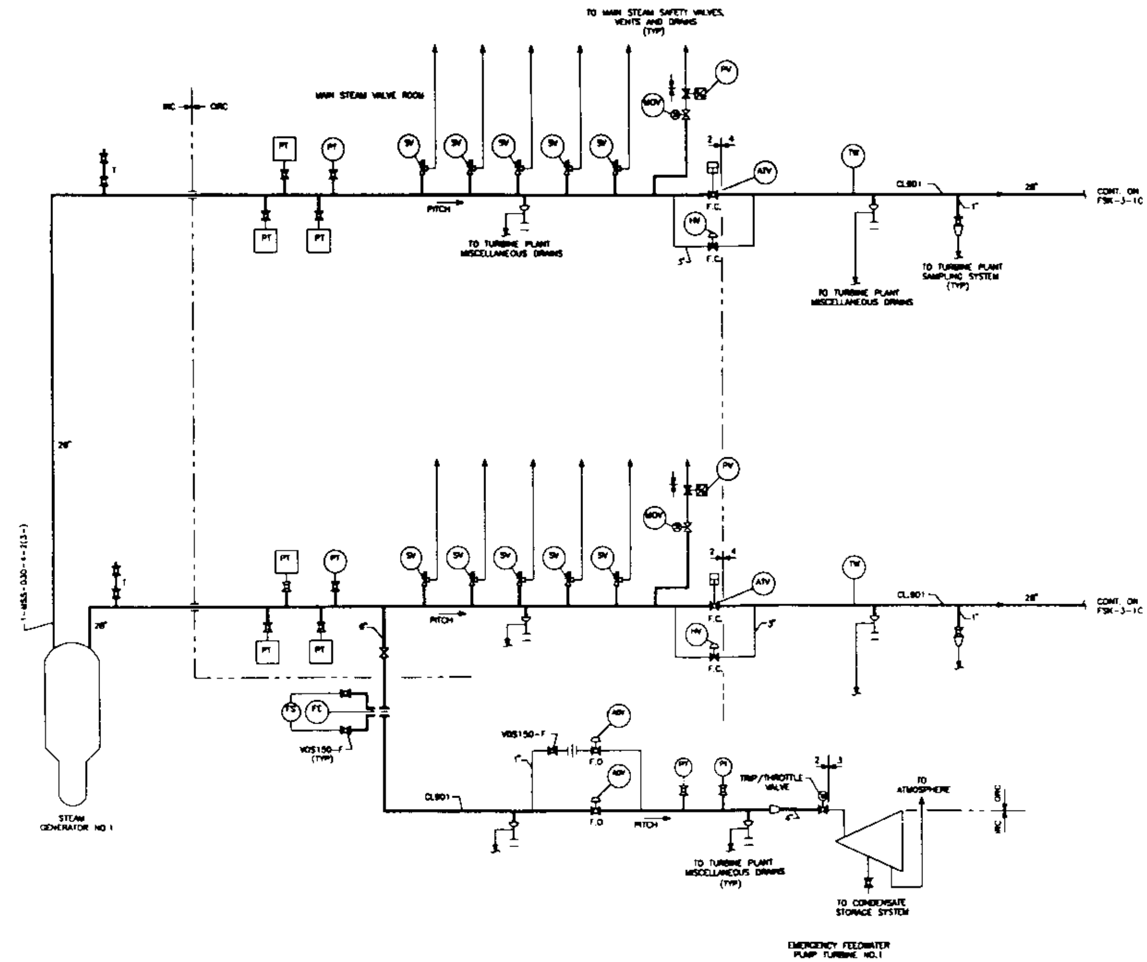


SYSTEM 80+
HEAT BALANCE DIAGRAM
FIGURE II-G-1
HP
LP1
LP2
LP3
MOPS S1+S2
RH1
RH2
FP
BP
GSC
1391633. KW
945.85 P 1191.3 H
876610. M 456.1 T
448.94 P 1142.0 H
157.77 P
362.5 T
1076.9 H
14093248. M
13.717 X
155.42 P
361.2 T
1.023 X
153.07 P
431.1 T
150.77 P
512.8 T
1281.0 H
11791313. M
970.11 P
540.9 T
1191.3 H
17636592. M
22388. M
11.916 X1
11.9 KA
0.885 P
97.7 S
984.6 H
3127830. M
11.853 X1
11.3 KA
1.131 P
105.9 S
990.0 H
3144450. M
11.657 X1
9.3 KA
1.450 P
114.5 S
1000.6 H
3227847. M
1.023 X1
155.42 P1
147.65 P
357.2 S
1185.8 H
501056. M
P = PRESSURE PSIA
T = TEMPERATURE °F
H = ENTHALPY BTU/LB
M = MASS FLOW LB/H
S = SATURATION TEMPERATURE °F
X = MOISTURE CONTENT %
INDEX 1: CONDITION AT TURBINE FLANGE
KA = EXHAUST LOSS BTU/LB
MECHANICAL LOSSES (TURBINE) = 2820.0 KW
GENERATOR EFFICIENCY = 98.758 %
THERMAL EFFICIENCY = 35.403 %
HR = (17636592. (1191.3- 430.8)
/1391633. = 9638.0 BTU/KWH
100%
PWR 3931 MWTH
970.1 PSIA 540.9 °F 1800 RPM
1.80/2.30/2.95 IN HGA
HT005
HTGD 583408

031B2.02-FSK-3-1A-1

NOTES:

1. ALL LOCAL VENT, DRAIN, TEST AND PRESSURE CONNECTIONS ARE 3/4" UNLESS OTHERWISE NOTED. ASSOCIATED VALVES ARE VOS150-F.
2. ALL PIPING IS CLB01 UNLESS OTHERWISE NOTED.
3. ALL TEMPERATURE CONNECTIONS ARE 1 1/2" SW.



PRELIMINARY

FIGURE II-G-2
MAIN STEAM
(SHEET 1 OF 4)

STONE & WEBSTER ENGINEERING CORP

031B2.02-FSK-3-1A-1

FILE NAME FSK031A

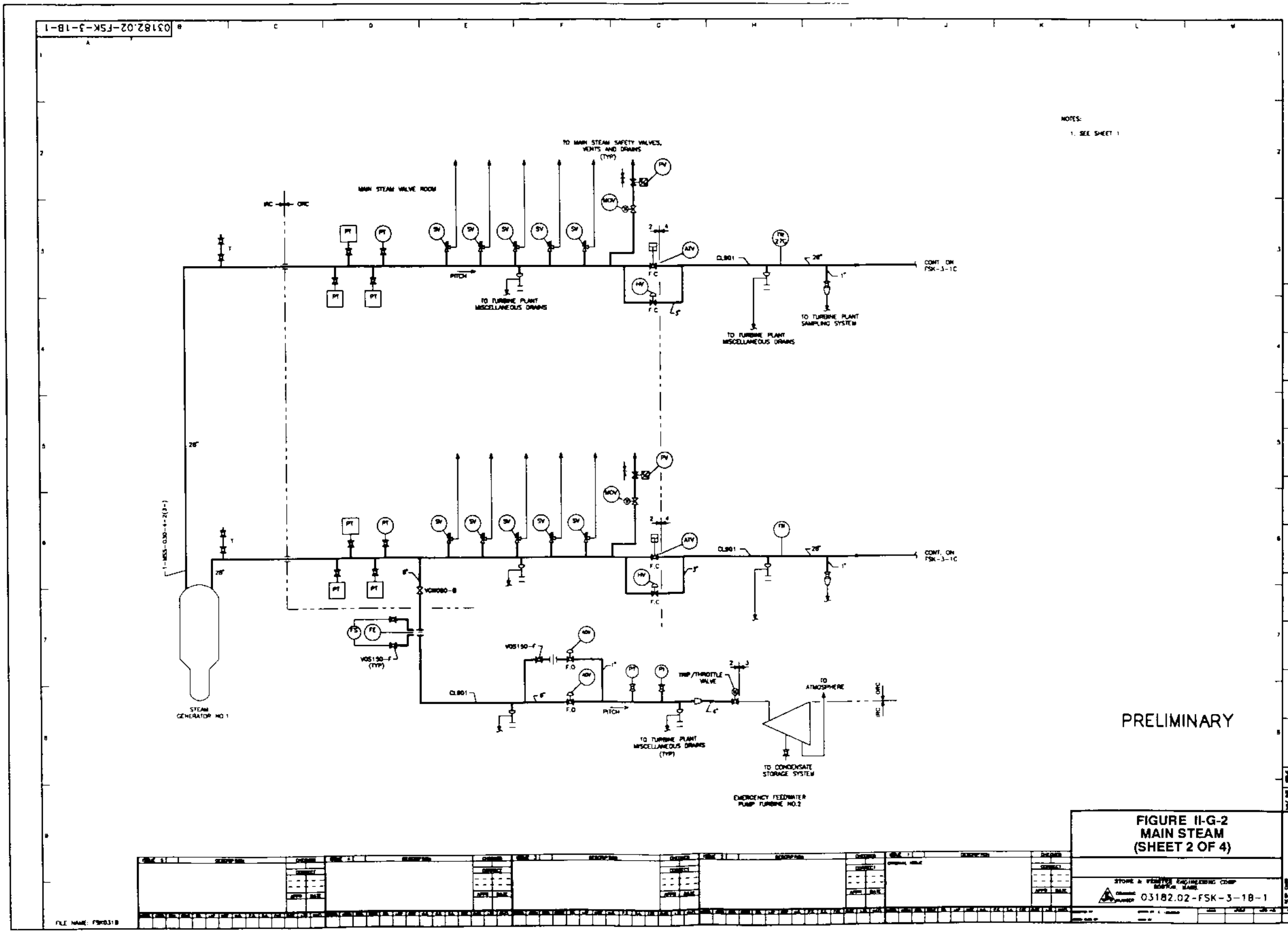


FIGURE II-G-2
MAIN STEAM
(SHEET 2 OF 4)

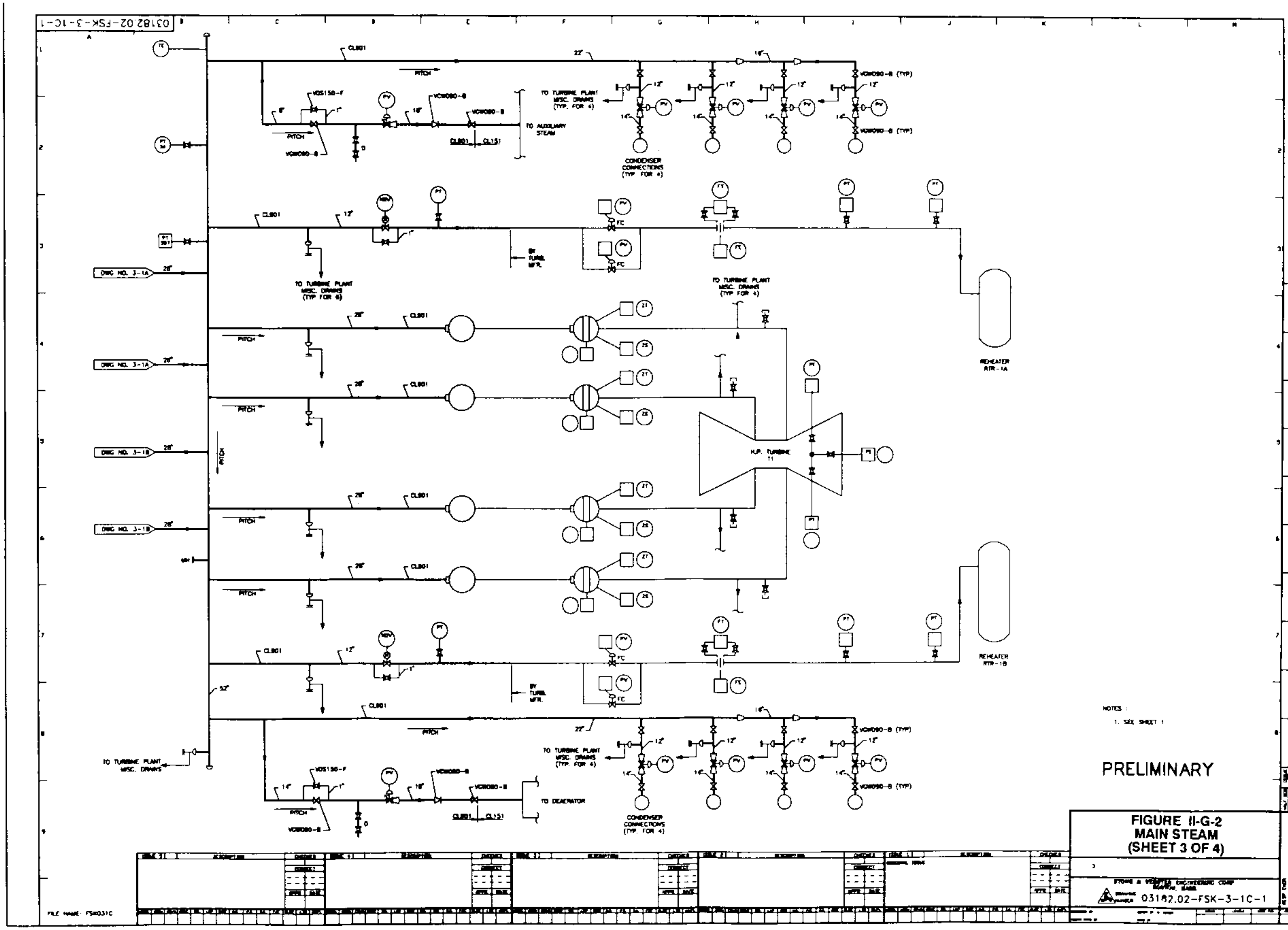


FIGURE II-G-2
MAIN STEAM
(SHEET 3 OF 4)

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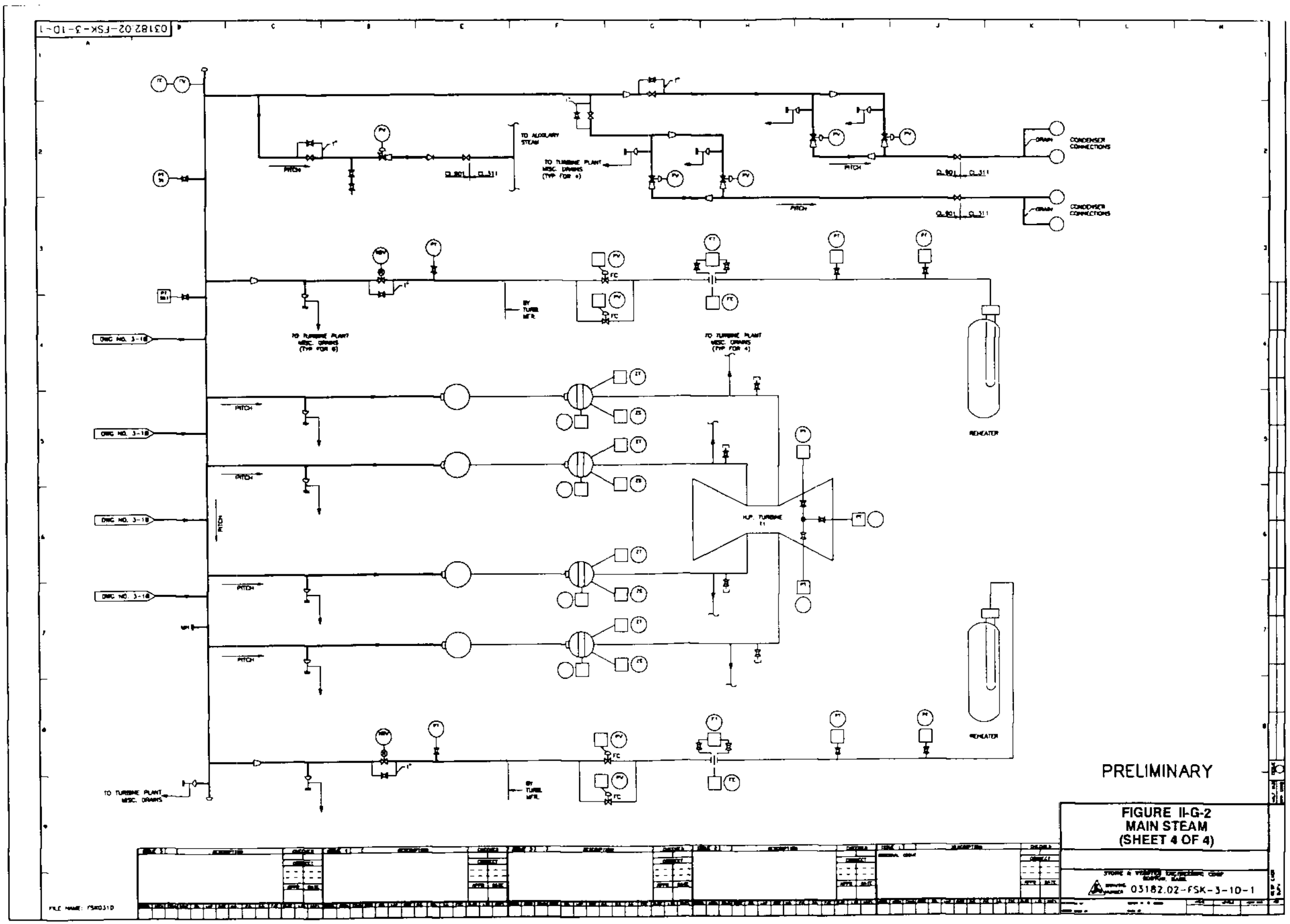
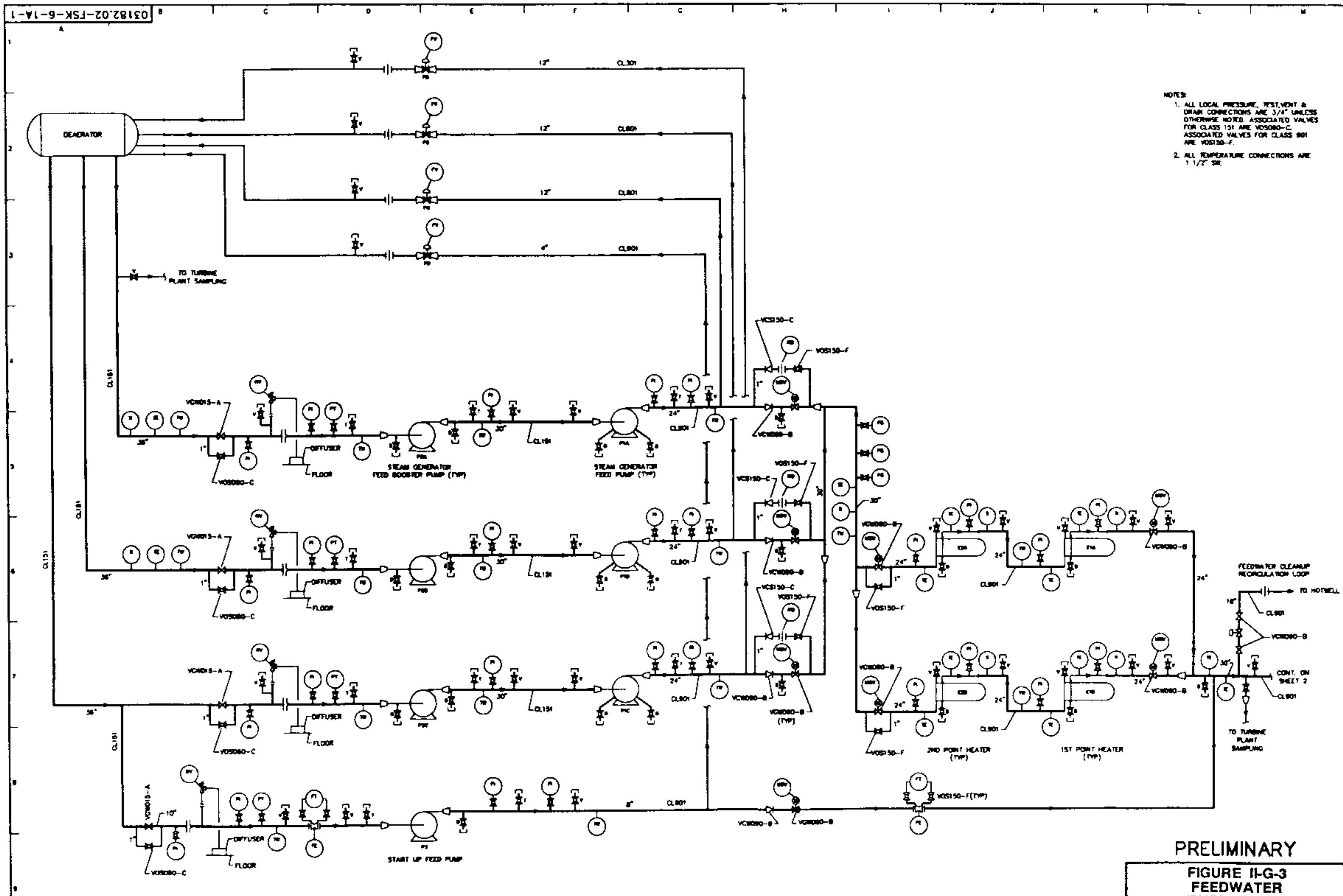


FIGURE II-G-2
MAIN STEAM
(SHEET 4 OF 4)



NOTES:
1. ALL LOCAL PRESSURE, TEST, VENT & DRAIN CONNECTIONS ARE 3/4" UNLESS OTHERWISE NOTED. ASSOCIATED VALVES FOR CLASS 151 ARE VOS080-C. ASSOCIATED VALVES FOR CLASS 801 ARE VOS150-F.
2. ALL TEMPERATURE CONNECTIONS ARE 1 1/2" SW.

PRELIMINARY

FIGURE II-G-3
FEEDWATER
(SHEET 1 OF 2)

03182.02-FSK-6-1A-1

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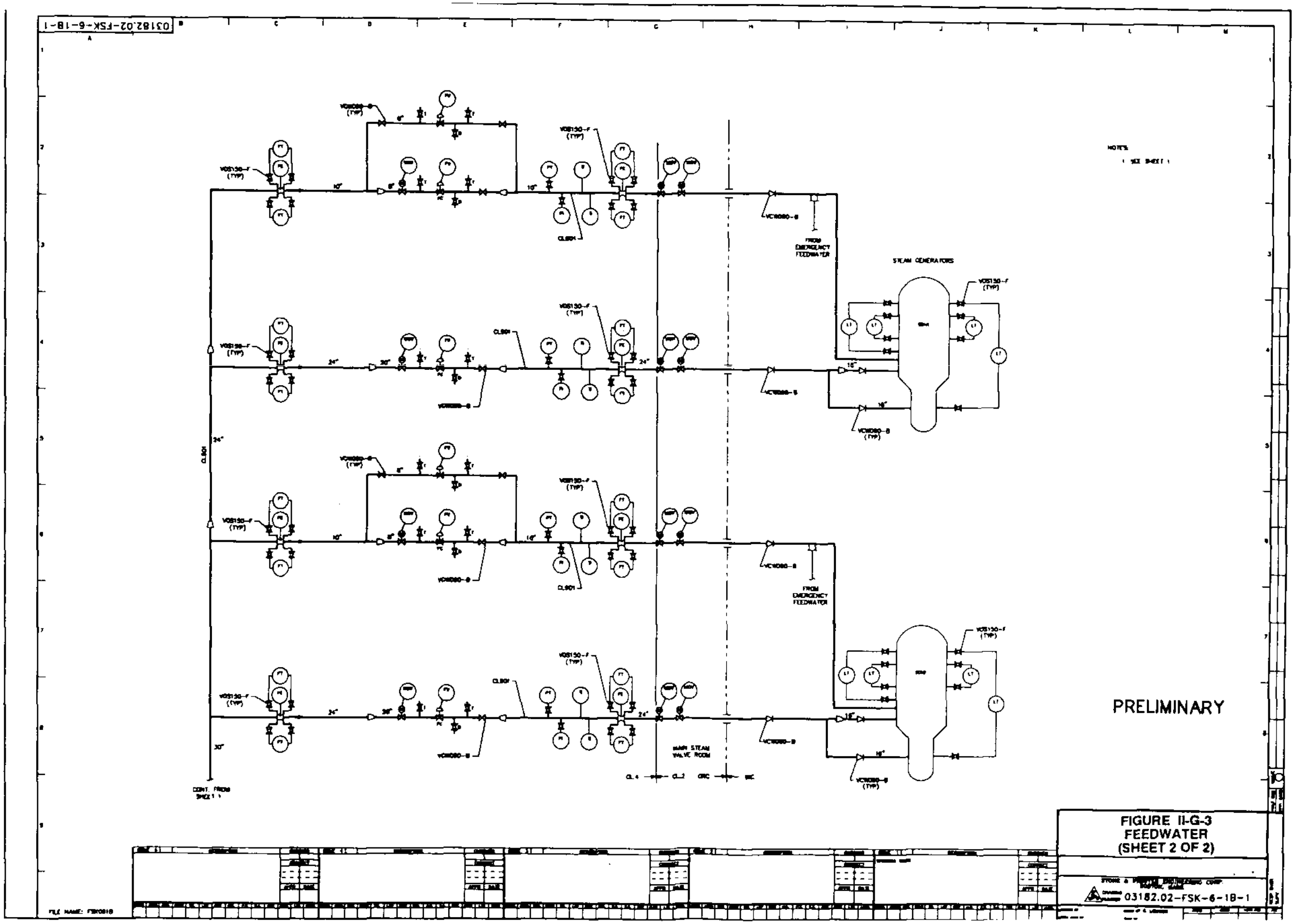

FIGURE II-G-3
FEEDWATER
(SHEET 2 OF 2)

03182.02-FSK-6-1B-1

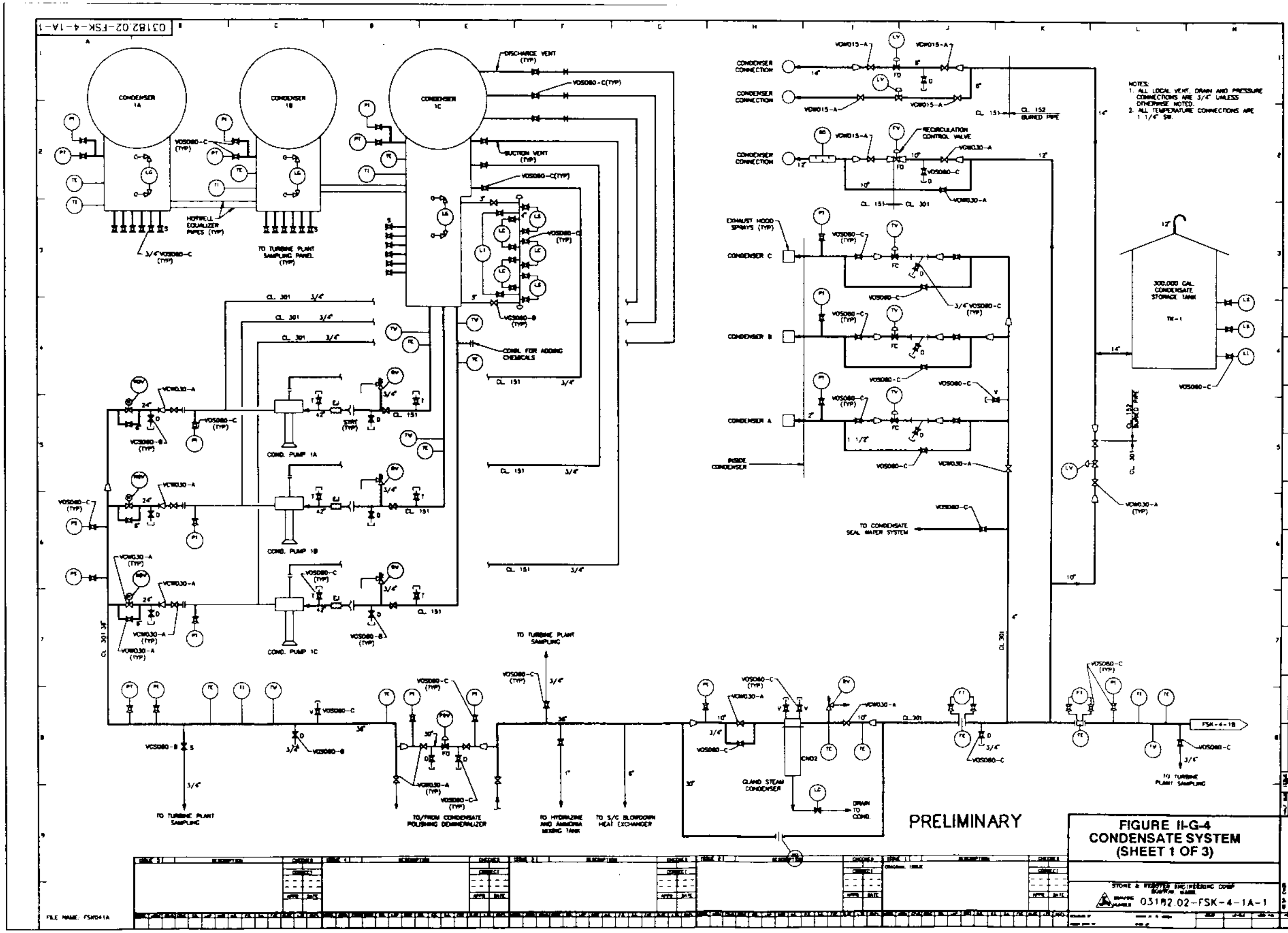

FIGURE II-G-4
CONDENSATE SYSTEM
(SHEET 1 OF 3)

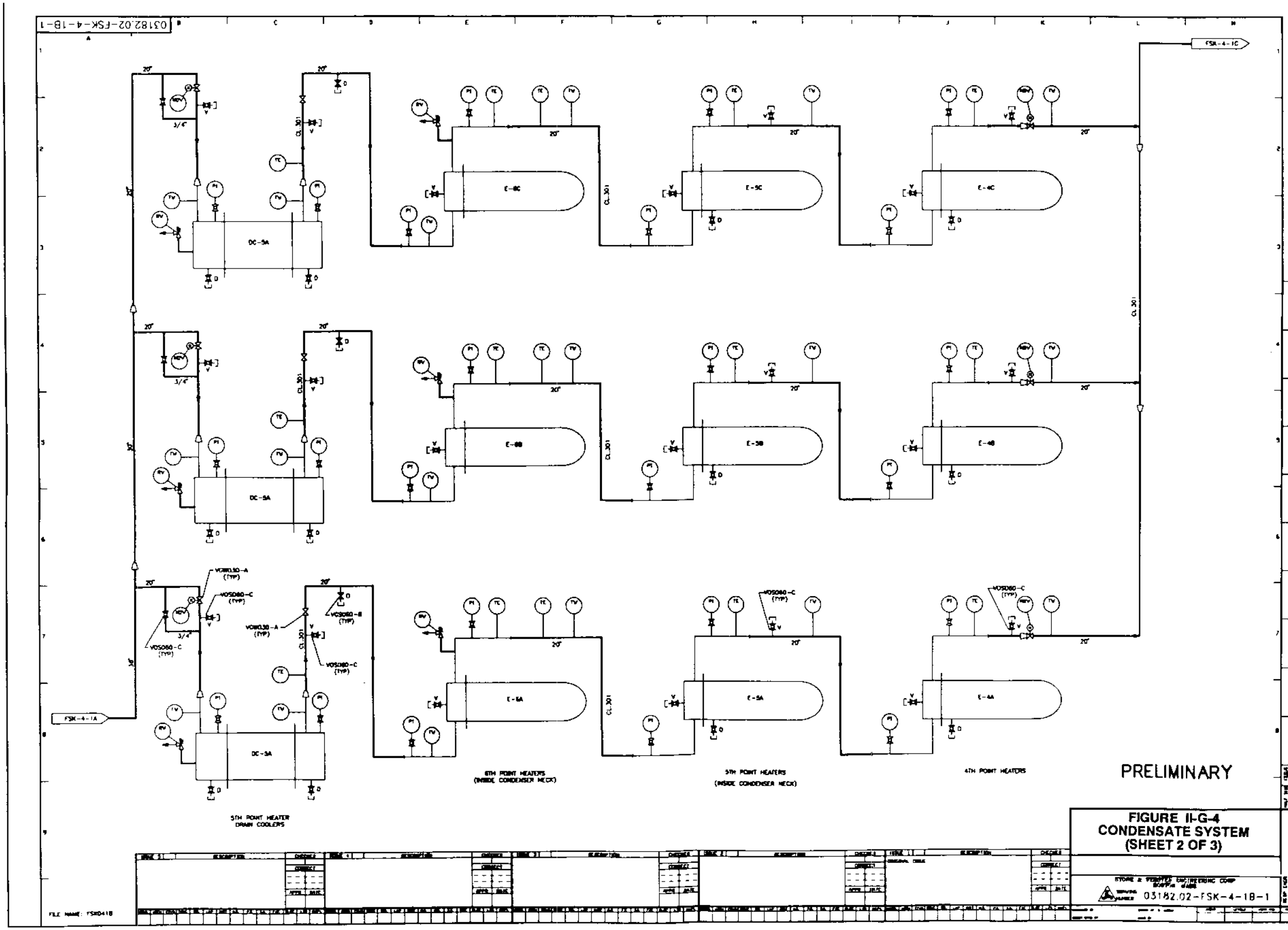

FIGURE II-G-4
CONDENSATE SYSTEM
(SHEET 2 OF 3)

03182.02-FSK-4-1B-1

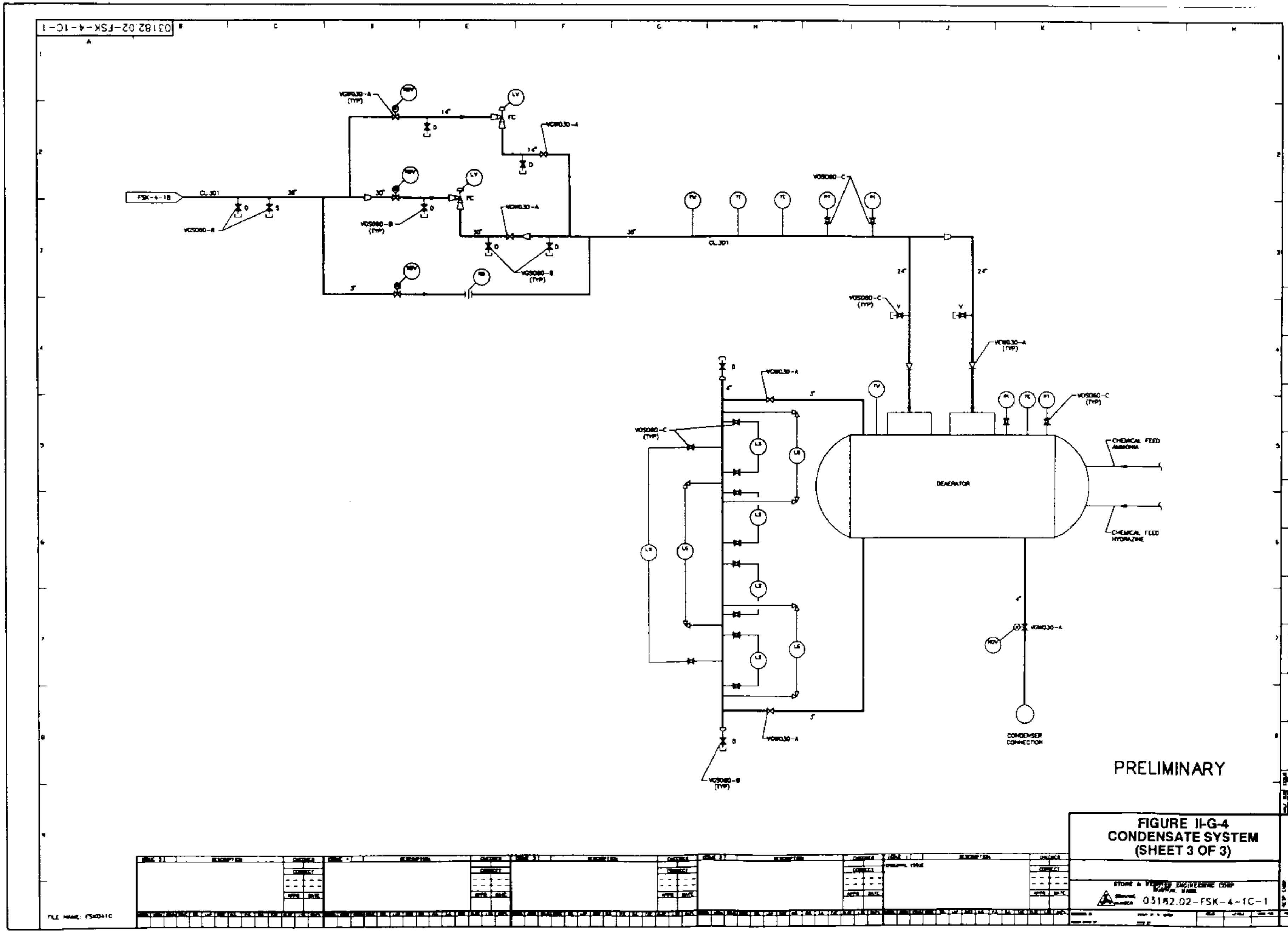

FIGURE II-G-4
CONDENSATE SYSTEM
(SHEET 3 OF 3)

The four main steam pipes are connected close to the turbine via the header, thereby achieving pressure balance in both lines before the admission valves of the turbine, a measure necessary when testing the turbine admission valves under load. From the header, the pipes branch off to the turbine bypass valves for reheating the steam after the HP turbine and for supporting steam supply to the deaerator/feedwater tank and the auxiliary steam header. The eight turbine bypass valves are sized for a maximum main steam mass flow of 55 percent of full load, so that the quantity of excess steam produced during start-up and shutdown functions and during load-shedding can be passed to the turbine condenser.

Each bypass assembly consists of a combined emergency-stop and control valve connected via intermediate pipe to an individual steam dump device. The cylindrical steam dump device is arranged in the condenser neck and consists of two chambers. The first chamber reduces the steam pressure via a perforated cone for the second chamber where main condensate is injected via spray nozzles to cool down the steam. The steam passes then through a perforated plate into the condenser steam space. The injection water is removed from the hotwell of the turbine condenser by the main condensate pumps.

After expansion in the HP turbine section, and before entering the LP turbine sections, moisture in the steam is drained via high-velocity moisture separators installed in the four exhaust steam pipes, in order to avoid high moisture and the associated risk of erosion. Following the second stage of high velocity moisture separation the steam is reheated with exhaust steam from the main steam header. After passing through the LP turbine section, the steam is condensed in a surface condenser. Condenser tubing for both sites is considered to be titanium.

(2) Gland Sealing System

The steam sealing system is designed to prevent air or steam leaking into or out of the clearances between the shaft ends and the casings, and between the valve stems and valve casings.

The sealing steam system is maintained slightly above atmospheric pressure by a pressure regulator.

Normally the system is supplied with steam from the high-pressure section shaft glands and then fed into the low-pressure shaft packing, thus preventing the ingress of air into the low pressure sections.

The pressure in the sealing steam system is maintained at a constant value by a pressure regulator. If the pressure rises (that is, if the high pressure section leaks off more steam than is required to seal the low pressure sections), the excess steam is passed by a valve to the condenser or to a low pressure feedwater heater. During starting and part load operation, when more sealing steam is required than the available leakage from the high-pressure section, a second valve maintains pressure by bleeding steam from the turbine inlet steam line.

The exhauster system is maintained at a pressure slightly below atmospheric and draws a mixture of steam and air from the lowest pressure glands. This mixture is passed to a gland steam condenser where the air is ejected. The condensate is then returned to the condenser hotwell.

d. Moisture Separator/Reheater

(1) General

In order to minimize the erosion of the low-pressure turbine blades, especially of the blades of the last stage, moisture separation takes place immediately after the high-pressure turbine. The moisture separator reduces the moisture content in the exhaust steam to nearly zero and the dried steam is reheated prior to its admission to the low-pressure turbine. High-velocity moisture separators and reheater units are provided.

The design of the moisture separator and reheater units is suitable for all operating conditions of the steam turbine. This especially applies to thermal stresses, drains and leakages.

The plant is capable of operating continuously with a balanced reduction in the heating steam to the two reheaters. Full-load operation with reduced superheat is allowed.

The extent of reheating is adjustable during operation in order to optimize the plant performance.

To prevent thermal distortion of the reheater at start-up, the pressure of the main steam flow to the reheater is controlled as a function of load in the load range of 20-50% (approx). Below 20% (approx). turbine load, the main steam flow to the reheater is shut off.

The moisture separator and reheater units are designed to resist erosion. Components especially prone to corrosion and erosion are of stainless steel or similar adequate material. The moisture separator and reheater internals are readily accessible for visual inspection and consideration is given to maintenance requirements.

The heating steam supply to the reheaters is controlled as an integral part of the turbine control system. The heating steam lines to the reheaters are provided with main stop and control valves. The reheaters are designed to prevent clogging of the tubes by condensate.

The condensate level in the reheater drain vessels is controlled automatically by a suitable level control system. This is adequate to cope with all conceivable operational situations without endangering either the reheaters proper operation or any other part of the plant while maintaining the maximum degree of operability of the plant.

The drain vessels have sufficient volume to collect drain during normal operation and transient conditions. During normal operation, the drains are led to the last HP heater stage. During start-up or low load, the emergency drain line enables trouble-free drainage to the condenser.

Non-condensable gases from the reheaters are vented to the working steam side of the reheater.

For moisture removal, an installation of ABB developed Moisture-Preseparators (MOPS) and Special Cross-Under Pipe Separators (SCRUPS) is supplied and installed in the cross-under pipes.

The separation equipment consists of one MOPS and two SCRUPS devices in each cross-under pipe. The MOPS device is installed in direct connection to the outlets of the high-pressure turbine in order to remove moisture from the pipe wall.

The MOPS consists basically of a concentric chamber around the cross-under pipe, whose upper end forms a gap through which the moisture is separated. Efficient moisture separation requires some transport steam to follow the separated moisture. To obtain optimum separation, the shape in the inlet section and the gap of the MOPS are designed for a given turbine geometry, moisture content and available transport steam flow.

The SCRUPS separation device comprises the turning vanes of the 90° elbow. The turning vanes are installed in the internal chamber in the SCRUPS. The chamber is installed in the outer shell which is cylindrical and has a slightly larger diameter than the cross-under pipe. The droplets in the steam flow cannot follow the steam when it is deflected in the turning vanes and impinge on the walls on the concave side of the vanes. The moisture forms a film on the turning vanes. The vanes are hollowed and provided with slots on the concave side to remove the moisture film.

Some transport steam is required to remove the moisture. The transport steam flows to the internal chamber where the moisture is separated.

The transport steam from the MOPS and SCRUPS is extracted to the feedwater tank, serving as heating steam of the deaerator, and the condensate from these MOPS and SCRUPS is led to drain vessels. From these drain vessels it is led via automatic level control to the LP 3 heaters or to the condenser during start-up of plant.

(2) Reheaters

After the last SCRUPS the steam is reheated in special reheaters.

The reheaters are of vertical design with straight tube bundles. Each reheater is equipped with two tube bundles. The tube bundles are located in the middle of the reheater shell.

Main steam is used as heating medium in the reheater tube bundles.

The tubes in each bundle are supported by a number of equal divided support plates. These are in one end flexibly fastened to longitudinal beams which are welded to the tube sheets. The support plates are also held by a number of staybolts and distance tubes. To accommodate expansion, the beams and staybolts are divided on the middle of the tube bundle. A T-ring between the beams prevents the tube bundles from inclining. The entire tube bundle is hanging in the upper tube plate, which rests on a console in the shell. At the lower tube plate, there are labyrinth sealings which give flexibility in the vertical direction.

Each transfer pipe between the reheaters and the LP turbines is provided with a hydraulically controlled butterfly valve, governed by the control of the turbine. The function of the valves is to limit the steam flow to the LP turbines in the event of load rejection and tripping of the turbine, in order to avoid excessive overspeed of the turbine. The stems of these valves will have a connection to the seal and leakage steam system.

The heating steam condensate is collected in level-controlled drain vessels and transported to the HP heaters or alternatively to the condenser. The system between HP turbine outlet and LP turbine inlet will be protected against high pressure by means of pressure transmitters in the turbine protection system.

e. <u>Multipressure Condenser</u>

A multipressure condenser design will be provided which will use a cooling tower for station heat rejection. The condenser will consist of three separate shells of two parallel tube bundles with each shell transversely oriented to the turbine shaft and located under each of the three hoods of the LP turbine. These separate condenser shells are generally referred to as low, intermediate and high pressure shells. The designations correspond to the magnitude of the turbine exhaust pressure each shell provides with the cool circulating water entering the low pressure shell, then flows successively through the intermediate shell and high pressure shells with the hot water leaving the latter for return back to the tower.

f. <u>Condenser Air Removal System</u>

The condenser air removal system consists of four packaged vacuum pump units and interconnecting piping. Normally three vacuum pump units are in operation. The fourth pump is a spare and is valved into the system whenever one of the three normally operating vacuum pumps is removed for maintenance or repair. The vacuum pump units have two modes of operation, a hogging mode and a holding mode. The hogging mode is used at plant startup to reduce the condenser pressure from normal atmospheric to approximately 5 to 10 in Hg absolute. The holding mode is used when the approximately 5 in Hg absolute pressure is reached to reduce the pressure to its operating value and then maintain the operating pressure during normal plant operation.

The condenser is a multipressure condenser having a low-pressure zone, an intermediate-pressure zone, and a high-pressure zone. Each zone operates at a different pressure during normal plant operation.

The condenser air removal system design provides a normally operating vacuum pump unit for each condenser zone and a common spare vacuum pump unit. Each normally operating vacuum pump takes suction from one of the condenser zones through two connections on the condenser shell. The two connections are connected to one header which goes to the suction connection of the respective normally operating vacuum pump. The normally operating vacuum pumps withdraw the air and noncondensable gases from the condenser shell zone, compress them and discharge them through an individual line from the discharge nozzle of each vacuum pump unit to a common header which carries the air and noncondensable gases to the unit vent.

The spare vacuum pump unit has a valved connection into each of the suction headers of the normally operating vacuum pumps that join into a common header which goes to the suction connection on the spare vacuum pump unit.

The compressant water used in the vacuum pump units is provided by the demineralized water system.

The vacuum pump cooling water is supplied by the turbine plant component cooling water system.

The pipelines from the condenser connections to the vacuum pump units are pitched toward the vacuum pump units and the pipelines from the vacuum pump units to the single vertical header that discharges to the unit vent. The vacuum pump units are designed for slugged-flow operation; therefore, the condensed moisture in the vacuum pump units suction lines will flow toward the pumps where it can be drained.

The condenser air removal system air and noncondensable gases quantities for removal are 1200 scfm for each operating vacuum pump during the hogging phase and 50 scfm for each operating vacuum pump during the holding phase. The hogging phase capacity is determined and based on the quantity of steam being condensed by the condenser in accordance with HEI standard for steam surface condensers. The phase capacity is based on the effective steam flow for each main steam exhaust opening into the condenser in accordance with HEI standard for steam surface condensers.

The condenser air removal connections to the condenser are connected to the condenser shells at the cold end.

Vent connections are provided on the vacuum pump coolers and high points of cooling water piping for venting of the condenser air removal system to prevent air entrainment.

The condenser air removal system is designed to permit isolation and removal of any vacuum pump from service.

The safety classification of the condenser air removal system piping, equipment, and components is non-nuclear safety in accordance with ANSI Standard 18.2.

The condenser air removal system piping is designed, fabricated, inspected, and erected in accordance with ANSI Standard B31.1.

g. <u>Feedwater System</u>

The feedwater system is shown on FSK-6-1A&B (Figure II-G-3-2 sheets).

The feedwater system consists of three, one-half capacity, variable speed, motor driven booster and feedwater pumps and a motor driven startup pump which take suction from the deaerator storage tank. The pumps discharge through check valves (which prevent backflow through an idle pump), and motor operated discharge valves to a common header. The discharge valves provide isolation for pump maintenance. A minimum flow recirculation control valve for each pump protects against undue heat buildup and vibration in the pump at reduced flow. The control valve in the recirculation line is located as close as practical to the deaerator in order to minimize the length of pipe subjected to two phase flow.

The feedwater flows from the common pump discharge header through two parallel high pressure feedwater heater strings, containing two heaters each. Each of the high pressure feedwater heater strings can be isolated for maintenance by two high pressure motor operated valves. A sampling connection is provided downstream of each heater string.

The discharge of the first point feedwater heaters is combined in a common manifold which supplies feedwater to two steam generator supply lines. A line back to the condenser comes off this manifold to allow cycle cleanup and recirculation. The heater bypass discharges to a second manifold which distributes the cooler feedwater through restriction orifices to the two steam generator supply lines. This manifold piping system provides adequate mixing to ensure that uniform temperature feedwater is supplied to each steam generator. Each of the two steam generator supply lines contains a feedwater flow control valve which is automatically positioned by a signal from the steam generator water level control system. A venturi type flow element, with associated flow transmitters, provides the feedwater flow signal used in the steam generator water level control system. The flow control valve, in conjunction with the variable speed feedwater pumps maintains steam generator level during steady state and transient operations. The feedwater pump speed is varied to maintain a programmed differential pressure between the feedwater supply header and the main steam header. The differential pressure setpoint varies directly with plant load (steam flow). The feedwater flow control valve can be bypassed by a remote manually controlled air operated valve to permit manual control of steam generator level at power levels below 25 percent.

The bypass control valve is in a separate line, which comes off the main feedwater line upstream of the main flow control valve, and penetrates the containment, going to the downcomer nozzle on the steam generator.

The steam generator chemical feed system can inject hydrazine and volatile amine into each steam generator downcomer supply line just upstream of the feedwater isolation valves.

Two quick closing feedwater isolation valves in each supply line are located as close as practical to the containment structure. A check valve in each supply line prevents uncontrolled blowdown from more than one steam generator in case a feedwater line breaks. These check valves are necessary to achieve abrupt, complete termination of reverse feedwater flow with the existence of a reverse flow condition and, in the downcomer line, to prevent backflow of emergency feedwater to other portions of the main feedwater system. The feedwater system from the steam generator inlet back to, and including, the feedwater isolation valves is designated Safety Class 2 and Seismic Category I to ensure the integrity of these valves.

The portion of the feedwater system extending from and including the feedwater isolation valves to the steam generator inlets meets Seismic Category I design requirements. The system is protected from missiles generated by various sources including tornadoes, rotating equipment, and high pressure system components.

Instrumentation and controls for the feedwater isolation valves are Seismic Category I. The remainder of the instrumentation and controls have no seismic design requirements.

The Seismic Category I portions of the feedwater system are protected from the effects of tornadoes. Whenever practical, the feedwater piping is separated from systems required for safe shutdown by remote location or structural enclosures.

The main feedwater flow is to the economizer nozzles, on the lower portion of each steam generator. The bypass control valve is in a separate line penetrating the containment, going to the downcomer nozzle, on the upper portion of each steam generator.

Feedwater temperature is equal to or greater than 200°F prior to initiation of feedwater flow to the economizer nozzles during plant startup. The 200°F feedwater temperature is achieved prior to reaching 15 percent power. All feedwater at a temperature lower than 200°F is directed to the downcomer feedwater nozzle, except for post turbine trip conditions.

The feedwater pumps are located on the ground floor of the turbine building. The pumps discharge to a common header which distributes the feedwater to the two high pressure heater strings on the mezzanine floor of the turbine building.

The high pressure heaters discharge to a manifold which distributes feedwater to the two steam generator supply lines. The steam generator supply lines run over the Nuclear Annex to the Main Steam Valve Houses.

The Main Steam Valve Houses are located on opposite sides of the containment in the Nuclear Annex. The feedwater isolation valves are also located in these Valve Houses. The feedwater lines run from the valve houses into the containment to the steam generator cubicles. Before entering the steam generator cubicles, the piping rises up to the nozzles to prevent steam from entering the feedwater piping and causing water hammer when a low water level exists in the steam generator.

3. Generator

a. <u>Construction, Materials and Cooling</u>

The generator is hydrogen cooled and rated for the following continuous operating conditions:

- Rated load and 0.90 power factor lag.
- Rated load and 0.95 power factor lead.

- Terminal voltages between 95 percent and 105 percent of rated voltage at a system frequency of 60 Hz.

The generator stator winding is star connected and insulated for operation with a high impedance neutral earthing system. The specific winding impulse level will be provided in the detailed design.

The busbar connections are designed to withstand the maximum value of peak asymmetrical fault current and are also capable of withstanding the maximum through fault symmetrical current for one second.

A generator output breaker is provided to permit offsite supply to the unit transformers by backfeeding the generator transformers for abnormal or plant startup conditions. This capability is depicted on the station one line diagram.

The generator excitation system is based on a proven static thyristor technique.

The static excitation system is capable of rapidly adapting air-gap flux to the load conditions, providing a stable energy supply when operating on a network and maintaining a constant generator voltage on no-load operation or station service operation. The static system consists of an excitation transformer, a thyristor rectifier bridge and a full current breaker.

The excitation system operation in combination with a generator step up transformer standard tap range of +6.66 percent to -13.33 percent will be adequate for the application. Specific aspects of the excitation range design are subject to detailed analysis reflecting grid and station operating design conditions.

Other systems in the turbine island which support operation of the turbine-generating system include the generator excitation and voltage regulator, seal oil system, stator cooling water system, hydrogen and carbon dioxide system, turbine controller, turbine plant supervision system, turbine plant protection system, turbine plant automatic operation system, on line condition monitoring system, generator busbars, and non-priority secondary cooling system.

b. <u>Excitation System and Automatic Voltage Regulator</u>

The static excitation equipment is a shunt supplied one, i.e. the excitation transformer is connected directly to the generator terminals. The secondary voltage is rectified by a fully controlled thyristor converter and fed to the generator field. The power converter is of redundant design with a number of parallel bridges. Even with one complete bridge out of service, all operational conditions can be met without restrictions. The field suppression equipment serves two purposes: the disconnection of the excitation equipment from the field winding and the fast field discharge.

The regulation and pulse electronics is of the dual channel type. Usually, the machine terminal voltage is controlled by the automatic voltage regulator in the AUTO channel. In case of a failure, e.g. loss of the PT sensing voltage or for maintenance reason there is a change - over facility to a MANUAL channel. In this mode, the field current is controlled. A follow-up control ensures a smooth change - over from one channel to the other.

The firing pulses for the thyristors are generated within two gate control modules and formed in the following pulse amplifiers. If the machine's operation limits are reached, then electronic limiters will take action. These limiters increase essentially the availability of the machine because they always operate before a protection relay may trip the unit.

For the voltage regulation and twice for the pulse generation, three separate micro-processors are provided.

Various monitoring and protection features prevent damages to excitation transformer, converter and fieldwinding resulting from internal and external faults. Another micro-processor takes cares of some of these functions.

An internal programmable logic control (PLC), for which a separate micro-processor is provided, takes care of the right sequences during start-up and shut-down. Further, it ensures the safe local operation of the equipment and its auxiliaries. In case of a failure, the fault is indicated on a local annunciation board which simplifies the trouble shooting.

3. Seismic Design

The turbine generator unit is designed for a seismic event of 0.4 g in the horizontal direction at turbine top table foundation (operating floor).

4. Condensate System

a. Flow Diagram

The Condensate System is shown on Flow Diagrams FSK-4-1A, B and C (Figure II-G-4-2 sheets).

b. System Description

Each of the three condenser shells receives the exhaust steam from one low pressure turbine. Effluents from the fifth point heater drain coolers and the sixth point feedwater heaters, the high pressure steam line drains, and other miscellaneous turbine plant drains are normally directed to the condenser. The turbine bypass system consisting of eight turbine bypass valves is connected to the main condenser.

The condensate system has sufficient water available to accommodate a steam loss to the atmosphere of 55 to 65 percent of the normal steam flow and to regain a steam generator level due to a collapse or shrinkage of the steam-water mixture when taking a 100 percent load rejection.

Three motor driven, vertical centrifugal condensate pumps, each capable of supplying 50 percent of condensate flow requirements, take suction from the condenser hotwell. Two condensate pumps are normally operating, with one in reserve, to ensure that the water supply to the feedwater system is not interrupted in the event of a condensate pump trip or the loss of feedwater heater effluents. Each pump has a butterfly isolation valve, a permanent simplex strainer, and an expansion joint in its suction line. The net positive suction head required by the pump is provided by the water level in the hotwell. The continuous vent in the suction line of each pump is connected to the condenser because the pump suction lines are under vacuum during operation. A relief valve is provided in the suction vent line to prevent possible overpressure from an idle pump. Another vent in each discharge line is used only when placing a pump in service after a section of the suction piping has been drained or if the system becomes air bound. These vents are connected to the condenser to minimize makeup water requirements. The discharge line of each pump has a check valve to prevent backflow through an idle pump and an isolation valve for pump maintenance.

The condensate pumps discharge to the main condensate header which flows through the condensate polishers. To meet the water chemistry requirements, the entire condensate flow is passed through side stream full flow condensate polisher cells and returns to the main condensate header. A bypass valve is provided between the two connections of the condensate polishing system to allow system cleaning and maintenance operations with condensate polishing isolation. Downstream of the return line from the condensate polishing system, a line branches to the chemical addition mixing tank. Hydrazine and a volatile amine are added to the condensate to maintain the desired pH and oxygen levels for corrosion control. Another branch line flows through the shell side of the steam generator blowdown heat exchanger.

Flow in the main condensate header then flows through the tube side of the gland steam condenser.

A branch line from the main header supplies water to the turbine exhaust hood sprays. A load controlled valve controls flow to the turbine exhaust hood sprays, which are required to cool the exhaust hoods during startup or low power operation.

A pressure valve controls flow to a line that branches from the main header. This line provides seal water for the extraction line non-return valve seals, the condensate pump suction line butterfly valve seals, the condensate pump shaft stuffing box, and condenser vacuum breaker valve stem seal. It also provides fill water to the loop seal on the sixth point heater drains and condenser vacuum breaker valves. The valve seals prevent air from being drawn into the condenser during initial startup and under low load conditions.

Sampling connections for the turbine plant sampling system are provided in all but one hotwell section, under each of the 12 tube sheets, and on the main header to detect and locate the source of any leakage into the condenser.

Flow in the main condensate header then passes through a flow element. The transmitter associated with this flow element provides a signal to the flow controller that regulates condensate recirculation to the main condenser. At low power levels this recirculation flow provides adequate cooling for the gland steam condenser and provides the condensate pumps with the minimum required recirculation flow.

A level valve actuated by level controllers in the condenser hotwell controls flow to the condensate makeup and drawoff system allowing the water inventory to be decreased to correct for a high hotwell level. Level controllers in the hotwell also control level valves in the condensate makeup and drawoff system to provide makeup water to the condenser.

The main condensate header then branches into three parallel flow, low pressure feedwater heater strings. Each feedwater heater string contains a fifth point heater drain cooler, two condenser neck-mounted feedwater heaters (fifth and sixth point), and an external fourth point feedwater heater. Each feedwater heater string has two motor driven isolation valves. A thermal relief valve in each feedwater heater string provides overpressure protection. The three heater strings discharge to a common header with a level control valve which runs to the deaerator.

One deaerator and deaerator storage tank are provided after the low pressure feedwater heaters.

Deaerator storage tank and condenser hotwell level are controlled as follows:

- Deaerator storage tank level is controlled by two pneumatic valves at the deaerator inlet which adjust condensate flow to the deaerator. During low load periods, one valve controls flow. At higher loads, one valve is open and the other valve controls flow.

- Condenser hotwell level is maintained by directing condensate flow to and from the condensate storage tank using normal and high capacity emergency lines. Makeup from the condensate storage tank is directed to the condenser for vacuum deaeration. Dissolved oxygen in the makeup water is minimized by the stainless steel floating cover on the condensate storage tank.

For heating purposes prior to startup or during hot standby conditions, the feedwater in the deaerator is maintained at about 221°F by reduced pressure steam from the main steam header or the auxiliary steam header.

H. MAINTENANCE

The nuclear maintenance program supports two basic types of maintenance-corrective maintenance (CM) and preventive maintenance (PM) (predictive, periodic, and planned). A proper ratio of these types of maintenance provides a high degree of confidence that prior to failure or malfunction, plant equipment degradation is identified and corrected, that equipment life is optimized, and the plant maintenance programs are cost effective.

The ABB System 80+ Nuclear Electric Generating Station Maintenance Program identifies the significant aspects of a total maintenance program including not only refueling outage maintenance but also maintenance training, corrective maintenance, preventive maintenance, maintenance management, and unplanned outage maintenance. This program is based on an aggressive approach that encourages all plant personnel to contribute to a safe and reliable plant and work place.

Any comprehensive maintenance program incorporates planning and preparation. The System 80+™ maintenance program progresses further and defines follow-up steps, pre and post maintenance documentation requirements, and analysis of maintenance problems. The program recognizes ALARA principles and supports minimizing personnel exposure to ionizing radiation. Lastly, this program defines a management organization that is equipped to successfully complete planned and unplanned outage maintenance activities.

The design of both the nuclear and the conventional island incorporates features that facilitate good maintenance practices and provides easy access for tools, equipment and maintenance personnel. Adequate lay-down areas and suitable lifting facilities are also include.

Pumps that require calibration flow tests, for example, High Pressure Safety Injection (HPSI) Pumps, have miniflow circuits permanently installed.

1. In-Service Inspections

In-service inspections of pressure vessels, piping, pumps, valves, and other equipment to prove the integrity and serviceability of these components are a major activity during refueling and maintenance outages and one of the largest contributors to personnel radiation exposure. In order to enhance the overall inspection process, save time, and reduce exposure, the plant design includes maintenance accommodations.

Piping and pipe supports are located such that adequate space exists for personnel and equipment access during in-service inspections. Insulation, hangers, stops and snubbers are designed so that they do not interfere or are quickly removable.

Where necessary, access platforms, removable pipe sections, jib cranes and any other necessary features are provided in high radiation areas to minimize exposure time.

Access for in-service inspection of the reactor pressure vessel, piping, pumps and valves is provided by using permanent access platforms to the maximum extent, and with adequate space for setup of inspection equipment. Clearances are provided to accommodate automated inspection devices and removable insulation panels.

Manholes and handholds are arranged to be clear of obstructions for inspection equipment or personnel.

Pumps and valves that must be disassembled for visual in-service inspection are provided with sufficient clearance and laydown space to permit disassembly with minimum removal of adjacent equipment or structures.

Permanent access platforms are provided outside reactor pressure vessel shield walls to facilitate in-service inspection of the reactor vessel and nozzles. This includes access to the vessel bottom or bottom nozzles.

Adequate space is provided around all inspection points for insulation and tool laydown.

Removable, reflective metal type insulation is used at locations on reactor system piping and equipment where access is required for in-service inspection. Snap on insulation is used for higher radiation areas to allow quicker removal and replacement.

Weld locations within a system boundary subject to in-service inspection is separated by at least three times the wall thickness or six inches from weld to weld, whichever is greater.

Weld locations on piping penetrating walls is at least six inches or three times the wall thickness, whichever is greater, away from the wall surface. Nozzle welds, support lugs or other attachments should also meet this criteria.

Radial clearances of at least six inches is provided around pipes or components for in-service inspection or more, if required, for specific automated equipment.

Where piping is run in a pipe chase, or as groups of pipes at different elevations, the pipe to be inspected is located in the outside layer or lower tier to ensure access without removing other piping.

Permanent working platforms, ladders or stairways are installed in in-service inspection areas where radiation levels will exceed 1 mSv/hr during the first 10 years interval.

Provisions for lighting, air, power, water, etc. are provided as required for the inspection method used in the immediate area of the inspection.

Permanent or temporary shielding for personnel are considered at inspection locations where radiation levels will exceed 2 mSv/hr. Provisions for removal and storage of temporary shielding will be available.

Mechanized inspections are considered for areas where radiation levels exceed 0.5 mSv/hr or physical limitations restrict or prevent manual methods.

Where manual inspections are required, sufficient clear space is provided for the upper body of a person working at arm's length 20 inches of the surface inspected.

Direct access routes are provided to locations requiring inspection.

The requirement, development, and control for special tools and equipment is established as an integral part of the plant maintenance program. These specialty items should be stored, maintained, and inventoried as a part of the plant tool and equipment program.

Inspections of the material condition of the nuclear power plant is the responsibility of management. This effort will depend on many factors including design, fabrication, modifications, ongoing maintenance, work control, and day-to-day operations. Following initial plant construction, control of maintenance and modification activities will be the prime method for keeping systems and equipment in an optimum condition for support of safe and reliable station operation.

2. Equipment to Minimize Refueling Outages

The following presents details of equipment and design features provided to minimize the length of the annual shutdown for scheduled refueling.

a. Multiple Stud Tensioner

A multiple stud tensioner (MST) and associated tooling is utilized for the complete and simultaneous removal and installation of the reactor vessel closure studs, nuts and washers. This design philosophy results in a decreased critical path time and reduces the manhours associated with these tasks. The MST allows for the sequential tensioning and detensioning of each half of the closure studs and the subsequent removal of all of the studs, nuts and washers.

b. Head Area Cable Tray Structure (HACTS)

The head area cable tray structure (HACTS) supports and provides channel separation for the reed switch position transmitter cable-conduit assemblies and the control element drive mechanism (CEDM) power-conduit assemblies. The design includes an integral missile shield for use during plant operation when the HACTS is located above the reactor vessel closure head. Utilization of this component minimizes handling time by allowing removal and replacement of the electrical cables to the reactor vessel head in one operation.

c. Permanent Pool Seal

The utilization of a permanently installed pool seal reduces the length of the refueling outage by removing the installation and removal of a temporary seal from the critical path. The permanently installed pool seal reduces the personnel radiation exposure which would normally occur during installation and removal, while also eliminating the requirement for a pool seal storage area in the containment building.

d. Automated Refueling Machine

A semi-automatic positioning system allows the refueling machine to automatically traverse to and from the transfer system upender to a preselected set of coordinates in the core and to automatically traverse between any two preset core coordinates. This is accomplished through the use of a computer based control console which is equipped with direct prompt operator control.

e. Concurrent CEA Removal

The upper guide structure (UGS) lift rig allows for concurrent removal and installation of all the control elements assemblies (CEAs), thereby significantly reducing the time and man-rem expenditures associated with refueling.

f. Parallel CEA Replacement

A CEA change platform is provided for CEA changeout independent of the refueling machine and therefore allows replacement of CEAs to be performed in parallel with the fuel handling operations and therefore off the critical path.

3. Maintenance During Outages

The principles of ALARA (As-Low-As-Reasonably-Achievable), and Radiation Protection when properly applied, allows maintenance personnel to perform their activities in a safe and efficient manner while minimizing exposure to ionizing radiation.

Maintenance management's role is to enforce the principles of ALARA, and Radiation Protection through preplanning of maintenance activities, coordination and preparation of maintenance activities, training and instruction of personnel, and the review of completed maintenance activities for ALARA improvements.

A typical outage plan includes:

1. Introduction: A planning summary of the outage.

2. Schedule:

   A. Outage start and end dates

   B. The outage duration in days

3. Work Summary:

   A. The Significant Job List provides a list of all significant jobs (including station modifications) that are scheduled to be done.

   B. Critical Path activities identify and discuss each critical path item, establish a beginning and end date for each critical path item, and a milestone plot shows the critical path items as each relates to the other.

   C. Personnel resource requirements identify requirements for each major work item. Personnel resource requirements should be in terms of craft, quantity required, and group supplying the people.

   D. RP/ALARA considerations are all activities that will result in a total person-rem exposure of >1. Estimate total exposure for the outage.

4. Areas of concern are items or activities that may require special attention during or before the outage.

5. Outage organization is identified on an Organizational Chart for the outage management team. Any special features of the organization are identified.

An outage is classified according to how far in advance it is scheduled. This affects the amount of time available for planning the outage and therefore the procedures used to plan, schedule, and manage the outage. All outages can be classified as being in one of three categories: planned, maintenance, or unplanned.

a. <u>Planned Outages</u>

A planned outage is one that is scheduled well in advance and is of a predetermined duration. Refueling outages fall into this category.

b. <u>Maintenance Outages</u>

A maintenance outage is one that can be deferred beyond the next weekend but requires the unit be removed from service before the next planned outage. Also, to be classified as a maintenance outage, a minimum of seven days to plan the outage is required. An example of this would be an outage to repair a piece of equipment that was limiting power level such as an inoperable feedwater pump.

4. On-load Maintenance

To establish the scope of the nuclear preventive maintenance programs, technical specification requirements, vendor recommended surveillance inspections, station operating experiences and problems, corrective maintenance problems, and code in-service inspections are considered.

a. <u>Maintenance Procedures</u>

Written procedures are established and properly used as one of the key elements to consistently perform maintenance in a safe and efficient manner. A balanced combination of written guidance, management support, the procedure users skills, training, and work site supervision will achieve quality workmanship essential to safe and reliable station operation.

A program for the preparation, verification, validation, review, approval, control, location, use, and the periodic review and revision of maintenance procedures is established.

b. <u>Planning, Scheduling, and Coordinating Maintenance</u>

Quality, productivity, and cost effectiveness is increased in the maintenance organization by detailed planning. The development of a well planned work package requires the assembly of information from various sources such as Operations, Health Physics/Radiation Protection, Quality Assurance, and within the Maintenance Planning Group itself.

One requirement of a well planned maintenance work package is to assure the proper scheduling of the package. The schedule should coordinate maintenance and support group resources.

Integration of maintenance schedules within the scope of total station activities is necessary to ensure that maintenance is conducted efficiently and within prescribed time limits.

c. Control of Maintenance Activities

A work control program should be established to assemble all the various parts of a maintenance activity. These parts range from work identification to maintenance work history. The heart of the work control program is the work request document and the associated computerized work management system. Controlling maintenance activities ensures that maintenance practices are effective in maintaining safe and reliable station operation. Control of maintenance activities will direct efforts toward achieving high-quality work performance, personnel safety, equipment and system protection, plant safety and reliability, and cost control.

Maintenance activities are normally scheduled on a 24-hour day basis to maximize the use of maintenance personnel, facilities, equipment and tools. A 24-hour maintenance schedule will minimize the outage time of critical equipment. Surveillance testing is maximized through more effective use of personnel.

Maximum plant reliability is established by the plant design and equipment selection. Preventive maintenance assures that the design life of the equipment is met. Establishing a regularly scheduled maintenance program improves plant availability and reduces the cost of unplanned maintenance.

A valve maintenance program is established to ensure the maximum possible reliability of each valve. This program includes frequent visual inspections of valves. The program also includes valve packing adjustment/repacking on an as needed basis. Routine servicing valves includes visual inspection, packing gland adjustment, lubrication and cleaning/preserving.

Periodic overhaul of valves is scheduled for valves that exhibit a gradual degradation over time that is not limited to a single part. Periodic overhauls are normally scheduled for a refueling outage. The valve maintenance program also includes predictive maintenance or diagnostics to predict and mitigate the onset of valve failure. Valve maintenance history records are considered in planning for valve corrective maintenance and preventive maintenance.

A lubrication program is established as a major factor in the total preventive maintenance program. Inspection and lubrication of equipment should be performed at appropriate intervals determined by vendor recommendations and/or the industry operating experience.

5. Testing

Post-Maintenance Testing is any appropriate testing or verification performed following maintenance to verify that a system, structure, or component is operable, is not degraded in performance, any original deficiency has been addressed, and a new deficiency has not been created.

Periodic and surveillance testing activities should be performed to provide assurance that plant equipment will perform within the required limits. This testing will be done under strict procedural controls and requires that prompt corrective action be taken when acceptance criteria are not met. Independent review of completed surveillance should be conducted to ensure that the acceptance criteria are met. Records of the testing program shall be computer maintained.

The requirements of the ASME Boiler and Pressure Vessel Code, Section XI is met through a test program designed to meet or exceed these requirements. The Section XI requirements and the results of the test program shall be computer maintained for verification of acceptable results and scheduling of tests.

6. Spares

A strong and effective maintenance program requires timely availability of parts, materials, and services. Parts and materials that meet the approved engineering design requirements are necessary for maintenance activities during normal station operation and for support of both unplanned and planned outages.

Proper care of parts, materials, and equipment is required from the time an item is received until it is installed in the plant. This includes all phases of receiving, inspecting, handling, inventory control, storing, retrieving, and issuing of materials.

The preventive maintenance associated with material, parts, and equipment during the storage phase is of primary importance.

7. Other Plant Maintenance Information

A maintenance training and qualification (T&Q) program is established to maintain the knowledge and skills needed by maintenance personnel to effectively perform maintenance activities. The program is designed to help optimize the effectiveness of maintenance personnel. The development of maintenance training programs is assisted by the industry's accreditation program (INPO), an ongoing process that provides for maintaining training programs.

Maintenance facilities directly affect maintenance personnel training and the ability to maintain the nuclear plant in an optimum state of readiness. Facilities include hot and cold shops and satellite work areas, laydown and staging areas, storage facilities, temporary facilities, tool and equipment storage, office equipment, and mockups for training facilities. Included with the facilities are the equipment, tools, supplies and parts to support them.

Inspections of the material condition of the nuclear power plant is the responsibility of management. This effort will depend on many factors including design, fabrication, modifications, ongoing maintenance, work control, and day-to-day operations. Following initial plant construction, control of maintenance and modification activities will be the prime method for keeping systems and equipment in an optimum condition for support of safe and reliable station operation.

The involvement of management and supervision in periodic station walkdowns and inspections will enhance the material condition of the station and clearly display management's standards to all personnel. The station program for identification of material condition deficiencies and housekeeping discrepancies is an important step in maintaining station equipment in a condition of maximum safety, reliability, and availability.

A document control program will provide correct and readily accessible information to support all plant maintenance requirements.

Technical manuals that provide vendor guidance for inspection, preventive maintenance and repair of all plant equipment are provided for use by plant personnel. Technical manuals are controlled documents and maintained in a condition that reflects the actual "as built" configuration of the equipment.

Drawings that provide guidance for maintenance inspection, preventive maintenance, and repair of plant equipment or systems are provided for use by plant personnel. These drawings are controlled documents and maintained in a condition that reflects the "as built" configuration that exists in the plant.

Approved procedures are used to ensure proper maintenance and testing to support safe efficient operation of the plant. The procedures are clear, concise, and contain adequate information for users to understand and perform their activities effectively. The preparation, review, approval and revision of procedures and documents shall be controlled in a manner that will assure their accuracy and availability.

Records to catalog individual personnel exposures are maintained by the owner and used for the life of the plant for the preventive maintenance program and also for possible design of modification or equipment updates during the life of the plant.

8. Steam Generator Replacement

The design of the System 80+ provides ample consideration for the replacement of major plant equipment including the plant's steam generators. ABB-CE steam generators have had an excellent performance record, and incorporate the latest advances in materials and design to maximize useful life. However, the overall industry experience record combined with the need to extend plant lifetimes well beyond 40 years dictates that today's designs incorporate provisions for steam generator replacement. The containment polar crane is designed such that it can be utilized to remove the steam generators from their cubicles to the area of the equipment hatch. The generators can then be skidded out the equipment hatch into the outage building where it will be lowered onto an awaiting transport vehicle.

Internal containment layout provides access to primary and secondary side piping to facilitate cutting and welding operations that will be required for this evolution.

9. Plant Capacity Factor

The plant is designed to achieve high capacity factors. This is accomplished by incorporating advanced control system designs and where appropriate, redundant design features. To the extent possible, redundant active components are provided in both the nuclear and turbine islands to achieve and maintain full power capabilities.

Historically, a large fraction of unanticipated trips result from faults initiated in the main feed and steam systems. To minimize the impact on capacity factors, the design incorporates a Reactor Power Cutback System (RPCS) designed to accommodate full load rejections and a loss of one main feed pump without initiating a reactor trip and opening primary/secondary safety valves. The design also incorporates an Extended Range Feedwater Control System which allows for automatic steam generator water level control from zero to full power. The automatic control of steam generator levels at low power operation eliminates the difficulty in manually controlling levels, which has also resulted in numerous reactor trips. These advanced design features eliminate the need for automatic start of a standby component should a loss of component function occur that might result in a turbine/reactor trip.

As an example, the design can accommodate a fault in the main feed system with insignificant impact on plant capacity factor. A fault with a main feed pump and/or controls will result in the actuation of RPCS, which will automatically reduce reactor power by insertion of CEAs and initiate turbine cutback and runback to prevent a reactor trip on low secondary or primary pressure. Once the plant has been stabilized at approximately 60%, the standby main feedwater pump can be placed in service and the plant can be quickly returned to full power operation. These design features have been demonstrated during testing and operations at PVNGS.

ABB-CE designed NSSS have consistently led the nuclear industry in the U.S. in performance, demonstrating the benefits of conservative plant design and prudent operation. For twelve of the past fourteen years, ABB-CE designed NSSS units have led the USA nuclear industry in capacity factors. In fact, eleven ABB-CE designed NSSS plants established new records for energy generation or for continuous operation during the past three years. As examples, Fort Calhoun set a world record of 477 days for continuous energy generation by a light water reactor; Palo Verde Unit 3 set world records for annual energy generation and first cycle continuous energy production and St. Lucie Unit 2 established annual capacity factor records among all USA nuclear power plants.

The historically good performance of these units (for fourteen years in the case of Calvert Cliffs 1) supports the view that the 1989 capacity factors reflect a set of unique conditions and that, once these conditions have been corrected, future capacity factors will again return to their historically high values.

The envious capacity factor record achieved with the ABB-CE nuclear plant designs are certainly indicative of the careful design, manufacture, installation and operation considerations incorporated into each plant. The continued reduction in unanticipated trips each year as the utility operators experience base increases indicates that the fundamental plant design is very sound and supportive of the requirements for high availability.

10. Spare Parts and Consumables

The Nuclear Spare Parts business unit of ABB Combustion Engineering has been a supplier of quality related spare and replacement parts to the nuclear industry for over twenty four (25) years. Through an approved supplier network consisting of both internal and external sources our clients needs for qualified spare and replacement items have been satisfied. ABB Combustion Engineering is dedicated to provide reliable (qualified) new and spare/replacement parts which are certified for use in safety related and non-safety related applications.

The spare parts and consumables stocking strategy is developed to support maximizing system and equipment reliability and availability while minimizing inventory costs. Evaluation of spare parts requirements is an integral part of the overall nuclear plant design process.

The emphasis on a comprehensive spare parts strategy produces a plant in which a low inventory cost has been specified during the design phase. During the technical evaluation and specification of equipment and components the cost of required inventory is also considered. Criteria is developed by which equipment and components can be compared relative to life of plant spare parts costs. The following aspects are considered:

- standardization of equipment to minimize stocking levels
- operational performance of equipment in nuclear applications
- vendor recommended spare parts lists
- spare parts lead times
- spare parts shelf life
- spare parts cost
- utilization of industry standard equipment

Emphasis is placed on the use of standard equipment where possible, which reduces parts costs and lead time over the life of the plant. The spare parts order is integrated with the initial equipment order whenever possible to reduce initial stocking costs.

Actual stocking levels for specific equipment is determined using the following approach. The starting point used to identify stocking levels is the equipment vendors recommended spare parts lists. These lists are compiled and reviewed, with the resulting list modified considering cost of spares, inflation factors, usage predictors, replacement lead times, parts criticality, shelf life, plant preventive or planned maintenance programs, and parts failure histories. A reliability centered maintenance (RCM) approach is considered to determine part criticality, probability of part failure and impact on the plant. The RCM methodology utilizes PRA and/or equipment failure history review to identify parts which have the highest potential for impacting overall plant reliability/availability. Specific criteria are developed to determine stocking levels considering these factors. From these criteria a program of periodic stock level review is created, which allows for continual control and improvement in inventory throughout the life of the plant.

I. ALARA

Radiation protection features important to assuring personnel and public radiation exposure is maintained ALARA are incorporated in the design. Many lessons have been learned from the current generation of nuclear power plants in the area of radiation protection. The System 80+ design incorporates these lessons to achieve the goal of limiting the collective personnel exposure to less than 1 man -Sv/yr (100 man-rem/yr) in accordance with the EPRI Advanced Light Water Reactor Utility Requirements Document. The radiation protection philosophy of ALARA anchors a fundamental commitment to the safe operation of a nuclear plant. This commitment includes not only the plant personnel, but also the general public who live and work in the surrounding communities.

1. Safety and Licensing Requirement

General

System 80+ is designed to satisfy stringent U.S. Regulatory Guides and requirements related to radiation protection. These regulatory guides provide guidance for the plant layout and operation, equipment design and selection, and system design to maintain doses to the plant personnel and the general public ALARA. These include:

- The requirements of the USNRC Regulatory Guide 8.8 for specific features which, if incorporated, will give high confidence that the ALARA principle will have been satisfied.
- Radiation dose criteria of U.S. 10CFR20, limiting the exposure of staff during normal plant operation, as well as limiting the maximum allowable concentration of gaseous and liquid wastes in the unrestricted area.
- Radiation dose criteria of U.S. 10CFR50, Appendix I, limiting the exposure of the public during plant operation.
- Radiation dose criteria of U.S. 10CFR50, Appendix A, General Design Criteria (GDC) 19, limiting the exposure of staff (control room operators) during accident conditions.
- Radiation dose guidelines of U.S. 10CFR100, designed to limit exposure of the public during accident conditions.

System 80+ design goals are to maintain the dose to the operating personnel and general public ALARA, in accordance with Regulatory Guides 8.8 and 8.10. The System 80+ design goal is to limit the collective dose to operating personnel to less than (100 man-rem/year) in accordance with the EPRI ALWR Utility Requirements Document. Individual dose reduction features used to minimize the dose to plant personnel and the general public are evaluated and implemented based on whether they are cost-effective and the radiation exposure is avoidable. Dose reduction techniques, such as increased shielding and distance between personnel and the radiation source and source term control are considered and implemented in plant design, equipment selection and design, and plant operation.

A cost-benefit analysis is performed to establish a favorable cost-benefit ratio to ensure reasonable efforts are taken to minimize the dose and maintain it ALARA. 10CFR50, Appendix I specifies a favorable cost-benefit ratio of at least $1000/man-rem to the whole body or critical organ (typically the thyroid) to minimize dose to the general public from radioactive gaseous and liquid effluents released to the unrestricted area.

To maintain occupational dose ALARA includes maintaining both annual individual and the collective occupational dose ALARA. A dose assessment is typically performed in accordance with Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Plants Design Stage Man-Rem Estimates," guidelines to identify major tasks with a significant contribution to the collective occupational dose (expressed in man-rems). The following sections provide examples of various radiation protection design features incorporated into the System 80+ design to maintain radiation exposures ALARA.

2. Design Features and Alternatives for Reduction

System 80+ incorporates various dose reduction features that significantly reduce occupational exposure. These include material selection, chemistry control, corrosion product reduction, shielding, and equipment selection discussed in the following section.

Source term control is an important aspect in the design in maintaining personnel exposure ALARA. Corrosion product production accounts for a significant portion of the total dose received by plant personnel. Incorporated are design features that not only reduce the production of corrosion products, but also minimizes the collection of corrosion products (crud) in crud traps. These crud traps result in localized hot spots that require additional maintenance and therefore increased personnel exposure. The implementation of corrosion product reduction features into the design reduces the overall dose due to operation, maintenance and inspection activities.

a. Material Selection

Materials have been selected to reduce the production of activation products, such as crud, in the reactor coolant system. One mechanism for corrosion product production is the erosion of metallic surfaces containing primarily cobalt and nickel impurities. These metallic fines can then be circulated through the reactor resulting in their activation. These corrosion products can then be deposited at low points or crud traps creating hot spots.

The design uses materials with low cobalt and nickel impurities ($\leq$ 0.020 w/o) and corrosion rates for components and piping in direct contact with primary system water to reduce in the production of radioactive corrosion products. Cobalt based alloys will be avoided except in cases where no proven alternative exists. Low cobalt alloys are utilized for control rod drive materials. In addition, the materials (such as stellite) used hard facing application for bearing journals and valve seats, which have a high content of cobalt, are minimized whenever possible. These materials are eroded or corroded from the base metal and transported to the reactor core region where it is activated.

Efforts are ongoing in the industry to develop cobalt free alloys for use in hard facing materials. For instance, current studies, sponsored by Electric Power Research Institute

(EPRI) in the United States, are evaluating cobalt-free hard facing materials for valves. Materials, such as NOREM, may be available in the near future.

The steam generator design uses Inconel 690, which has a lower nickel and cobalt content than Inconel 600 used in the current generation of nuclear power plants. This will reduce the buildup of corrosion products in the steam generator requiring sludge lancing during refueling outages.

b. Chemistry Control

The chemistry control program for the reactor coolant system (RCS) also minimizes radiological consequences. The EPRI primary chemistry guidelines, as currently revised, are incorporated in this program. During the pre-core operation period, the RCS is operated at temperatures above 350°F with an elevated pH to form a protective oxide film on metal surfaces. This oxide film resists chemical attack during subsequent plant operations, thereby limiting further corrosion of RCS materials and consequent formation of activated corrosion products. Chemistry control during normal plant operation requires an adequate hydrogen inventory to scavenge corrosion-inducing oxygen, and coordinated lithium-boron control to maintain pH at a level which minimizes the precipitation of corrosion products on the core. This reduces the activation rate of these corrosion products and the subsequent formation of out-of-core radiation fields. Additionally, studies have shown that a sufficiently high pH minimizes the corrosion rate of RCS components. The pH of the reactor coolant is increased from 6.9 to 7.4 to reduce the equilibrium corrosion rates and the buildup of corrosion products on the primary system surfaces. These features reduce the out-of-core radiation fields and therefore the personnel exposure associated with maintenance and inspection activities.

c. Corrosion Product Accumulation Avoidance

The design minimizes corrosion product accumulation in systems in contact with reactor coolant. These systems are designed to avoid low points or stagnant pipe legs where accumulation of corrosion products can occur. In addition, piping in contact with reactor have smooth internal surfaces to prevent the accumulation of radioactive corrosion products in internal crevices.

d. Decontamination

Flushing capability will be provided in the primary system to facilitate decontamination of the piping. Additional methods of decontamination include chemical decontamination. The primary system components are designed to be compatible with the chemical decontamination techniques to reduce personnel exposure for ALARA objectives. Current nuclear plants have successfully performed decontaminations of primary system components, such as steam generator primary heads and reactor coolant pump impellers. The low oxidation-state metal ions (LOMI) soft decontamination process is used as a model process for design considerations. All chemical concentrations are less than 3000 ppm. Subboiling temperatures are used throughout, alleviating the need to pressurize the system. In addition, all decontamination solutions can be borated as a precaution against possible boron dilution accidents.

e. Shielding and Plant Layout

The plant is divided into radiation zones to maintain personnel exposure ALARA and aid in plant layout. Sources are identified and their associated source strengths are developed based on 0.25% failed fuel cladding rate in accordance with the USNRC NUREG-0800, the Standard Review Plan, Section 12. From this shielding codes, are used to evaluate the dose rate to surrounding areas from these sources based on their source strength, shielding provided, and geometry. Careful attention has been given to the location of penetrations so that they are not in a direct line of sight with the source. This minimizes the potential for streaming of radiation through a penetration and the adverse impact on equipment qualification and personnel exposure.

In addition, access is controlled by the use of locked doors to high radiation areas. These areas are also provided with shield doors or labyrinth entrances which ensure at least two reflections of radiation from the source to the entrance. This reduces the personnel exposure due to the scatter of radiation.

Shielding is provided between redundant components. Redundant components are located in separate cubicles whenever possible to minimize personnel exposure received during maintenance activities. The cubicle walls, which are typically four feet thick, provide an effective means of shielding. Similarly, highly radioactive components, such as ion exchangers, are located in separate compartments or cubicles. Components, such as valves, are located in valve galleries so that operational and maintenance activities can be performed in a lower radiation area.

Permanent shielding is provided whenever possible. Temporary shielding is provided, as necessary, during operational and maintenance activities.

The general arrangement design incorporates ALARA principles to minimize personnel exposure. They include system and component location, spacing, and pipe routing. For instance, the general arrangement or the plant layout provides for the physical separation of radioactive systems from nonradioactive systems. This helps control the spread of contamination and minimize the necessity for routing radioactive piping through personnel corridors, as well as facilitating radiation area access control. Radioactive piping is routed through shielded pipe chases whenever possible. The number of active components located in the pipe chases are minimized to reduce the frequency of access required into the pipe chase for maintenance activities. In addition, adequate spacing around equipment for easy access to facilitate maintenance and inspection activities is provided. This includes provisions for equipment laydown and pull areas, platforms, as well as transport paths to facilitate removal, transport, or replacement of equipment or portable shielding during maintenance activities.

f. Special Tools and Job Preplanning

Dose can be reduced during operation by the use of remotely operated equipment, such as reach rods and robotics. These tools will be used whenever there is a direct dose reduction benefit or there is no other means to perform the task. These tools can reduce personnel exposure by enabling personnel to perform activities more efficiently or by increasing the distance between the personnel and the radiation source.

Provisions have been made in the design of various components to facilitate the use of special tools, such as robotics, for maintenance activities. For instance, the size of the manways for the steam generator has been increased to 21 inches to permit accessibility for robotic equipment to form routine maintenance and inspection activities.

Included in job preplanning is a total dose estimate of person-rems required to complete the job. The plant layout will be designed on a three dimensional graphics program called PASCE. This model can be used in future plant operations for developing three dimensional dose maps. These maps can be generated using PASCE which integrates the plant layout graphics with specific area information, such as dose rate and source location(s) as measured by health physics personnel. This information can be readily used by health physics personnel to estimate dose, as well as by personnel in the field to effectively implement ALARA principles of time, distance, and shielding during maintenance activities. In general, much work will be performed outside the radiation area including reading manuals or maintenance procedures, adjusting tools, repairing valve internal, and prefabricating components.

g. Waste Management Systems

The waste management systems are designed to segregate waste based on the radiation level, physical and chemical characteristics, and the type of waste (solid, liquid, or gaseous). By segregating waste streams, processes can be tailored to the unique characteristics of each waste stream. This improves the efficiency of the process and prevents the mixing of waste streams, thus minimizing the radiation exposure to the personnel.

The waste management systems are designed to process waste streams resulting from 1% failed fuel cladding during normal operations so that liquid and gaseous discharges are maintained less than 10CFR20 limits. These systems are also designed with sufficient surge capacity to accommodate the maximum expected production rate of waste during normal operating, refueling, and shutdown conditions. The liquid waste management system is designed with additional decontamination and recirculation capabilities. The waste management systems are not designed to provide decontamination of post-accident source terms during post-fault conditions. These systems are typically isolated during post-accident conditions by the control room operators.

h. Fault Mitigation

Radiological consequences of faults (accidents) are mitigated by the processing of the inplant atmosphere and effluents discharged by ventilation systems designed in accordance with Regulatory Guides 1.52 and 1.140, Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants" and "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants," respectively. These standards specify the system operation and filter efficiencies for removal of organic, particulate, and elemental iodine. These ventilation systems include:

- Containment Annulus Ventilation System is used to mitigate the offsite dose consequences resulting from a design basis accident. The filters provided have efficiencies of:

elemental iodine	0.95
organic iodine	0.95
particulates	0.99

- Control Room Ventilation System operation considered in control room habitability during post accident conditions. This system maintains a positive pressure on the control room to minimize unfiltered inleakage of radioactive gases. In addition, the filters mitigate dose to control room operators. The filters provided have efficiencies of:

elemental iodine	0.95
organic iodine	0.95
particulates	0.99

- Fuel Building Ventilation System is used to mitigate the dose consequences of a fuel handling accident in the fuel building. The filter provided have efficiencies of:

elemental iodine	0.95
organic iodine	0.95

The filter efficiencies listed are dependent on the flow rate and moisture content of air being processed. Provisions are made in these systems to provide adequate flow rate and moisture content control to ensure the above filter efficiencies are maintained in accident conditions.

In addition, containment spray provides for the reduction of the concentration of radioiodine in the containment atmosphere in post accident conditions via fission product scrubbing by the containment spray water. The design ensures that there is 65% coverage of the total area of containment. This ensures sufficient removal capability for radioiodine so that radiological consequences can be effectively mitigated.

3. Radiation Control

Radiation Protection Design

Lessons have been learned through the operation of the current generation of nuclear power plants. These lessons have been incorporated into the System 80+ plant design, layout, equipment design and selection in accordance with Regulatory Guide 8.8 and 8.10 guidelines.

a. Civil, Plant, and Equipment Design

The plant is designed so that radioactive and nonradioactive components are segregated. This minimizes the potential for the spread of contamination and the need to route radioactive piping through or adjacent to personnel corridors. Radioactive piping is routed through shielded pipe chases whenever possible to minimize personnel exposure. Redundant radioactive components are located in separate cubicles so that maintenance can be performed in a lower radiation area. Adequate spacing is provided to ensure sufficient laydown or equipment pull area for maintenance to be formed. Shield walls are generally at least four feet thick.

Platforms are provided around equipment such as the steam generators and reactor coolant pumps to facilitate accessibility during maintenance and inspection activities. Penetrations are located so that they are not in a direct line of sight with the source. This minimizes the adverse impact of radiation streaming on equipment qualification and personnel exposure. Locked doors are provided to high radiation areas to ensure sufficient access control to these areas. Labyrinth entrances are also provided to minimize the personnel exposure due to scattered radiation.

A shielding analysis is formed for both normal and post accident conditions to ensure accessibility to vital areas and personnel exposures are maintained ALARA. Sources are identified and evaluated based on associated source strengths, geometry, and shield thickness and composition using accepted shielding codes. These analyses calculate the reactor coolant equilibrium source term assuming 0.25% failed fuel cladding. This source term is then used to calculate various component source terms taking credit for decontamination through filters and ion exchangers as appropriate. In addition, the concentration in filter and decontamination media is calculated to determine the solid waste management system isotopic inventory for the shielding analysis.

b. Pipework

It is a design objective of System 80+ to minimize dead legs, flow restrictors, etc., which may lead to crud traps.

Experience from past designs and inservice inspection programs has resulted in design features being incorporated that reduce occupational radiation exposure. The most significant improvement for performing inservice inspection is the reduction of linear feet of weld in the major components. The reduction in weld footage has been accomplished by component redesign, use of forged sections versus forged-welded plate sections, and increasing the size of certain sections.

c. Radioactive Vents and Drains

In System 80+, Radioactive vents and drains are segregated from nonradioactive vents and drains to prevent the spread of contamination in the plant. Radioactive fluids collected will be routed to the liquid waste management system to be processed and monitored prior to release to the environment. Nonradioactive drains will be monitored prior to release to the environment. Nonradioactive drains will be monitored for radioactivity. If they are radioactive, they will be directed to the liquid waste management system for further processing prior to release. Radioactive vents from tanks, such as the equipment and reactor drain tanks, are routed to the gaseous waste management system for processing via a charcoal delay system prior to release. This system is designed to delay the release of radioactive gases so that the concentration of radioactive gases at the exclusion area boundary is within 10CFR20 limits during normal operation. Ventilation systems are provided for containment, reactor building subsphere, the nuclear annex, the fuel building, and the control complex to ensure gaseous effluents are filtered prior to release. All release pathways are monitored prior to release.

d. Floor Drains

In the System 80+, a floor drain system is provided to collect equipment leakage and condensate. This system is segregated into potentially radioactive floor drains and nonradioactive floor drains. These drains are physically separated to prevent the inadvertent contamination of the nonradioactive drains by contaminated liquid. These radioactive drains are directed to the liquid waste management system for processing prior to release. The nonradioactive drains are monitored prior to release to ensure there is no inadvertent release of unprocessed radioactive liquid. If this drain in radioactive, these fluids are directed to the liquid waste management system for processing.

e. Ventilation (HVAC) Provisions

Plant ventilation provisions are provided for containment, the reactor subsphere, the nuclear annex, the fuel building, and the control complex. These systems are designed so that flow is from areas of lower to areas of higher potential activity. This design minimizes the potential for the spread of contamination. The System 80+ design ensures the inplant concentration of airborne contamination is within 10CFR20 limits.

f. Radiation Monitoring

The System 80+ design provides area, airborne, and process radiation monitoring in all areas that are potentially radioactive or contaminated. These monitors provide indication and alarms of abnormal condition in the plant. Process monitors are used to provide control actions, as necessary, to terminate releases or divert flow through filters or to an appropriate system for processing prior to release. Alarms are provided both locally and in the main control room. The activity levels and the status of the monitors are monitored via the DIAS and DPS by the operators. Area monitors are provided with both visual and audible alarms to alert personnel to abnormal radiological conditions in the plant.

4. Radwaste Processing Leading to Safe Storage/Disposal/Discharge

The principal design objectives of the radioactive waste management systems are:

- Collection of wastes generated during anticipated plant operations which potentially contain radioactive nuclides.
- Provision of sufficient processing capability such that wastes, both liquid and gaseous, may be discharged to the environment at concentrations well below regulatory limits specified in 10CFR20 and consistent with ALARA guidelines specified in 10CFR50, Appendix I.
- To provide collection for the collection, processing, solidification, packaging, temporary storage, and preparation for shipment of wastes generated during plant operations and maintenance, while maintaining personnel exposure

ALARA in accordance with Regulatory Guide 8.8 and 8.10, 10CFR20 and 10CFR50.

a. **System 80+ Gaseous Waste Management System Design**

The Gaseous Waste Management System (GWMS) provides means to collect, store, process, sample, and monitor radioactive gaseous waste prior to release. Gaseous waste is collected from the Reactor Coolant System (RCS), the Chemical Volume and Control System (CVCS), Liquid Waste Management System (LWMS), and aerated gases from various vents in the plant. These gases are processed in charcoal delay system. The GWMS contains conditioning equipment to minimize moisture and contamination in the charcoal absorbers, as well as charcoal absorbers to delay the passage of noble gases through the equipment. The effluent is discharge through the radwaste facility vent via HEPA filters. Dual hydrogen analyzers are used to provide indication in the control room of high concentrations of hydrogen and oxygen in the GWMS. These concentrations are maintained less than 4% by use of nitrogen purge to preclude the buildup of an explosive mixture of hydrogen and oxygen.

The GWMS effluent is continuously monitored to ensure the concentration of radionuclides at the unrestricted area are within 10CFR20 limits. The radiation monitor terminates release from the GWMS if a preset limit is exceeded. This provision prevents the inadvertent discharge of effluent in excess of regulatory limits.

The current effluent analysis is performed based on a gaseous source term assuming 0.25% failed fuel cladding.

b. **System 80+ Liquid Waste Management System Design**

Liquid Waste Management System (LWMS) provides the capability to collect, segregate, store, process, sample, and monitor radioactive liquid waste. The LWMS is designed to segregate waste to minimize the potential for mixing waste streams. This enables the operator to tailor the system based on the properties of the waste stream. The LWMS is divided into subsystems to process each of the following categories of waste:

- Equipment drain waste which includes, for example, degassed reactor grade radioactive liquid waste

- Floor drain waste which includes, for example, non-reactor grade radioactive liquid waste

- Detergent waste which includes, for example, laundry and hot shower waste liquids

- Chemical waste which includes, for example, non-detergent liquid waste.

These waste streams are not interconnected prior to collection and processing. The LWMS is provided with filtration, decontamination by mixed bed ion exchangers, as necessary. The LWMS is provided with quick disconnects which enable the operator to connect additional ion exchangers, mounted on skids, as needed. The waste is batch sampled prior to processing and release and recirculated for further processing as needed. Waste specific pretreatments, such chemical addition to adjust pH and flocculent addition, oil and crud removal from floor drain waste are provided as necessary.

A radiation monitor is provided at the radwaste plant discharge to monitor the concentration of the effluent to ensure it is in compliance with 10CFR20 limits. This monitor terminates release if a preset limit is exceeded. This minimizes the potential for an inadvertent release of effluent to the environment in excess of 10CFR20 limits.

The current effluent analysis is formed based on a gaseous source term assuming 0.25% failed fuel cladding.

c. <u>System 80+ Solid Waste Management System Design</u>

The Solid Waste Management System (SWMS) is designed to collect and process waste for shipment to a licensed burial site. The solid waste management system is provided with the capability to dewater or solidify wet solid wastes, and compact dry compactible waste. Noncompactible dry solid waste is simply placed in a shipping container for shipment. The wet solid wastes are processed in accordance with 10CFR61 requirements. Wet solid wastes are dewatered to ensure that the liquid content is less than 1% by volume prior to shipment. The owner operator will develop procedures to establish initial boundary conditions and processing requirements, such as settling time, dewater times, etc. The processed waste is shipped offsite to a licensed burial site in accordance with 10CFR72 and Department of Transportation requirements. The interim radwaste storage facility is designed to provide additional storage capacity for up to 6 months. This facility is located in close proximity to facilitate transport of shipping containers from the radwaste facility to interim radwaste storage facility.

d. <u>Radwaste Facility</u>

The LWMS, GWMS, and the SWMS are housed in the Radwaste Facility which is designed in accordance with Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear power Plants." This regulatory guide requires curbing and drains to prevent or minimize the potential for an uncontrolled release of liquid effluent from the building. This facility is also designed to withstand an Operating Basis Earthquake (OBE).

e. <u>Radioactive Waste</u>

The overall radiowaste waste generated by System 80+ is shown in Tables III.K-8a, b and c.

J. SAFETY (Including PRA)

1. Safety of Design

The engineered safety features of the System 80+ design are designed to prevent and/or mitigate the effects of core damage accidents. As an evolutionary ALWR, System 80+ not only satisfies USNRC regulatory requirements in this regard, but also addresses the safety philosophy and requirements set forth in the EPRI ALWR Utility Requirements Document (ALWR URD). This section describes the System 80+ design philosophy for engineered safety systems that responds to these two sets of requirements.

A basic principle of the ALWR design philosophy used in the development of the System 80+ engineered safety systems is the differentiation between core damage prevention and core damage mitigation. This distinction is based on the overall ALWR concept of defense-in-depth, which includes fault tolerant design, reducing the frequency and consequence of faults and mitigating the consequences of faults that may occur. The highest priority is assigned to prevention of core damage accidents, followed by mitigation.

The System 80+ reactor vessel and primary reactor systems are housed in a steel containment structure which is surrounded by a concrete reactor shield building. This redundant containment design provides a means of preventing fission products from being released to the environment in the unlikely event of an accident. Other physical barriers are employed to prevent or limit fission product release as part of the traditional defense-in-depth philosophy.

As a first level of defense-in-depth, the reactor, reactor coolant, and auxiliary systems are designed to be accident resistant. Substantial design margins are utilized in systems design to ensure that unanticipated demand for actuation of safety systems (e.g., transients and external events) is infrequent. The next level of defense-in-depth further emphasizes core damage prevention, which is also a function of the System 80+ engineered safety systems. Thus, these safety systems are high capability, high margin, and simple in design. The next level of defense-in-depth emphasizes mitigation systems, which are designed to contain and control fission product activity released to the containment in the very unlikely event of a core damage accident. Together, these systems ensure that the public and the environment are protected from such an accident.

a. <u>Safety Design Basis</u>

The complete set of requirements which form the foundation of the System 80+ engineered safety features design is collectively referred to as the safety design basis. The EPRI ALWR Program found it convenient to further differentiate the safety design basis into two categories, since the URD requirements were intended to enhance safety beyond that required by USNRC regulations. These categories are referred to as the Licensing Design Basis (LDB) and a Safety Margin Basis (SMB).

The Licensing Design Basis requirements are those which are necessary to satisfy the USNRC requirements for nuclear power generating plants as defined in the Code of Federal Regulations (CFR) and in associated regulatory guidance. Therefore, conservative NRC-approved calculation methods and assumptions are used for safety systems analyses, and the systems meet NRC-mandated acceptance criteria. For the purposes of meeting regulatory limits for Licensing Design Basis events, only safety-related equipment is assumed to be available.

The System 80+ design incorporates features responsive to the ALWR Safety Margin Basis requirements in order to provide safety margins beyond the minimum required by the USNRC. The Safety Margin Basis is intended to provide additional safety assurance for investment protection and severe accident (beyond design basis) protection. The concept of increased investment protection focuses on minimizing an owner's financial risk, with the added value of improved plant safety by improving accident prevention. Severe accident protection incorporates NRC policy-level guidance; the impetus is to provide an increased assurance of containment integrity and low leakage of radioactivity during a severe accident. While conservative, NRC-specified design methods and acceptance criteria are required under the LDB, best-estimate methods may be used to confirm the adequacy of design margin under the SMB. Further, this approach factors in industry operating experience and accepted good engineering practice.

Within this context, design requirements for safety may be associated with each of the three levels of defense-in-depth described above. For example, the System 80+ LDB addressed accident resistance by adopting regulatory imposed margins, in-service inspection and testing requirements, and requirements associated with ensuring reactor coolant system boundary integrity. Core damage prevention is provided by safety systems that meet USNRC regulatory requirements for specified accidents, without exceeding regulatory fuel limits. Mitigation is based on the NRC requirements for Loss-of-Coolant Accidents (LOCA), the design basis for containment and assumed source terms.

The System 80+ SMB provides for additional accident resistance through the use of increased safety margins, greater simplicity, and specification of imposed system and component reliability. Core damage prevention goes beyond regulatory limits, to encompass such features as increased margins to fuel damage and no fuel damage for a medium LOCA. For mitigation, provisions include conservative, rugged containment systems and a challenging large release requirement for less than 25 rem whole body dose at the site boundary for accident sequences with cumulative frequency greater than 10^{-6} per year. This large release requirement affords a substantial margin to the NRC safety goal, and will be confirmed by the PRA. The mean annual core damage frequency goal is less than 1×10^{-5} events per year.

The following paragraphs describe the functions and features provided for core damage prevention and mitigation, and indicate the consideration given to hypothetical severe accident scenarios.

b. <u>Core Damage Prevention</u>

Safety functions that act to prevent initiating events (e.g., transients, equipment failures) from progressing to the point of core damage are collectively referred to as core damage prevention function. Core damage prevention addresses not only the direct safety functions of the engineered safety features, but also support functions provided by non-safety auxiliary systems and on the safe operation of normal plant operating systems. The direct core damage prevention functions includes:

- The core coolant inventory function - the System 80+ Safety Injection System (SIS) provides makeup water from the in-containment refueling water storage tank (IRWST) to the Reactor Coolant System (RCS) to prevent significant core heatup. A related system, the Containment Spray System (CSS) is used to provide containment cooling and containment atmosphere fission product removal.

- The decay heat removal function - the System 80+ Emergency Feedwater (EFW) and Shutdown Cooling (SC) Systems provide for the transfer of sensible and decay heat from the RCS to the atmospheric dump valves and the Ultimate Heat Sink (UHS).

- The diverse reactivity control function - the System 80+ Safety Injection System (SIS) provides a normal means of inserting negative reactivity into the reactor core beyond that of the control rods; a backup means is provided by the Chemical and Volume Control System (CVCS).

- The RCS pressure control function - the System 80+ Safety Depressurization System (SDS) is designed to preclude RCS overpressure and to allow for RCS depressurization. This depressurization function is included in the core coolant inventory control function.

Specific features of the System 80+ engineered safety systems which are intended to prevent core damage are:

- Safety system functions are assured by use of redundant divisions. This ensures that plant design meets the single failure criterion as specified in 10 CFR Part 50, Appendix A, General Design Criteria, Definitions and Explanations. The number of divisions employed for engineered safety systems is determined using the occurrence of the most limiting single failure as the basis for analysis. Each division of a system important to safety is totally independent and separated both mechanically and electrically, except for those areas where it is physically not practicable or is less safe to do so, such as in the control room and at the reactor vessel.

- Systems are provided to maintain the plant in a safe condition during a station blackout (loss of off-site and on-site AC power). For station blackout, a safe condition is defined as a plant condition in which the reactor is subcritical, the core is covered with water, and no design limits have been exceeded. The condition must be such that upon restoration of AC power to one division, plant recovery can commence.

- The capability to depressurize and cool down the primary system is provided using safety grade equipment, assuming an initiating event and the most limiting single failure. Reactor core cooling systems, including the Residual Heat Removal System, are provided with sufficient instrumentation to provide the operator with accurate indication of core cooling conditions.

- The System 80+ safety-related core cooling systems are sized such that adequate inventory of cooling water is available for core cooling in compliance with 10 CFR Part 50.46 fuel cladding temperature limits for design basis events, with a design margin to account for flexibility in operations.

- Safety-related systems are designed such that no fuel damage is predicted to occur, based on best-estimate calculations with full system operation, for a postulated near instantaneous pipe break with an area equivalent to up to six inches in diameter in the reactor coolant boundary.

c. <u>Core Damage Mitigation</u>

In the unlikely event of core damage, the System 80+ mitigation functions are designed to limit on-site and off-site radioactivity releases to acceptable levels. The systems and structures which provide this function are operable during normal, off-normal and design basis events to limit radionuclide release as well as protect personnel and equipment from both internal and external environmental hazards. The principal mitigation functions are:

- The containment integrity function. This assures high containment integrity under core damage condition. This function includes containment boundary testing, heat removal to maintain containment integrity following events that release mass and energy to the containment, isolation in the event of conditions which could lead to core damage, and combustible gas control, which ensures that the hydrogen gas which could be generated as a consequence of core damage does not threaten containment integrity.

- The fission product control function. This ensures that radioactive fission products released from a damaged core are controlled so that specified allowable leakage from the containment does not pose a significant off-site dose threat. This function includes both removal of fission products from inside primary containment, from within the annulus between the primary and secondary containment structures, and control of fission product leakage through containment.

Principal System 80+ design features which contribute to the mitigation of core damage in conformance with the licensing design basis (LDB) are:

- Containment systems are designed to provide for retention of radionuclides for LDB events. Maximum off-site dose levels are calculated to be within US regulatory limits and accepted values for source terms.

- The System 80+ plant is designed with both a primary containment and secondary containment. The Annulus Ventilation System (AVS) serves the space between them, and is expressly designed to mitigate the consequences of airborne products of radiation that might otherwise become an environmental hazard during and following an accident. The AVS serves no normal ventilation function; its purpose is specifically to provide additional assurance against the release of fission products to the environment.

- Active systems used for the containment safety function are designed in accordance with the single failure criterion and specify safety-grade components.

- The containment and containment cooling systems are designed to contain or remove the energy associated with energy sources within the containment as a result of a Design Basis Accident, including sensible energy and energy generated as a result of the event, without exceeding design limits.

- The System 80+ containment design leakage rate meets applicable off-site dose limits for conditions in the containment using conservative US regulatory methods. The Containment Isolation System assures that containment leakage is within design values and regulatory limits. Provisions are made for containment leak rate testing to demonstrate that design leakage rate is in accordance with NRC requirements.

- The Hydrogen Mitigation System provides for combustible gas control following a core damaging accident. The System 80+ plant meets the regulatory limits imposed by 10 CFR Part 50.34, which requires that for an assumed 100% metal-water reaction of active fuel cladding, the resulting uniformly distributed hydrogen concentration in containment is kept below 10% by volume or the containment is inerted. Containment integrity is maintained during an accident with accompanied hydrogen release.

- The pH level in containment sumps is maintained at a level such that, in the event that fission product iodine should collect there after an accident, formation of elemental iodine from radiolysis in water does not preclude meeting the off-site dose limits allowed by the EPRI ALWR program.

Principal System 80+ design features which contribute to the mitigation of core damage in conformance with the safety margin basis (SMB) are:

- Reliable containment isolation is provided, such that containment bypass due to isolation failure is improbable.

- The containment structure is designed with sufficient margin to provide protection against both short-term overpressure failure (several hours) and longer-term overpressure failure and temperature failure. This conforms with EPRI ALWR provisions for severe accident protection.

- The overall size of the containment and the designed lack of compartmentalization within containment allow for prevention of hydrogen buildup by providing for good mixing of containment air through natural circulation; in enclosed spaces within containment, hydrogen igniters are provided. The System 80+ containment is sized to assure that the uniformly distributed hydrogen gas concentration does not exceed 13% under dry conditions for an amount of hydrogen equivalent to that generated by oxidation of 75% of the fuel cladding surrounding the active fuel.

To accommodate severe accident scenarios, the following design considerations are employed in the System 80+ design for mitigation of core damage:

- The System 80+ Severe Accident Analyses are predicated on the assumption that the containment will be inaccessible for a long term period after a severe accident.
- Reliance on active equipment inside the containment is minimized for severe accident analyses.
- The reactor cavity can be flooded after a severe accident to prevent core-concrete interaction. Adequate cooling of core debris is provided.
- Decay heat can be removed from core debris after the severe accident for an extended period, up to the time at which active cooling means are no longer required.
- The containment is designed to retain fission products after a severe accident.
- Plant instrumentation is designed to provide an operator with sufficient information to determine whether core damage has occurred, such that the operator may take necessary actions for severe accident mitigation.

The overall System 80+ design program for fault mitigation is based on the philosophies and features outlined above. The System 80+ incorporates sufficient design margins to optimize capabilities for post-fault operations while ensuring continued plant safety. As detailed plant design progresses, final systems setpoints, technical specifications, detailed operating procedures and post-accident management procedures will be developed, including accident management procedures for severe accidents.

2. Approach to Hazards

The System 80+ standard nuclear plant is designed to meet or exceed the requirements to which it is being reviewed by the USNRC under the Design Certification Program for ALWRs. To ensure meeting the objectives in performance, reliability, and safety, the System 80+ is designed in anticipation of certain external events which are unrelated to the energy conversion/electricity generation processes, but whose occurrence could damage the plant. These events are referred to as hazards. These hazards, US regulations which govern them, and the System 80+ design features used to accommodate them are discussed in this section.

The overall approach to treatment of hazards used in developing the System 80+ design consists of identification and evaluation of those events which are potential hazards. A set of potential hazard events is defined based on USNRC requirements and utility industry operating experience. These events are then considered individually or are placed in groups as appropriate for further design analyses. This approach is consistent with the extensive experience base of US operating reactors and the reliance on proven design inherent to the evolutionary System 80+. Consequently, events with potential for frequent occurrence are subject to deterministic evaluations which are based on design criteria developed from USNRC regulatory guidance and previous power plant design, construction and operations experience. Hazard events, including those which may occur infrequently, are also evaluated using probabilistic risk assessment methods. The combination of deterministic and probabilistic assessments provides a comprehensive method of assuring that the frequency and consequences of hazard events are appropriately limited.

a. Hazard Definition

A hazard is any event which may damage plant structures, systems, or equipment and has the potential to cause singly, or in combination:

- one or more initiating faults which are within the design bases
- a significant reduction in the reliability and availability of plant safeguards
- a more severe initiating fault than that assumed for design bases calculations
- an initiating fault which is outside the design bases.

Design basis hazards are those hazards which may occur with sufficient frequency that they must be considered in the facility design. Particular hazard events are assigned this classification based on deterministic criteria developed using USNRC regulatory requirements and drawing heavily on US nuclear power industry design and operating experience. Hazards which may occur infrequently and are not considered in plant design are called beyond design basis hazards. Beyond design bases hazards are evaluated using a Probabilistic Risk Assessment (PRA) analysis. Although it may be determined that occurrence of particular hazard events is unlikely for certain potential plant sites, System 80+ is designed on the basis of a set of assumed site-related parameters selected to envelop most potential nuclear power plant sites in the US.

For the System 80+ design, hazards have been classified as either internal or external. Internal hazards are defined as those hazards which originate from within the site boundary. External hazards are those which affect the station but originate outside the site boundary; external hazards may be further subdivided into those which are man-made and those which are natural phenomena.

b. General Requirements for Design Against Hazards

This subsection describes general requirements and methods of analyses used in the System 80+ design.

(1) Single Failure Criterion

The System 80+ is designed to ensure that it can be brought to and maintained at safe shutdown conditions following the occurrence of a design basis hazard. The plant is designed to the single failure criterion as required by the USNRC and the codes and standards applied to US design of safety systems. Analysis of initiating events within the design basis begins with application of this criterion; the single failure criterion is applied to both active and passive failures.

Within the US regulatory framework, 10 CFR Part 50, Appendix A, General Design Criteria), a single failure is defined as "*... an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly) results in a loss of the capability of the system to perform its safety functions.*"

In addition to the requirements found in the US Code of Federal Regulations 10 CFR Part 50, other US regulatory requirements are provided separately to address specific USNRC safety concerns. Depending on the safety issue involved, these deterministic requirements may be more limiting than the single failure criterion. For example, station blackout caused by a loss of offsite power is a result of multiple failures. The System 80+ is designed to meet the more limiting regulations in order to obtain US design certification.

(2) Internal Hazards

Potential hazards originating from within the station site boundary are considered in the plant design. This includes, but is not limited to, fires, explosions, disruptive failure of pressure parts or rotating machinery, flooding, releases of potentially damaging substances, dropped loads and failure of static structures.

(3) External Hazards

Potential hazards originating from outside the station site boundary are considered in the plant design. 10 CFR Part 50 Appendix A, General Design Criteria, GDC 2, requires that structures, systems and components important to safety be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. This includes, but is not limited to, earthquakes, tornadoes, hurricanes and other extreme winds, flooding from tsunami and seiche, extreme ambient temperatures, precipitation, and lightning. Man-made hazards, such as aircraft and other transportation, and proximate military and industrial installations, and requirements for their treatment in nuclear plant design are also covered by USNRC regulations, notably NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition". The System 80+ plant is designed to be in compliance with applicable US regulations.

Plant design features must be analyzed with respect to impact of hazard events. In conducting such an analysis of any hazard event, damage to safety related systems, structures and components is considered; failure of non-essential buildings and structures is also factored into the hazard analysis. Fires or missiles which may result from the hazard event are considered in assessing the degree of protection and separation required for essential plant features. The plant is designed such that, in the event of any of the specified hazards, safe shutdown of the reactor can be achieved.

c. System 80+ Design Against Hazards

The System 80+ plant is designed to meet or exceed applicable USNRC regulatory requirements, making it eligible for Design Certification in the US. US nuclear industry operating experience and owner-operator preferences for advanced plant design have been factored into the System 80+™ design. Thus, the overall design approach for protection against hazard events is a result of the integration of these regulations, experiences and good engineering practice.

(1) Plant Arrangement

The System 80+™ plant has been arranged to provide flood protection for safety-related structures, systems, and components, to provide fire protection in compliance with US regulations, and to prevent adverse system interactions from water intrusion, flooding, seismic events, pipe rupture, and missile generation. To provide supporting documentation for the information contained in CESSAR-DC, numerous studies have been conducted to analyze and/or define specific plant design features. Of special interest in relation to design against hazards are the "System 80+™ Flood Protection Assessment", the "System 80+™ Fire Hazards Assessment", the "System 80+™ Distribution Systems Design Guide", and the "System 80+™ Development of General Arrangements" documents.

The first-line protection from the effects of hazard events for safety-related structures, systems, and components is provided by the overall plant arrangement philosophy. The System 80+™ plant employs the use of divisional and building quadrant separation for the purposes of flood, fire, and missile hazards protection. A barrier is provided between divisions of safety-related equipment located in the reactor building and nuclear annex. Mechanical and electrical systems/components of one division are completely separated from the other division by this barrier. The reactor building subsphere is further divided into quadrants, which aids in internal flooding control. The divisional and quadrant walls also function as three hour fire-barriers, and serve to prevent interdivisional system/component interaction due to seismic movements, pipe whip, or jet impingement.

The "System 80+™ Distribution Systems Design Guide" has been created as a specification of appropriate guidelines for piping, HVAC duct, and electrical cable routing. This design guide also contains design and analysis requirements which will maintain the arrangement philosophy of divisional separation to prevent adverse structure, system and component interactions. Applicable portions of this design guide are included as Design Acceptance Criteria in the Inspection, Test, Analysis Acceptance Criteria (ITAAC) documents prepared by ABB-CE in conjunction with the US nuclear plant Design Certification licensing process. ITAAC documents are US regulatory commitments that the plant will be constructed as certified; they provide minimum acceptance criteria for startup.

(2) Flood Protection

Flood protection measures are designed in accordance with USNRC Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants." To ensure that safe shutdown capabilities are not compromised, elimination or minimization of potential sources of flooding within safety-related areas of the plant is used extensively as a means of flood protection. Interdivisional flooding from internal sources is prevented by the divisional barrier described above; flooding potential is diminished by locating major water system components outside the Nuclear Annex and by reducing the lengths of high energy and moderate energy pipe runs. Flood barriers are provided in the reactor building subsphere to protect the electrical equipment from any flooding which might occur in the mechanical equipment areas. Flood barriers are provided between quadrants within a division. Effective use of sumps and sump pumps, and curbing around equipment is made throughout the plant.

Drainage systems, suitable for collecting water reaching the site from any source, are provided to prevent any adverse effect on plant safety systems. Analysis of site flooding hazards includes water from rainfall, flood defense leakage, overtopping of flood defenses by waves, and spray; reasonable simultaneous ingress of water from these sources is considered.

Protection from external flooding is provided by elevated building entrances. Secondary flooding sources located in the Turbine Building are confined to that building; entrances from the Turbine Building to the Nuclear Annex are elevated sufficiently to allow 30 minutes before operator action is required to isolate a break in the Condenser Circulating Water system before the water level from the Turbine Building Flood reaches the Nuclear Annex entrance. The control complex is protected from flooding in that no water lines are routed above or through the control room or computer room. Water lines routed to HVAC air handling units around the control room are contained in rooms with curbs which preclude the potential for water leakage from entering the control room or computer room.

(3) Fire Protection

The System 80+ plant is designed in accordance with NUREG 0800, Standard Review Plan, Section 9.5.1, "Fire Protection Program", and BTP CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." The System 80+™ fire protection design ensures:

- compliance with US regulatory requirements
- an acceptable level of reactor safety
- protection of plant personnel
- protection of capital investment

To minimize the adverse effects of fires on plant structures, systems, and components important to safety, appropriately sized fire fighting and fire detection systems are included in the System 80+ design. Fire fighting systems are designed such that inadvertent operation or rupture of system piping will not significantly impair the capability of safety related structures, systems, or components.

The plant is divided into fire zones to facilitate fire control. Structural barriers in addition to those used for flood protection are used as appropriate to isolate and contain fires within these defined fire zones. It is assumed that a fire may occur within any zone wherein combustible substances are located or transported. Other than fire fighting equipment, all essential plant components in a particular fire zone are assumed to be incapacitated by the occurrence of a fire in that zone. Effects on passive components located within the area also are considered.

(4) Missile Protection

Possible consequences of failure of a pressure component are considered, including external thermal damage, overpressure and impulse loadings, jets, pipe whip, pressure and rarefaction waves and missiles. Measures are taken to minimize the potential for damage to safety related structures, systems, and components due to failure of any pressure component to the extent practicable. Pressure parts whose failure could be of consequence are located such that they are not exposed to the risk of impact loads; where this is not possible they are protected by missile shields or other barriers. It is assumed that sudden failure of a pressure barrier is credible, unless adequate justification of incredibility can be provided. Mechanisms of sudden failure, including fast fracture, fire, dropped loads, and sabotage, must be addressed in any demonstration of incredibility.

Similarly, it is assumed that any load that is lifted or transported, inclusive of the lifting and transporting equipment, may be inadvertently dropped or may cause impact damage, unless it can be demonstrated that the probability of such an occurrence is sufficiently low that it can be ignored. Such demonstration must show that mechanisms of sudden failure (including fire and fast fracture) can be excluded, and that adequate redundancy and/or inspection or early detection of incipient failure of the lifting/ transporting system is provided.

Devices such as mechanical and electrical turbine overspeed trips are provided wherever necessary to prevent the overspeeding of rotating machinery whose disruptive failure could cause a safety hazard. Alarms, such as excessive vibration alarms, are used to warn plant operators of incipient failures where possible. Safety related plant features are sited such that they are not subject to disintegration of adjacent rotating machinery, or are provided with missile barriers of adequate integrity. The likely size, velocity, and trajectory of potential missiles are estimated; the overall site layout is arranged such that safety related equipment and combustibles are not located within the path of turbine-generated missiles (turbine missile zone).

(5) Protection from Hazardous Materials

Adequate protection is provided against leakage, failure, explosion, missile or fire which could occur as a result of an incident involving hazardous materials generation or storage. US regulations concerning and listings of hazardous materials are located in 10 CFR Part 40 and in 40 CFR 116. In the course of plant design for compliance with those portions dealing with protection from hazardous materials, particular attention is given to:

- protection of the plant and personnel
- separation and isolation of hazardous substances from one another and from the nuclear plant
- the necessity for storage in bulk
- reasonable limitation of the size of bulk storage
- the provision of monitoring and alarm equipment

- the provision of appropriate countermeasures for use in emergencies
- inspection, testing, and maintenance of each part of the plant containing hazardous substances

(6) Seismic Design Considerations

Two classifications of earthquake excitation are used for plant design reference: the Safe Shutdown Earthquake (SSE) and the Operating Basis Earthquake (OBE). The plant is designed to withstand the effects of ground motions corresponding to the SSE. Reactor design is such that it can be safely shut down and cooled following SSE-level ground motions without causing an unacceptable release of radiation. The plant is designed to ensure the safety of the reactor, fuel storage and radioactive waste storage in the event of repeated OBE-level ground motions. Safety features are designed for a minimum of five (5) OBE events. For purposes of analysis and design, the SSE and the OBE are each assumed to occur concurrently with the most adverse normal plant operating conditions. Possible common mode effects are considered. Analysis of effects of a seismic event on a plant feature includes the subsequent analysis of effects of failure of that specific feature on any other plant feature which is important to safety.

(7) Other Design Considerations

The effects of abnormal wind loading on safety-related structures are assessed and, where appropriate, the structures are designed to withstand the effects of abnormal wind to preclude an unacceptable radiological hazard. Abnormal wind loadings are assumed to occur simultaneously with other adverse meteorological conditions, such as accumulated ice deposits on surfaces, high rainfall, and/or heavy snowfall.

The most probable effect on plant operations from lightning is the loss of off-site power, caused by a strike in the switchyard. These occurrences are accounted for in plant safety related systems design through provisions for loss of offsite power. Offsite power supply and its interface with the onsite power system are designed in accordance with IEEE Standard 765-1983. The switchyard design minimizes the probability of a single equipment failure causing the simultaneous loss of both preferred power supply circuits.

The System 80+ plant is designed for full-power operation and for safe shutdown within the range of ambient air temperatures and water temperatures specified in the design bases.

Projected and planned development around the nuclear site is considered in plant design. Possible impacts to plant safety due to any incident in an installation, means of transport, or pipeline outside the nuclear site are analyzed in compliance with requirements set forth in NUREG 0800 Section 2.2.1 - 2.2.2 and the plant is designed accordingly.

(8) Probabilistic Risk Assessment

The PRA evaluation of plant hazards consists of three steps: identification, categorization, and analyses. Events which may occur for a given site and plant are identified. This identification process is exhaustive and not constrained by limitations on the size or intensity of the events, which permits inclusion of design basis hazards as well as beyond design basis hazards. Each identified hazard is then appropriately categorized as internal or external, and if external then further classed as man-made or naturally occurring. The identified hazard list is reduced by combining events with similar consequences and plant effects for analysis purposes (e.g., hurricane winds have similar plant effects as do tornado winds and thus would be grouped together). The resultant final list of hazard events is included in the plant Probabilistic Risk Assessment (PRA).

On a generic basis for principal sites in the US, ABB-CE has completed the identification and categorization of hazards for the System 80+ design. This was performed in conjunction with the US Department of Energy's Advanced Reactor Severe Accident Program (ARSAP).

d. <u>Beyond Plant Capability Faults</u>

These events are beyond the design basis for System 80+ although in many cases the fault may be able to be handled or its effect minimized by taking advantage of System 80+ plant features.

Examples of initiating events that are beyond the plant's capabilities include:

- Rupture of the reactor vessel.
- Rupture of a main steamline in the annulus between the primary containment sphere and the secondary shield building.
- Due to assumptions of "leak before break", certain double ended pipe breaks are considered to be incredible with regard to the need for pipe whip restraints.

e. <u>Role of the Operators</u>

The licensed control room operator plays a key role in the prevention of faults. This is accomplished through a clear definition of the operator's role in monitoring and controlling the plant during both normal and emergency situations. The crew size, use of operating procedures, and operator error are also important considerations. Overview of design and operational philosophy is necessary here.

(1) <u>The role of operators in the plant and the control room</u>.

All control room actions pertinent to plant monitoring and control are to be taken only by duly licensed individuals. Control tasks are limited to the control room operators themselves, while monitoring tasks may also be performed by the other licensed individuals, namely the Shift Technical Advisor, Control Room Supervisor, and Shift Supervisor. Other individuals may monitor plant status but not as part of plant operations scenarios.

Actions outside the control room are not performed by the control room staff (except for remote shutdown, in the unlikely event of a control room evacuation). Licensed auxiliary operators will perform such extra control room monitoring and control activities as are necessary for normal operation of the plant. In the event of an emergency, all necessary monitoring and control activities can be performed in the main control room. The role of the operators in relation to the balance of the plant is one of communication and co-ordination. Such activities are considered in the human factors of the design process through the establishment of human factors standards and guidelines for the plant, control room layout, and the allocation of necessary and sufficient control and monitoring capability in the main control room.

(2) <u>The approach to staffing</u>.

Three basic staffing issues were considered in the design of the Nuplex 80+ Advanced Control Complex:

- the complex should be able to accommodate various crew sizes based on the owner/operator's preference and operating philosophy
- the complex should meet the staffing goals set forth in Chapter 10 of the EPRI ALWR Utilities Requirement Document
- the complex should accommodate crew sizes as required by pertinent regulatory agencies

The Nuplex 80+ control complex meets all of these considerations in its design. Crew sizes of one, three and six are considered feasible. Task analysis has been performed on the limiting crew size (one) to assure that the task loading on the operator could be adequately handled from hot standby to full power operations, and immediately post trip. This one person crew is part of the EPRI requirements and does not necessarily reflect the actual crew size which would be present in the control complex and would operate the plant during start-up, shut down, or emergency operations. Rather, it reflects a goal of one person staffing in the immediate controlling workspace, specifically at the main control console in the main control room.

Task analysis and subsequent availability and suitability verifications show that the control complex is adequate for all operations with a variety of crew sizes. In addition, many features of the Nuplex 80+ complex are designed to facilitate flexible crew sizes. Some of these are enumerated below:

The use of CRT displays allows full plant monitoring to be done from any CRT. Hence, individuals at any control panel, the control room supervisor's console, or in the shift supervisor's office can monitor the plant. This, combined with the control room floor plan, allows for effective communications between operators and for crew members to perform complementary functions without getting in each other's way.

In short, the approach to staffing has been to design a control room which allows flexibility while meeting requirements. The assumptions and features designed to support staffing are verified and validated using accepted human factors engineering techniques such as task analysis, function analysis, availability and suitability verification, the use of mock-ups, review by human factors professionals, and the participation of experienced operators in the design process. Final validation will be accomplished at an integrated test facility, using dynamic simulation.

(3) The role of operators in monitoring the status of plant during operation

Operator monitoring of plant status has been addressed in detail during the design of the System 80+ plant. Due to the evolutionary nature of the plant systems, function allocation (the decision of whether to assign monitoring tasks to the operator or to automate them) was not changed significantly from that employed at existing System 80 plants. Based on operating experience, function allocation was changed only in those areas which were identified as problems. Hence, the monitoring tasks assigned to the operator in the Nuplex 80+ control room remain approximately the same as at conventional plants.

Characteristics and requirements on information for the operators were identified as part of the function and task analysis process. The suitability of this information was later checked during a verification analysis. In short, the design process employed during the development of the Nuplex 80+ control complex assured that the operators had all necessary and sufficient information to perform their assigned monitoring tasks and that information was presented in a suitable format.

The evolutionary technology used for information presentation in the Nuplex 80+ control complex was employed such that operators would be able to retain the best features of conventional operating plants, such as spatial dedication, analog trend formats, etc. while achieving a workload reduction and receiving information in highly salient groupings appropriate to plant condition. The use of software backed display devices and a large screen display made this possible.

In short, operator monitoring tasks were analyzed using accepted human factors practices supplemented by operating experience. Appropriate information for monitoring was then selected for the main control room. This information was then formatted such that it appeared in highly useable states, at appropriate locations.

(4) Operating rules and procedures during normal operation.

The System 80+ Technical Specifications Operating Rules provide the limiting conditions for operations (LCO's) consistent with the design bases (safety analyses) for the plant. The Technical Specifications reflect the restructuring effort in the US intended to clarify and simplify the operating requirements for the operator. The Technical Specifications also specify the actions to be taken when an LCO is not met and the surveillance necessary to demonstrate operability of required systems and components. The Technical Specifications specifically address shutdown operations, which is a component of the overall risk of plant operations. Of special interest are the operating rules that address shutdown operations with a reduced water level in the reactor vessel (also called mid-loop operation). The System 80+ Technical Specifications are improved with regard to understandability and consistency as compared to those now presently used by US operating plants.

Operating procedures for normal operation are consistent with the requirements of the Technical Specifications and provide direction to operate the plant consistent with the intent of the plant designer. Operating procedures cover all aspects of acceptable plant evolutions. The procedures are verified and validated through their use on the plant simulator and during plant startup.

Abnormal operating procedures provide direction to respond to and recover from non-accident, but not normal operations. Conditions requiring response with abnormal operating procedures include equipment failures or malfunctions that do not pose an immediate threat to plant safety.

Postulated accident conditions are responded to through Emergency Operating Procedures (EOPs). The EOPs provide the instructions and directions to diagnose the cause of the accident and to select the correct pathway to bring the plant to a safe condition. For severe accident conditions, the EOPs provide directions and guidance to bring and maintain the plant in a condition that minimizes the potential hazards to the plant staff and the public.

The structure and content of the Technical Specifications and operating procedures reflect the improved safety features of the System 80+ plant and the advances in man-machine interfaces and computer systems incorporated into the Nuplex 80+ Advanced Control Complex.

Concentration has been placed on developing a control room design which supports operating procedures. The large screen display indicates emergency operating procedures which are appropriate to a given plant situation. The layout and organization of control and indicators has been done to support station operation. This has been achieved through the use of existing operating guidelines during task analysis.

Further, the continual presence of experienced operators on the Nuplex 80+ Human System Interface design team has aided in the development of a control room which supports operating practice and Technical Specifications.

5. <u>Operator error</u>

Operator error is primarily assessed in the design through the use of Probabilistic Risk Assessment and Human Reliability Analysis to evaluate operational features. The actual reduction of the probability of human error is the ultimate design goal of human factors engineering. Many techniques are used to achieve these reductions. The main ones are described briefly below.

Levels of human error are frequently small in power plant operation, and hence, difficult to measure. The approach which has been taken in the design of the Nuplex 80+ complex has been to incorporate lessons learned based on human errors committed in the overall industry and to employ sound ergonomic design practice as an integrated part of the design process. This application of human factors engineering good practice throughout the design process has as its outcome, a human-system interface with a very low probability of human error, despite the difficulty in actually assessing such error rates.

The principal empirical method of reducing human error has been the continuous presence of experienced human factors engineers and operators on the design team. Through full participation in the design process, these individuals have actually had the lead in man-machine interface design. In portions of the design where they have not provided the lead, they have reviewed the design feature to assure sound ergonomics.

The next empirical method for reducing the probability of human error in the operation of the Nuplex 80+ control complex has been the development of comprehensive human factors standards and guidelines. Based on the very best human factors references, as well as utility and industry guidance and regulatory documents, they provide an exhaustive compendium of correct design practice, in order to reduce the probability of human error. They are complete, unambiguous, and cover a much wider scope than just normal control room operations or 'knobs and dials'. For instance, maintenance issues, software concerns, and balance-of-facility man-machine interface are also addressed.

In summary, human error for the System 80+ design is assessed via PRA, HRA and RAMI. The design has been developed through the use of human factors engineering techniques, both formal analyses and empirical practice, which have been employed to reduce the probability of operator error.

f. <u>Role of Human Factors</u>

Human interactions with the plant are assessed in two basic categories: quantitative and qualitative. Quantitative assessments are made by the Probabilistic Risk Assessment. To a lesser extent, the Function and Task Analyses employed during the process for the Nuplex 80+ design quantify human interactions with the plant in that they place numeric values on reaction time and provide a gross assessment of workload.

To a far greater extent, human interactions with the plant are assessed qualitatively, through a determination of adequate operator performance under the full range of possible plant operating scenarios. This assessment is carried out throughout the design process. The assessment begins with the establishment of a practical and sound human factors engineering program for the design. This program is described in the System 80+ Human Factors Program Plan. By following this program, the plant design team engages in a set of prespecified evaluations and analyses which are able to qualitatively determine the adequacy of human-plant interactions.

The term "human-plant interactions" is essentially synonymous with Human-System Interface or Man-Machine Interface. Therefore, the qualitative assessment of the human-plant interactions is essentially the assessment of the adequacy of the man-machine interface. The System 80+ review criteria for the design process and design products of the man-machine interface provide an objective and testable set of requirements which must be met to assure good human factors in the design of the main control room and other operating stations. As such, these review criteria provide the ultimate vehicle for assessing the human-plant interactions. By following design process and product requirements outlined in criteria for the design of the main control room and other operating stations, the design team assures that the human will not be placed in error-likely situations during his interactions with the plant.

Several elements must be met to assess whether the design process will result in a design which includes good human-plant interactions. The first of these is good Human Factors Engineering Program Management. This includes not only the aforementioned program plan but also a design team structured and qualified to implement good ergonomics and a method of tracking human factors open issues.

The next element is the incorporation of industry experience. As noted in the previous section, errors of commission and omission are so infrequent in realistic simulation of nuclear power plant operation that there is an insufficient data base to objectively assess error types and to note all human engineering deficiencies. By incorporating wider industry experience into the System 80+ design, known human-plant interaction problems can be rectified. The use of the industry data base also allows a wider experience than that normally available to the design team to be accessed.

To qualitatively assess human-plant interactions, a number of formal human factors analyses and evaluations must be included in the design process. These include Evaluation and Allocation of System Functions, Task Analyses, Availability Verification, Suitability Verification, and Validation of Ensemble. These evaluations span the design process from the more conceptual stages to the final simulations. Throughout these analyses, emphasis has been placed on operating experience and building on safe existing designs. The purpose of these analyses is to assure that the man-machine interface is necessary and sufficient. That is, the operator must have information and control sufficient to operate the plant safely and efficiently under various plant conditions, but not be overburdened with unnecessary data and controls.

The final element in an adequacy assessment of the human-plant interactions, is the assessment of the man-machine interface design itself. To a great extent, the qualitative and quantitative assessments of the process assures an adequate human-plant interface.

An assessment of the design product is performed by assuring adherence to an objective set of unambiguous criteria for various hardware and software portions of the overall design. These criteria have been developed for all aspects of the System 80+ human-plant interface. They include, for example, criteria for alarms, operator aids, parameter indications, integrated displays, discrete component control and indication, modulating component control and indication, special controls, and control room monitoring and control function location. Further criteria are provided for control room configuration, individual control panels, work space and environment, print and text format, local control panels, and maintainability.

These criteria are founded on the System 80+ Human Factors Standards and Guidelines. The Standards and Guidelines, which are disseminated to all applicable design team members, represent a prescriptive set of guidance to be adhered to in order to assure the adequacy of the man-machine interface. The guidance and standards provided therein is a distillation of industry and regulatory reference materials and basic human factors references. Design practice and requirements to be followed for System 80+ design are explained and referenced to their source documents through the inclusion of bases material. The Standards and Guidelines are the yardstick against which the design is measured by the Design Product Requirements used for assessment of the human-plant interactions.

In summary, the design follows human factors good practice based on a systematic application of ergonomics throughout the design process. Human factors are also applied comprehensively to the design product. The adequacy of the process and the resulting design are quantitatively assessed through the application of PRA and HRA. The qualitative assessment of human interactions with the plant occurs through the use of review criteria to assess whether the design process and product are adequate from a human factors viewpoint.

g. Special Design Provisions

(1) Interlocks: mechanical, electrical or administrative provisions

Interlocks for fluid, mechanical and electrical systems in the System 80+ design are discussed below. Administrative provisions are addressed.

(a) Shutdown Cooling System (SCS) suction line valve interlocks and Safety Injection Tank (SIT) isolation valve interlocks are provided in the design.

The interlocks on the SCS and on the SIT are designed to act as permissives. The SCS suction line valve interlocks permit the isolation valves to be opened below a certain pressure and automatically closes them above a certain pressure. The SIT isolation valve interlocks are designed to permit the operator to isolate the SITs at low pressure thereby allowing the SITs to be maintained at a given pressure when the balance of the RCS is depressurized.

(b) In the reactor coolant system, since there are no reactor coolant loop isolation valves, there will always be some induced flow in an idle loop. There is, therefore, no need for a cold water interlock.

(c) In the refueling mechanical equipment, the refueling electrical interlocks prevent improper refueling machine movement and movement of heavy loads over vulnerable equipment.

(d) For electrical equipment in the Reactor Protective and Engineered Safety Features Actuation Systems, interlocks are provided to ensure RPS and ESFAS availability for fault mitigation. They include: Bistable Trip Channel Bypass Interlock, Manual Bistable Test Interlock, Initiation Circuit Test Interlock, Nuclear Instrumentation Test Interlock, and Trip Logic Calculator Test Interlock.

(e) Administrative Provisions

Administrative provisions provide a means of ensuring that operation, maintenance, and periodic testing is conducted in accordance with the design bases. Various requirements are intended to be controlled by the plant's administrative procedures rather than the more formal prescriptive Technical Specifications. The administrative controls generally apply to manually operated or actuated systems where the controls are intended to ensure changes are not inadvertently made. These include tagging and physically locking valves and bypasses, having key control to areas where the equipment is located, among others. Use of administrative controls rather than inclusion in Technical Specifications provides the plant operator greater flexibility in applying the controls without compromising plant safety.

(2) Facilities for monitoring during operation

The Discrete Indication and Alarm System (DIAS) Channel N and Data Processing System (DPS) include alarms for a limited number of operational occurrences for which no specific automated actuation of a safety system is required. Both of these systems provide alarm checking, but DIAS Channel N activates Priority 1 alarm tiles for both systems.

Alarms are provided for Reactor Coolant Pump, Cooling Water Supply Monitoring and Safety Injection Tank Pressure.

(3) Other design provisions to reduce fault frequency or to ensure availability of safeguards features

(a) Reactor Power Cutback System (RPCS) allows reactor to remain operational following Loss of Load and Loss of Feedwater Pump events.

(b) Steam Bypass Capacity that matches steam bypass requirements to RPCS capabilities.

(c) Steam and Feedwater Flow venturi to limit secondary coolant losses for applicable events.

(d) Reactor Power Shape Monitors to assist operators with axial xenon control during reactor power changes (load maneuvers).

(e) Reactor Level Monitoring Instrumentation for enhancing safe operations during mid-loop draindowns for steam generator and reactor coolant pump maintenance.

(f) Fast transient data recording for post-trip root cause analysis to avoid subsequent trips and downtimes.

(g) The reactor vessel is fabricated using ring forgings with material specifications that result in a sixty year end-of-life RT_{NDT} which is well below the current US NRC screening criteria. Use of such forgings reduces the number of welds and eliminates concern for pressurized thermal shock.

(h) The pressurizer volume is increased, which improves transient response.

(i) The steam generators use Inconel 690 tubes, have improved steam dryers, and incorporate a ten percent margin for future potential tube plugging. There is a large feedwater inventory to extend "boil-dry" time, and other design modifications increase access and facilitate maintenance.

(j) The Chemical and Volume Control System (CVCS) has been redesigned as a non-safety related system, and has thereby been simplified and improved.

(k) The Safety Injection System is simple and reliable. It contains four trains for injection, direct-to-vessel connections, and an in-containment refueling water storage tank, which eliminates the need to switch supply from the external Refueling Water Storage Tank to the containment sump, which are now integrated.

(l) A Safety Depressurization System provides the capability to depressurize the Reactor Coolant System rapidly so that primary system feed-and bleed can be used to remove decay heat following a total loss of feedwater event. Discharge is routed to the in-containment refueling water storage tank, not to containment volume, which prevents contamination of the containment due to system activation.

(m) A dedicated Emergency Feedwater System with a cavitating venturi (to limit feed flow to a steam generator with a ruptured feed or steam line) is independent of the main feed system. This system eliminates the need for automatic isolation of feed flow to a disabled steam generator.

(n) The Shutdown Cooling System is designed to a 900 psig design pressure, to provide greater flexibility of operation, and has been further modified to increase system reliability.

(o) The Advanced Control Complex (NUPLEX 80+) incorporates state-of-the-art advances, including distributed digital processing and remote multiplexing, fiber optic data communications, and touch-sensitive video displays. NUPLEX 80+ also features an enhanced man-machine interface, including a large wall-mounted color graphic display which is called the Integrated Process Status Overview (IPSO) panel, electroluminescent displays, and dedicated indicators which replace analog devices on each panel.

(p) Self-testing features have been designed into the plant instrumentation and control systems utilizing the latest in continuous on-line computer-aided circuitry and diagnostics. This simplifies operation and reduces operations and maintenance efforts, with resulting cost savings and enhanced safety.

Station documentation provided by the designer also has a safety role. The documentation scope includes the spectrum of activities from construction through commissioning to normal operations. Operational guidance and procedural guidance is provided by the designer for the plant owner to develop plant specific procedures. Included are procedures derived from emergency procedures guidelines for the operator to follow during abnormal operations.

h. Analysis

(1) <u>Level and Type of Analysis</u>

Two types of analyses have been performed - the traditional deterministic safety analysis required by the USNRC as the basis for licensing and a Level 1 Probabilistic Risk Assessment (PRA), also required by the USNRC and used to assess the potential for severe accidents with significant consequences to the public. The fault schedule for the deterministic analyses is prescribed by the USNRC. It is based upon the evolution of LWR design and licensing experience. Events are grouped into seven categories and three coarse frequency groups for moderate, infrequent and limiting faults. Typically, the most limiting events are selected for analyses that yield the most adverse consequences for each category and group. The methods and computer codes employed in the analyses have been approved by the USNRC. Results of these analyses are compared to acceptance criteria to demonstrate the safety of the plant and to establish its licenseability.

The PRA fault tree schedule is derived from general core damage initiators, again based on prior LWR design experience. Event trees are developed that all result in core damage. These are collapsed into a tractable set for which fault tree analyses yield event sequence probabilities. The PRA fault schedule and frequencies complement the deterministic analyses. They serve to measure effectiveness of plant design evolutions and they serve to confirm the adequacy of the deterministic fault schedule.

Quantitative frequencies for events on the deterministic fault schedule are estimated with the methods and results from the PRA. The difference between the two schedules is that events on the deterministic fault schedule are bounded by a set of prescribed assumptions for which the plant safety features are shown to yield acceptable consequences, whereas events on the PRA schedule are deliberately extended beyond the plant design basis by assumptions of system, component or structural failures until core damage is achieved. Frequencies reported for the PRA event sequences are therefore generally lower than frequencies of events on the deterministic fault schedule.

(2) Safety Standards

Safety standards include mechanical standards on fuel rod and plant integrity for which violation may represent precursors to radioactive release and radiological dose standards applicable to plant personnel and the public. Physical standards are given in Table II-J-1. They are limits on the fuel rod cladding to prevent release of radioactive fission products from fuel rods, limits on the primary and secondary coolant pressure boundaries to prevent release of radioactive coolant and limits on containment temperature and pressure equivalent to the design values, which for the containment are for faulted conditions.

A LOCA event is by definition a breach of the primary coolant pressure boundary, so a separate set of acceptance criteria are specified and are summarized in Table II-J-2. Satisfaction of the LOCA acceptance criteria assures maintenance of a coolable core geometry and limits the potential for containment failure by ignition of event generated hydrogen gas.

Safety standards for radiological dose include limits during normal operation and faulted conditions. Table II-J-3 summarizes the dose assessment levels that are compared with calculated doses for faulted conditions.

(3) Analysis Methods

The analysis methods employed in assessing the safety of the System 80+ design have been developed by ABB over a number of years. They have been applied to fifteen operating nuclear plants licensed in the US and to four nuclear plants under construction in

Korea. Documentation for the methods and computer codes employed in the deterministic safety analyses has been submitted to the USNRC and the methods are approved for nuclear licensing in the US.

Validation of methods and codes is performed by comparisons with a variety of tests. Results from the NSSS transient and performance code, CESEC III, employed for the non-LOCA transient analyses, were compared with actual plant performance during power ascension testing, including coastdown tests and material convection cooling, and were also compared with data from abnormal events on operating reactors. In addition, NRC audits of CESEC included benchmarking against RELAP by Argonne National Laboratory.

The calculation of DNBR is supported by reactor vessel scale flow model tests that yield the pressure and flow distributions throughout the vessel internals and core. Pressure losses are combined with the pump characteristics determined from reactor coolant pump tests at full temperature, pressure and flow rate to predict reactor core flow rate. Core flow distributions from the scale model tests and the system flow rate are input boundary conditions in calculating DNBR with TORC. The DNBR correlation is derived from CHF testing on heated rod arrays of the System 80+ dimensions and configuration. The CHF tests are evaluated with the same TORC code employed in the transient analyses.

(4) Safety During Normal Operations

Radiological dose limits for normal operation are summarized in Tables II-II-J-4 and II-J-5. System 80+ features that minimize doses from normal operation include material selection to minimize coolant transport of radioactive material, plant configurations to minimize local crud depositions and equipment design to reduce maintenance times.

(5) Analysis of Faults

(a) Fault Schedule

Transient analyses are performed for the event sequences taken from the deterministic fault schedule and given in Table II-J-6. The consequences of the events in the table encompass the consequences from all the deterministic events considered. These are considered the design basis events for which the limits in Tables II-J-1 and II-J-2 shall be satisfied. In addition, three events are shown in Table II-J-6 which are beyond the design basis. Analyses show that the limits are satisfied for these events also.

(b) Fault Sequence Analysis

The fault sequence for faults on the deterministic schedule is prescribed so as to assure that the calculated consequences of the fault initiator are a conservative measure of the performance capability of the plant and safety systems. Analyses of the deterministic faults include the event initiator, adverse initial conditions, loss of off site power, worst single failure, credit for only safety grade systems and no operator action for 30 minutes.

The fault sequence for events on the PRA fault schedule may be similar, but assume additional failures in order to force the sequence to achieve core damage. The dominant accident sequences in the PRA are summarized in Table II-J-7.

(c) Transient Analysis

Each transient analysis has been analyzed in detail. Tables of the fault sequences show the timing of the automatic and manual actions credited to mitigate the event and of the calculated values of pressure and DNBR to compare with the safety standards. Tables II-J-8a through II-J-8c are samples of the sequence tables that have been developed.

For each initiating fault, the sequence table provides the following information. The Reactor Protection System (RPS) parameter that trips the reactor is stated. The term, analysis setpoint, used with the value of the trip parameter, implies that the trip point used in the analysis is conservatively biased relative to the nominal setpoint because of instrument error, time delay and/or event induced errors. The safety equipment that functions to mitigate the event is stated. The magnitude of the parameters representing fuel or plant integrity standards are shown. They include DNBR and the fraction of failed rods, primary pressure and secondary pressure. Comparison with the limit values in Table II-J-1 demonstrate the System 80+ design margins for the event. Actuation of the secondary safety or atmosphere dump valve is particularly important because they are a radiation release pathway. For events with significant release, the mass of fluid released is shown. Finally, most of the events are essentially concluded by the operator after 1800 sec (30 min) when the operator first takes action, which correctly suggests that quicker operator action can significantly reduce the severity of these events. The Nuplex 80+ Advanced Control Complex facilitates the operator's recognition and mitigating actions in such circumstances.

(d) Radiological Consequence Analysis

Sources of radioactivity released during the design basis events on the deterministic fault schedule include 1) radioactivity in the primary coolant initially and from failed fuel, 2) radioactivity in the secondary coolant from a leaking or event-caused break in a steam generator tube, 3) radioactivity released into the containment atmosphere following a LOCA and 4) radioactivity in ex-core components.

The radiological sources comprise groups of iodine, krypton and xenon isotopes. For each isotope and quantity released, the dose to the public is determined at 500m (150 ft). The summed doses are presented in Table II-J-6. Doses are calculated at eight hours, after which the release pathways for non-LOCA events will have been isolated and the plant put into a stable, shutdown mode. Timing of the event sequences is evident in Table II-J-8.

LOCA doses are calculated for longer periods because it is assumed that leakage continues from the containment atmosphere. The System 80+ filtered containment annulus vent system substantially reduces the LOCA dose contribution from this pathway. LOCA doses are given in Table II-J-6 for seven days.

(e) Probabilistic Analyses

A schedule of traditional fault initiators was developed from prior PRA evaluations and was augmented with specific considerations of the System 80+ design features. Event trees were prepared for each initiator to develop sequences that would lead to core damage. Fault trees established the quantitative frequencies for each fault sequence. Condensation of the sequences resulted in a tractable set of initiators and sequences from which the total core damage frequency was determined.

(f) Frequency of Off Site Releases and Plant Damage

The probabilistic analysis yields the frequency of core damage, although detailed transient analyses to quantify the core damage have not been performed for the events on the probabilistic fault schedule since these events are beyond the design basis for the System 80+ plant. The total core damage frequency for events on the probabilistic fault schedule is 1.87×10^{-6} (Figure II-J-9). Typical procedures for Level 2 and 3 PRA to be completed at a later stage would be to combine the Level 1 events according to the characteristics of the plant state related to core damage and to continue the PRA to include containment releases.

(g) Assessment Against Deterministic Standards

The faults on the deterministic fault schedule are analyzed assuming the worst single failure. Table II-J-10 lists the failures from which the worst single failure is selected. The selection is based on the analyst's experience and, where necessary, multiple analyses for comparison.

In analyzing the design basis events on the deterministic fault schedule, no credit is taken for the diversity provided in the safety systems. Credit may be taken for diverse systems when analyzing beyond design basis events, depending on the event and the intended use of the results in the regulatory process.

Diversity exists in the reactor trip function, through the Alternate Protection System (APS) to open the trip breakers. The deterministic Analysis of the ATWT event demonstrated acceptable consequences without crediting the APS.

Diversity exists in the emergency feedwater actuation independent of the ESFAS actuation portion of the Plant Protection System. Diversity exists in the emergency feedwater pumps that are both electrically and steam powered. Deterministic analyses of the total loss of feedwater event do not credit these diverse features.

Diversity exists in the Alternate AC combustion turbine driven generator. It is diverse from the two redundant emergency diesel generators. Credit is taken for the Alternate AC during Station Black Out (SBO). The USNRC defines the SBO event to include loss of all offsite and redundant emergency (two diesel generators) on site electrical power. The diverse Alternate AC power is assumed to be available.

(3) Conclusion

The System 80+ design benefits from the evolution of safety system development and safety evaluation for LWR power plants. Equipment and component design is based upon clear performance objectives and accepted industry standards. Justification of safety performance is confirmed with matured procedures that combine deterministic and probabilistic methodologies. Deterministic evaluations demonstrate that System 80+ satisfies the NRC acceptance criteria. Probabilistic evaluations quantify the frequency of events and demonstrate that System 80+ satisfies the safety assessment levels presented below. Further, the evaluations show that the total mean core damage frequency from internal and external events is 1.87 x 10^{-6} per year.

The following Fundamental Safety Principles are satisfied by the System 80+ plant design.

i) No person shall receive doses of radiation in excess of the statutory dose limits as a result of normal operation.

ii) The exposure of any person to radiation shall be kept as low as reasonably practicable.

iii) The collective effective dose to operators and to the general public as a result of operation of the nuclear installation shall be kept as low as reasonably practicable.

iv) All reasonably practicable steps shall be taken to prevent accidents.

v) All reasonably practicable steps shall be taken to minimize the radiological consequences of any accident.

TABLE II-J-1

PLANT AND FUEL ROD INTEGRITY LIMITS FOR DESIGN BASIS EVENTS

Fault Frequency Designation	Integrity Limits
Moderate and Infrequent Faults	1. Coolant pressure boundary stress less than 110% of ASME design value.
	2. CE-1 DNBR greater than 1.24.
	3. Containment pressure and temperature less than design.
Limiting Faults	1. Coolant pressure boundary stress less than 120% of ASME design value.
	2. Fuel pellet enthalpy less than 280 cal/gm.
	3. Containment pressure and temperature less than design.
Beyond Design Basis Faults	1. Coolant pressure boundary stress less than ASME Level C stress limits.

TABLE II-J-2

LOCA ACCEPTANCE CRITERIA

1. Peak clad temperature less than 1204°C (2200°F).
2. Local clad oxidation less than 17%.
3. Core wide oxidation over active fuel cladding length less than 1%.
4. Core geometry shall remain coolable.
5. Core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core.

TABLE II-J-3

DOSE ASSESSMENT LEVELS FOR FAULTED CONDITIONS

Dose Band	Releases Giving Rise to an Individual Whole Body Dose at 500m, mSv	Total Acceptable Frequency per Year
1	0.1 to 1.0	10^{-2}
2	1 to 10	10^{-3}
3	10 to 100	10^{-4}
4	100 to (no upper limit)	10^{-5}

TABLE II-J-4

DOSE LIMITS FOR NORMAL OPERATING CONDITIONS (FOR OPERATORS)

Assessment Level	Population	System 80+™ Annual Dose Limit	
1	Plant personnel	[1] Whole Body:	10 mSv (1 Rem)
2	Average plant personnel	[1] Whole Body:	2 mSv (0.2 Rem)
3	Collective Plant Personnel	[2] 0.8 man-mSv/MWe (or 100 person-rem/1300 MWe plant)	

[1] Section 12.4, CESSAR-DC.
[2] EPRI ALWR Utility Document, Chapter 11 goal.
[3] Statutory limits of non-stochastic exposure. Compliance ensured by above limits for whole body and extremities.

TABLE II-J-5

DOSE LIMITS FOR NORMAL OPERATING CONDITIONS (FOR THE PUBLIC)

Assessment Level	Population	System 80+™ Annual Dose Limit	[3] Calculated Dose for System 80+
4	General Public	[1] Whole Body: 10 mSv (0.5 Rem)	N/A
		[2] Liquid Discharges: Whole Body: 30 μSv (3.0 mrem) Organ: 100 μSv (10.0 mrem)	Liquid Discharges: Whole Body: 0.011 μSv/yr (1.1E-03 mrem/y
		[2] Gaseous Discharges: Whole Body: 50 μSv (5.0 mrem) Skin: 150 μSv (15.0 mrem) Organ: 150 μSv (15.0 mrem)	Gaseous Discharges: Whole Body: 2.9 μSv/yr (0.29 mrem/yr)
		N/A	N/A

[1] 10 CFR 20 limits.
[2] 10 CFR 50, Appendix I.
[3] Calculated based System 80+™ design parameters using on methodology per Chapter 5, Part 1 of the OTS.

TABLE II-J-6

DOSE LIMITS BASED ON FREQUENCY OF FAULT SEQUENCES FROM DETERMINISTIC FAULT SCHEDULE

Event Description[3]	Dose Band	Dose Criterion (mSv)	Dose (mSv)[2]
1.) Steamline Break-Full Power	2	$1 < dose \leq 10$	4.20
2.) Steamline Break-Zero Power	1	$0.1 < dose \leq 1$	0.137
+ GIS[1]	2	$1 < dose \leq 10$	1.24
+ PIS[1]	1	$0.1 < dose \leq 1$	0.578
3.) Loss of Condenser Vacuum	2	$1 < dose \leq 10$	2.19
4.) Feedwater Line Break	1	$0.1 < dose \leq 1$	0.134
+ GIS[1]	2	$1 < dose \leq 10$	1.24
+ PIS[1]	1	$0.1 < dose \leq 1$	0.575
5.) Locked Rotor	2	$1 < dose \leq 10$	2.40
6.) CEA Ejection	2	$1 < dose \leq 10$	6.44
7.) Letdown Line Break	1	$0.1 < dose \leq 1$	0.248
+ GIS[1]	1	$0.1 < dose \leq 1$	0.507
+ PIS[1]	2	$1 < dose \leq 10$	1.83
8.) SGTR w/o LOOP	1	$0.1 < dose \leq 1$	0.329
+ GIS[1]	1	$0.1 < dose \leq 1$	0.511
+ PIS[1]	1	$0.1 < dose \leq 1$	0.663
9.) SGTR w/LOOP	1	$0.1 < dose \leq 1$	0.318
+ GIS[1]	1	$0.1 < dose \leq 1$	0.468
+ PIS[1]	1	$0.1 < dose \leq 1$	0.659
10.) SGTR/LOOP + SF	2	$1 < dose \leq 10$	1.88
+ GIS[1]	3	$10 < dose \leq 100$	11.5
+ PIS[1]	2	$1 < dose \leq 10$	5.29

TABLE II-J-6 (Continued)

DOSE LIMITS BASED ON FREQUENCY OF
FAULT SEQUENCES FROM DETERMINISTIC FAULT SCHEDULE

Event Description[3]	Dose Band	Dose Criterion (mSv)	Dose (mSv)[2]
11.) LOCA - 7 Day + PIS[1]	3	$10 < \text{dose} \leq 100$	24.9
12.) ATWT	2	$1 < \text{dose} \leq 10$	<2.19
13.) Station Blackout	2	$1 < \text{dose} \leq 10$	<2.19
14.) Total Loss of Feedwater + PIS[1]	2	$1 < \text{dose} \leq 10$	1.16

(1) GIS - Dose includes event generated iodine spike
PIS - Dose includes pre-existing iodine spike

(2) All doses are taken at 8 hours, unless noted.

(3) Events 1 through 11 are design basis events.
Events 12 through 14 are beyond design basis events.

TABLE II-J-7

CORE DAMAGE FREQUENCY CONTRIBUTIONS FOR DOMINANT ACCIDENT SEQUENCES

BY INITIATING INTERNAL EVENT

SEQUENCE	MEAN CORE DAMAGE FREQUENCY CONTRIBUTION EVENTS/YEAR	ERROR FACTOR
(ATWS)(Adverse MTC)	1.67E-07	5.66
(LOOP)(Fail to Reseat PSV [PSV LOCA])(Injection Fails)	8.57E-08	9.02
(Medium LOCA1/2)(Injection Fails)(Containment Spray [CS] OK)	8.53E-08	5.62
(SGTR)(Injection Fails) (Aggressive Secondary Cooldown Fails)	7.30E-08	10.61
(SLOCA)(Injection Fails) (Aggressive Secondary Cooldown Fails)	4.34E-08	11.22
(Large LOCA)(SIT OK)(Injection Fails)(Containment Spray [CS] Fails)	3.95E-08	5.22
(Large LOCA)(SIT OK)(Injection OK)(Containment Spray [CS] Fails)	1.05E-08	6.67
(SGTR)(Injection OK)(FW to Intact SG OK)(RCS Pressure Control Fails)(Unisolable Leak in Ruptured SG)	7.02E-09	11.08

TABLE II-J-7 (Cont'd)

CORE DAMAGE FREQUENCY CONTRIBUTIONS FOR DOMINANT ACCIDENT SEQUENCES

BY INITIATING INTERNAL EVENT

SEQUENCE	MEAN CORE DAMAGE FREQUENCY CONTRIBUTION EVENTS/YEAR	ERROR FACTOR
(LOFW)(Fail to Deliver Emergency FW)(Bleed Fails)	5.52E-09	8.23
(Medium LOCA1/2)(Injection OK) (Containment Spray [CS] Fails)	2.64E-09	5.52
((TOTH)(Fail to Deliver Feedwater)(Bleed Fails)	4.42E-09	7.83
(CCWB)(Fail to Deliver Emergency FW)(Bleed Fails)	3.67E-09	11.57
(ATWS)(Emergency FW OK)(Fail to deliver Boron to RCS(Bleed OK) (Feed Fails)	3.28E-09	12.87
(CCWB)(Emergency FW OK) (Long-Term Decay Heat Removal Fails) (Bleed Fails	2.75E-09	11.30
(LOOP)(Fail to Reseat PSV [PSV LOCA])(Injection OK)(Fail to Cool IRWST)	2.14E-09	12.71
(CCWB)(Emergency FW OK)(Long-Term Decay Heat Removal Fails) (Feed & Bleed OK)(Fail to Cool IRWST)	1.87E-09	9.11
(ATWS)(Consequent SGTR) (Emergency FW to Intact SG OK) (Injection Fails)	1.03E-09	7.23

TABLE II-J-8a

FAULT SEQUENCE FOR FULL POWER INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE (IOSGADV)

Time (sec)	Event	Setpoint or Value
0.0	One atmospheric dump valve opens fully	--
1800	Hot channel DNBR	1.24
1800	Operator initiates manual trip	--
1800.4	Manual reactor trip signal generated	--
1800.55	Reactor trip breakers open	--
1800.9	Steam generator water level reaches emergency feedwater actuation analysis setpoint, %WR	19.9
1805	Void begins to form in RV upper head	--
1808	Main steam safety valves open, bar (psia)	83.6 (1212)
1817	Main steam safety valves close, bar (psia)	79.4 (1151)
1860.9	Emergency feedwater delivered to generator	--
2032.9	Steam generator pressure reaches main steam isolation signal (MSIS) analysis setpoint, bar (psia)	49.6 (719)
2039.2	MFIVs close completely	--
2039.2	MSIVs close completely	--
2064.7	Pressurizer pressure reaches low pressurizer safety injection actuation analysis setpoint, bar (psia)	107.2 (1555)
2104.7	Safety injection pumps reach full speed	--
2744	Affected steam generator dries out	--
3000	Operator manually closes ADV	--
3600	Operator initiates plant cooldown	--

TABLE II-J-8b

FAULT SEQUENCE FOR FULL POWER INADVERTENT OPENING OF A STEAM GENERATOR ATMOSPHERIC DUMP VALVE WITH LOSS OF CEDMC TRIP SIGNAL

Time (sec)	Event	Setpoint or Value
0.0	One atmospheric dump valve opens fully	--
1800	Hot channel DNBR	1.24
1800	Operator initiates manual trip	--
1800.4	Manual reactor trip signal generated	--
1800.55	Reactor trip breakers open	--
1801.8	Minimum transient DNBR	1.24
1805	Void begins to form in RV upper head	--
1816.2	Steam generator pressure reaches main steam isolation signal (MSIS) analysis setpoint, bar (psia)	49.6 (719)
1821.8	Steam generator water level reaches emergency feedwater actuation analysis setpoint, %WR	19.9
1822.55	MFIVs close completely	--
1822.55	MSIVs close completely	--
1881.8	Emergency feedwater delivered to generator	--
1890.9	Pressurizer pressure reaches low pressurizer pressure safety injection actuation analysis setpoint, bar (psia)	107.2 (1555)
1930.9	Safety injection pumps reach full speed	--
2465	Affected steam generator dries out	--
3000	Operator manually closes ADV	
3600	Operator initiates plant cooldown	--

TABLE II-J-8c

FAULT SEQUENCE FOR A STEAM LINE BREAK OUTSIDE CONTAINMENT DURING FULL POWER OPERATION WITH OFFSITE POWER AVAILABLE (SLBFPD)

Time (sec)	Event	Setpoint or Value
0.0	Steam Line Break Occurs	--
5.76	CPC Variable Overpower Trip Condition Reached	115
5.76	EFW Initiated to Both Steam Generators	
6.16	CPC Variable Overpower Trip Signal Generated	--
6.31	Reactor Trip Breakers Open	--
8.87	Minimum Transient DNBR	1.18
13.38	Steam Generator Pressure Reaches Main Steam Isolation Signal Analysis Setpoint, bar (psia)	49.6 (719)
17.07	Voids Begin to Form in RV Upper Head	--
19.73	MFIVs Close Completely	--
19.73	MSIVs Close Completely	--
124.7	Maximum Post-trip Transient Reactivity, 10^{-2} $\Delta\rho$	-1.72
203.6	Pressurizer Pressure Reaches Safety Injection Actuation Signal (SIAS) Analysis Setpoint, bar (psia)	107.2 (1555)
243.6	Safety Injection Flow Begins	--
310	Safety Injection Boron Begins to reach Reactor Core	--
1800	Operator Initiates Cooldown	--

TABLE II-J-9

CORE DAMAGE FREQUENCY CONTRIBUTION BY INITIATING EVENT

INTERNAL INITIATING EVENTS	MEAN CORE DAMAGE FREQUENCY	ERROR FACTOR	PERCENT OF TOTAL
Large Loss-Of-Coolant Accident (LLOCA)	5.00E-8	5.63	7.43
Medium Loss-Of-Coolant-Accident (MLOCA1)	4.53E-8	5.36	6.73
Medium Loss-Of-Coolant-Accident (MLOCA2)	4.53E-8	5.36	6.73
Small Loss-Of-Coolant-Accident (SLOCA)	4.36E-8	10.73	6.48
Large Secondary Side Break (LSSB)	2.00E-10	14.58	0.03
Steam Generator Tube Rupture (SGTR)	8.04E-8	9.71	11.95
Loss of Feedwater Flow (LOFW)	5.66E-9	7.53	0.84
Other Transients (TOTH)	4.51E-9	6.88	0.67
Loss Of Offsite Power (LOOP) including Station Blackout with Battery Depletion	1.00E-7	8.23	14.86
Loss of Component Cooling Water (CCW) Div 2	9.00E-9	8.00	1.34
Loss of 4.16 Kv Bus	2.52E-11	9.82	<0.01
Loss of 125 VDC Vital Bus	2.79E-12	6.23	<0.01
Anticipated Transient Without Scram (ATWS)	1.72E-7	6.44	25.56
Interfacing System LOCA	3.01E-9	18.50	0.45
Loss of HVAC	1.40E-8	18.85	2.08
Vessel Rupture	1.00E-7	10.00	14.86
Internal Events - Total	6.73E-7		100.00
EXTERNAL INITIATING EVENTS			
Seismic Events	1.19E-6	9.59	98.5
Tornado Strike Event	1.21E-8	3.50	1.5
External Events - Total	1.20E-6		100.0
COMBINED EVENT FREQUENCY			
Internal Events	6.73E-7		43.0
External Events	1.20E-6		57.0
TOTAL	1.87E-6		100.0

TABLE 1II-J-10

(Sheet 1 of 2)

SINGLE FAILURES

A. STEAM BYPASS CONTROL SYSTEM

1. Failure to Modulate Open
2. Failure to Quick Open
3. One Bypass Valve Fails to Quick Close
4. Excessive Steam Bypass Flow
5. Failure to Generate Automatic Withdrawal Prohibit Single During Steam Bypass Operation
6. Failure to Generate the Reactor Power Cutback Signal

B. REACTIVITY CONTROL SYSTEM

7. Regulating Group(s) Fail(s) to Insert or Withdraw
8. A Single CEA Stuck*
9. A CEA Subgroup Stuck*
10. Failure to Initiate or Execute the Reactor Power Cutback
11. CEA's Withdraw upon Automatic Withdrawal Prohibit and/or CEA Withdrawal Prohibit

C. FEEDWATER CONTROL SYSTEM

12. Failure of Reactor Trip Override
13. Failure of High Level Override

D. TURBINE-GENERATOR CONTROL SYSTEM

14. Setback w/o Cutback
15. Failure to Modulate the Turbine Control Valves
16. Failure to Setback Given a Cutback (100% _ Initial Power 75%)
17. Failure to Setback (75% _ Initial Power 60%)
18. Failure to Runback (60% Initial Power)
19. Failure to Trip the Turbine

E. PRESSURIZER PRESSURE CONTROL SYSTEM (PPCS)

20. Failure of Spray Control Valves to Open
21. Failure of Spray Control Valves to Close
22. Failure of Backup Heaters to Turn On

---

* Control Element Drive Mechanism does not respond to control signal. Release of CEA(s) on trip is not inhibited.

TABLE II-J-10 (Cont'd)

(Sheet 2 of 2)

SINGLE FAILURES

23. Failure of Backup Heaters to Turn Off

F. PRESSURIZER LEVEL CONTROL SYSTEM

24. Backup Charging Pump Fails to Turn On
25. Backup Charging Pump Fails to Turn Off
26. Letdown Flow Control Valve Fails to Close
27. Letdown Flow Control Valve Fails to Open

G. MAIN FEEDWATER SYSTEM

28. One MFIV Fails to Close
29. One Back-flow Check Valve Fails to Close

H. MAIN STEAM SYSTEM

30. One MSIV Fails to Close
31. One Atmospheric Dump Valve Fails to Open
32. One MSSV Fails to Reclose

I. EMERGENCY FEEDWATER SYSTEM

33. Failure of any One Emergency Feed Pump to Start

J. SAFETY INJECTION SYSTEM

34. Failure of One SI Pump

K. ELECTRICAL POWER SOURCES

35. Loss of Offsite Power Greater than 3 Seconds After Turbine trip Caused by Reactor Trip
36. Failure of One Emergency Generator to Start, Run, or Load (Two SI pumps are powered from one Emergency Generator.)

L. INTERACTIVE CONTROL SYSTEM FAILURES

37. Loss of CEDMC Reactor Tripped Signal

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TABLE II-1

SUMMARY COMPARISON OF PLANT PARAMETERS

ITEM	UNIT	Reference System 80+ Plant with Base UO_2 Core	Full Plutonium Core load	Plutonium Core with Tritium
1. POWER OUTPUT AND MAIN PLANT PARAMETERS				
NO. OF PRIMARY COOLANT LOOPS		2	2	2
REACTOR THERMAL OUTPUT	MW(t)	3931	3817	3427
GROSS GENERATED OUTPUT	MWe	1412	1350	1200
NET POWER STATION OUTPUT	MWe	1326	1256	1115
RCS OPERATING PRESSURE	PSIA	2250	2250	2250
RCS COOLANT FLOW	GPM	445,600	445,600	445,600
REACTOR INLET TEMP.	°F	552	552	552.5
REACTOR OUTLET TEMP.	°F	611	609.5	604
STEAM PRESSURE AT S.G. OUTLET	PSIA	1012	1014	1023
STEAM QUALITY		.9975	.9975	.9975
TOTAL STEAM FLOW	Mlb/hr	17.66	17.08	15.15
2. REACTOR CORE				
CORE THERMAL OUTPUT	MW(t)	3914	3800	3410
NO. OF FUEL ASSEMBLIES		241	241	241
TYPE OF FUEL ASSEMBLY		16x16	16X16	16X16
NO. OF ZIRCALOY 4 GRIDS		10	10	10
NO. OF INCONEL 625 GRIDS		1	1	1
TOTAL NUMBER OF CONTROL ROD ASSEMBLIES		101	101	101
NUMBER OF 12 ELEMENT FULL STRENGTH CONTROL ASSEMBLIES		48	48	48
NUMBER OF 4 ELEMENT FULL STRENGTH CONTROL ASSEMBLIES		53	53	53
REFUELING INTERVAL	MO	18	12	12
FUEL CYCLE LENGTH	EFPD	432	274	274
NO. OF ZONES		3	3	3

TABLE II-1

SUMMARY COMPARISON OF PLANT PARAMETERS

ITEM	UNIT	Reference System 80+ Plant with Base UO_2 Core	Full Plutonium Core load	Plutonium Core with Tritium
CORE MECHANICAL DESIGN PARAMETERS				
FUEL ASSEMBLIES		16X16	16X16	16X16
ROD BUNDLE ARRANGEMENT		0.506	0.506	0.506
ROD PITCH	IN	7.98X7.98	7.98X7.98	7.98X7.98
CROSS SECTION DIMENSIONS	IN	11	11	11
NUMBER OF GRIDS PER				
ASSEMBLY	#	56,876	54,956	49,164
FUEL RODS	IN	0.382	0.382	0.382
LOCATIONS	IN	0.007	0.007	0.007
OUTSIDE DIAMETER	IN	0.025	0.025	0.025
DIAMETRAL GAP		ZIRCALOY-4	ZIRCALOY-4	ZIRCALOY-4
CLAD THICKNESS				
CLAD MATERIAL		UO_2	UO_2-PuO_2-Er_2O_3	UO_2-PuO_2-Er_2O_3
FUEL PELLETS	IN	0.325	0.325	0.325
MATERIAL (SINTERED)	IN	0.390	0.390	0.390
DIAMETER				
LENGTH		INCONEL 625	INCONEL 625	INCONEL 625
CONTROL ASSEMBLIES	IN	0.035	0.035	0.035
CLADDING MATERIAL		B_4C	B_4C	B_4C
CLAD THICKNESS				
ABSORBER MATERIAL	IN	157/162	157/162	157/162
CORE STRUCTURE				
CORE BARREL ID/OD				

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TABLE II-1

SUMMARY COMPARISON OF PLANT PARAMETERS

ITEM	UNIT	Reference System 80+ Plant with Base UO_2 Core	Full Plutonium Core load	Plutonium Core with Tritium
3. REACTOR COOLANT SYSTEM				
REACTOR VESSEL				
OVERALL HEIGHT (INCL. CEDM NOZZLES)	FT.	48	48	48
OVERALL DIAMETER (INCL. RCS	FT	23.3	23.3	23.3
NOZZLES)	FT	15.2	15.2	15.2
INTERNAL DIAMETER	IN	9	9	9
CYLINDRICAL SHELL THICKNESS				
MATERIALS		C.S	C.S	C.S
BASE		S.S. + INCONEL	S.S. + INCONEL	S.S. + INCONEL
CLADDING	IN	.125	.125	.125
CLAD THICKNESS				
NOZZLES (LOCATION, NUMBER, SIZE)	#/IN	101/2	101/2	101/2
CLOSURE HEAD: CEDM'S	#/IN	4/30	4/30	4/30
VESSEL: INLET	#/IN	2/42	2/42	2/42
VESSEL: OUTLET	#/IN	4/10	4/10	4/10
DIRECT VESSEL INJECTION				
STEAM GENERATOR:		2	2	2
NUMBER		VERTICAL V.TUBE	VERTICAL V.TUBE	VERTICAL V.TUBE
TYPE	FT	77.8	77.8	77.8
OVERALL HEIGHT	FT	20.5	20.5	20.5
OVERALL DIAMETER		12,300	12,300	12,300
NUMBER OF TUBES/S.G.	IN	0.75	0.75	0.75
TUBE SIZE O.D.	IN	0.042	0.042	0.042
TUBE THICKNESS	FT^2	150,895	150,895	150,895
SURFACE AREA/S.G.				
MATERIALS		IN.690	IN.690	IN.690
TUBING		C.S. CLAD WITH S.S	C.S. CLAD WITH S.S	C.S. CLAD WITH S.S
PRIMARY SIDE		C.S	C.S	C.S
SECONDARY SIDE				

TABLE II-1

SUMMARY COMPARISON OF PLANT PARAMETERS

ITEM	UNIT	Reference System 80+ Plant with Base UO_2 Core	Full Plutonium Core load	Plutonium Core with Tritium
PRESSURIZER				
DESIGN PRESSURE	PSIA	2500	2500	2500
DESIGN TEMPERATURE	°F	700	700	700
OVERALL HEIGHT	FT	54.4	54.4	54.4
OVERALL DIAMETER	FT	8.9	8.9	8.9
VOLUME	FT^3	2400	2400	2400
SAFETY VALVES	#/IN	4/6	4/6	4/6
CAPACITY EACH	#/Hr	$4.6 + 10^5$	$4.6 + 10^5$	$4.6 + 10^5$
SPRAY FLOW RATE	GPM	375	375	375
HEATERS-IMMERSION TYPE	#/KW	48/50	48/50	48/50
REACTOR COOLANT PUMP				
NUMBER		4	4	4
TYPE		VERTICAL SINGLE STAGE CENTRIFUGAL	VERTICAL SINGLE STAGE CENTRIFUGAL	VERTICAL SINGLE STAGE CENTRIFUGAL
DESIGN FLOW (EACH)	GPM	111,400	111,400	111,400
DESIGN HEAD	FT	365	365	365
MOTOR				
TYPE		A-C INDUCTION	A-C INDUCTION	A-C INDUCTION
RATING, COLD	H.P	12,000	12,000	12,000
SEAL-TYPE		MECHANICAL FACE	MECHANICAL FACE	MECHANICAL FACE
NUMBER		3	3	3

TABLE II-1

SUMMARY COMPARISON OF PLANT PARAMETERS

ITEM	UNIT	Reference System 80+ Plant with Base UO_2 Core	Full Plutonium Core load	Plutonium Core with Tritium
4. SHUTDOWN COOLING SYSTEM				
MAX. OPERATING TEMP.	°F	350	350	350
MAX. OPERATING PRES.	PSIA	400	400	400
DESIGN TEMP.	°F	400	400	400
DESIGN PRES	PSIA	915	915	915
HEAT EXCHANGERS				
NUMBER		2	2	2
HEAT REMOVAL CAPACITY (EA)	BTU/HR	$3.38X10^7$	$3.38X10^7$	$3.38X10^7$
FLOW-TUBE SIDE-REACTOR	GPM	5000	5000	5000
COOLANT (EA)	GPM	10940	10940	10940
FLOW-SHELL SIDE-COMPONENT				
COOLING (EA)		2	2	2
PUMPS	GPM	5000	5000	5000
NUMBER	FT	400	400	400
DESIGN FLOW - EA.				
DESIGN HEAD				
	#	4	4	4
5. SAFETY INJECTION SYSTEM	GPM	815	815	815
PUMPS-CENTRIFUGAL, HORIZONTAL,	FT	2850	2850	2850
MULTISTAGE	°F	350	350	350
DESIGN FLOW - EA.	PSIA	2065	2065	2065
DESIGN HEAD	#	4	4	4
DESIGN TEMP.	FT^3	2406	2406	2406
DESIGN PRES.	FT^3	1858	1858	1858
TANKS (SAFETY INJECTION)	PSIA	700	700	700
TOTAL VOLUME (EA)	#	1	1	1
OPERATING LIQUID VOLUME (EA)	FT^3	116,000	116,000	116,000
DESIGN PRES.	GAL	495,000	495,000	495,000
IN-CONTAINMENT REFUELING WATER				
STORAGE TANK				
TOTAL VOLUME				
VOLUME FOR SAFETY				
INJECTION				

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TABLE II-1

SUMMARY COMPARISON OF PLANT PARAMETERS

	ITEM	UNIT	Reference System 80+ Plant with Base UO_2 Core	Full Plutonium Core load	Plutonium Core with Tritium
6.	CONTAINMENT SPRAY SYSTEM				
	PUMPS-CENTRIFUGAL, VERTICAL,	#	2	2	2
	SINGLE STAGE	GPM	5000	5000	5000
	DESIGN FLOW - EA	FT	400	400	400
	DESIGN HEAD - EA	°F	400	400	400
	DESIGN TEMP	PSIA	900	900	900
	DESIGN PRES.	#	2	2	2
	HEAT EXCHANGERS	GPM	5000	5000	5000
	FLOW-TUBE SIDE-REACTOR	GPM	8000	8000	8000
	COOLANT (EA)	LBM	658	658	658
	FLOW-SHELL SIDE-COMPONENT	GPM	15.2	15.2	15.2
	COOLING (EA)	μ	600	600	600
	SPRAY NOZZLES				
	FLOW/NOZZLE				
	DROPLET SIZE				
		#	4	4	4
7.	EMERGENCY FEEDWATER SYSTEM	#	2	2	2
		#	2	2	2
	PUMPS-MULTI-STAGE, HORIZONTAL	GPM	500	500	500
	CENTRIFUGAL	FT	3620	3620	3620
	MOTORS-A.C.	#	2	2	2
	STEAM TURBINES	GAL	175,000	175,000	175,000
	DESIGN FLOW (EA)	PSIG	0.5	0.5	0.5
	DESIGN HEAD	°F	140	140	140
	TANKS				
	MINIMUM USABLE VOLUME, EA				
	DESIGN PRES.				
	DESIGN TEMP.				

TABLE II-1

SUMMARY COMPARISON OF PLANT PARAMETERS

	ITEM	UNIT	Reference System 80+ Plant with Base UO_2 Core	Full Plutonium Core load	Plutonium Core with Tritium
8.	**CONTAINMENT**		DUAL-SPHERICAL STEEL W REINFORCED CONCRETE SHIELD BLDG.		
	TYPE				
	DIAMETER-INSIDE-SPHERE	FT	200	200	200
	FREE VOLUME	FT^3	$3.4X10^6$	$3.4X10^6$	$3.4X10^6$
	STEEL THICKNESS	IN	1.75	1.75	1.75
	DESIGN PRESSURE	PSIA	53	53	53
	DESIGN TEMP.	°F	290	290	290
	DIAMETER-OUTSIDE-SHIELD BLDG.	FT	216	216	216
	THICKNESS-SHIELD BUILDING	FT	3	3	3
9.	**BACKUP ELECTRICAL SYSTEMS**				
	EMERGENCY ON SITE GENERATORS				
	DIESELS-SAFETY GRADE	#	2	2	2
	RATING-EA	H.P.	8575	8575	8575
	COMBUSTION TURBINES-NON-SAFETY	#	1	1	1
	GRADE	H.P.	5000	5000	5000
	RATING				
10.	**FEEDWATER SYSTEM**				
	MAIN FEEDWATER PUMPS	#	3	3	3
	CAPACITY OF MAIN FEED PUMPS	%	50	50	50
	FEED PUMP MOTOR TYPE	DUTY	VARIABLE SPEED	VARIABLE SPEED	VARIABLE SPEED
	FEED PUMP MOTOR POWER	H.P.	17500	17500	17500
	MAIN FEED FLOW (EA)	GPM	17,160	17,160	17,160
	MAIN FEED TEMP.	°F	450	450	450
11.	**MAIN STEAM SYSTEM**				
	STEAM FLOW - EA.	LBM/HR	$8.56X10^6$	$8.54X10^6$	$7.57X10^6$
	STEAM TEMP	°F	544.6	544.6	544.6
	MINIMUM PRESSURE	PSIA	1000	1000	1000
	MAXIMUM MOISTURE	%	0.25	0.25	0.25

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TABLE II-1

SUMMARY COMPARISON OF PLANT PARAMETERS

	ITEM	UNIT	Reference System 80+ Plant with Base UO_2 Core	Full Plutonium Core load	Plutonium Core with Tritium
12.	<u>TURBINE GENERATOR</u>	#	1	1	1
	TURBINE TYPE		REACTION	REACTION	REACTION
	ROTATIONAL SPEED	RPM	1500	1800	1800
	MAIN STEAM PRES. AT TURBINE STOP	PSIA	947	947	947
	VALVES	%	0.3	0.3	0.3
	MAIN STEAM WETNESS AT T.S.V.'S		TITANIUM	TITANIUM	TITANIUM
	CONDENSER TUBE MATERIAL				
13.	<u>CHEMICAL & VOLUME CONTROL SYSTEM</u>				
		GPM	200	200	200
	MAXIMUM LETDOWN RATE	GPM	200	200	200
	MAXIMUM CHARGING RATE	°F	419	419	419
	NORMAL CHARGING TEMP.	GPM	200	200	200
	MAXIMUM PURIFICATION RATE	GPM	100	100	100
	NORMAL PURIFICATION RATE	GPM	33	33	33
	NORMAL R.C. PUMP SEAL INJECTION	GPM	200	200	200
	FLOW	GAL.	4914	4914	4914
	MAXIMUM DEGASIFICATION RATE	GPM	100	100	100
	VOLUME CONTROL TANK VOLUME	#/HR	208	208	208
	CHARGING PUMP CAPACITY - EA.				
	CHEMICAL ADDITION FLOW RATE				