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1.8 SYMBOLS AND TERMS USED IN ENGINEERING AND TEXT

8202020071.

1.8.1 TEXT DEFINITIONS AND ABBREVIATIONS

Definitions used throughout the FSAR are listed in the Glossary of Terms, Table 1.8-1. Ancronyms and technical abbreviations are listed in Tables 1.8-2 and 1.8-3, respectively.

1.8.2 DRAWING INDEX AND SYMBOLS

Design drawings which have been used in this FSAR have been listed with a cross reference to the FSAR figure number in Table 1.8-4. Abbreviations used on these drawings are listed in Table 1.8-5.

Symbols used on GE supplied Piping and Instrument Diagrams (P&ID's) are shown on figure 1.8-1. Symbols for other P&ID's are shown on figures 1.8-2a, 1.8-2b, and 1.8-2c. Logic Symbols and Instrument Symbols are shown on figures 1.8-3 and 1.8-4 respectively.

1.8.3 PIPING IDENTIFICATION

Piping is identified on the Piping and Instrument Diagrams (P&ID's) by a three-group identifier where the first group is the nominal pipe size in inches; the second is a three-letter group for the pipe class; and the third is a three-digit group sequentially assigned within a pipe class.

Example:

6"-HBD-117

Size Class Sequence

The three letter group for the pipe class is described in detail in Table 1.8-6.

The three digit sequence number is assigned consecutively to identify specific lines in a pipe class as follows:

Piping common to both units	0-99 and 3001-3999
Piping for Unit 1	100-199 and 1000-1999
Piping for Unit 2	200-299 and 2000-2999

1.8.4 VALVE IDENTIFICATION

All manual and remotely operated valves will have unique identification numbers for tracking purposes and will be shown on the P&ID's.

Listed below are the numbering systems used for each group of valves:

All manual valves, except those which have a GE Master Parts List (MPL) number, and those valves supplied by vendors as part of the equipment package and not installed by Bechtel will be identified by the following method:

```
                                  1       52     006
                                  :       :      :
                                  :       :      :
Unit No.----------------------:       :      :
0-Common                              :      :
1-Unit 1                              :      :
2-Unit 2                              :      :
                                      :      :
System Identification                 :      :
(last 2 digits of P&IDs)--------------:      :
                                             :
Sequence No.-------------------------------:
(3 digit numbers)
```

Remote operated valves which do not have a GE MPL number, are identified by the operator number, eg:

```
                                  HV      1       52      40
                                  :       :       :       :
                                  :       :       :       :
Valve type-------------------------:       :       :       :
                                           :       :       :
Unit No.-----------------------------------:       :       :
                                                   :       :
P&ID No.-------------------------------------------:       :
(last 2 digits)                                            :
                                                           :
Sequence No.-------------------------------------------------:
```

Those valves in GE's MPL are identified by the GE numbering system, eg:

```
                              E11     HV     1     F031

                               '      '      '      '
                               '      '      '      '
MPL System No.----------------'       '      '      '
(Referenced on figure notes)          '      '      '
                                      '      '      '
Valve Type---------------------------'      '      '
                                             '      '
Unit No.-----------------------------------'       '
                                                    '
GE Valve No.-------------------------------------------'
```

Valves that are not numbered but are supplied as part of a vendor mounted equipment will be identified in the vendor's operation and maintenance manuals. This is to avoid duplication of numbering these valves.

1.8.5 INSTRUMENT IDENTIFICATION

1.8.5.1 Instrument Components

Identification of instruments and control devices is made by the use of one of the following numbering systems:

1. Instruments and devices within GE scope of design are numbered in accordance with the GE MPL system. Associated devices shown on P&ID's but without any numerical identity are numbered as in 2 below.

2. Except as in 1 above, instrument identifications are based on Instrument Society of America (ISA) Standard S5.1-1973, as modified by Figures 1.8-2a through 1.8-2c.

In general, each instrument or device in a measurement loop is assigned the same number, however, loops containing instruments and devices identified in the GE MPL system are an exception to this rule.

When a loop contains more than one instrument component of the same functional type, a suffix letter will be added and used to establish a unique identity for those components.

Redundant measurement loops will be identified by the addition of a suffix letter to each instrument component or device in the loop. In the case of redundant loops containing more than one

instrument of the same functional type, the suffix letter will be followed by a number.

Instrument and device numbers are constructed as follows:

```
                          FE      1     11     78     A1
                           :      :      :      :      :
Functional Identification----:    :      :      :      :
Per 8856-M-100               :    :      :      :      :
                             :    :      :      :      :
Unit Number------------------------:      :      :      :
                                         :      :      :
Last 2 Digits of                         :      :      :
P&ID Number-------------------------------:      :      :
                                                :      :
Loop Number-------------------------------------------:
                                                       :
Suffix-----------------------------------------------------:
```

A zero in the unit number position indicates that the instrument or device is common to both units.

1.8.5.2 Instrument Location

Instrument components and devices are mounted on racks and panels which are identified by a 5 character, alpha-numberic code. This code is marked adjacent to the instrument component identifier, as shown on Figure 1.8-2a.

The code numbers identify the unit number and the general location of the rack or panel by the following block-number assignment:

C001 - C099	NSSS Local Panels and Racks
C101 - C199	Turbine Building
C201 - C299	Reactor Building
C301 - C399	Radwaste Building
C401 - C499	Primary Containment
C501 - C599	Miscellaneous Locations
C601 - C699	Control Structure
C701 - C799	Administration Building

A prefix digit is used to identify the unit or common plant assignment.

With each block-number assignment above, the series from 076 thru 099 are reserved for local racks and panels in heating and ventilation service. The following examples illustrate typical rack or panel assignments:

```
                                        1    C    6    50
                                        '    '    '    '
                                        '    '    '    '
Unit Number-----------------------------'    '    '    '
                                             '    '    '
"C" = Rack or Panel--------------------------'    '    '
"CB" = Component Box-------------------------'    '    '
"Z" = Plant Computer-------------------------'    '    '
                                                  '    '
Control Structure-------------------------------------'    '
                                                           '
Panel Number-------------------------------------------------'
```

1.8.6 ELECTRICAL COMPONENT IDENTIFICATION

This section describes the methods used to identify electrical equipment locations and to number electrical schemes, cables, and raceways. Additional information is contained in Section 8.3.

1.8.6.1 Equipment Location Numbers

Each piece of electrical equipment is identified by an equipment number. To facilitate cable routing from one equipment location to another, a location number is also assigned to each piece of electrical equipment. Generally, the equipment number and equipment location number for a specific piece of electrical equipment are identical. For large pieces of electrical equipment, such as switchgear, load centers, and motor control centers, which are compartmentalized, the equipment location number consists of the basic equipment number plus additional suffixed information to identify a location within the equipment itself. The following two examples illustrate equipment location numbers:

```
                                          1    X    1   01
                                          |    |    |   |
Unit Number - (1)------------------------|    |    |   |
                                          |    |    |   |
Equipment Classification-(Transformer)---------|    |   |
                                          |    |    |   |
Plant Area - (Turbine Building)----------------------|   |
                                          |    |    |   |
Sequential Number-(Transformer No. 1)-------------------|
```

```
                                     1    B    1    11   12   1
                                     |    |    |    |    |    |
Unit Number - (1)--------------------|    |    |    |    |    |
                                     |    |    |    |    |    |
Equipment Classification - (Motor    |    |    |    |    |    |
Control Center (MCC))----------------|    |    |    |    |    |
                                     |    |    |    |    |    |
Plant Area - (Turbine Building)------------------|    |    |
                                     |    |    |    |    |    |
Sequential number - (MCC No. 11)-----------------------|    |
                                     |    |    |    |    |    |
Stack Number - (12, left to right)-------------------------|
                                     |    |    |    |    |    |
Cubicle Number - (1, top to bottom)-------------------------------|
```

In the first example, the equipment number and equipment location number for transformers 1X101 are identical. In the second example, the basic MCC equipment number 1B111 is suffixed to establish an equipment location number, 1B111121, which identifies a specific compartment within the MCC.

To distinguish one piece of electrical equipment from other duplicate equipment used in the same service, a suffix letter is added to the basic equipment number to establish individual equipment location numbers. For example, if main transformer 1X101 is composed of a bank of three single phase transformers, the transformers for phases A, B and C are identified with equipment location numbers 1X101A, 1X101B and 1X101C, respectively.

Equipment location numbers are generally assigned to items listed in the circuit and raceway schedules. Accordingly, most electrical equipment related to systems such as lighting, communications, and cathodic protection is not included.

Electrical equipment which is an integral part of mechanical equipment is assigned the same number as the mechanical equipment.

All major pieces of electrical equipment are listed in an equipment index. The equipment index provides a description of

the equipment and identifies pertinent drawings such as applicable electrical layout drawings and P&ID's.

1.8.6.2 Scheme Numbers

Each electrical scheme is identified by a six character number. The first character is numeric and refers to the plant unit number for which the scheme is applicable. The second character is alphabetic and classifies the scheme by major plant system. The last four characters are numeric, with the exception of GE supplied cables, and provide a sequential, but arbitrary, identity for each scheme. Given below is an example of a typical scheme number.

```
                                          1      0     0501
                                          :      :      :
Unit Number - (1)------------------------:      :      :
                                                :      :
Plant System - (Nuclear Steam Supply System)----:      :
                                                       :
Scheme Sequential Number - (Arbitrary No.              :
for RHR Pump 1A)-------------------------------------:
```

A log of all schemes is maintained in the scheme number index which contains pertinent information such as scheme description, scheme drawing number and source drawing number.

1.8.6.3 Scheme Cable Numbers

Except for cabling associated with the plant lighting, communications, and cathodic protection systems, each cable in the plant is identified by a scheme cable number composed of nine characters. The first character is alphabetic and indicates the separation group to which the cable belongs. The second character is also alphabetic and denotes the system voltage level. Characters three through eight identify the six character scheme number to which the cable is assigned. The ninth and final character is alphabetic, except for GE supplied cables, and provides a distinctive identity to each cable in the block diagram shown on the scheme drawings. The following two examples illustrate typical scheme cable numbers:

```
                                                A    K    1Q0501    R
                                                |    |    |         |
Separation Group-(Safeguard Channel A)-------'    |    |         |
                                                     |    |         |
Voltage Level-(120 Vac to 250 Vdc Control)-------'    |         |
                                                          |         |
Scheme Number-(1Q0501 for RHR Pump 1A)-----------------'         |
                                                                    |
Cable Identity-(Cable R in Block Diagram)----------------------'

                                                N    M    1R3007    D
                                                |    |    |         |
Separation Group-(Non-Safety Related)---------'    |    |         |
                                                     |    |         |
Voltage Level-(Low Level Instrumentation)---------'    |         |
                                                          |         |
Scheme Number-(Radwaste Bldg. Sump Pump A)-------------'         |
                                                                    |
Cable Identity (Cable D in Block Diagram)----------------------'
```

An alpha-numeric listing of all scheme cable numbers is maintained in the electrical circuit schedule. The circuit schedule also identifies the cable type, quantity of conductors, from and to equipment location numbers, and the cable routing. The circuit schedule uses the first two characters of the scheme cable number as a facility code, ensures that separation and voltage criteria are not violated.

A cable marker is affixed to each end of the cable for permanent identification. Cable markers for Class IE cables have distinguishing colors for each separation group. Additionally, all Class IE cables are marked at regular intervals along their length with colors corresponding to the cable marker colors.

1.8.6.4 Raceway Numbers

All electrical cable trays, ducts, conduits, manholes, conduit sleeves and junction boxes are identified by six character raceway numbers. The two examples given below illustrate typical raceway numbers for engineered safety feature and non-safety feature cable trays, respectively.

	A	1	K	B	99
Separation Group - (Safeguard Channel A)					
Unit Number - (1)					
Voltage Level - (120 VAC to 250 VDC Control)					
Main or Branch Run - (B)					
Section Number - (Tray Section 99)					

	1	P	B	C	85
Unit Number - (1)					
Voltage Level - (250 VDC to 480 VAC Power)					
Main Run - (Main Tray B)					
Branch Run - (Branch Tray C)					
Section Number - (Tray Section 85)					

The first character of each engineered safety feature cable tray is an alphabetic letter that relates to the first character of each engineered safety feature scheme cable eligible for routing, therein. Non-safety feature cable tray, whose first character is numeric representing the unit number, may only contain scheme cable numbers prefixed by the letter N. This same practice was followed for conduit numbers as shown below.

	A	1	P	999
Separation Group - (Safeguard Channel A)				
Unit Number - (1)				
Voltage Level - (250 VDC to 480 VAC Power)				
Conduit Sequential Number - (Arbitrary No.)				

```
                                                  1    H    B    005
                                                  :    :    :    :
Unit Number - (1)----------------------------:    :    :    :
                                                  :    :    :
Voltage Level - (13.8kV Power)-------------------:    :    :
                                                       :    :
Plant Area - (Turbine Bldg. Elev. 656')---------------:    :
                                                            :
Conduit Sequential Number - (Arbitrary No.)-----------------:
```

An alphanumeric listing of all raceway numbers is maintained in the electrical Raceway Schedule, which also contains the raceway type, length from end point locations, percent fill, and list of included cables.

Raceway markers are affixed to each raceway for permanent identification. Identification markers for Class IE raceways are marked at regular intervals along the length of the raceway with unique and distinguishing colors for each separation group corresponding to the cable marker colors.

TABLE 1.8-2

ACRONYMS

Name	Abbreviation
American Concrete Institute	ACI
American Institute of Steel Construction, Inc.	AISC
American National Standards Institute	ANSI
American Society of Civil Engineers	ASCE
American Society of Mechanical Engineers	ASME
American Society of Mechanical Engineers Boiler and Pressure Vessel Code	ASME B&PV Code
American Society for Testing and Materials	ASTM
American Welding Society	AWS
American Petroleum Institute	API
American Water Works Association	AWWA
Area Radiation Monitor	ARM
Automatic Depressurization	ADS
Average Power Range Monitor	APRM
Balance of Plant	BOP
Bechtel Power Corporation (San Francisco)	Bechtel
Beginning of core life	BOL
Boiling Water Reactor	BWR
Closed Cooling Water	CCQ
Control Rod Drive	CRD
Control Rod Position Indicator	CRPI
Core Spray	CS
Critical Power Ratio	CPR

TABLE 1.8-2

ACRONYMS

Departure from Nucleate Boiling	DNB
Design Basis Accident	DBA
Diesel Engine Generator	DG
Dye Penetrant Test/Liquid Penetrant Test	PT
East	E
Electrohydraulic Control	EHC
Emergency Core Cooling System	ECCS
End of core life	EOL
End of Cycle	EOC
Engineered Safety Features	ESF
Engineering Change Authorization	ECA
Engineering Change Notice	ECN
Equivalent full power years	EFPY
Excess Flow Check Valve	EFCV
Field Deviation Disposition Request	EDDR
Final Safety Analysis Report	FSAR
Fuel Pool Cooling and Cleanup	FPCC
Full Arc(Mode of TCV Operation)	FA
Full-Length Emergency Cooling Heat Transfer	FLECHT
Functional Control Diagram	FCD
General Electric Company	GE
Heat Exchanger	HX
Heating and Ventilating	H&V
Heating Vintilating and Air-Conditioning	HVAC
High efficiency particulate air-filter	HEPA

TABLE 1.8-2

ACRONYMS

High Pressure Coolant Injection	HPCI
Hydraulic Control Unit	HCV
Instrument Data Sheet	IDS
Institute of Electrical and Electronics Engineers	IEEE
Instrument Society of America	ISA
Insulated Power Cable Engineers Association	IPCEA
Interim Acceptance Criteria (NRC)	IAC
Intermediate Range Monitor	IRM
Leakage Control System	LCS
Leak-Detection System	LDS
Limiting Condition of Operation	LCO
Limiting Safety System Setting	LSSS
Local Power Range Monitor	LPRM
Loss-Of-Coolant Accident	LOCA
Low Pressure Coolant Injection	LPCI
Low Population Zone	LPZ
Magnetic Particle Test	MT
Main Steam Isolation Valve	MSIV
Main Steam Insulation Valve Leakage Control System	MSIV-LCS
Main Steam Line	MSL
Manufacturers Standardization Society	MSS
Maximum Average Planar Linear Heat Generation Rate	MAPLHGR
Mean Low Water Datum	MLD

TABLE 1.8-2

ACRONYMS

Mean Sea Level	MSL
Minimum Critical Power Ratio	MCPR
Motor Control Center	MCC
Motor-Generator Set	MG
National Electrical Manufacturers Association	NEMA
Neutron-Monitoring System	NMS
Nil Ductility Transition Temperature	NDTT
Nondestructive Examination	NDE
Nondestructive Testing	NDT
North	N
Nuclear Boiler	NB
Nuclear Boiler Rated (power)	NBR
Nuclear Energy Division (GE)	GED
Nuclear Regulatory Commission	NRC
Nuclear Safety Operational Analysis	NSOA
Nuclear Steam Supply Shutoff System	NSSSS
Nuclear Steam Supply System	NSSS
Operating Basis Earthquake	OBE
Peak Cladding Temperature	PCT
Pennsylvania Power and Light Co.	PP&L
Piping and Instrumentation Diagram	P&ID
Plant Vent Stack	PVS
Power Range Monitor	PRM
Preconditioning Cladding Interim Operating Management Recommendation	PCIOMR
Preliminary Safety Analysis Report	PSAR

TABLE 1.8-2

ACRONYMS

Probable Maximum Flood	PMF
Process Computer System	PCS
Public Address System	PA
Quality Assurance	QA
Quality Control	QC
Radiographic Test	RT
Reactor Coolant Pressure Boundary	RCPB
Reactor Core Isolation Cooling	RCIC
Reactor Manual Control	RMC
Reactor Pressure Vessel	RPV
Reactor Protection System	RPS
Reactor System Outline	RSO
Reactor Water Cleanup	RWC
Regulatory Guide (NRC) (formerly Safety Guide)	RG
Residual Heat Removal	RHR
Rod Block Monitor	RBM
Rod Sequence Control System	RSCS
Rod Position Information System	RPIS
Rod Worth Minimizer	RWM
Safe Shutdown	SS
Safe Shutdown Earthquake	SSE
Safety Analysis Report	SAR
Safety/Relief Valve	SRV
Seismic Category I or II	SC I or II

TABLE 1.8-2

ACRONYMS

Service Water	SW
Source Range Monitor	SRM
South	S
Standby Gas Treatment System	SGTS
Standby Liquid Control	SLC
Traversing Incore Probe	TIP
Turbine Control Valve	TCV
Turbine-Generator	TG
Ultrasonic Testing	UT
West	W

TABLE 1.8-4

FIGURE INDEX FOR PLANT SYSTEMS

P&ID NUMBER*	SYSTEM	FSAR FIGURE NUMBER
M-100	P&ID Legend & Symbols, Shts. 1,2 & 3	1.8-2a thru 1.8-2c
M-101	Main Steam	10.4-1
M-102	Extraction Steam	10.4-6
M-103	Vents & Drains, Heaters 1,2 & Drain Cooler	10.4-7
M-104	Vents & Drains, Heaters 3,4 & 5	10.4-8
M-105	Condensate	10.4-4
M-106	Feedwater	10.4-5
M-107	Air Removal & Sealing Steam	10.4-9
M-108	Condensate & Refueling Water Storage	9.2-9
M-109	Service Water	9.2-1a
M-110	Service Water	9.2-1b
M-111	Emergency Service Water, Sheets 1 & 2	9.2-5a,9.2-5b
M-112	RHR Service Water	9.2-6
M-113	Reactor Building Closed Cooling Water	9.2-2
M-114	Turbine Building Closed Cooling Water	9.2-3
M-115	Circulating Water	NOT REFERENCED
M-116	Condensate Demineralizer, Sheets 1 & 2	10.4-2, 10.4-3
M-117	Raw Water Treatment, Sheets 1 & 2	9.2-7a, 9.2-7b
M-118	Make-Up Demineralizer	9.2-8
M-119	Lube Oil	NOT REFERENCED
M-120	Diesel Oil Storage & Transfer	9.5-19
M-121	Auxiliary Steam	NOT REFERENCED
M-122	Fire Protection, Shts. 1,2,3 & 4	9.5-9 thru 9.5-12

TABLE 1.8-4

FIGURE INDEX FOR PLANT SYSTEMS

P&ID NUMBER*	SYSTEM	FSAR FIGURE NUMBER
M-123	Process Sampling, Shts. 1,2,3 & 4 Process Sampling, Sheet 5	9.3-6 thru 9.3-9 NOT REFERENCED
M-124	Chlorination	NOT REFERENCED
M-125	Compressed Air, Shts. 1,2 & 3	9.3-1, 9.3-2, 9.3-4
M-2125	Compressed Air, Unit 2, Sheets 1 & 2 (later)	9.3-3a,9.3-3b (later)
M-126	Containment Instrument Gas	9.3-5
M-127	Feed Pump Turbine Steam	NOT REFERENCED
M-128	Make-Up Water Supply System	NOT REFERENCED
M-129	Process Valve Steam Leakoff Collection	NOT REFERENCED
M-130	ASME Test	NOT REFERENCED
M-131	Gaseous Radwaste Recombiner Closed Cooling Water	9.2-4
M-132	Acid Injection for the Circulating Water System	NOT REFERENCED
M-133	Hydrogen Storage	NOT REFERENCED
M-134	Diesel Engine Auxiliaries	9.5-20
M-136	Primary Coolant Degasifier Package	NOT REFERENCED
M-137	Area Radiation Monitoring	12.3-29
M-138	Cooling Tower Blowdown Treatment	NOT REFERENCED
M-139	MSIV Leakage Control System	6.7-1
M-140	Reactor Recirculation Motor Generator Set	NOT REFERENCED
M-141	Nuclear Boiler	5.1-3a
M-142	Nuclear Boiler Vessel Instrumentation	5.1-3b
M-143	Reactor Recirculation	5.4-2b

TABLE 1.8-4

FIGURE INDEX FOR PLANT SYSTEMS

P&ID NUMBER*	SYSTEM	FSAR FIGURE NUMBER
M-144	Reactor Water Clean-Up	5.4-16
M-145	Clean-Up Filter-Demineralizer	5.4-18
M-146	Control Rod Drive - Part A	4.6-5a
M-147	Control Rod Drive - Part B	4.6-5b
M-148	Standby Liquid Control	7.4-3, 9.3-13
M-149	Reactor Core Isolation Cooling	5.4-9a, 7.4-1 sh. 1
M-150	RCIC Turbine - Pump	5.4-9b, 7.4-1 sh. 2
M-151	Residual Heat Removal, Sheets 1 & 2	5.4-13a, 5.4-13b
M-152	Core Spray	6.3-4
M-153	Fuel Pool Cooling & Clean-Up	9.1-5
M-154	Fuel Pool Filter-Demineralizer	9.1-6
M-155	High Pressure Coolant Injection	6.3-1a
M-156	HPCI Turbine - Pump, Shts. 1,2 & 3 (later)	6.3-1b
M-157	Containment Atmosphere Control, Sheets 1, 2, & 3 (later)	6.2-55a, 6.2-55b, 6.2-55c (later)
M-159	Primary Containment Leakage Rate Testing	6.2-67
M-160	Miscellaneous Drainage	9.3-12
M-161	Liquid Radwaste Collection Sheets 1 & 2	9.3-10, 9.3-11
M-162	Liquid Radwaste Processing Sheets 1 & 2	11.2-9, 11.2-10
M-163	Liquid Radwaste Chemical Processing	11.2-11
M-164	Liquid Radwaste Laundry Processing	11.2-12
M-166	Solid Radwaste Collection	11.4-1
M-167	Radwaste Solidification	11.4-2

TABLE 1.8-4

FIGURE INDEX FOR PLANT SYSTEMS

P&ID NUMBER*	SYSTEM	FSAR FIGURE NUMBER
M-169	Offgas Recombiner System, Sheets 1 & 2	11.3-3a, 11.3-3b
M-171	Ambient Temperature Charcoal Off Gas Treatment System	11.3-4
M-173	Circulating Water Pump House Air Flow Diagram	9.4-20
M-174	Turbine Building Air Flow Diagram	9.4-13
M-175	Reactor Building Air Flow Diagram Zone III	9.4-5
M-176	Reactor Building Air Flow Diagram Zone	9.4-4
M-177	Drywell Air Flow Diagram	9.4-15
M-178	Control Structure Air Flow Diagram	9.4-1
M-179	Radwaste Building Air Flow Diagram, Sheets 1 & 2	9.4-10, 9.4-11
M-181	Miscellaneous Buildings Air Flow Diagram	NOT REFERENCED
M-182	Diesel Generator & ESSW Pump House Air Flow Diagram	9.4-19
M-183	Misc. HV & AC Equipment Drainage System	NOT REFERENCED
M-184	NGH, SGH, & SCC Air Flow Diagram	NOT REFERENCED
M-186	Control Structure Chilled Water	9.2-11
M-187	Reactor Building Chilled Water, Sheets 1 & 2	9.2-13a, 9.2-13b
M-188	Turbine Building Chilled Water	9.2-12
M-189	Radwaste Building Chilled Water	9.2-14
M-190	NGH, SGH, & SCC Refrigerant	NOT REFERENCED

* Numbers correspond to Unit 1 and Common System P&ID's. Unit 2 P&ID's are preceded by the numeral "2".

3.3 WIND AND TORNADO LOADINGS

3.3.1 WIND LOADINGS

All exposed structures are designed for wind loading.

3.3.1.1 Design Wind Velocity

The design wind velocity for all structures is 80 mph at 30 ft above ground for a 100-year recurrence interval. The design wind velocity is based on Figure 5 of Ref 3.3-1. (References are listed in Subsection 3.3.3).

The vertical velocity distribution is based on Table 1(a) of Ref 3.3-2. The velocity distribution is tabulated in Table 3.3-1.

A gust factor of 1.1, as given in Ref 3.3-2, is used.

3.3.1.2 Determination of Applied Forces

The procedure used to transform the wind velocity into an effective pressure applied to exposed surfaces of structures is as described in Ref 3.3-2 and is summarized as follows:

The dynamic pressure is given by:

$q = 0.002558\ V^2$ where,

q = Dynamic pressure in psf

V = Wind velocity in mph (design wind velocity x gust factor).

The local pressure at any point on the surface of a building is equal to:

$q \times Cp$ where

Cp = Pressure coefficient

The total pressure on a building is equal to:

$q \times C_D$ where,

C_D = Shape coefficient.

The Susquehanna SES structures have sloping roofs with a pitch less than 20 degrees. The following are values for Cp and C : (See Ref 3.3-2, p. 1151 and Figure 7)

Cp for windward wall = 0.8 (pressure)

Cp for leeward wall = -0.5 (suction)

Cp for windward slope = 0

Cp for leeward slope = -0.6 (suction)

C_D = 1.3 (pressure).

Wind loads on structures are tabulated in Table 3.3-1.

Exposed tanks are designed to resist a minimum wind load of 30 psf on the vertical projection, based on Ref 3.3-3. For cylindrical tanks, wind is considered acting on six-tenths of the vertical projection. No increases in allowable working stresses are permitted for these structures for loading conditions involving wind.

3.3.2 TORNADO LOADINGS

Table 3.3-2 lists the systems that are protected against tornadoes and the enclosures which provide this protection. This table is based on NRC Regulatory Guide 1.117 (Ref 3.3-4).

3.3.2.1 Applicable Design Parameters

The following design parameters are used for the design of tornado-resistant structures and are based on Ref 3.3-5:

a) Dynamic Wind Loading

Tangential speed: 300 mph

Translational speed: 60 mph

b) Pressure Differential Between the Inside and Outside of a Building

A pressure drop of 3 psi at the rate of 1 psi per second.

c) Tornado-Generated Missiles

These are discussed in Subsection 3.5.1.4.

3.3.2.2 Determination of Forces on Structures

The following procedures are used to transform the tornado loadings into effective loads on structures:

a) Dynamic Wind Loading

A procedure the same as the one utilized to transform the wind velocity into an effective pressure, as described in Subsection 3.3.1.2, is used with the following exceptions:

1) Velocity and velocity pressure are assumed not to vary with height.

2) The gust factor is taken as unity.

As shown in Figure 5 of Ref 3.3-5, and as explained therein, the equivalent uniform tornado wind velocity on the building due to a tangential component of 300 mph and a translational component of 60 mph is 220 mph. On Susquehanna SES the pressure loads are calculated on the basis of a uniform 300 mph wind velocity and are as follows:

Windward pressure on walls:	185 psf
Leeward suction on walls:	115 psf
Total design pressure:	300 psf
Suction (uplift) on roof:	140 psf.

"The turbine building is designed to resist the tornado loading assuming 2/3 of the metal siding and the roof deck being blown away. However, all the frames are designed for the full tornado loading.

The metal siding and the roof deck of all structures are not designed to resist full tornado loading."

b) Differential Pressure Loading

Differential pressure loading is calculated using the following pressure-time function:

The differential pressure is assumed to vary from zero to 3 psi at the rate of 1 psi/sec, remain at 3 psi for 2 seconds and then return to zero at 1 psi/sec.

Blowout panels are used as necessary on safety related structures to minimize differential pressure.

c) Tornado-Generated Missiles

The procedure used for transforming the tornado-generated missile loadings into effective static loads is described in Subsection 3.5.3.

Loadings a), b), and c) are combined in the following manner to obtain the total tornado loading:

(i) $W' = Ww$

(ii) $W' = Wp$

(iii) $W' = Wm$

(iv) $W' = Ww+0.5Wp$

(v) $W' = Ww+Wm$

(vi) $W' = Ww+0.5Wp+Wm$

where,

W' = Total tornado load

Ww = Tornado wind load

Wp = Tornado differential pressure load, and

Wm = Tornado missile load

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads

Structures not designed for tornado loads are checked to ensure that during a tornado they will not generate missiles that have more severe effects than those listed in Table 3.5-2.

The modes of failure of these structures are analyzed to verify that they will not collapse on safety related structures.

3.3.3 REFERENCES

3.3-1. H.C.S. Thom, "New Distributions of Extreme Winds in the United States", Journal of the Structural Division, ASCE (July 1968), pp 1787.

3.3-2. "Wind Forces on Structures", ASCE Paper No. 3269, Transactions, Volume 126, Part II (1961), p 1124.

3.3-3. "Steel Tanks, Standpipes, Reservoir, and Elevated Tanks for Water Storage", AWWA Standard, D100-73.

3.3-4. "Tornado Design Classification", US NRC Regulatory Guide 1.117, (June 1976).

3.3-5. J.A. Dunlap and Karl Wiedner, "Nuclear Power Plant Tornado Design Considerations",Journal of the Power Division, ASCE, (March 1971).

APPENDIX 3.8A

Computer Programs

This appendix contains a description of the computer programs used for the structural analysis of all Seismic Category I structures. For each computer program, there is a brief description of the program's theoretical basis, the assumptions and references used in the program, and the extent of the application. Examples of verification procedures are included for each Bechtel in-house program.

3.8A.1 3D/SAP

3D/SAP is a finite element program used to perform the static analysis of arbitrary, three-dimensional, elastic solids subjected to concentrated or distributed (pressure) loadings thermal expansion and/or arbitrarily directed static body forces. 3D/SAP is a mathematical version of "SAP" (Reference 3.8A-1) which is a general purpose structural analysis computer code.

3D/SAP was developed by the Control Data Corporation and is in the public domain.

3.8A.2 ASHSD

ASHSD (Axisymmetric Shell And Solid) is a special-purpose program which can be used in the elastic, static or dynamic analysis of structural systems capable of being represented as axisymmetric shells and/or solids.

This program is a refinement of the original ASHSD code developed at the University of California at Berkeley. The present program has been highly modified for the special purpose of static and dynamic analysis of nuclear containment structures. The modified program has the following features:

- The code has a shell finite element which uses an interaction stiffness that allows analysis of layered shells.
- Since shell layers may be <u>bonded</u> or <u>unbonded</u> from each other, it is possible to describe concrete shells in their actual geometric form. For example, it is possible to describe liner plate, concrete, reinforcing steel, and post tensioning steel in their real spatial locations.

- Post tension forces may be applied to the shell by subjecting only the unbonded post tensioning elements to a pseudothermal loading.

- Isotropic or orthotropic elastic constants are possible for both shell and solid elements. The orthotropic material properties may be used to describe the different stiffness of reinforcing steel in the hoop and meridional directions, for example.

- Nonuniform thermal gradients through the wall thickness may be imposed.

- Eigenvalues and eigenvectors may be computed by the program.

- Three dynamic response routines are available in the program. They are:

  - Arbitrary dynamic-loading or earthquake-base excitation using an uncoupled (modal) technique.

  - Arbitrary dynamic-loading or earthquake-base excitation using a coupled (direct integration) technique.

  - Response spectrum modal analysis for absolute and square root of the sum of the squares displacements and element stresses.

- The coupled time-history solution has the capability to allow an arbitrary damping matrix.

- The stiffness and mass matrices may be obtained as punched output for input into other programs.

This program allows a useful study of the interaction between a typical nuclear containment structure modeled as an axisymmetric shell and the subsoil modeled as an axisymmetric solid.

This program was verified by comparing the computer results with hand calculations and published references. Three sample problems are presented as examples of verification.

Sample Problem: Closed Cylinder Under Internal Pressure

This problem demonstrated the membrane state of stress in a closed cylinder subjected to a uniformly distributed internal pressure. Hand calculations were used to verify this aspect of the program.

The selected problem was a cylinder with closed ends subjected to internal pressure. Only one half of the cylinder was required in the model because of symmetry. Furthermore, it was assumed that

the closed ends were distant from the section being analyzed and they were excluded.

Two models of the cylinder were actually analyzed. One model used the thin shell elements and the other used the axisymmetric solid elements. These models are shown in Figures 3.8A-1 and 3.8A-2 with their key dimensions.

The problem parameters for both test cases are as follows:

Boundary Conditions:

Node 1: Z displacement = 0
θ displacement = 0
Rotation in R-Z plane = 0
(free to move radially)

Node 16: θ displacement = 0
(free to move axially, radially and to rotate about the θ axis)

Numerical Data:

Material: concrete
Modulus of Elasticity = E = 4.031×10^6 psi
Thickness = t = 36"
Radius = R = 900"
Poisson's Ratio = ν = 0.17
Pressure = p = 60 psi
Length = L = 1800"
N = 27,000 lb/in (an equivalent node load applied at Node 16)

The theoretical values for the membrane force resultants were calculated to be pR/2 (= 27,000 lb/in) axial force, and pR (= 54,000 lb/in) for the circumferential force (hoop direction).

The results obtained from the ASHSD program are presented in Table 3.8A-1, both for the thin shell and the layered shell models. Analytical computations indicated maximum errors at Node 16 of .4% for the longitudinal force and 3.2% for the circumferential force.

Sample Problem: Cylindrical Shell Subjected to Internal Pressure and Uniform Temperature Rise

This test example demonstrated the use of a combined static load and thermal load condition. A short circular cylindrical shell clamped at both ends was subjected to an internal pressure and a uniform temperature rise.

The theoretical solutions given in Reference 3.8A-2 were used to verify this analysis.

This test used a short cylinder that was clamped at both ends. The cylinder had an internal pressure applied and was subjected to a uniform temperature increase. The general arrangement is shown in Figure 3.8A-3.

Because of symmetry, only one-half of the cylinder was used for the finite element model. This is shown in Figure 3.8A-4 with Node 1 located at the middle of the cylinder. For the purpose of inputting the thermal coefficient of expansion of this isotropic shell, it was required to identify the shell material as orthotropic.

Boundary Conditions:

At center of cylinder, Node 1: Z displacement = 0
θ displacement = 0
Rotation in the R-Z plane = 0

At end of cylinder, Node 26: R displacement = 0
Z displacement = 0
θ displacement = 0 (tangential)
Rotation in the R-Z plane = 0

Numerical Data

Material: concrete
Modulus of Elasticity = E = 4,030,508 psi
Poisson's Ratio = ν = 0.17
Thermal Coefficient of Expansion = α = 55 x 10^{-7} in/in/°F
Thickness = t = 30"
Radius = R = 600"
Length = L = 1200"
Pressure = p = 60 psi
Temperature = T = 150°F
R/t = 20
L/R = 2

The theoretical results are shown in Figure 3.8A-5. These values were obtained by using the following equations from Reference 3.8A-2:

Axial Moment: $M_x = 2\mu^2 D_x (\frac{pR^2}{Et} + R\alpha T)$

where $\mu^2 = [\frac{3\ (1-\nu^2)}{R^2 t^2}]\ 1/2$

and $$D_X = \frac{Et^3}{12(1-\nu^2)}$$

Normalized length: $Ln. = (Z/R)\ (L/2R)$

Figure 3.8A-5 compares the results obtained from the ASHSD program and the theoretical solution. The results of ASHSD agree well with those of the reference.

Sample Problem: Asymmetric Bending of a Cylindrical Shell

The purpose of this test example was to illustrate the use of higher harmonics for asymmetric loading cases. As a comparison to the computer output, results for this problem were taken from B. Budiansky and P. P. Radkowski's Numerical Analysis of Unsymmetric Bending of Shells of Revolution (Reference 3.8A-3).

The cylindrical shell that was analyzed was a short, wide cylinder as shown in Figure 3.8A-6. The finite element idealization of the cylinder and the pertinent data are illustrated in Figure 3.8A-7. At each end of the cylinder, moments of the form $M = M_o \cos n\theta$ were input for harmonics n = 0, 2, 5, 20.

The problem parameters are as follows:

Material: steel
$E = 29 \times 10^6$ psi
$t = 1.25''$
$R = 60.0''$
$\nu = 0.3$
$L = 60.0''$
$\therefore = L/R = 1$
$R/t = 48$

$$M_o = \frac{Et^2}{100\ (1-\nu^2)}$$

$$= 497939.56 \text{ lb - in/in}$$

The comparison results were taken directly from the reference. Those results were plotted in Figure 3.8A-8.

The comparison of the computer results to the reference results are shown in Figure 3.8A-8. (Note that the longitudinal moments and radial displacements are expressed as nondimensional ratios.)

The reference and computer results showed good agreement. This verified the accuracy of the program for this type of analysis.

3.8A.3 CECAP

CECAP computes stresses in a concrete element under thermal and/or nonthermal (real) loads, considering effects of concrete cracking. The element represents a section of a concrete shell or slab, and may include two layers of reinforcing, transverse reinforcing, prestressing tendons, and a liner plate.

CECAP assumes linear stress-strain relationships for steel and concrete in compression. Concrete is assumed to have no tensile strength. The solution is an iterative process, whereby tensile stresses found initially in concrete are relieved (by cracking) and redistributed in the element. Equilibrium of nonthermal loads is preserved. For thermal effects, the element is assumed free to expand inplane, but fixed against rotation. The capability for expansion and cracking generally results in a reduction in thermal stresses from the initial condition.

To verify this program, example problems were analyzed by CECAP and compared with hand calculation solutions. These example problems considered a reinforced concrete beam as shown in Figure 3.8A-9. The problem parameters are as follows:

Concrete modulus of elasticity,	$E_c = 3 \times 10^6$ psi
Rebar modulus of elasticity,	$E_s = 30 \times 10^6$ psi
Concrete Poisson's ratio,	$\nu_c = .22$
Concrete coefficient of thermal expansion,	$\alpha_c = 6 \times 10^{-6}$ in/in/°F
Temperature difference,	$\Delta T = 100°F$
Rebar coefficient of thermal expansion,	$\alpha_R = \alpha_c$

Three sample problems are presented as examples of verification.

Sample Problem: Beam With a Thermal Moment

The analysis of a reinforced concrete beam subjected to a linear thermal gradient was performed to test the redistribution of thermal stresses due to the relieving effect of concrete cracking. The results were compared with hand calculations.

Figure 3.8A-10 shows the reinforced concrete beam and the corresponding CECAP concrete element used in the analysis. Boundary conditions, geometry, and applied loads are illustrated.

The following illustrates how thermal loads are treated in a cracked section analysis of a reinforced concrete beam. The main assumptions pertaining to thermal boundary conditions are:

(1) The beam is allowed to expand freely axially.

(2) There is no rotation of the initial thermal stress slope.

The beam cross-section and initial thermal stress distribution are shown in Figure 3.8A-11. For $\Delta T = 100°F$, the equivalent thermal moment and concrete and rebar stresses are:

$$M = \Delta T\, \alpha_c\, E_c b t^2/12 = (100)(6\times10^{-6})(3\times10^{6})(12)(42)^2/12 = 3{,}175{,}000 \text{ in-lbs}$$

$$\sigma_c = \Delta T\, \alpha_c\, E_c/2 = (100)(6\times10^{-6})(3\times10^{6})/2 = 900 \text{ psi (compression)}$$

$$\sigma_c' = \frac{(t/2-2)}{t/2}\,\sigma_c = \frac{(21-2)}{21}\,900 = 814 \text{ psi (tension)}$$

The stress diagram used for the cracked section analysis with thermal loading is shown in Figure 3.8A-12. The assumptions of free movement axially and constant thermal stress slope are maintained by a lateral translation of the initial reference axis to a final cracked position.

From force equilibrium: $F_{rebar} + F_{concrete} = 0$

$$\underbrace{1.0\,(814 + \Delta\sigma_c)\,10}_{F_{rebar}} - \underbrace{900\left(\frac{42}{2}\right)\left(\frac{12}{2}\right) + \frac{\Delta\sigma_c(12)}{2}\left[21 + \left(\frac{900-\Delta\sigma_c}{900}\right)21\right]}_{F_{concrete}} = 0$$

Solving for $\Delta\sigma_c$,

$\Delta\sigma_c = 582$ psi

Rebar and concrete stresses are:

$f_s = (814+582)10 = 13{,}970$ psi (Tension)

$f_c = 900-582 = 318$ psi (Compression)

Location of cracked neutral axis is:

$$kd = x = \left(\frac{900-582}{900}\right) 21 = 7.42 \text{ in}$$

Self-relieved thermal moment is:

$$M_T = \frac{f_s A_s (d - \frac{X}{3})}{12} = \frac{13970(1)(40-2.47)}{12} = 43{,}690 \frac{\text{in-lb}}{\text{in}}$$

The rebar and concrete stresses, self-relieved thermal moment and neutral axis location obtained from the CECAP program are compared with the hand calculations in Table 3.8A-2. It can be seen that the CECAP results compare favorably with the hand calculations.

Sample Problem: Beam With a Real Moment

The analysis of a reinforced concrete beam subjected to a real moment was performed to test the CECAP program for non-thermal moments. The results were compared with hand calculations.

Figure 3.8A-13 shows the loading and geometry for the reinforced concrete beam and the corresponding CECAP concrete element model.

The following illustrates the working stress analysis of reinforced concrete beams. The beam cross-section, stress block, and transformed sections are shown in Figure 3.8A-14. The resultant forces and moment are:

$$C = f_c \ (kd)(b)/2$$

$$T = A_s f_s$$

$$M = Cjd = Tjd$$

Equating the first moments of the compression and tension areas about the neutral axis of the transformed section,

$$kd(b)\left(\frac{kd}{2}\right) = nA \ (d - kd)$$

which yields

$$kd^2 + 1.67kd - 66.67 = 0$$

Solving for kd;

kd = 7.37 in.

The resultant forces are:

$$C = T = \frac{M}{jd} = \frac{3,175,000}{(40 - \frac{7.37}{3})}$$

C = T = 84,570 lb.

Rebar and concrete stresses are:

$$f_s = \frac{T}{A_s} = 84,574 \text{ psi (tension)}$$

$$f_c = \frac{2C}{kdb} = \frac{2(84,574)}{(7.37)(12)} = 1,193 \text{ psi (compression)}$$

Table 3.8A-3 shows a comparison of rebar and concrete stresses and neutral axis locations obtained from the CECAP program and hand calculations. The CECAP results are shown to compare to hand calculations within the force accuracy limits in the program.

Sample Problem: Beam with a Real Moment and a Real Axial Load

This verification problem involves the analysis of a reinforced concrete beam subjected to both a real moment and a real axial compressive load. A hand calculation solution using the equations presented in Reference 3.8A-4 was obtained and compared with the CECAP results.

The loading and geometry for the reinforced concrete beam and corresponding CECAP model are illustrated in Figure 3.8A-15.

The following illustrates the working stress analysis of reinforced concrete beams subjected to both moments and axial compressive loads. The beam cross-section and stress block are shown in Figure 3.8A-16. The analysis uses the equations presented in Reference 3.8A-4, which are simplified to the following:

(1) $$(kd)^3 + 3\left(\frac{M}{N} - \frac{t}{2}\right)(kd)^2 + \frac{6nA_s}{b}\left(d - \frac{t}{2} + \frac{M}{N}\right)(kd) - \frac{6nA_s d}{b}\left(d - \frac{t}{2} + \frac{M}{N}\right) = 0$$

(2) $$f_s = \frac{N}{A_s} \quad \frac{\left(\frac{M}{N} + \frac{kd}{3} - \frac{t}{2}\right)}{\left(d - \frac{kd}{3}\right)}$$

(3) $$f_c = \frac{f_s kd}{n(d-kd)} \quad \text{for } \frac{M}{N} \geq t/6$$

Equation (1) becomes:

$$kd^3 + 55.8kd^2 - 293kd = 11720 = 0$$

$$M/N = \frac{3175000}{101000} = 31.4 \geq t/6 = \frac{42}{6} = 7$$

Solving the above equations by iteration for kd yields:

kd = 12.7 in.

The resulting rebar and steel stresses are:

$$f_s = \frac{101000}{1.0} \quad \frac{(31.4 + 12.7/3 - 21)}{(40 - 12.7/3)} = 41{,}320 \text{ psi (Tension)}$$

$$f_c = \frac{41320\ (12.7)}{10\ (40-12.7)} = 1{,}922 \text{ psi (Compression)}$$

The rebar and concrete stresses and neutral axis location obtained from the CECAP program are compared with the hand calculations in Table 3.8A-4. The results for the two solution methods agree very closely.

3.8A.4 CE 668

This program performs the linear elastic analysis of a plate with arbitrary shape and supports, stiffener beams, and elastic subgrade, under loads normal to the middle plane of the plate.

This program was verified by comparing selected hand calculated values to CE 668 values with the deflections and moments of a rectangular plate for different loading and support conditions.

Sample Problem: Rectangular Plate with a Concentrated Load at the Center

The simply supported rectangular plate, shown in Figure 3.8A-17 was subjected to a concentrated load of 300 lbs. at the center. Because of symmetry only half of the plate was modelled by the finite elements. The boundary conditions were zero displacement with free normal rotation at the simply supported edges and free displacement with zero normal rotation at the symmetry axis. The plate had isotropic structural properties.

The problem parameters are as follows:

Poisson's Ratio	$\nu = 0.3$
Young's Modulus	$E = 2.9 \times 10^7$ psi
Thickness	$h = 0.5$ in.
Concentrated Load	$P = 300$ lb.

The formulae for the deflections and moments were taken from Reference 3.8A-5.

a) Deflection

$$@ \text{ center } \omega = .01695 \frac{Pa^2}{D} = .01695 \left[\frac{300\ (100)\ \ 12\ (1-(.3)^2)}{(2.9 \times 10^7)\ (.5)^3} \right]$$

$$\omega = .00153 \text{ in. @ Node 116}$$

b) Moments

M_X : (for b >> a)

$$@\ x = 2, y = 0 \qquad M_X = \frac{-P(1+\nu)}{8\pi} \ln \left[\frac{1 - \sin \frac{\pi X}{a}}{1 + \sin \frac{\pi X}{a}} \right]$$

$$M_X = \frac{-300(1.3)}{8\pi} \ln \left[\frac{1 - \sin \frac{\pi}{5}}{1 + \sin \frac{\pi}{5}} \right] = (-15.52)\ (-1.348)$$

$$M_X = 20.92 \text{ lb - in @ Node 113}$$

My : (for b >> a)

$$@X = 6, y + 0 \qquad My = \frac{-P(1+\nu)}{8\pi} \ln \left[\frac{1 - \sin \frac{\pi X}{a}}{1 + \sin \frac{\pi X}{a}} \right]$$

$$= \frac{-300(1.3)}{8\pi} \ln \left[\frac{1 - \sin \frac{3\pi}{5}}{1 + \sin \frac{3\pi}{5}} \right]$$

$$= (-15.52)\ (-3.685)$$

$$My = 57.198 \text{ lb-in @ Node 117}$$

The hand calculated values for deflections and moments are compared with the CE 668 values in Table 3.8A-5. The results are very close with the greatest difference being 1.55%.

Sample Problem: Uniform Load on a Rectangular Plate With Various Edge Conditions

The rectangular plate had one edge fixed, one edge free, and two edges simply supported as shown in Figure 3.8A-18. It was subjected to a uniformly distributed load of intensity q = 2.0 psi. Because of symmetry only half of the plate was modelled by finite elements. Boundary conditions were specified according to the appropriate edge support conditions.

The problem parameters are as follows:

Poisson's Ratio	$\nu = 0.3$
Young's Modulus	$E = 2.9 \times 10^7$ psi
Thickness	$h = 0.2$ in.
Load Intensity	$q = 2.0$ psi

The formulae used to calculate the deflections and moments were taken from Reference 3.8A-5.

a) Deflection

$$@\ \chi = 15,\ y = 15\quad \omega = .0582\left(\frac{qb^4}{D}\right) = .0582\left[\frac{2(15)^4(12)(1-(.3)^2)}{(2.9 \times 10^7)\ (.2)^3}\right]$$

$$\omega = .277 \text{ in. @ Node 11}$$

b) Moments

$M\chi$:

$@\ \chi = 15,\ y = 15 \qquad M\chi = .0293\ qa^2 = .0293\ (2)\ (30)^2$

$M\chi = 52.74$ in-lbs @ Node 11

My:

$@\chi = 15,\ y = 0 \qquad My = .319\ qb^2 = .319\ (2)(15)^2$

$My = 143.55$ in-lbs. @ Node 121

The hand calculated values for the deflection and moments are compared to the CE 668 results in Table 3.8A-6. The results agree closely, with the largest difference being 3.4%.

3.8A.5 EASE

EASE (Elastic Analysis for Structural Engineering) performs static analysis of two- and three-dimensional trusses and frames, plane elastic bodies and plate and shell structures. The finite

element approach is used with standard linear or beam elements, a plane stress triangular element or a triangular plate bending element. The EASE program accepts thermal loads as well as pressure, gravity, or concentrated loads.

The program output includes joint displacements, beam forces and triangular element stresses and moments.

EASE was developed by the Engineering Analysis Corporation, Redondo Beach, California, in 1969 and is in the public domain. The version currently used by Bechtel is maintained by the Control Data Corporation, Cybernet Service.

3.8A.6 E0119

This program performs an analysis of a bolted flange. Flange dimensions reflect the corroded condition. Symbols, terms, and mathematics are in accordance with Appendix XI of the ASME Code Section III. Stress values for both design (operating) and bolt-up conditions are printed. Both allowable and actual stresses are printed out for bolts, longitudinal flange stress, radial flange stress, and tangential flange stress. The shape constants and moments are printed out for information only.

Two program solutions are included in verifying Program E0119. A welding neck flange design and a slip-on flange design have been prepared. Also attached are solutions of the same problems as published in Bulletin 502, Modern Flange Design from Gulf & Western Manufacturing Company (Reference 3.8A-6).

The problem parameters for the two sample problems are as follows:

Design pressure = 400 psi
Design temperature = 500°F
Atmospheric temperature = 75°F
Poisson's ratio = 0.30
Corrosion allowance = 0
Gasket width = 0.75"
Effective gasket width = 0.306"
Gasket Factor = 2.75
Gasket seating strength = 3700 psi

Sample Problem: Welding Neck Flange

Figure 3.8A-19 shows the dimensions of the welding neck flange. Table 3.8A-7 compares the results of E0119 computer program with those published in Reference 3.8A-6. The results compare very closely.

Sample Problem: Slip-on Flange

Figure 3.8A-20 shows the dimensions of the slip-on flange. Table 3.8A-8 compares the results of E0119 computer program with those published in Reference 3.8A-6. The results compare very closely.

3.8A.7 E0781

The Shells of Revolution Program was developed by Aerturs Kalnin while at Yale University. The Mathematics are based on a method of analysis contained in his paper "Analysis of Shells of Revolution Subjected to Symmetrical and Non-Symmetrical Loads" published in the Journal of Applied Mechanics, Vol. 31, September, 1964 (Reference 3.8A-7).

This program calculates the stresses and displacements in thin walled elastic shells of revolution when subjected to static edge, surface, and/or temperature loads with arbitrary distribution over the surface of the shell. The Geometry of the shell must be symmetric, but the shape of the median is arbitrary. It is possible to include up to three branch shells with the main shell in a single model. In addition, the shell wall may consist of different orthotropic materials, and the thickness of each layer and the elastic properties of each layer may vary along with the median.

Program E0781 numerically integrates the eight ordinary first order differential equations of thin shell theory derived by H. Reissner. The equations are derived such that the eight variables are chosen which appear on the boundaries of the axially symmetric shell so that the entire problem can be expressed in these fundamental variables.

Kalnin's program has been altered such that a 4 x 4 force-displacement relation can be used as a boundary condition as an alternative to the usual procedure of specifying forces or displacements. This force-displacement relation can be used to describe the forces at the boundary in terms of displacements at the boundary, or the displacements at the boundary in terms of forces or some compatible combination of the two. In this manner, it is possible to study the behavior of a large complex structure. It is also possible to introduce a "Spring Matrix" at the end of any part of the stress model. This matrix must be expressed in the form, Force = Spring Matrix X Displacement. In addition, to the above changes, the Kalnin's Program has been modified to increase the size of the problem that can be considered and to improve the accuracy of the solution.

This program was verified by comparing the computer results with experimental measurements and published references. Two sample problems are presented as examples of verification.

Sample Problem: Comparison of 2:1 Ellipsoidal and Torispherical Heads Subjected to an Internal Pressure Load

This problem illustrates Program E0781's ability to generate cylindrical, torispherical, and ellipsoidal shapes.

A comparison is made to an experimental investigation of 2:1 ellipsoidal heads subjected to internal pressure (see Reference 3.8A-8).

The problem consists of comparing a 2:1 ellipsoidal head to an equivalent torispherical head subjected to the same uniformly distributed internal pressure. An equivalent torisphere will be defined as one having the same height above the tangent line as the ellipsoid and a minimal L/b ratio (thus having the least possible discontinuity between the torus and the sphere). For the geometry shown in Figure 3.8A-21:

$$(L-b)\ \sin\phi_o = A-r \qquad (1)$$

$$(L-b)\ \cos\phi_o = L-B \qquad (2)$$

Minimizing L/b using (1) and (2):

$$\tan\phi_o = B/A = 0.5019$$

$$\phi_o = 26.653^\circ$$

$$L/A = \frac{C \pm \sqrt{C^2-2C}}{2}$$

$$C = B/A + A/B = 2.494$$

$$L = \frac{18.19}{2}\ [\ 2.5 + \sqrt{6.22 - 4.99}\ ] = 32.778''$$

$$b = B\ [\ B/A - L/A\ ] + A = 9.13\ [\ .5019 - 1.80198\ ] + 18.19 = 6.32''$$

Note: For purpose of calculation

$A = 18.19''$

$B = 9.13''$ from Figure 3.8A-21

Segment lengths used are:

cylinder - $\sqrt{rt} = \sqrt{18.16\ (0.31)} = 2.37$

torisphere

5° to 10° - 4 @ 1.25°

10° to 26.567° - 4 @ 4.13°

26.567° to 90° - 6 @ 10.57°

ellipsoid

5° to 10° - 4 @ 1.25°

10° to 30° - 4 @ 5°

30° to 90° - 6 @ 10°

Boundary Conditions:

It will be assumed that at 5° from the pole a membrane state of stress exists in both the ellipsoid and the torisphere:

$Q = M\phi = 0$

$N\phi = \frac{pr}{2 \sin\phi}$

where r = distance to pole = 32.778"

Q = tranverse shear in ϕ direction.

$M\phi$ = moment resultant in ϕ direction.

$N\phi$ = membrane force in ϕ direction.

Letting p = 680 psi

Then for the torisphere:

$N\phi = (680/2)(32.778) = 11,144.5$ lb/in.

If $N\phi = 11,144.5$ lb/in., a preliminary run yields Q = 95.202 lb/in., so a new value for $N\phi$ for the torisphere was calculated: $\Delta N = \frac{\Delta Q}{\tan\phi}$

$N\phi = 11,144.5 + \Delta N = 10056.3$ lb/in. and an appropriate membrane state was generated.

For the ellipsoid

$$r = \frac{A \sin \phi}{R}$$

where $R = \sqrt{C_1 + (1-C_1) \sin^2 \phi}$

$$C_1 = (B/A)^2 = 0.2519$$

$$R = \sqrt{.2519 + .7481\ (0.0871557)^2} = 0.5075$$

$$N\phi = \frac{A \sin\phi}{R} \frac{P}{2 \sin\phi} = \frac{18.19\ (680)}{2\ (0.5075)} = 12{,}185.78 \text{ lb/in.}$$

To better compare the heads it seemed desirable to have the longitudinal displacement at the center of the cylinder 0 ($\mu_\phi = 0$). So the problem was run twice, the first run yielding the radial displacement, W required for 0 displacement at the center (W = 0.0966")

1. Start W = 0.0966" $N\phi$ = 10,056 lb/in $M\phi$ = N = 0

2. End Q = N = $M\phi$ = 0 $N\phi$ = 12,186 lb/in

Figure 3.8A-24 shows the analytical model with boundary conditions.

Results

To check the results, first the answers at the boundaries should be examined. It was assumed that there was a membrane state of stress at the boundaries and, therefore, at the edges Q and M must be approximately 0.

	Q (lbs/in)	M ϕ (in. - lbs/in.)
Start	- 0.01027	0.0
End	- 0.0008613	-0.0001487

Also to satisfy equilibrium in the cylinder, $N\phi \simeq 0.5pr$ = 6169 lb/in.

Plots of the hoop force and longitudinal bending from E0781 results compare the ellipsoidal and torispherical heads. Even though the change in radii has been minimized the disturbance at the junction of the sphere and torus is considerable (see Figure 3.8A-25.

Comparison to the experimental ellipsoidal head shows good correlation of stress values. See Figures 3.8A-26 through 3.8A-30 for plots of $\nabla\phi$ and $\nabla\theta$ on the inside, outside, and meridian of the head. Deviations are caused by the changes in thickness and the experimental head's variation from a true 2:1 ellipsoidal head.

Sample Problem: Cylindrical Water Tank with Tapered Walls

This problem illustrates Program E0781's capability to analyze a pressure load with one fixed boundary condition and one free boundary condition.

The problem used for this verification is "Shell of Variable Thickness" taken from "Stresses in Shells", by W. Flugge, pp. 289-295 (Reference 3.8A-9)

The problem consists of a tapered shell filled with water. The shell has a radius of 9'-0" and is 12'-0" high. The shell thickness varies from 11" at the bottom to 3" atthe top. See Figure 3.8A-31 for location of the Z axis. The length of a segment is 18" ($\sqrt{rt}$) .

Taking the weight of water as 62.5 lb/ft³, the pressure at the bottom of the tank is

$$p = \frac{(12\ \text{ft})\ (62.5\ \text{lb/ft}^3)}{144\ \text{in.}^2\ /\text{ft}^2} = 5.2083\ \text{psi}$$

The pressure at the top is zero. The pressure varies linearly so that only two points are needed in the function generator in order to fully describe the function.

Boundary Conditions

W - displacement normal to surface

$U\phi$ - displacement component in ϕ direction

$B\phi$ - rotation of reference surface in ϕ direction

Q - transverse shear in ϕ direction

$N\phi$ - membrane force in ϕ direction

$M\phi$ - moment resultant in ϕ direction

1. fixed at start $W = U\phi = B\phi = 0$

2. free at end $Q = N\phi = M\phi = 0$

Results

Table 3.8A-9 lists the Program E0781 results and compares them with the theoretical solutions from Reference 3.8A-9at two locations.

Program E0781 gives a maximum hoop force, $N\theta$ = 346.8 lb/in. = 4160 lb/ft at 54" from the base. This value differs from the theoretical solution of 4180 lb/ft by 0.48%.

Program E0781 gives a maximum moment of the base, $M\phi$ = -1539 in-lb/in. = -1539 ft-lb/ft. This value differs from the theoretical solution of -1470 ft-lb/ft by 4.69%.

3.8A.8 FINEL

This program performs the static analysis of stresses and strains in plane and axisymmetric structures by the finite element method. In this method, the structure is idealized as an assemblage of two-dimensional finite elements of triangular or quadrilateral shapes having arbitrary material properties. Reinforcement of concrete materials is included by adjusting the element material properties. Special emphasis is made on bilinearity in compression and bilinearity or cracking in tension. FINEL computes the displacements of the corners of each element and the stresses and strains within each element.

To verify this program, example problems were analyzed by FINEL and compared to experimental and/or hand calculated solutions. Three sample problems are presented as examples of verification.

Sample Problem: Simply Supported Beam with a Concentrated Load at the Center

The beam shown in Figure 3.8A-32 has been the subject of an experimental and analytical investigation. The purpose of this investigation is to compare results obtained from the FINEL program with those obtained from References 3.8A-13 and 3.8A-14.

The finite element mesh used in Reference 3.8A-14 and in the FINEL analysis are shown in Figures 3.8A-33 and 3.8A-34, respectively. The FINEL analysis required a finer mesh because it used linear displacement elements while Reference 3.8A-14 used quadratic displacement elements.

The material properties of the concrete and reinforcing steel, and the loading history used in the FINEL analysis are given in Tables 3.8A-10 and 3.8A-11, respectively.

This problem was not continued beyond the yield point of the reinforcing steel due to an error in the FINEL program. The stiffness of an element which yielded should have been determined according to:

$$E_{eff} = \frac{[Ty + n\,(T - Ty)]\,E_o}{T}$$

where,

E_o = initial material stiffness or modulus

T_y = yield stress

T = element stress, in yield direction, at end of previous cycle (<Ty)

n = E_{plast}/E_o ; E_{plast} = plastic stiffness

E_{eff} = effective stiffness, in yield direction, to use in next cycle

A new E_{eff} should be calculated after each cycle. The FINEL program calculated an E_{eff} only after the first cycle following yielding, (or first cycle in a restart run), and used the value of E_{eff} for all subsequent cycles in the same computer run. (This error could be overcome by making a series of one cycle restart runs.)

The cracking patterns obtained from Reference 3.8A-14 and FINEL are shown in Figure 3.8A-35. The load-deflection curves from References 3.8A-13 and 3.8A-14 and the FINEL analysis are shown in Figure 3.8A-36. The load deflection curve obtained from the FINEL analysis shows very good agreement with the experimental results. The cracked region grows faster in the FINEL analysis and more slowly in Reference 3.8A-14, since the FINEL and Reference 3.8A-14 load-deflection curves show different gradients (stiffnesses).

The results of analytical, experimental, and FINEL solutions are shown in Figure 3.8A-36. The FINEL analysis agrees well with the experimental results up to the point where the reinforcing steel in the beam yields. After the yield point, the FINEL analysis incorrectly calculated the effective stiffness of elements which have yielded. Therefore, the solution was not valid for further loadings. However, since all reinforcing steel remains elastic for the containment analysis, the FINEL program is verified and restricted for that application.

<u>Sample Problem: Axially Constrained Hollow Cylinder With a Distributed Pressure Loading</u>

This verification involves the response of an axially constrained hollow cylinder to internal pressure. A hand calculated solution yields values of tangential, axial, and radial stresses at various radii from the center of the cylinder, which are then compared to the FINEL values.

The finite element model is illustrated in Figure 3.8A-37. Nodal points are free to move only in the radial direction, representing the conditions of axisymmetry and plane strain.

The problem parameters are as follows:

Poisson's Ratio	$\nu = 0.25$
Young's Modulus	$E = 4.32 \times 10^5$ ksf
Number of nodal points	= 22
Number of elements	= 10
Internal Pressure	$P = 1.0$ ksf

From Reference 3.8A-15, the following equations were used:

hoop or tangential stress, T_θ :

$$T_\theta = p \frac{a^2 (b^2 + r^2)}{r^2 (b^2 - a^2)}$$

axial stress, T_z :

$$T_z = \frac{p}{z} \quad \frac{a^2}{b^2 - a^2}$$

radial stress, T_R:

$$T_R = p \frac{a^2(b^2 - r^2)}{r^2(b^2 - a^2)}$$

where $a = 65.0$ ft.
$b = 68.75$ ft.
$p = 1.0$ ksf
$a \leq r \leq b$

The results from FINEL for tangential, axial, and radial stresses of the hollow cylinder are compared with the hand calculated values in Table 3.8A-12. The results are exactly the same except for one value where there is only 4.17% difference.

<u>Sample Problem: Axially Constrained Hollow Cylinder With A Linear Temperature Gradient</u>

The response of an axially constrained hollow cylinder to a radially varying linear temperature gradient was the problem used for this verification. The tangential, axial, and radial stresses were determined by hand calculations and compared to the FINEL results.

Figure 3.8A-38 illustrates the finite element mesh. The conditions of axisymmetry and plane strain were imposed by using the axisymmetric quadrilateral element and restraining all nodes against axial displacement.

The temperature profile is shown in Figure 3.8A-39.

The problem parameters are as follows:

Poisson's Ratio	$\nu = 0.25$
Young's Modulus	$E = 4.32 \times 10^5$ ksf
Coefficient of Thermal Expansion	$\alpha = 6 \times 10^{-6}$ ft/ft/°F
Number of nodal points	= 22
Number of elements	= 10

From References 3.8A-16 and 3.8A-17, the following equations were used:

hoop or tangential stress, δ_θ :

$$\delta_\theta = \frac{\alpha E}{1-\nu} \frac{1}{r^2} \left[\left(\frac{r^2 + a^2}{b^2 - a^2} \right) \int_a^b Tr\,dr + \int_a^r Tr'\,dr' - Tr^2 \right]$$

axial stress, δz:

$$\delta_z = \frac{\alpha E}{1-\nu} \left[\frac{2\nu}{b^2 - a^2} \int_a^b Tr\,dr - T \right]$$

radial stress, δ_r :

$$\delta_r = \frac{\alpha E}{1-\nu} \frac{1}{r^2} \left[\left(\frac{r^2 - a^2}{b^2 - a^2} \right) \int_b^a Tr\,dr - \int_a^r Tr'\,dr' \right]$$

where: $a = 65.0$ ft.
$b = 68.75$ ft.
$T = T(r)$ = temperature above reference ($T_{REF} = 100°F$)

Expression for the temperature field:

$$T(r) = C_2 r + C_1$$

$$T(a) = 25 = C_1 + 65.0C_2$$

$$T(b) = -25 = C_1 + 68.75C_2$$

solving,

$$C_2 = \frac{-50}{68.75-65} = -13.33$$

$$C_1 = -25 - 68.75(-13.33) = 891.67$$

then

$$T(r) = -13.33r + 891.67$$

Evaluation of the integral:

$$\int Trdr = \int(-13.33r + 891.67)\, rdr$$

$$= \frac{-13.33r^3}{3} + \frac{891.67r^2}{2} + C$$

$$= -4.44r^3 + 445.83r^2 + C$$

$$\int_a^b Trdr = -4.44(b^3-a^3) + 445.83(b^2-a^2)$$

$$\int_a^r Tr'dr' = -4.44(r^3-a^3) + 445.83(r^2-a^2)$$ 4

The results from FINEL for the tangential, axial, and radial stresses are compared with the values obtained by hand calculations in Table 3.8A-13. The results between the two methods of solution agree very closely.

3.8A.9 ME 620

The heat conduction program, ME 620, is used to determine the temperature distribution, as a function of time, within a plane or axisymmetric solid body subjected to step-function temperature or heat flux inputs. The program is also used for steady-state temperature analysis.

The program utilizes a finite element technique coupled with a step-by-step time integration procedure as described in "Application of the Finite Method to Heat Conduction Analysis" by E. L. Wilson and R. E. Nickell (Reference 3.8A-18).

The program was developed at the University of California, Berkeley, by Professor E. L. Wilson and subsequently modified by Bechtel Corporation to incorporate the save and restart capabilities.

To verify this program, example problems were analyzed by ME 620 and compared with program data. Two sample problems are presented as examples of verification.

Sample Problem: Heat Conduction in a Square Plate With One Edge Quenched

This problem tested the ability of the program to solve the temperature changes in a plane region subjected to conduction boundary conditions. The plate was brought to an equilibrium temperature and one edge was quenched while the other three edges were kept insulated.

A square plate was brought to equilibrium at a given initial temperature, T_o. Three edges were perfectly insulated while a third edge was suddenly brought to a lower temperature, T_1. This quench was kept constant for the entire analysis. A temperature time history was then obtained for the corner farthest from the quenched edge.

Figure 3.8A-40 shows the actual plate arrangement, while Figure 3.8A-41 shows a diagram of the finite elements.

The problem parameters are as follows:

Nomenclature

L = length of longest heat flow path

T_o = initial temperature of slab (°F)

T_1 = quenching temperature of edge (°F)

Data:

The plate was 10" x 10" square.

T_o = 100°F

T_1 = 0°F

Diffusivity α = 1.0 in²/sec (chosen for convenience)

Time increment ΔT = 1 second for numerical solution

At any time t during the transient state, the time factor T (or characteristic time) is given by $T = \alpha t/L^2$. The time to reach

steady-state is given when T = 1.0, hence the transient time is t = L^2/α = 100 seconds. The results derived from Reference 3.8A-19 are plotted in Figure 3.8A-42.

The temperature variation at point A was plotted in Figure 3.8A-42 according to the results of ME 620 and compared with the theoretical transient change. The curves are seen to agree quite well. Deviations are due to the selected finite element mesh size and to the selected time step for the analysis.

Sample Problem: Heat Conduction in a Surface Quenched Sphere

This problem tested the ability of ME 620 to analyze the temperature distribution in an axisymmetric solid with given temperature boundary conditions. The results of the program analysis were compared to a closed-form solution derived from Reference 3.8A-20.

This problem considered a solid steel sphere (shown in Figure 3.8A-43) that was brought to an equilibrium temperature, and then its surface was suddenly quenched to a lower uniform temperature. The quenching environment was held at a constant temperature. A temperature-time history for three seconds was obtained from the program for all node points. The points used for the comparison were at a radius of 0.2 inches, and only one time period was checked. The finite element model is shown in Figure 3.8A-44.

The problem parameters are as follows:

Nomenclature:

L = length of the longest heat flow path (radius of sphere)

T_o = initial temperature of sphere (°F)

T_1 = quenching temperature of outer surface (°F).

Data

Radius of sphere = R = .59 in.

T_o = 1472 °F

T_1 = 68 °F

Conductivity = 6.02×10^{-4} Btu/in-sec-°F)

Diffusivity = α = .0193 in^2/sec

Specific heat = .11 Btu/(lb-°F)

Density = ρ = .284 lb/in^3

Time increment = .2 sec

At any time, t, during the transient state, the time factor T (or characteristic time) is given by $T = \alpha t/L^2$. The time to reach steady-state is given when T = 1.0, hence the transient time is $t = L^2/\alpha = 3.0$ seconds. The result from Reference 20 for the temperature at a radius of 0.2 inches at time t = 1.8 seconds; was 933.8°F.

The temperatures from both the program and the reference are shown in Table 3.8A-14. There is an error of 1.1%.

3.8A.10 SUPERB

SUPERB is a general-purpose, isoparametric, finite element computer program. The program determines the displacement and stress characteristics of complex structures subjected to concentrated loads, pressure distributions, enforced displacements, and thermal gradients, as well as the temperature distribution due to steady-state heat transfer. Isoparametric elements with curved boundaries and high-order strain variations permit curved regions and area with high stress concentrations to be accurately represented with a minimum number of elements.

The SUPERB program is a recognized program in the public domain and has had sufficient history of use to justify its application and validity without further demonstration. The version of the program currently used by Bechtel is maintained by the Control Data Corporation, Cybernet Service.

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APPENDIX 3.9A

COMPUTER PROGRAMS

3.9A.1 ME101

Program Description

ME101 is a finite element computer program which performs linear elastic analysis of piping systems using standard beam theory techniques. The input data format is specifically designed for pipe stress engineering, and the English system of units is used. A thorough checking of the input has been co-ordinated in the program. In addition, modifications aimed at achieving an improved model are performed automatically.

The output may be used directly for piping design and for conformation to code and other regulatory requirements. Two piping codes, ASME BPV code 1974 and B31.1 Summer 1973 addenda are incorporated in the program to the extent of computing flexibility factors, stress intensification factors and stresses.

ME101 may be used for static and seismic load analysis of piping systems and also performs effective weight calculations.

Static analysis considers one or more of the following: thermal expansion, dead weight, uniformly distributed loads, and externally applied forces, moments, displacements and rotations, or individual force loads.

Seismic analysis is based on standard normal mode techniques and uses response spectrum data. Two methods of eigenvalue solution are available. Determinant Search or Subspace Iteration considers all data points as mass points. Kinematic Reduction and Householder QR considers masses only at specified data points in designated directions. Differential seismic anchor movement analysis and static seismic analysis are also provided.

ME101 generates isometric plots of the piping configuration with optional node numbering. The plots are obtained by either ZETA or CALCOMP 1036 plotter.

The program uses out-of-core solution techniques for both static and dynamic analysis, and has no practical limitations to the number of equations or band width. However, very large systems may become prohibitive due to cost of computation. The maximum number of mode shapes allowable is currently 125.

Program Version and Computer

The current UNIVAC version (C3) of ME101 is being used by Bechtel Power Corporation.

Extent of Application

ME101 is a piping program developed by Bechtel Power Corporation (BPC). Its development began in July 1975 and is being continuously supported by BPC. It has been used by various projects in the BPC.

Test Problems

The ASME Benchmark Problem No. 1 demonstrates the solution for natural frequencies of a three dimensional structure as described in Reference 3.9A-1.

The following table lists the natural frequencies from ME101 and Reference 3.9A-1:

Natural Frequency Comparisions, CPS.

Mode No.	Reference 3.9A-1	ME 101
1	110	112
2	117	116
3	134	138

Additional test problems can be found in Reference 3.9A-2.

3.9A.2 ME632

Program Description

ME632 performs stress analysis of 3-dimensional piping systems. The effects of thermal expansion, uniform load of the pipe, pipe contents and insulation, concentrated loads, movements of the piping system supports, and other external loads, such as wind and snow, may be considered. The input data format is specifically designed for pipe stress engineering, and the English system of units is used. A thorough checking of the input has been coordinated in the program.

The output may be used directly for piping design and for conformation to code and other regulatory requirements. Piping codes, ASME BPV code, B31.1 code and B31.3 code have been incorporated in the program to the extent of computing flexibility factors, stress intensification factors and stresses.

A response spectrum analysis may be performed to analyze the effect of earthquake forces on the piping system, and transient effects of water hammer, steam hammer, or other impulsive type dynamic loading are also handled by the program. Also, a plot of piping geometry and/or response spectrum curves may be obtained to verify the accuracy of the model.

Program Version and Computer

The current UNIVAC version (B9) of ME632 is being used by Bechtel Power Corporation.

Extent of Application

ME632 is a piping program developed by Bechtel Power Corporation (BPC). Its development began in 1970 and is being continuously supported by BPC. It has been used by various projects in the BPC.

Test Problems

The ASME Benchmark Problem No. 1 demonstrates the solution for natural frequencies of a three-dimensional strucure as described in Reference 3.9A-1.

The following table lists the natural frequencies for ME632 and Reference 3.9A-1:

Natural Frequency Comparison, CPS

Mode No.	Reference 3.9A-1	ME632
1	110	111
2	117	116
3	134	137

Additional test problems can be found in Reference 3.9A-3.

3.9A.3 ME912

Program Description.

Finite-difference representation of the heat diffusion equation is used for the pipes or component wall section in contact with fluid of specified temperature and flow rate time histories. The program is quasi-two-dimensional, so that reduction of severity of a given transient with distance from inlet is accounted for.

Thermal properties of water, liquid sodium, stainless and carbon steel are built in the program. Film transfer coefficients for water or liquid sodium are computed by the program for each time step and pipe section. For other fluids such as steam, the program is used on a one-dimensional basis with user supplied film coefficients. Sequential computations are done for pipe lengths of different diameters or wall thicknesses. Fluid outlet temperature data from one pipe length are stored for use as inlet to the next length downstream. Average temperature differences $T_a - T_b$ are thus calculated for structural discontinuity.

Program Version and Computer

The ME912 program has been used by Bechtel Power Corporation in Gaithersburg, and San Francisco offices on various BPC projects. A Univac 1110 computer is used to run the ME912 program.

Extent of Application

The ME912 program was developed from References 3.9A-4, 3.9A-5 and 3.9A-6 by the Stress Group of Gaithersburg and San Francisco offices of BAC. The ME912 program has been extensively used since 1975 for nuclear Class I component design on FFTF project.

Test Problem

For local gradients, the program has been compared with analytical flat plate data of Ref. 3.9A-5 and numerical results by in-house program ME643, Ref. 3.9A-7. The results were acceptable. For axial variations of fluid and wall temperatures, the program agrees closely with the analytical solution of Ref. 3.9A-6. Table 3.9A-2 shows the comparison of ME912 with ME 643 and analytical results.

The ME643 program was developed from References 3.9A-11 and 3.9A-12 by the Stress Group of Los Angeles Power Division of BPC. The results of ME 643 transient temperature responses on both inside and outside surfaces are compared with Chart 36 of Reference 3.9A-13 and plotted Figure 3.9A-1.

3.9A.4 ME 913

Program Description

ME913 can determine stress intensity levels for Class 1 nuclear power piping components for Equations 9 through 14 of subarticle NB-3650, ANALYSIS OF PIPING COMPONENTS of Section III, ASME Boiler and Pressure Vessel Code. Before attempting to exercise

this program, the user shoudl be familiar with the requirements and procedures set forth in subarticle NB-3650.

Prior to using this program, the user should have the following information external to the program.

1. Piping configuration

2. Piping and piping component properties

3. Moment reactions due to

   a. Thermal expansion loads

   b. Weight looads and

   c. Earthquake loads

4. The thermal response of the piping system due to the specified transients:

   ΔT_1, ΔT_2, and the $(T_a - T_b)$ values for the key points during system life.

Program Version and Computer

The current ME913 version is being used by Bechtel Power Corporation in its Gaithersburg, Los Angeles, Ann Arbor and San Francisco offices. A Univac 1100 computer is used to run the ME913 program.

Extent of Application

ME913 is the revised and expanded version of the LOTEMP program which was originally developed by the pipe stress group of the San Francisco Power Division of BPC and made available for use through the CDC 6600 computer. The LOTEMP program has been extensively used by the Bechtel Fast Flux Test Facility (FFTF) Systems Analysis Group since 1972 in the preliminary design of FFTF Class I piping. The ME913 program has been used to analyze nuclear Class I piping for Bechtel nuclear power plant projects.

Test Problesm

The Grand Gulf Project feedwater line was selected as a test problem. Hand calculations of a selected component in the piping system were performed in accordance with the sample problem (Reference 3.9A-8). Their results were compared with the computer output for code equations 9 through 14 in ME 913 (Reference 3.9A-10).

Table 3.9A-1 shows the comparison between the ASME sample problem (Reference 3.9A-8) and ME913 results (Butt Welding Tee, Location 10).

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3.12 SEPARATION CRITERIA FOR SAFETY RELATED MECHANICAL AND ELECTRICAL POWER EQUIPMENT

3.12.1 INTRODUCTION

This section describes the separation criteria used for Auxiliary Support System to safety-related systems that are identified in Chapter 7. The section addresses separation for both mechanical and electrical equipment.

For discussion of NSSS scope, see Section 7.1.

3.12.2 SYSTEMS

The Auxiliary Support Systems to which the separation criteria described in this section applies are identified in Table 3.12-2. Mechanical descriptions of the systems are given in Chapter 6, and 9, while actuation systems and electrical systems are described in detail in Chapters 7 and 8 respectively.

3.12.2.1 Criteria

3.12.2.1.1 General Criteria

Redundant systems are separated from each other so that single failure of a component or channel will not interfere with the proper operation of its redundant/diverse counterpart.

The affected mechanical systems and equipment are separated so that systems important to safety are protected from the following hazards:

a) The pipe break dynamic effects outlined in Section 3.6

b) Environmental effects as a result of pipe breaks and as outlined in Section 3.11

c) Flooding effects as a result of pipe breaks and as outlined in Section 3.4

d) Missiles as defined in Section 3.5

e) Fires capable of damaging redundant mechanical safety equipment.

The need for and adequacy of separation to protect the safety equipment from the above hazards are determined in conjunction with the criteria specified in Sections 3.4, 3.5, 3.6, 3.11, and 9.5.

3.12.2.1.2 System Separation Criteria

Piping for a redundant safety system is run independently of its counterparts, unless it can be shown that no single credible event, eg LOCA, is capable of causing piping failure that could prevent reactor shutdown. Supports and restraints of redundant mechanical components and piping are not shared, unless such sharing does not significantly impair their ability to perform their safety function.

Penetrations to the primary containment are separated or other adequate provisions are made so that the initial break of one piping branch of a system does not render its redundant counterpart(s) inoperable.

3.12.2.1.3 Physical Separation Criteria

Mechanical equipment and piping are separated from each other so that single failure of a device or component will not interfere with the proper operation of its redundant counterpart.

3.12.2.2 Separation Techniques

The methods used to protect redundant Auxiliary Support Systems from the above hazards (Subsection 3.12.2.1.1) fall into four categories of separation techniques: plant arrangement, barriers, spatial separation, and alternatives.

a) Plant Arrangement

A basic design consideration of plant layout is that redundant divisions of a safety system should not share common equipment areas. However, equipment common to a particular safety system division can share a common area if that equipment does not constitute a hazard within itself to another safety system of the same division.

Failure of any nonsafety related structure system or component shall not result in failure of any safety related structures, system, or component.

To accomplish Auxiliary Support Systems separations through plant arrangement, redundant division of a safety system may be placed in different compartments or even on different elevations. Nonsafety equipment, components, or piping should not be run above safety equipment unless they are adequately restrained or it can be demonstrated that failure will not impair function of the safety equipment.

b) Barriers

Barriers are most often used in restricted areas where a particular hazard (eg, small turbine missiles) is more easily identified or where other techniques are inappropriate (eg, separation between control boards). Separation by barriers is an extension of separation by the use of compartments in plant arrangement. Separation was also accomplished through the use of suitably designed equipment that in itself acts as a barrier. Examples would be heavily constructed control boards or heavy wall conduits and enclosed cable trays. In many cases, the barrier may enclose the hazard (eg, a compartment around a high speed turbine driven pump) in lieu of effecting a direct separation between redundant systems.

c) Spatial Separation

Spatial separation is another method of separating redundant safety systems and protecting them from the hazards described in Subsection 3.12.2.1.1.

For example, in areas where a barrier would be impractical, piping has been rerouted so that jet impingement resulting from a break would be dissipated by the distance traveled. In this example, partial barriers or restraints could also be used, as well as by hardening design (eg, heavier housing construction) of system components within the hazard area. When it can be shown that a hazard would have only a certain sphere of effectiveness (eg, for pipe whip, a rotation about a plastic hinge at the next restraint), spatial separation was considered adequate.

d) Alternatives

When one of the above techniques is impractical, a suitable alternative was used, some of which are additional restraints, hardening design, or temporary system isolation under accident conditions. When the redundant safety component cannot be held safe from common hazards by the alternatives outlined above, more resistant components were selected. An example would be the use of high pressure piping in a low pressure safety system to ensure its ability to withstand the effect of a break in adjacent high pressure lines.

3.12.3 ELECTRICAL SYSTEMS AND EQUIPMENT SEPARATION CRITERIA

3.12.3.1 Affected Systems

The electrical portions of the following systems are designed to the criteria of Subsection 3.12.3.2.

a) Standby diesel generator and auxiliaries

b) Class 1E 4160 V switchgear

c) Class 1E 480 V load centers

d) Class 1E 480 V MCCs

e) Class 1E 120 V ac control and instrument supply system

f) Class 1E 125 V dc supply system

g) Class 1E 250 V dc supply system

h) 480 V swing bus and associated motor generator set

Equipment covered by the requirements of this section includes instrument channels, trip systems, and trip actuators.

3.12.3.2 General Criteria

The resulting installations satisfy the criteria of IEEE 279-1971, 10CFR50 Appendix A, General Design Criteria 3, 17, and 21, as further clarified and limited below.

~~Single Failure Criteria~~

The affected electrical systems and equipment are separated that systems important to safety are protected from the following hazards:

a) Fires in cable raceways due to an electrical fault that could cause failure of insulation on other cables.

b) Mechanical damage of electrical equipment in a single location.

c) Single Design Base Event (DBE) should not disable essential automatic or manual protective function, ie, reactor scram, primary containment isolation, core cooling, etc.

~~Identification~~

Identification and division/channels conform to the following:

a) Panels and racks, not part of the PGCC, are labeled with distinctive marker plates. The marker plates include identification of the proper division/channel, as listed in Table 3.12-1.

b) Junction and/or pull boxes, not part of the PGCC, have identification similar to and compatible with the panels and racks considered above.

c) Cables external to cabinets and/or panels, not part of the PGCC, are marked to distinguish them in color from other cables and to identify their separation division/channel as applicable.

d) Raceways, not part of the PGCC, identified as described in Subsection 3.12.3.4.2.1.b.

e) For PGCC panels and racks refer to Section 7.1.

f) For cables external to panels and racks but within the PGCC, refer to Section 7.1.

~~3.12.3.3 System Separation Criteria~~

See Section 7.1.

3.12.3.4 Electrical Physical Separation Criteria

3.12.3.4.1 General Separation Criteria

Methods of Separation

The separation of circuits and equipment is achieved by separate safety class structures, distance, or barriers, or any combination thereof.

Compatibility with Mechanical Systems

Class 1E circuits are routed and/or protected such that failure of related mechanical equipment of one redundant system cannot disable Class 1E circuits or equipment essential to the operation of the other redundant system(s).

Raceway Sharing of Class 1E and non-Class 1E Circuits

See Subsection 8.1.6.1.

3.12.3.4.2 Specific Separation Criteria

3.12.3.4.2.1 Cables and Raceways

a) General

The minimum separation distances specified in paragraphs d) and e) are based on open ventilated trays. Where these distances are used to provide adequate physical separation:

1) Cable splices in raceways are prohibited

2) Cables and raceways involved are flame retardant

3) The design basis is that the cable trays will not be filled above the side rails

4) Hazards will be limited to failures or faults internal to the electric equipment or cables.

b) Identification of Non-PGCC Cables and Raceways

Exposed Class 1E raceways are identified in a distinct and permanent manner at intervals not to exceed 15 ft. In addition, these raceways are also identified where they pass through walls and/or floors, and enclosed

areas. Class 1E raceways are identified prior to the installation of their cables.

Cables installed in these raceways are identified at intervals not exceeding 5 ft to facilitate initial verification that the installation conforms to the separation criteria. These cable identifications are applied prior to or during their installation.

Class 1E cables are identified by a permanent marker at each end in accordance with the design drawings or cable schedule.

Color coding is used to meet the above requirements and to distinguish between redundant Class 1E cables and non-Class 1E cables.

c) Identification of PGCC Cables and Raceways

Refer to Subsection 7.1.2a.3.2.

d) Cable Spreading Areas/Control Structure Complex

The control structure complex consists of two elevations of relay rooms, two cable spreading areas, and the main control room. Below the main control room is the lower cable spreading room, which facilitates cable convergence from the computer room and the lower relay rooms (which are below the lower cable spreading room) to the general plant areas, and to the cable entrance areas at the bottom of the control room panels. The lower relay rooms consist mainly of control and instrument panels of non-Class 1E systems and one division (i.e., DIvision II) of redundant systems as listed in Subsection 3.12.3.1. The main control room panels are mounted on a raised floor assembly with cable trays and wire way gutters that enter the bottom of the main control room panels. Above the main control room are the upper relay rooms and the upper cable spreading area. The upper cable spreading area facilities cable convergence from the upper relay room to the general plant areas, to the top of the main control room panels and to the control room raised floor. The upper relay room consists mainly of control and instrument panels of non-class 1E systems and the other division (ie Division I) of the redundant systems listed in Subsection 3.12.3.1. The relay room panels and cabinets are integrated with a module type floor assembly with lateral and longitudinal ducts that act as raceways and barriers. The cabling interface between the PGCC and the spreading area is made at

termination cubicles on the periphery of the relay room floor assemblies.

The relay rooms and spreading room areas do not contain high energy equipment (such as switchgear, transformers) or potential sources of missiles or pipe whip and are not used for storing flammable materials.

Circuits in the relay room and main control room are limited to control functions, instrument functions, and those power supply circuits and facilities serving the main control room and instrument systems.

Where for operational reasons redundant channel/division Class 1E cables are not separated by different safety class structures (eg, two relay rooms and spreading areas), the minimum separation distance between the redundant Class 1E cable trays is 1 ft horizontally and 3 ft vertically. Where 1 ft horizontal separation is not possible, one of the two following requirements are met: a fire barrier is placed between the redundant cable trays 1 ft above the trays or to the ceiling; or cables of each channel/division are installed in rigid steel conduit up to a point where the 1 ft spacing requirement is met. Where cables of redundant channel/ divisions must be stacked one above the other with less than 3 ft vertical spacing, one of the following requirements are met: a) a fire barrier is placed between the trays and extended to 6 in. of each side of the tray system or to the wall, or b) a solid steel tray cover is installed on the lower cable tray and the upper tray has a solid bottom up to a point where 3 ft vertical separation is met; or c) the cables of each redundant channel/division are installed in rigid steel conduit to a point where the 3 ft vertical separation exists.

Separation requirements between Class 1E and non-Class 1E circuits are the same as separation of redundant channel/division.

e) General Plant Areas

In plant areas from which potential hazards such as missiles, external fires, and pipe whip are excluded, the minimum separation distance between redundant Class 1E cable trays is 3 ft between trays separated horizontally if no physical barrier exists between trays. If a horizontal separation of less than 3 ft exists, alternate methods as stated in paragraph d) above are required. Vertical stacking of trays is avoided wherever possible; however, where cable trays

of redundant channel/divisions are stacked, a minimum vertical separation distance of 5 ft is required, or alternate methods as stated in paragraph d) above are required. Where a cross-over of one tray over another carrying redundant channel/division is made, and minimum vertical separation distance cannot be maintained; a) a solid cover is installed on the lower tray to extend 1 ft 0 in. minimum either side of upper tray, b) fire barriers are installed minimum 1 in. from upper tray and extend 1 ft 0 in. minimum beyond the crossing tray.

Separation requirements between Class 1E and non-Class 1E circuits are the same as separation of redundant channel/division.

f) Power Generation Control Complex - (PGCC)

Refer to Subsection 7.1.2a.3.3.6.

g) An exception to the above subsections d) and e) is the 450 MHZ radio antenna cable network.

The 450 MHZ radio antenna is the plant security communication radio system (non-Class 1E). This system utilizes an antenna cable network consisting of installed exposed (not enclosed in raceway) on the cable raceway supports throughout the plant. The jacketing material of the antenna cable is made of flame retardant RULAN. The cable has been tested and passed IEEE 383 and ASTM Proc. D2633 part 30.

Separation between the 450 MHZ radio antenna cable and those safe-shutdown Class 1E raceways listed in Appendix A of the Susquehanna SES Fire Protection Review Report is provided in accordance with Regulatory Guide 1.75. Separation between this antenna cable and other class 1E raceways is not required because: 23

1) The antenna cable is a low energy circuit. A short circuit of the antenna cable would not produce enough energy to cause degradation of any other circuits.

2) The antenna cable is not routed with any other cables.

3) The antenna cable jacket is made of flame retardant material.

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4) The antenna cable does not terminate in close proximity or routed through any equipment with voltage level higher than 120V AC.

5) The maximum radio frequency (rf) power output level of the antenna cable is 37.5 watts.

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3.12.3.4.2.2 - Standby Power Supply

a) Standby Generating Units

Redundant Class 1E standby generating units are located in separate safety class structures and have independent air and fuel supplies.

b) Auxiliaries and Local Controls

The auxiliaries and local controls for redundant standby generating units are in the same safety class structure as the unit they serve, except for the fuel oil transfer pumps that are located in separate safety class structures at the fuel oil storage tanks (See Subsection 9.5.4).

3.12.3.4.2.3 - DC System

a) Batteries

Redundant Class 1E batteries are placed in separate safety class structures. The structures are served by redundant ventilation equipment.

b) Battery Chargers

Battery chargers for redundant Class 1E batteries are placed in separate safety class structures with their respective switchgears.

3.12.3.4.2.4 - Distribution System

a) Switchgear

Redundant Class 1E distribution switchgear groups are placed in separate safety class structures.

b) Motor Control Centers

Redundant Class 1E motor control centers are physically separated in accordance with the requirements of Subsection 3.12.3.4.1.

listed in Subsection 3.12.3.1. The main control room panels are mounted on a raised floor assembly with cable trays and wire way gutters that enter the bottom of the

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c) Distribution Panels

Redundant Class 1E distribution panels are physically separated in accordance with the requirements of Subsection 3.12.3.4.1.

3.12.3.4.2.5 Primary Containment Electrical Penetrations

Redundant Class 1E primary containment electrical penetrations are physcially separated in accordance with the requirements of Subsection 3.12.3.4.1. The minimum physical separation for redundant penetrations meets the requirements for cables and raceways given in Subsections 3.12.3.4.2.1 through 3.12.3.4.2.6.

3.12.3.4.2.6 Main Control Room and Relay Room Panels

A. For NSSS panels see Subsection 7.1.2a.3.1.1. 17

B. All non-NSSS panels containing safety-related equipment and circuits are provided as follows:

1. Panels are divisionalized, i.e, are devoted to one (1) division only, and are physically separated from the redundant divisions panels.

2. Panels which contain the redundant circuits of both divisions are physically divided by metal barriers with only one (1) division on each side of the barrier. 12

3. All requirements for connection of circuits between separated divisions are accomplished with MDR relays to provide positive isolation of the circuits.

TABLE 3.12-1

ESF DIVISION SEPARATION

Division I	Division II
Core Spray Loop A	Core Spray Loop B
Automatic Depressurization System A	Automatic Depressurization System B
Residual Heat Removal Loop A	Residual Heat Removal Loop B
Reactor Core Isolation Cooling System	High Pressure Coolant Injection System
Nuclear Steam Supply Shutoff System (Inboard Valves)	Nuclear Steam Supply Shutoff System (Outboard Valves)
Recirculation Pump Trip Loop A	Recirculation Pump Trip Loop B
Emergency Service Water Loop A	Emergency Service Water Loop B
RHR Service Water Loop A	RHR Service Water Loop B
Containment Instrument Gas Loop A	Containment Instrument Gas Loop B
Main Steam Isolation Valve Leakage Control System Div I	Main Steam Isolation Valve Leakage Control System Div II
Containment Atmospheric Control System A	Containment Atmospheric Control System B
Standby Gas Treatment System Train A	Standby Gas Treatment System Train B
Reactor Building HVAC Isolation and Recirculation System A	Reactor Building HVAC Isolation and Recirculation System B
Drywell HVAC System A	Drywell HVAC System B
Control Structure HVAC System Train A	Control Structure HVAC System Train B
Control Structure Chilled Water System Loop A	Control Structure Chilled Water System Loop B
Battery Room Ventilation System A	Battery Room Ventilation System B

TABLE 3.12-1 (Continued)

Division I	Division II
HVAC Coolers for Div I	HVAC Coolers for Div II
Standby Liquid Control System Pumps A(1) and B(1) and Explosive Valves A(1) and B(1)	
Class 1E 250 V dc Supply System I	Class 1E 250 V dc Supply System II
480 V Swing Bus and Associated Motor-Generator Set Div I	480 V Swing Bus and Associated Motor-Generator Set Div II
Class 1E 480 V ac MCCs	Class 1E 480 V ac MCCs
Class 1E 120 V ac Distribution Panels	Class 1E 120 V ac Distribution Panels
Class 1E 125 V dc Distribution Panels	Class 1E 125 V dc Distribution Panels

(1) The redundant standby liquid control pumps and explosive valves are powered from different electrical buses.

CHAPTER 4

REACTOR

TABLE OF CONTENTS

5

TABLES

NOTE: Additional tables are presented in NEDE 20944/20944-P.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

"Functional design of the control rod drive system (CRD) is discussed below. Functional designs of the recirculation flow control system and standby liquid control system are described in subsections 5.4.1 and 9.3.5, respectively."

4.6.1 Information for CRDS

4.6.1.1 Control Rod Drive System Design

4.6.1.1.1 Design Bases

4.6.1.1.1.1 General Design Bases

4.6.1.1.1.1.1 Safety Design Bases

The control rod drive mechanical system shall meet the following safety design bases:

(1) Design shall provide for a sufficiently rapid control rod insertion that no fuel damage results from any abnormal operating transient.

(2) Design shall include positioning devices, each of which individually supports and positions a control rod.

(3) Each positioning device shall:

a. Prevent its control rod from initiating withdrawal as a result of a single malfunction.

b. Be individually operated so that a failure in one positioning device does not affect the operation of any other positioning device.

c. Be individually energized when rapid control rod insertion (scram) is signaled so that failure of power sources external to the positioning device does not prevent other positioning devices' control rods from being inserted.

4.6.1.1.1.2 Power Generation Design Basis

The control rod system drive design shall provide for positioning the control rods to control power generation in the core.

4.6.1.1.2 Description

The control rod drive system (CRD) controls gross changes in core reactivity by incrementally positioning neutron absorbing control rods within the reactor core in response to manual control signals. It is also required to quickly shut down the reactor (scram) in emergency situations by rapidly inserting withdrawn control rods into the core in response to a manual or automatic signal. The control rod drive system consists of locking piston control rod drive mechanisms, and the CRD hydraulic system (including power supply and regulation, hydraulic control units, interconnecting piping, instrumentation and electrical controls).

4.6.1.1.2.1 Control Rod Drive Mechanisms

The CRD mechanism (drive) used for positioning the control rod in the reactor core is a double-acting, mechanically latched, hydraulic cylinder using water as its operating fluid. (See Figure 4.6-1, 4.6-2, 4.6-3, and 4.6-4.) The individual drives are mounted on the bottom head of the reactor pressure vessel. The drives do not interfere with refueling and are operative even when the head is removed from the reactor vessel.

The drives are also readily accessible for inspection and servicing. The bottom location makes maximum utilization of the water in the reactor as a neutron shield and gives the least possible neutron exposure to the drive components. Using water from the condensate system as the operating fluid eliminates the need for special hydraulic fluid. Drives are able to utilize simple piston seals whose leakage does not contaminate the reactor water but provides cooling for the drive mechanisms and their seals.

The drives are capable of inserting or withdrawing a control rod at a slow, controlled rate, as well as providing rapid insertion when required. A mechanism on the drive locks the control rod at 6-inch increments of stroke over the length of the core.

A coupling spud at the top end of the drive index tube (piston rod) engages and locks into a mating socket at the base of the control rod. The weight of the control rod is sufficient to

engage and lock this coupling. Once locked, the drive and rod form an integral unit that must be manually unlocked by specific procedures before the components can be separated.

The drive holds its control rod in distinct latch positions until the hydraulic system actuates movement to a new position. Withdrawal of each rod is limited by the seating of the rod in its guide tube. Withdrawal beyond this position to the over-travel limit can be accomplished only if the rod and drive are uncoupled. Withdrawal to the over-travel limit is annunciated by an alarm.

The individual rod indicators, grouped in one control panel display, correspond to relative rod locations in the core. A separate, smaller hardwired display is located on the standby information panel. A CRT presentation is available on the unit operating benchboard. This latter display presents the position of the control rod selected for movement as well as the other rods in the affected rod group.

For display purposes the control rods are considered in groups of four adjacent rods centered around a common core volume. Each group is monitored by four LPRM strings (see Subsection 7.6.1.5). Rod groups at the periphery of the core may have less than four rods. The small rod display shows the positions, in digital form, of the rods in the group to which the selected rod belongs. A white light indicates which of the four rods is the one selected for movement.

4.6.1.1.2.2 Drive Components

Figure 4.6-2 illustrates the operating principle of a drive. Figures 4.6-3 and 4.6-4 illustrate the drive in more detail. The main components of the drive and their functions are described below.

4.6.1.1.2.2.1 Drive Piston

The drive piston is mounted at the lower end of the index tube. This tube functions as a piston rod. The drive piston and index tube make up the main moving assembly in the drive. The drive piston operates between positive end stops, with a hydraulic cushion provided at the upper end only. The piston has both inside and outside seal rings and operates in an annular space between an inner cylinder (fixed piston tube) and an outer cylinder (drive cylinder). Because the type of inner seal used is effective in only one direction, the lower sets of seal rings are mounted with one set sealing in each direction.

A pair of nonmetallic bushings prevents metal-to-metal contact between the piston assembly and the inner cylinder surface. The outer piston rings are segmented step-cut seals with expander springs holding the segments against the cylinder wall. A pair of split bushings on the outside of the piston prevents piston contact with the cylinder wall. The effective piston area for downtravel, or withdrawal, is approximately 1.2 sq. in. versus 4.1 sq. in. for uptravel, or insertion. This difference in driving area tends to balance the control rod weight and assures a higher force for insertion than for withdrawal.

4.6.1.1.2.2.2 Index Tube

The index tube is a long hollow shaft made of nitrided stainless steel. Circumferential locking grooves, spaced every 6 inches along the outer surface, transmit the weight of the control rod to the collet assembly.

4.6.1.1.2.2.3 Collet Assembly

The collet assembly serves as the index tube locking mechanism. It is located in the upper part of the drive unit. This assembly prevents the index tube from accidentally moving downward. The assembly consists of the collet fingers, a return spring, a guide cap, a collet housing (part of the cylinder, tube, and flange), and the collet piston.

Locking is accomplished by fingers mounted on the collet piston at the top of the drive cylinder. In the locked or latched position the fingers engage a locking groove in the index tube.

The collet piston is normally held in the latched position by a force of approximately 150 lb supplied by a spring. Metal piston rings are used to seal the collet piston from reactor vessel pressure. The collet assembly will not unlatch until the collet fingers are unloaded by a short, automatically sequenced, drive-in signal. A pressure, approximately 180 psi above reactor vessel pressure, must then be applied to the collet piston to overcome spring force, slide the collet up against the conical surface in the guide cap, and spread the fingers out so they do not engage a locking groove.

A guide cap is fixed in the upper end of the drive assembly. This member provides the unlocking cam surface for the collet fingers and serves as the upper bushing for the index tube.

If reactor water is used during a scram to supplement accumulator pressure, it is drawn through a filter on the guide cap.

4.6.1.1.2.2.4 Piston Tube

The piston tube is an inner cylinder, or column, extending upward inside the drive piston and index tube. The piston tube is fixed to the bottom flange of the drive and remains stationary. Water is brought to the upper side of the drive piston through this tube. A buffer shaft, at the upper end of the piston tube, supports the stop piston and buffer components.

4.6.1.1.2.2.5 Stop Piston

A stationary piston, called the stop piston, is mounted on the upper end of the piston tube. This piston provides the seal between reactor vessel pressure and the space above the drive piston. It also functions as a positive end stop at the upper limit of control rod travel. Piston rings and bushings, similar to those used on the drive piston, are mounted on the upper portion of the stop piston. The lower portion of the stop piston forms a thinwalled cylinder containing the buffer piston, its metal seal ring, and the buffer piston return spring. As the drive piston reaches the upper end of the scram stroke it strikes the buffer piston. A series of orifices in the buffer shaft provides a progressive water shutoff to cushion the buffer piston as it is driven to its limit of travel. The high pressures generated in the buffer are confined to the cylinder portion of the stop piston, and are not applied to the stop s steel tube. The switches are actuated by a ring magnet located at the bottom of the drive piston.

The drive piston, piston tube, and indicator tube are all of nonmagnetic stainless steel, allowing the individual switches to be operated by the magnet as the piston passes. One switch is located at each position corresponding to an index tube groove and one switch is located at the midpoint between each latching point. Thus, indication is provided for each 3 inches of travel. Duplicate switches are provided for the full-in and full-out positions. Redundant overtravel switches are located at a position below the normal full-out position. Because the limit of downtravel is normally provided by the control rod itself as it reaches the backseat position, the drive can pass this position and actuate the overtravel switches only if it is uncoupled from its control rod. A convenient means is thus provided to verify that the drive and control rod are coupled after installation of a drive or at any time during plant operation.

4.6.1.1.2.2.6 Flange and Cylinder Assembly

A flange and cylinder assembly is made up of a heavy flange welded to the drive cylinder. A sealing surface on the upper face of this flange forms the seal to the drive housing flange. The seals contain reactor pressure and the two hydraulic control pressures. Teflon coated, stainless steel rings are used for these seals. The drive flange contains the integral ball, or two-way, check (ball-shuttle) valve. This valve directs either the reactor vessel pressure or the driving pressure, whichever is higher, to the underside of the drive piston. Reactor vessel pressure is admitted to this valve from the annular space between the drive and drive housing through passages in the flange.

Water used to operate the collet piston passes between the outer tube and the cylinder tube. The inside of the cylinder tube is honed to provide the surface required for the drive piston seals.

Both the cylinder tube and outer tube are welded to the drive flange. The upper ends of these tubes have a sliding fit to allow for differential expansion.

The upper end of the index tube is threaded to receive a coupling spud. The coupling (see Figure 4.6-1) accommodates a small amount of angular misalignment between the drive and the control rod. Six spring fingers allow the coupling spud to enter the mating socket on the control rod. A plug then enters the spud and prevents uncoupling.

Two means of uncoupling are provided. With the reactor vessel head removed, the lock plug can be raised against the spring force of approximately 50 pounds by a rod extending up through the center of the control rod to an unlocking handle located above the control rod velocity limiter. The control rod, with the lock plug raised, can then be lifted from the drive.

4.6.1.1.2.2.7 Lock Plug

The lock plug can also be pushed up from below, if it is desired to uncouple a drive without removing the reactor pressure vessel head for access. In this case, the central portion of the drive mechanism is pushed up against the uncoupling rod assembly, which raises the lock plug and allows the coupling spud to disengage the socket as the drive piston and index tube are driven down.

The control rod is heavy enough to force the spud fingers to enter the socket and push the lock plug up, allowing the spud to enter the socket completely and the plug to snap back into place.

Therefore, the drive can be coupled to the control rod using only the weight of the control rod. However, with the lock plug in place, a force in excess of 50,000 lb is required to pull the coupling apart.

4.6.1.1.2.3 Materials of Construction

Factors that determine the choice of construction materials are discussed in the following subsections.

4.6.1.1.2.3.1 Index Tube

The index tube must withstand the locking and unlocking action of the collet fingers. A compatible bearing combination must be provided that is able to withstand moderate misalignment forces. The reactor environment limits the choice of materials suitable for corrosion resistance. The column and tensile loads can be satisfied by an annealed, single phase, nitrogen strengthened, austenitic stainless steel. The wear and bearing requirements are provided by Malcomizing the complete tube. To obtain suitable corrosion resistance, a carefully controlled process of surface preparation is employed.

4.6.1.1.2.3.2 Coupling Spud

The coupling spud is made of Inconel 750 that is aged for maximum physical strength and the required corrosion resistance. Because misalignment tends to cause chafing in the semispherical contact area, the part is protected by a thin chromium plating (Electrolized). This plating also prevents galling of the threads attaching the coupling spud to the index tube.

4.6.1.1.2.3.3 Collet Fingers

Inconel 750 is used for the collet fingers, which must function as leaf springs when cammed open to the unlocked position. Colmonoy 6 hard facing provides a long wearing surface, adequate for design life, to the area contacting the index tube and unlocking cam surface of the guide cap.

4.6.1.1.2.3.4 Seals and Bushings

Graphitar 14 is selected for seals and bushings on the drive piston and stop piston. The material is inert and has a low friction coefficient when water-lubricated. Because some loss of Graphitar strength is experienced at higher temperatures, the drive is supplied with cooling water to hold temperatures below 250°F. The Graphitar is relatively soft, which is advantageous when an occasional particle of foreign matter reaches a seal. The resulting scratches in the seal reduce sealing efficiency until worn smooth, but the drive design can tolerate considerable water leakage past the seals into the reactor vessel.

4.6.1.1.2.3.5 Summary

All drive components exposed to reactor vessel water are made of austenitic stainless steel except the following:

(1) Seals and bushings on the drive piston and stop piston are Graphitar 14.

(2) All springs and members requiring spring action (collet fingers, coupling spud, and spring washers) are made of Inconel-750.

(3) The ball check valve is a Haynes Stellite cobalt-base alloy.

(4) Elastomeric O-ring seals are ethylene propylene.

(5) Metal piston rings are Haynes 25 alloy.

(6) Certain wear surfaces are hard-faced with Colmonoy 6.

(7) Nitriding by a proprietary new Malcomizing process and chromium plating are used in certain areas where resistance to abrasion is necessary.

(8) The drive piston head is made of Armco 17-4Ph.

Pressure-containing portions of the drives are designed and fabricated in accordance with requirements of Section III of the ASME Boiler and Pressure Vessel Code.

4.6.1.1.2.4 Control Rod Drive Hydraulic System

The control rod drive hydraulic system (Figures 4.6-5a, b) supplies and controls the pressure and flow to and from the drives through hydraulic control units (HCU). The water discharged from the drives during a scram flows through the HCUs to the scram discharge volume. The water discharged from a drive during a normal control rod positioning operation flows through the HCU, the exhaust header, through the other HCV's to combine with the cooling water flow at the CRD's, and into the reactor vessel. There are as many HCUs as the number of control rod drives.

4.6.1.1.2.4.1 Hydraulic Requirements

The CRD hydraulic system design is shown in Figures 4.6-5a, b, and 4.6-6. The hydraulic requirements, identified by the function they perform, are as follows:

(1) An accumulator hydraulic charging pressure of approximately 1400 to 1500 psig is required. Flow to the accumulators is required only during scram reset or system startup.

(2) Drive pressure of approximately 250 psi above reactor vessel pressure is required. A flow rate of approximately 4 gpm to insert a control rod and 2 gpm to withdraw a control rod is required. 23

(3) Cooling water to the drives is required at approximately 30 psi above reactor vessel pressure and at a flow rate of 0.20 to 0.34 gpm per drive unit. (Cooling water to a drive can be interrupted for short periods without damaging the drive.) 23

(4) The scram discharge volume is sized to receive and contain all the water discharged by the drives during a scram; a minimum volume of 3.34 gal. per drive is required.

4.6.1.1.2.4.2 System Description

The CRD hydraulic systems provide the required functions with the pumps, filter, valves, instrumentation, and piping shown in Figures 4.6-5a, b and described in the following subsection.

Duplicate components are included, where necessary, to assure continuous system operation if an in-service component requires maintenance.

4.6.1.1.2.4.2.1 Supply Pump

One supply pump pressurizes the system with water from the condensate system. One spare pump is provided for standby. A discharge check valve prevents backflow through the nonoperating pump. A portion of the pump discharge flow is diverted through a minimum flow bypass line to the condensate storage tank. This flow is controlled by an orifice and is sufficient to prevent immediate pump damage if the pump discharge is inadvertently closed.

Condensate water is processed by two filters in the system. The pump suction filter is a disposable element type with a 25-micron absolute rating. The drive water filter downstream of the pump is a cleanable element type with a 50-micron absolute rating. Differential pressure indicators and control room alarms monitor the filter elements as they collect foreign materials.

4.6.1.1.2.4.2.2 Accumulator Charging Pressure

Accumulator charging pressure is established by the discharge pressure of the system supply pump. During scram the scram inlet (and outlet) valves open and permit the stored energy in the accumulators to discharge into the drives. The resulting pressure decrease in the charging water header allows the CRD supply pump to "run out" (i.e., flow rate to increase substantially) into the control rod drives via the charging water header. The flow sensing system upstream of the accumulator charging header detects high flow and decreases flow returning to the RPV. This action maintains increased flow through the charging water header.

Pressure in the charging header is monitored in the control room with a pressure indicator and high pressure alarm.

During normal operation the flow control valve maintains a constant system flow rate. This flow is used for drive flow, drive cooling, and system stability.

4.6.1.1.2.4.2.3 Drive Water Pressure

Drive water pressure required in the drive header is maintained by the pressure control valve, which is manually adjusted from the control room. A flow rate of approximately 6 gpm (the sum of the flow rate required to insert and withdraw a control rod) normally bypasses the drive water pressure stage through two solenoid operated stabilizing valves (arranged in parallel). The flow through one stabilizing valve equals the drive insert flow; that of the other stabilizing valve equals the drive withdrawal flow. When operating a drive, the required flow is diverted to that drive by closing the appropriate stabilizing valve. Thus, flow through the drive pressure control valve is always constant.

Flow indicators in the drive water header and in the line downstream from the stabilizing valves allow the flow rate through the stabilizing valves to be adjusted when necessary. Differential pressure between the reactor vessel and the drive pressure stage is indicated in the control room.

4.6.1.1.2.4.2.4 Cooling Water Header

All water passing through the pressure control valve and the stabilizing valves is routed to the reactor via the cooling water header. Without flow in the drive and charging water headers the cooling water flow is equal to the flow passing through the flow control valve.

The flow through the flow control valve is virtually constant. Therefore, once adjusted, the drive pressure control valve can maintain the required pressure independent of reactor pressure. Changes in setting of the pressure control valve is required only to adjust for changes in the cooling requirements of the drives, as their seal characteristics change with time. A flow indicator in the control room monitors cooling water flow. A differential pressure indicator in the control room indicates the difference between reactor vessel pressure and drive cooling water pressure. Although the drives can function without cooling water, seal life is shortened by long term exposure to reactor temperatures. The

temperature of each drive is recorded in the reactor building, and excessive temperatures are annunciated in the control room.

4.6.1.1.2.4.2.5 Scram Discharge Volume

The scram discharge volume consists of header piping which connects to each HCU and drains into an instrument volume. The header piping is sized to receive and contain all the water discharged by the drives during a scram, independent of the instrument volume.

During normal plant operation the scram discharge volume is empty, and vented to atmosphere through its open vent and drain valve. When a scram occurs, upon a signal from the safety circuit these vent and drain valves are closed to conserve reactor water. Lights in the control room indicate the position of these valves.

During a scram, the scram discharge volume partly fills with water discharged from above the drive pistons. After scram is completed, the control rod drive seal leakage from the reactor continues to flow into the scram discharge volume until the discharge volume pressure equals the reactor vessel pressure. A check valve in each HCU prevents reverse flow from the scram discharge header volume to the drive. When the initial scram signal is cleared from the reactor protection system, the scram discharge volume signal is overridden with a keylock override switch, and the scram discharge volume is drained and returned to atmospheric pressure.

Remote manual switches in the pilot valve solenoid circuits allow the discharge volume vent and drain valves to be tested without disturbing the reactor protection system. Closing the scram discharge volume valves allows the outlet scram valve seats to be leak-tested by timing the accumulation of leakage inside the scram discharge volume.

Six liquid-level switches connected to the instrument volume, monitor the volume for abnormal water level. They are set at three different levels. At the lowest level, a level switch actuates to indicate that the volume is not completely empty

during post-scram draining or to indicate that the volume starts to fill through leakage accumulation at other times during reactor operation. At the second level, one level switch produces a rod withdrawal block to prevent further withdrawal of any control rod, when leakage accumulates to half the capacity of the instrument volume. The remaining four switches are interconnected with the trip channels of the Reactor Protection System (RPS) and will initiate a reactor scram should water accumulation fill the instrument volume.

4.6.1.1.2.4.3 Hydraulic Control Units

Each hydraulic control unit (HCU) furnishes pressurized water, on signal, to a drive unit. The drive then positions its control rod as required. Operation of the electrical system that supplies scram and normal control rod positioning signals to the HCU is described in Subsection 7.7.1.2. The basic components in each HCU are manual, pneumatic, and electrical valves; an accumulator; related piping; electrical connections; filters; and instrumentation (see Figures 4.6-5b, 4.6-6, and 4.6-7). The components and their functions are described in the following paragraphs.

4.6.1.1.2.4.3.1 Insert Drive Valve

The insert drive valve is solenoid-operated and opens on an insert signal. The valve supplies drive water to the bottom side of the main drive piston.

4.6.1.1.2.4.3.2 Insert Exhaust Valve

The insert exhaust solenoid valve also opens on an insert signal. The valve discharges water from above the drive piston to the exhaust water header.

4.6.1.1.2.4.3.3 Withdraw Drive Valve

The withdraw drive valve is solenoid-operated and opens on a withdraw signal. The valve supplies drive water to the top of the drive piston.

4.6.1.1.2.4.3.4 Withdraw Exhaust Valve

The solenoid-operated withdraw exhaust valve opens on a withdraw signal and discharges water from below the main drive piston to the exhaust header. It also serves as the settle valve, which opens following any normal drive movement (insert or withdraw) to allow the control rod and its drive to settle back into the nearest latch postion.

4.6.1.1.2.4.3.5 Speed Control Units

The insert drive valve and withdraw exhaust valve have a speed control unit. The speed control unit regulates the control rod insertion and withdrawal rates during normal operation. The manually adjustable flow control unit is used to regulate the water flow to and from the volume beneath the main drive piston. A correctly adjusted unit does not require readjustment except to compensate for changes in drive seal leakage.

4.6.1.1.2.4.3.6 Scram Pilot Valves

The scram pilot valves are operated from the reactor protection system. A scram pilot valve with two solenoids controls both the scram inlet valve and the scram exhaust valve. The scram pilot valves are three-way, solenoid-operated, normally energized valves. On loss of electrical signal to the solenoids, such as the loss of external a-c power, the inlet port closes and the exhaust port opens. The pilot valves (Figures 4.5-5a and b) are designed so that the trip system signal must be removed from both solenoids before air pressure can be discharged from the scram valve operators. This prevents inadvertent scram of a single drive in the event of a failure of one of the pilot valve solenoids.

4.6.1.1.2.4.3.7 Scram Inlet Valve

The scram inlet valve opens to supply pressurized water to the bottom of the drive piston. This quick opening globe valve is operated by an internal spring and system pressure. It is closed by air pressure applied to the top of its diaphragm operator. A main control room indicating light is energized when both the scram inlet valve and the scram exhaust valve are fully open.

4.6.1.1.2.4.3.8 Scram Exhaust Valve

The scram exhaust valve opens slightly before the scram inlet valve, exhausting water from above the drive piston. The exhaust valve opens faster than the inlet valve because of the higher air pressure spring setting in the valve operator.

4.6.1.1.2.4.3.9 Scram Accumulator

The scram accumulator stores sufficient energy to fully insert a control rod at lower vessel pressures. At higher vessel pressures the accumulator pressure is assisted or supplanted by reactor vessel pressure. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water on top from the nitrogen below. A check valve in the accumulator charging line prevents loss of water pressure in the event supply pressure is lost.

During normal plant operation, the accumulator piston is seated at the bottom of its cylinder. Loss of nitrogen decreases the nitrogen pressure, which actuates a pressure switch and sounds an alarm in the control room.

To ensure that the accumulator is always able to produce a scram, it is continuously monitored for water leakage. A float type level switch actuates an alarm if water leaks past the piston barrier and collects in the accumulator instrumentation block.

4.6.1.1.2.5 Control Rod Drive System Operation

The control rod drive system performs rod insertion, rod withdrawal, and scram. These operational functions are described below.

4.6.1.1.2.5.1 Rod Insertion

Rod insertion is initiated by a signal from the operator to the insert valve solenoids. This signal causes both insert valves to open. The insert drive valve applies reactor pressure plus approximately 90 psi to the bottom of the drive piston. The insert exhaust valve allows water from above the drive piston to discharge to the exhaust header.

As is illustrated in Figure 4.6-3, the locking mechanism is a ratchet-type device and does not interfere with rod insertion.

The speed at which the drive moves is determined by the flow through the insert speed control valve, which is set for approximately 4 gpm for a shim speed (nonscram operation) of 3 in./sec. During normal insertion, the pressure on the downstream side of the speed control valve is 90 to 100 psi above reactor vessel pressure. However, if the drive slows for any reason, the flow through, and pressure drop across, the insert speed control valve will decrease; the full differential pressure (260 psi) will then be available to cause continued insertion. With 260-psi differential pressure acting on the drive piston, the piston exerts an upward force of 1040 lb.

4.6.1.1.2.5.2 Rod Withdrawal

Rod withdrawal is, by design, more involved than insertion. The collet finger (latch) must be raised to reach the unlocked position (see Figure 4.6-3). The notches in the index tube and the collet fingers are shaped so that the downward force on the index tube holds the collet fingers in place. The index tube must be lifted before the collet fingers can be released. This is done by opening the drive insert valves (in the manner described in the preceding paragraph) for approximately 1 sec. The withdraw valves are then opened, applying driving pressure above the drive piston and opening the area below the piston to the exhaust header. Pressure is simultaneously applied to the collet piston. As the piston raises, the collet fingers are cammed outward, away from the index tube, by the guide cap.

The pressure required to release the latch is set and maintained at a level high enough to overcome the force of the latch return spring plus the force of reactor pressure opposing movement of the collet piston. When this occurs, the index tube is unlatched and free to move in the withdraw direction. Water displaced by the drive piston flows out through the withdraw speed control valve, which is set to give the control rod a shim speed of 3 in./sec. The entire valving sequence is automatically controlled and is initiated by a single operation of the rod withdraw switch.

4.6.1.1.2.5.3 Scram

During a scram the scram pilot valves and scram valves are operated as previously described. With the scram valves open, accumulator pressure is admitted under the drive piston, and the area over the drive piston is vented to the scram discharge volume.

The large differential pressure (initially approximately 1500 psi and always several hundred psi, depending on reactor vessel pressure) produces a large upward force on the index tube and control rod. This force gives the rod a high initial acceleration and provides a large margin of force to overcome friction. After the initial acceleration is achieved, the drive continues at a nearly constant velocity. This characteristic provides a high initial rod insertion rate. As the drive piston nears the top of its stroke the piston seals close off the large passage (buffer orifices) in the stop piston tube, providing a hydraulic cushion at the end of travel.

Prior to a scram signal the accumulator in the Hydraulic Control Unit has approximately 1450-1510 psig on the water side and 1050-1100 psig on the nitrogen side. As the inlet scram valve opens, the full water side pressure is available at the control rod drive acting on a 4.1 sq in. area. As CRD motion begins, this pressure drops to the gas side pressure less line losses between the accumulator and the CRD. At low vessel pressures the accumulator completely discharges with a vaulting gas side pressure of approximately 575 psi. The control rod drive accumulators are required to scram the control rods when the reactor pressure is low, and the accumulators retain sufficient stored energy to ensure the complete insertion of the control rods in the required time.

The ball check valve in the drive flange allows reactor pressure to supply the scram force whenever reactor pressure exceeds the supply pressure at the drive. This occurs, due to accumulator pressure decay and inlet line losses, during all scrams at higher vessel pressures. When the reactor is close to or at fully operating pressure, reactor pressure alone will insert the control rod in the required time, although the accumulator does provide additional margin at the beginning of the stroke.

The control rod drive system, with accumulators, provides the following scram performances at full power operation, in terms of average elapsed time after the opening of the reactor protection system trip actuator (scram signal) for the drives to attain the scram strokes listed:

Percent of full stroke	5	20	50	90
Stroke in inches	7.2	28.8	72.0	129.6
Average time in sec	0.375	0.90	2.0	3.5

4.6.1.1.2.6 Instrumentation

The instrumentation for both the control rods and control rod drives is defined by that given for the manual control system. The objective of the reactor manual control system is to provide the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution can be controlled. The system allows the operator to manipulate control rods.

The design bases and further discussion are covered in Chapter 7.

4.6.1.2 Control Rod Drive Housing Supports

4.6.1.2.1 Safety Objective

The control rod drive (CRD) housing supports prevent any significant nuclear transient in the event a drive housing breaks or separates from the bottom of the reactor vessel.

4.6.1.2.2 Safety Design Bases

The CRD housing supports shall meet the following safety design bases:

(1) Following a postulated CRD housing failure, control rod downward motion shall be limited so that any resulting nuclear transient could not be sufficient to cause fuel damage.

(2) The clearance between the CRD housings and the supports shall be sufficient to prevent vertical contact stresses caused by thermal expansion during plant operation.

4.6.1.2.3 Description

The CRD housing supports are shown in Figure 4.6-8. Horizontal beams are installed immediately below the bottom head of the reactor vessel, between the rows of CRD housings. The beams are supported by brackets welded to the steel form liner of the drive room in the reactor support pedestal.

Hanger rods, approximately 10 ft long and 1-3/4 in. in diameter, are supported from the beams on stacks of disc springs. These springs compress approximately 2 inches under the design load.

The support bars are bolted between the bottom ends of the hanger rods. The spring pivots at the top, and the beveled, loose fitting ends on the support bars prevent substantial bending moment in the hanger rods if the support bars are overloaded.

Individual grids rest on the support bars between adjacent beams. Because a single piece grid would be difficult to handle in the limited work space and because it is necessary that control rod drives, position indicators, and in-core instrumentation components be accessible for inspection and maintenance, each grid is designed for in-place assembly or disassembly. Each grid assembly is made from two grid plates, a clamp, and a bolt. The top part of the clamp guides the grid to its correct position directly below the respective CRD housing that it would support in the postulated accident.

When the support bars and grids are installed, a gap of approximately 1 inch at room temperature (approximately 70°F) is provided between the grid and the bottom contact surface of the control rod drive flange. During system heatup, this gap is reduced by a net downward expansion of the housings with respect to the supports. In the hot operating condition, the gap is approximately 1/4 inch.

In the postulated CRD housing failure, the CRD housing supports are loaded when the lower contact surface of the CRD flange contacts the grid. The resulting load is then carried by two grid plates, two support bars, four hanger rods, their disc springs, and two adjacent beams.

The American Institute of Steel Construction (AISC) Manual of Steel Construction, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," was used in designing the CRD housing support system. However, to provide a structure that absorbs as much energy as practical without yielding, the allowable tension and bending stresses used were 90% of yield and the shear stress used was 60% of yield. These design stresses are 1.5 times the AISC allowable stresses (60% and 40% of yield, respectively).

For purposes of mechanical design, the postulated failure resulting in the highest forces is an instantaneous circumferential separation of the CRD housing from the reactor vessel, with an internal pressure of 1250 psig (reactor vessel design pressure) acting on the area of the separated housing. The weight of the separated housing, control rod drive, and blade, plus the pressure of 1250 psig acting on the area of the

separated housing, gives a force of approximately 35,000 lb. This force is multiplied by a factor of 3 for impact, conservatively assuming that the housing travels through a 1-in. gap before it contacts the supports. The total force (105,000 lb) is then treated as a static load in design.

All CRD housing support subassemblies are fabricated of commonly available structural steel, except for the disc springs, which are Schnorr, Type BS-125-71-8.

4.6.2 Evaluations of the CRDS

This subject is covered under nuclear safety and operational analysis (NSOA) in Appendix 15A, Subsection 15A.6.5.3. 22

4.6.2.3 Safety Evaluation

Safety evaluation of the control rods, CRDS, and control rod drive housing supports is described below. Further description of control rods is contained in Section 4.2.

4.6.2.3.1 Control Rods

4.6.2.3.1.1 Materials Adequacy Throughout Design Lifetime

The adequacy of the materials throughout the design life was evaluated in the mechanical design of the control rods. The primary materials,

B_4C powder and 304 austenitic stainless steel, have been found suitable in meeting the demands of the BWR environment.

4.6.2.3.1.2 Dimensional and Tolerance Analysis

Layout studies are done to assure that, given the worst combination of part tolerance ranges at assembly, no interference exists which will restrict the passage of control rods. In addition, preoperational verification is made on each control blade system to show that the acceptable levels of operational performance are met.

4.6.2.3.1.3 Thermal Analysis of the Tendency to Warp

The various parts of the control rod assembly remain at approximately the same temperature during reactor operation, negating the problem of distortion or warpage. What little differential thermal growth could exist is allowed for in the mechanical design. A minimum axial gap is maintained between absorber rod tubes and the control rod frame assembly for the purpose. In addition, dissimilar metals are avoided to further this end.

4.6.2.3.1.4 Forces for Expulsion

An analysis has been performed which evaluates the maximum pressure forces which could tend to eject a control rod from the core. The results of this analysis are given in Subsection 4.6.2.3.2.2.2. In summary, if the collet were to remain open, which is unlikely, calculations indicate that the steady-state control rod withdrawal velocity would be 2 ft/sec for a pressure-under line break, the limiting case for rod withdrawal.

4.6.2.3.1.5 Functional Failure of Critical Components

The consequences of a functional failure of critical components have been evaluated and the results are covered in Subsection 4.6.2.3.2.2.

4.6.2.3.1.6 Precluding Excessive Rates of Reactivity Addition

In order to preclude excessive rates of reactivity addition, analysis has been performed both on the velocity limiter device and the effect of probable control rod failures (see Subsection 4.6.2.3.2.2).

4.6.2.3.1.7 Effect of Fuel Rod Failure on Control Rod Channel Clearances

The control rod drive mechanical design ensures a sufficiently rapid insertion of control rods to preclude the occurrence of fuel rod failures which could hinder reactor shutdown by causing significant distortions in channel clearances.

4.6.2.3.1.8 Mechanical Damage

Analysis has been performed for all areas of the control system showing that system mechanical damage does not affect the capability to continuously provide reactivity control.

In addition to the analysis performed on the control rod drive (Subsection 4.6.2.3.2.2 and Subsection 4.6.2.3.2.3) and the control rod blade, the following discussion summarizes the analysis performed on the control rod guide tube.

The guide tube can be subjected to any or all of the following loads:

(1) Inward load due to pressure differential

(2) Lateral loads due to flow across the guide tube

(3) Dead Weight

(4) Seismic (Vertical and Horizontal)

(5) Vibration

In all cases analysis was performed considering both a recirculation line break and a steam line break. These events result in the largest hydraulic loadings on a control rod guide tube.

Two primary modes of failure were considered in the guide tube analysis; exceeding allowable stress and excessive elastic deformation. It was found that the allowable stress limit will not be exceeded and that the elastic deformations of the guide tube never are great enough to cause the free movement of the control rod to be jeopardized.

4.6.2.3.1.9 Evaluation of Control Rod Velocity Limiter

The control rod velocity limiter limits the free fall velocity of the control rod to a value that cannot result in nuclear system

process barrier damage. This velocity is evaluated by the rod drop accident analysis in Chapter 15.

4.6.2.3.2 Control Rod Drives

4.6.2.3.2.1 Evaluation of Scram Time

The rod scram function of the control rod drive system provides the negative reactivity insertion required by safety design basis in Subsection 4.6.1.1.1.1.1. The scram time shown in the description is adequate as shown by the transient analyses of Chapter 15.

4.6.2.3.2.2 Analysis of Malfunction Relating to Rod Withdrawal

There are no known single malfunctions that cause the unplanned withdrawal of even a single control rod. However, if multiple malfunctions are postulated, studies show that an unplanned rod withdrawal can occur at withdrawal speeds that vary with the combination of malfunctions postulated. In all cases the subsequent withdrawal speeds are less than that assumed in the rod drop accident analysis as discussed in Chapter 15. Therefore, the physical and radiological consequences of such rod withdrawals are less than those analyzed in the rod drop accident.

4.6.2.3.2.2.1 Drive Housing Fails at Attachment Weld

The bottom head of the reactor vessel has a penetration for each control rod drive location. A drive housing is raised into position inside each penetration and fastened by welding. The drive is raised into the drive housing and bolted to a flange at the bottom of the housing. The housing material is seamless, Type 304 stainless steel pipe with a minimum tensile strength of 75,000 psi. The basic failure considered here is a complete circumferential crack through the housing wall at an elevation just below the J-weld.

Static loads on the housing wall include the weight of the drive and the control rod, the weight of the housing below the J-weld, and the reactor pressure acting on the 6-in. diameter cross-sectional area of the housing and the drive. Dynamic loading results from the reaction force during drive operation.

If the housing were to fail as described, the following sequence of events is foreseen. The housing would separate from the vessel. The control rod drive and housing would be blown downward against the support structure by reactor pressure acting on the cross-sectional area of the housing and the drive. The downward motion of the drive and associated parts would be determined by the gap between the bottom of the drive and the support structure and by the deflection of the support structure under load. In the current design, maximum deflection is approximately 3 in. If the collet were to remain latched, no further control rod ejection would occur (Reference 4.6-1); the housing would not drop far enough to clear the vessel penetration. Reactor water would leak at a rate of approximately 220 gpm through the 0.03-inch diametral clearance between the housing and the vessel penetration.

If the basic housing failure were to occur while the control rod is being withdrawn (this is a small fraction of the total drive operating time) and if the collet were to stay unlatched, the following sequence of events is foreseen. The housing would separate from the vessel. The drive and housing would be blown downward against the control rod drive housing support. Calculations indicate that the steady-state rod withdrawal velocity would be 0.3 ft/sec. During withdrawal, pressure under the collet piston would be approximately 250 psi greater than the pressure over it. Therefore, the collet would be held in the unlatched position until driving pressure was removed from the pressure-over port.

4.6.2.3.2.2.2 Rupture of Hydraulic Line(s) to Drive Housing Flange

There are three types of possible rupture of hydraulic lines to the drive housing flange: (1) pressure-under line break; (2) pressure-over line break; and (3) coincident breakage of both of these lines.

4.6.2.3.2.2.2.1 Pressure-under Line Break

For the case of a pressure-under line break, a partial or complete circumferential opening is postulated at or near the point where the line enters the housing flange. Failure is more likely to occur after another basic failure wherein the drive housing or housing flange separates from the reactor vessel. Failure of the housing, however, does not necessarily lead directly to failure of the hydraulic lines.

If the pressure-under line were to fail and if the collet were latched, no control rod withdrawal would occur. There would be no pressure differential across the collet piston and, therefore, no tendency to unlatch the collet. Consequently, the associated control rod could not be withdrawn, but if reactor pressure is greater than 600 psig, it will insert on a scram signal.

The ball check valve is designed to seal off a broken pressure-under line by using reactor pressure to shift the check ball to its upper seat. If the ball check valve were prevented from seating, reactor water would leak to the atmosphere. Because of the broken line, cooling water could not be supplied to the drive involved. Loss of cooling water would cause no immediate damage to the drive. However, prolonged exposure of the drive to temperatures at or near reactor temperature could lead to deterioration of material in the seals. High temperature would be indicated to the operator by the thermocouple in the position indicator probe. A second indication would be high cooling water flow.

If the basic line failure were to occur while the control rod is being withdrawn the hydraulic force would not be sufficient to hold the collet open, and spring force normally would cause the collet to latch and stop rod withdrawal. However, if the collet were to remain open, calculations indicate that the steady-state control rod withdrawal velocity would be 2 ft/sec.

4.6.2.3.2.2.2.2 Pressure-over Line Break

The case of the pressure-over line breakage considers the complete breakage of the line at or near the point where it enters the housing flange. If the line were to break, pressure over the drive piston would drop from reactor pressure to atmospheric pressure. Any significant reactor pressure (approximately 600 psig or greater) would act on the bottom of the drive piston and fully insert the drive. Insertion would occur regardless of the operational mode at the time of the failure. After full insertion, reactor water would leak past the stop piston seals. This leakage would exhaust to the atmosphere through the broken pressure-over line. The leakage rate at 1000 psi reactor pressure is estimated to be 4 gpm nominal but not more than 10 gpm, based on experimental measurements. If the reactor were hot, drive temperature would increase. This situation would be indicated to the reactor operator by the drift alarm, by the fully inserted drive, by a high drive temperature (indicated and printed out on a recorder in the control room), and by operation of the drywell sump pump.

For the simultaneous breakage of the pressure-over and pressure-under lines, pressures above and below the drive piston would drop to zero, and the ball check valve would close the broken pressure-under line. Reactor water would flow from the annulus outside the drive, through the vessel ports, and to the space below the drive piston. As in the case of pressure-over line breakage, the drive would then insert (if the reactor were above 600 psi) at a speed dependent on reactor pressure. Full insertion would occur regardless of the operational mode at the time of failure. Reactor water would leak past the drive seals and out the broken pressure-over line to the atmosphere, as described above. Drive temperature would increase. Indication in the control room would include the drift alarm, the fully inserted drive, the high drive temperature printed out on a recorder in the control room, and operation of the drywell sump pump.

4.6.2.3.2.2.3 All Drive Flange Bolts Fail in Tension

Each control rod drive is bolted to a flange at the bottom of a drive housing. The flange is welded to the drive housing. Bolts are made of AISI-4140 steel, with a minimum tensile strength of 125,000 psi. Each bolt has an allowable load capacity of 15,200 pounds. Capacity of the 8 bolts is 121,600 pounds. As a result of the reactor design pressure of 1250 psig, the major load on all 8 bolts is 30,400 pounds.

If a progressive or simultaneous failure of all bolts were to occur, the drive would separate from the housing. The control rod and the drive would be blown downward against the support structure. Impact velocity and support structure loading would be slightly less than that for drive housing failure, because reactor pressure would act on the drive crosssectional area only and the housing would remain attached to the reactor vessel. The drive would be isolated from the cooling water supply. Reactor water would flow downward past the velocity limiter piston, through the large drive filter, and into the annular space between the thermal sleeve and the drive. For worst-case leakage calculations, the large filter is assumed to be deformed or swept out of the way so it would offer no significant flow restriction. At a point near the top of the annulus, where pressure would have dropped to 350 psi, the water would flash to steam and cause choke-flow conditions. Steam would flow down the annulus and out the space between the housing and the drive flanges to the drywell. Steam formation would limit the leakage rate to approximately 840 gpm.

If the collet were latched, control rod ejection would be limited to the distance the drive can drop before coming to rest on the support structure. There would be no tendency for the collet to

unlatch, because pressure below the collet piston would drop to zero. Pressure forces, in fact, exert 1435 pounds to hold the collet in the latched position.

If the bolts failed during control rod withdrawal, pressure below the collet piston would drop to zero. The collet, with 1650 pounds return force, would latch and stop rod withdrawal.

4.6.2.3.2.2.4 Weld Joining Flange to Housing Fails in Tension

The failure considered is a crack in or near the weld that joins the flange to the housing. This crack extends through the wall and completely around the housing. The flange material is forged, Type 304 stainless steel, with a minimum tensile strength of 75,000 psi. The housing material is seamless, Type 304 stainless steel pipe, with a minimum tensile strength of 75,000 psi. The conventional, full penetration weld of Type 308 stainless steel has a minimum tensile strength approximately the same as that for the parent metal. The design pressure and temperature are 1250 psig and 575°F. Reactor pressure acting on the cross-sectional area of the drive; the weight of the control rod, drive, and flange; and the dynamic reaction force during drive operation result in a maximum tensile stress at the weld of approximately 6000 psi.

If the basic flange-to-housing joint failure occurred, the flange and the attached drive would be blown downward against the support structure. The support structure loading would be slightly less than that for drive housing failure, because reactor pressure would act only on the drive cross-sectional area. Lack of differential pressure across the collet piston would cause the collet to remain latched and limit control rod motion to approximately 3 inches. Downward drive movement would be small, therefore, most of the drive would remain inside the housing. The pressure-under and pressure-over lines are flexible enough to withstand the small displacement and remain attached to the flange. Reactor water would follow the same leakage path described above for the flange bolt failure, except that exit to the drywell would be through the gap between the lower end of the housing and the top of the flange. Water would flash to steam in the annulus surrounding the drive. The leakage rate would be approximately 840 gpm.

If the basic failure were to occur during control rod withdrawal (a small fraction of the total operating time) and if the collet were held unlatched, the flange would separate from the housing. The drive and flange would be blown downward against the support structure. The calculated steady-state rod withdrawal velocity

would be 0.13 ft/sec. Because pressure-under and pressure-over lines remain intact, driving water pressure would continue to the drive, and the normal exhaust line restriction would exist. The pressure below the velocity limiter piston would drop below normal as a result of leakage from the gap between the housing and the flange. This differential pressure across the velocity limiter piston would result in a net downward force of approximately 70 pounds. Leakage out of the housing would greatly reduce the pressure in the annulus surrounding the drive. Thus, the net downward force on the drive piston would be less than normal. The overall effect of these events would be to reduce rod withdrawal to approximately one-half of normal speed. With a 560-psi differential across the collet piston, the collet would remain unlatched; however, it should relatch as soon as the drive signal is removed.

4.6.2.3.2.2.5 Housing Wall Ruptures

This failure is a vertical split in the drive housing wall just below the bottom head of the reactor vessel. The flow area of the hole is considered equivalent to the annular area between the drive and the thermal sleeve. Thus, flow through this annular area, rather than flow through the hole in the housing, would govern leakage flow. The housing is made of Type 304 stainless steel seamless pipe, with a minimum tensile strength of 75,000 psi. The maximum hoop stress of 11,900 psi results primarily from the reactor design pressure (1250 psig) acting on the inside of the housing.

If such a rupture were to occur, reactor water would flash to steam and leak through the hole in the housing to the drywell at approximately 1030 gpm. Choke-flow conditions would exist, as described previously for the flange-bolt failure. However, leakage flow would be greater because flow resistance would be less, that is, the leaking water and steam would not have to flow down the length of the housing to reach the drywell. A critical pressure of 350 psi causes the water to flash to steam.

No pressure differential across the collet piston would tend to unlatch the collet; but the drive would insert as a result of loss of pressure in the drive housing causing a pressure drop in the space above the drive piston.

If this failure occurred during control rod withdrawal, drive withdrawal would stop, but the collet would remain unlatched. The drive would be stopped by a reduction of the net downward force action on the drive line. The net force reduction would occur when the leakage flow of 1030 gpm reduces the pressure in the annulus outside the drive to approximately 540 psig, thereby

reducing the pressure acting on top of the drive piston to the same value. A pressure differential of approximately 710 psi would exist across the collet piston and hold the collet unlatched as long as the operator held the withdraw signal.

4.6.2.3.2.2.6 Flange Plug Blows Out

To connect the vessel ports with the bottom of the ball check valve, a hole of 3/4-inch diameter is drilled in the drive flange. The outer end of this hole is sealed with a plug of 0.812 inch diameter and 0.25 inch thickness. A full-penetration, Type 308 stainless steel weld holds the plug in place. The postulated failure is a full circumferential crack in this weld and subsequent blowout of the plug.

If the weld were to fail, the plug were to blow out, and the collet remained latched, there would be no control rod motion. There would be no pressure differential across the collet piston acting to unlatch the collet. Reactor water would leak past the velocity limiter piston, down the annulus between the drive and the thermal sleeve, through the vessel ports and drilled passage, and out the open plug hole to the drywell at approximately 320 gpm. Leakage calculations assume only liquid flows from the flange. Actually, hot reactor water would flash to steam, and choke-flow conditions would exist. Thus, the expected leakage rate would be lower than the calculated value. Drive temperature would increase and initiate an alarm in the control room.

If this failure were to occur during control rod withdrawal and if the collet were to stay unlatched, calculations indicate that control rod withdrawal speed would be approximately 0.24 ft/sec. Leakage from the open plug hole in the flange would cause reactor water to flow downward past the velocity limiter piston. A small differential pressure across the piston would result in an insignificant driving force of approximately 10 lb, tending to increase withdraw velocity.

A pressure differential of 295 psi across the collet piston would hold the collet unlatched as long as the driving signal was maintained.

Flow resistance of the exhaust path from the drive would be normal because the ball check valve would be seated at the lower end of its travel by pressure under the drive piston.

4.6.2.3.2.2.7 Ball Check Valve Plug Blows Out

As a means of access for machining the ball check valve cavity, a 1.25 inch diameter hole has been drilled in the flange forging. This hole is sealed with a plug of 1.31 inch diameter and 0.38 inch thickness. A full-penetration weld, utilizing Type 308 stainless steel filler, holds the plug in place. The failure postulated is a circumferential crack in this weld leading to a blowout of the plug.

If the plug were to blow out while the drive was latched, there would be no control rod motion. No pressure differential would exist across the collet piston to unlatch the collet. As in the previous failure, reactor water would flow past the velocity limiter, down the annulus between the drive and thermal sleeve, through the vessel ports and drilled passage, through the ball check valve cage and out the open plug hole to the drywell. The leakage calculations indicate the flow rate would be 350 gpm. This calculation assumes liquid flow, but flashing of the hot reactor water to steam would reduce this rate to a lower value. Drive temperature would rapidly increase and initiate an alarm in the control room.

If the plug failure were to occur during control rod withdrawal, (it would not be possible to unlatch the drive after such a failure) the collet would relatch at the first locking groove. If the collet were to stick, calculations indicate the control rod withdrawal speed would be 11.8 feet per second. There would be a large retarding force exerted by the velocity limiter due to a 35 psi pressure differential across the velocity limiter piston.

4.6.2.3.2.2.8 Drive Pressure Control Valve Closure (Reactor Pressure, 0 psig)

The pressure to move a drive is generated by the pressure drop of practically the full system flow through the drive pressure control valve. This valve is either a motor operated valve or a standby manual valve; either one is adjusted to a fixed opening. The normal pressure drop across this valve develops a pressure 260 psi in excess of reactor pressure.

If the flow through the drive pressure control valve were to be stopped, as by a valve closure or flow blockage, the drive pressure would increase to the shutoff pressure of the supply pump. The occurrence of this condition during withdrawal of a drive at zero vessel pressure will result in a drive pressure increase from 260 psig to no more than 1750 psig. Calculations

indicate that the drive would accelerate from 3 in./sec to approximately 6.5 in./sec. A pressure differential of 1670 psi across the collet piston would hold the collet unlatched. Flow would be upward, past the velocity limiter piston, but retarding force would be negligible. Rod movement would stop as soon as the driving signal was removed.

4.6.2.3.2.2.9 Ball Check Valve Fails to Close Passage to Vessel Ports

Should the ball check valve sealing the passage to the vessel ports be dislodged and prevented from reseating following the insert portion of a drive withdrawal sequence, water below the drive piston would return to the reactor through the vessel ports and the annulus between the drive and the housing rather than through the speed control valve. Because the flow resistance of this return path would be lower than normal, the calculated withdrawal speed would be 2 ft/sec. During withdrawal, differential pressure across the collet piston would be approximately 40 psi. Therefore, the collet would tend to latch and would have to stick open before continuous withdrawal at 2 ft/sec, could occur. Water would flow upward past the velocity limiter piston, generating a small retarding force of approximately 120 pounds.

4.6.2.3.2.2.10 Hydraulic Control Unit Valve Failures

Various failures of the valves in the HCU can be postulated, but none could produce differential pressures approaching those described in the preceding paragraphs and none alone could produce a high velocity withdrawal. Leakage through either one or both of the scram valves produces a pressure that tends to insert the control rod rather than to withdraw it. If the pressure in the scram discharge volume should exceed reactor pressure following a scram, a check valve in the line to the scram discharge header prevents this pressure from operating the drive mechanisms.

4.6.2.3.2.2.11 Collet Fingers Fail to Latch

The failure is presumed to occur when the drive withdraw signal is removed. If the collet fails to latch, the drive continues to withdraw at a fraction of the normal speed. This assumption is made because there is no known means for the collet fingers to become unlocked without some initiating signal. Because the

collet fingers will not cam open under a load, accidental application of a down signal does not unlock them. (The drive must be given a short insert signal to unload the fingers and cam them open before the collet can be driven to the unlock position.) If the drive withdrawal valve fails to close following a rod withdrawal, the collet would remain open and the drive continue to move at a reduced speed.

4.6.2.3.2.2.12 Withdrawal Speed Control Valve Failure

Normal withdrawal speed is determined by differential pressures in the drive and is set for a nominal value of 3 in./sec. Withdrawal speed is maintained by the pressure regulating system and is independent of reactor vessel pressure. Tests have shown that accidental opening of the speed control valve to the full-open position produces a velocity of approximately 6 in./sec.

The control rod drive system prevents unplanned rod withdrawal and it has been shown above that only multiple failures in a drive unit and in its control unit could cause an unplanned rod withdrawal.

4.6.2.3.2.3 Scram Reliability

High scram reliability is the result of a number of features of the CRD system. For example:

(1) Two reliable sources of scram energy are used to insert each control rod: individual accumulators at low reactor pressure, and the reactor vessel pressure itself at power.

(2) Each drive mechanism has its own scram and a dual solenoid scram pilot valve so only one drive can be affected if a scram valve fails to open. Both valve solenoids must be deenergized to initiate a scram.

(3) The reactor protection system and the HCUs are designed so that the scram signal and mode of operation override all others.

(4) The collet assembly and index tube are designed so they will not restrain or prevent control rod insertion during scram.

(5) The scram discharge volume is monitored for accumulated water and the reactor will scram before the volume is reduced to a point that could interfere with a scram.

4.6.2.3.2.4 Control Rod Support and Operation

As described above, each control rod is independently supported and controlled as required by safety design bases.

4.6.2.3.3 Control Rod Drive Housing Supports

Downward travel of the CRD housing and its control rod following the postulated housing failure equals the sum of these distances: (1) the compression of the disc springs under dynamic loading, and (2) the initial gap between the grid and the bottom contact surface of the CRD flange. If the reactor were cold and pressurized, the downward motion of the control rod would be limited to the spring compression (approximately 2 in.) plus a gap of approximately 1 in. If the reactor were hot and pressurized, the gap would be approximately 1/4 in. and the spring compression would be slightly less than in the cold condition. In either case, the control rod movement following a housing failure is substantially limited below one drive "notch" movement (6 in.). Sudden withdrawal of any control rod through a distance of one drive notch at any position in the core does not produce a transient sufficient to damage any radioactive material barrier.

The CRD housing supports are in place during power operation and when the nuclear system is pressurized. If a control rod is ejected during shutdown, the reactor remains subcritical because it is designed to remain subcritical with any one control rod fully withdrawn at any time.

At plant operating temperature, a gap of approximately 1/4 in. exists between the CRD housing and the supports. At lower temperatures the gap is greater. Because the supports do not contact any of the CRD housing except during the postulated accident condition, vertical contact stresses are prevented.

4.6.3 Testing and Verification of the CRDs

4.6.3.1 Control Rod Drives

4.6.3.1.1 Testing and Inspection

4.6.3.1.1.1 Development Tests

The development drive (prototype) testing included more than 5000 scrams and approximately 100,000 latching cycles. One prototype was exposed to simulated operating conditions for 5000 hours. These tests demonstrated the following:

(1) The drive easily withstands the forces, pressures, and temperatures imposed.

(2) Wear, abrasion, and corrosion of the nitrided stainless parts are negligible. Mechanical performance of the nitrided surface is superior to that of materials used in earlier operating reactors.

(3) The basic scram speed of the drive has a satisfactory margin above minimum plant requirements at any reactor vessel pressure.

(4) Usable seal lifetimes in excess of 1000 scram cycles can be expected.

4.6.3.1.1.2 Factory Quality Control Tests

Quality control of welding, heat treatment, dimensional tolerances, material verification, and similar factors is maintained throughout the manufacturing process to assure reliable performance of the mechanical reactivity control components. Some of the quality control tests performed on the control rods, control rod drive mechanisms, and hydraulic control units are listed below:

(1) Control rod drive mechanism tests:

a. Pressure welds on the drives are hydrostatically tested in accordance with ASME codes.

b. Electrical components are checked for electrical continuity and resistance to ground.

c. Drive parts that cannot be visually inspected for dirt are flushed with filtered water at high velocity. No significant foreign material is permitted in effluent water.

d. Seals are tested for leakage to demonstrate correct seal operation.

e. Each drive is tested for shim motion, latching, and control rod position indication.

f. Each drive is subjected to cold scram tests at various reactor pressures to verify correct scram performance.

(2) Hydraulic control unit tests:

a. Hydraulic systems are hydrostatically tested in accordance with the applicable code.

b. Electrical components and systems are tested for electrical continuity and resistance to ground.

c. Correct operation of the accumulator pressure and level switches is verified.

d. The unit's ability to perform its part of a scram is demonstrated.

e. Correct operation and adjustment of the insert and withdrawal valves is demonstrated.

4.6.3.1.1.3 Operational Tests

After installation, all rods and drive mechanisms can be tested through their full stroke for operability.

During normal operation, each time a control rod is withdrawn a notch, the operator can observe the in-core monitor indications to verify that the control rod is following the drive mechanism. All control rods that are partially withdrawn from the core can be tested for rod-following by inserting or withdrawing the rod one notch and returning it to its original position, while the operator observes the in-core monitor indications.

To make a positive test of control rod to control rod drive coupling integrity, the operator can withdraw a control rod to the end of its travel and then attempt to withdraw the drive to

the overtravel position. Failure of the drive to overtravel demonstrates rod-to-drive coupling integrity.

Hydraulic supply subsystem pressures can be observed from instrumentation in the control room. Scram accumulator pressures can be observed on the nitrogen pressure gages.

4.6.3.1.1.4 Acceptance Tests

Criteria for acceptance of the individual control rod drive mechanisms and the associated control and protection systems will be incorporated in specifications and test procedures covering three distinct phases: (1) pre-installation, (2) after installation prior to startup, and (3) during startup testing.

The pre-installation specification will define criteria and acceptable ranges of such characteristics as seal leakage, friction and scram performance under fixed test conditions which must be met before the component can be shipped.

The after installation, prestartup tests will be performed as outlined in Chapter 14.

As fuel is placed in the reactor, the power test procedure will be performed as outlined in Chapter 14.

4.6.3.1.1.5 Surveillance Tests

The surveillance requirements for the control rod drive system are described below.

(1) Sufficient control rods shall be withdrawn, following a refueling outage when core alterations are performed, to demonstrate with a margin of 0.25% Δk that the core can be made subcritical at any time in the subsequent fuel cycle with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.

(2) Each partially or fully withdrawn control rod shall be exercised one notch at least once each week.

In the event that operation is continuing with three or more rods valved out of service, this test shall be performed at least once each day.

The weekly control rod exercise test serves as a periodic check against deterioration of the control rod system and also verifies the ability of the control rod drive to scram. If a rod can be moved with drive pressure, it may be expected to scram since higher pressure is applied during scram. The frequency of exercising the control rods under the conditions of three or more control rods valved out of service provides even further assurance of the reliability of the remaining control rods.

(3) The coupling integrity shall be verified for each withdrawn control rod as follows:

a. When the rod is first withdrawn, observe discernible response of the nuclear instrumentation; and

b. When the rod is fully withdrawn the first time, observe that the drive will not go to the overtravel position.

Observation of a response from the nuclear instrumentation during an attempt to withdraw a control rod indicates indirectly that the rod and drive are coupled. The overtravel position feature provides a positive check on the coupling integrity, for only an uncoupled drive can reach the overtravel position.

(4) During operation, accumulator pressure and level at the normal operating value shall be verified.

Experience with control rod drive systems of the same type indicates that weekly verification of accumulator pressure and level is sufficient to assure operability of the accumulator portion of the control rod drive system.

(5) At the time of each major refueling outage, each operable control rod shall be subjected to scram time tests from the fully withdrawn position.

Experience indicates that the scram times of the control rods do not significantly change over the time interval between refueling outages. A test of the scram times at each refueling outage is sufficient to identify any significant lengthening of the scram times.

4.6.3.1.1.6 Functional Tests

The functional testing program of the control rod drives consists of the 5 year maintenance life and the 1.5X design life test programs as described in Subsection 3.9.4.4.

There are a number of failures that can be postulated on the CRD but it would be very difficult to test all possible failures. A partial test program with postulated accident conditions and imposed single failures is available.

The following tests with imposed single failures have been performed to evaluate the performance of the CRDs under these conditions.

Simulated Ruptured Scram Line Test
Stuck Ball Check Valve in CRD Flange
HCU Drive Down Inlet Flow Control Valve (V122) Failure
HCU Drive Down Outlet Flow Control Valve (V120) Failure
CRD Scram Performance with V120 Malfunction
HCU Drive Up Outlet Control Valve (V121) Failure
HCU Drive Up Inlet Control Valve (V123) Failure
Cooling Water Check Valve (V138) Leakage
CRD Flange Check Valve Leakage
CRD Stabilization Circuit Failure
HCU Filter Restriction
Air Trapped in CRD Hydraulic System
CRD Collet Drop Test
CR Qualification Velocity Limiter Drop Test

Additional postulated CRD failures are discussed in Subsections 4.6.2.3.2.2.1 through 4.6.2.3.2.2.11.

4.6.3.2 Control Rod Drive Housing Supports

4.6.3.2.1 Testing and Inspection

CRD housing supports are removed for inspection and maintenance of the control rod drives. The supports for one control rod can be removed during reactor shutdown, even when the reactor is pressurized, because all control rods are then inserted. When the support structure is reinstalled, it is inspected for correct assembly with particular attention to maintaining the correct gap between the CRD flange lower contact surface and the grid.

4.6.4 Information for Combined Performance of Reactivity Systems

4.6.4.1 Vulnerability to Common Mode Failures

The reactivity control systems have been located in accordance with the separation criteria described in Section 3.12. The locations of the equipment for these systems are shown on the figures in Section 1.2.

4.6.4.2 Accidents Taking Credit for Multiple Reactivity Systems

There are no postulated accidents evaluated in Chapter 15 that take credit for two or more reactivity control systems preventing or mitigating each accident.

4.6.5 EVALUATION OF COMBINED PERFORMANCE

As indicated in Subsection 4.6.4.2, credit is not taken for multiple reactivity control systems for any postulated accidents in Chapter 15.

4.6.6 REFERENCES

4.6-1 Benecki, J.E., "Impact Testing on Collet Assembly for Control Rod Drive Mechanism 7RD B144A," General Electric Company, Atomic Power Equipment Department, APED-5555, November 1967.

CHAPTER 5.0

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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CHAPTER 5.0

Tables

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section discusses measures employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) for the plant design lifetime.

5.2.1 COMPLIANCE WITH CODES AND CODE CASES

5.2.1.1 Compliance with 10CFR50, Section 50.55a

A table which shows compliance with the rules of 10CFR50 is included in Section 3.2, (See Table 3.2-1). Code edition, applicable addenda, and component dates are in accordance with 10CFR50.55a. Table 5.2-10 lists those RCPB components which comply with the rules of 10CFR50 in accordance with 10CFR50.55(a)(2)(ii).

5.2.1.2 Applicable Code Cases

The reactor pressure vessel and appurtenances, and the RCPB piping, pumps and valves, have been designed, fabricated, and tested in accordance with the appplicable edition of the ASME Code, including addenda that were mandatory at the order date for the applicable components. Section 50.55a of 10CFR50 requires code case approval only for Class 1 components. These code cases contain requirements or special rules which may be used for the construction of pressure-retaining components of Quality Group Classification A. The various ASME code case interpretations that were applied to components in the RCPB are listed in Table 5.2-1.

5.2.2 OVERPRESSURE PROTECTION

This section provides evaluation of the system that protects the RCPB from overpressurization.

5.2.2.1 Design Basis

Overpressure protection is provided in conformance with 10CFR50, Appendix A, General Design Criteria 15. Preoperational and startup instructions are given in Chapter 14.

5.2.2.1.1 Safety Design Bases

The nuclear pressure-relief system has been designed:

(1) To prevent overpressurization of the nuclear system that could lead to the failure of the reactor coolant pressure boundary.

(2) To provide automatic depressurization for small breaks in the nuclear system occurring with maloperation of the high pressure coolant injection (HPCI) system so that the low pressure coolant injection (LPCI) and the core spray (CS) systems can operate to protect the fuel barrier.

(3) To permit verification of its operability.

(4) To withstand adverse combinations of loadings and forces resulting from normal, upset, emergency and faulted conditions.

5.2.2.1.2 Power Generation Design Bases

The nuclear pressure relief system safety/relief valves have been designed to meet the following power generation bases:

(1) Discharge to the containment suppression pool.

(2) Correctly reclose following operation so that maximum operational continuity can be obtained.

5.2.2.1.3 Discussion

The ASME Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from overpressure under upset conditions. The code allows a peak allowable pressure of 110% of vessel design pressure under upset conditions. The code specifications for safety valves require that: (1) the lowest safety valve set point be set at or below vessel design pressure and (2) the highest safety valve set point be set so that total accumulated pressure does not exceed 110% of the design pressure for upset conditions. The safety/relief valves are designed to open via either of two modes of operation as discussed in Chapter 15. The Safety (spring lift) set points are listed in Table 5.2-2. These setpoints satisfy the ASME Code

specifications for safety valves, because all valves open at less than the nuclear system design pressure of 1250 psig.

The automatic depressurization capability of the nuclear system pressure relief system is evaluated in section 6.3 and in section 7.3.

The following detailed criteria are used in selection of safety relief valves:

(1) Must meet requirements of ASME Code, Section III;

(2) Valves must qualify for 100% of nameplate capacity credit for the overpressure protection function;

(3) Must meet other performance requirements such as response time, etc., as necessary to provide relief functions.

The safety/relief valve discharge piping is designed, installed, and tested in accordance with the ASME Code, Section III.

5.2.2.1.4 Safety Valve Capacity

The safety valve capacity of this plant is adequate to limit the primary system pressure, including transients, to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels (up to and including Summer 1970 Addenda for Unit 1 and Unit 2).

It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure relieving devices may not independently provide complete protection. The safety valve sizing evaluation assumes credit for operation of the scram protective system which may be tripped by one of two sources; i.e., a direct or flux trip signal. The direct scram trip signal is derived from position switches mounted on the main steam line isolation valves or the turbine stop valves or from pressure switches mounted on the dump valve of the turbine control valve hydraulic actuation system. The position switches are actuated when the respective valves are closing and following 10% travel of full stroke. The pressure switches are actuated when a fast closure of the turbine control valves is initiated. Further, no credit is taken for power operation of the pressure relieving devices. Credit is taken for the dual purpose safety/relief valves in their ASME Code qualified (spring lift) mode of safety operation.

The rated capacity of the pressure relieving devices shall be sufficient to prevent a rise in pressure within the protected vessel of more than 110% of the design pressure (1.10 x 1250 psig = 1375 psig) for events defined in Subsection 4.3.1.

Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All combination safety/relief valves discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve; thus providing flow independence to discharge piping losses.

Table 5.2-6 lists the systems which could initiate during the design basis overpressure event.

5.2.2.2 Design Evaluation

5.2.2.2.1 Method of Analysis

To design the pressure protection for the nuclear boiler system, extensive analytical models representing all essential dynamic characteristics of the system are simulated on a large computing facility. These models include the hydrodynamics of the flow loop, the reactor kinetics, the thermal characteristics of the fuel and its transfer of heat to the coolant, and all the principal controller features, such as feedwater flow, recirculation flow, reactor water level, pressure, and load demand. These are represented with all their principal nonlinear features in models that have evolved through extensive experience and favorable comparison of analysis with actual BWR test data.

A detailed description of each model is documented in References 5.2-1 and 5.2-6. Safety/relief valves are simulated in a nonlinear representation, and the model thereby allows full investigation of the various valve response times, valve capacities, and actuation setpoints that are available in applicable hardware systems.

The typical valve characteristic as modeled is shown in Figure 5.2-2 for the spring mode of operation. The associated bypass, turbine control valve, and main steam isolation valve characteristics are also simulated in the model.

5.2.2.2.2 System Design

A parametric study was conducted to determine the required steam flow capacity of the safety/relief valves based on the following assumptions.

5.2.2.2.2.1 Operating Conditions

(1) operating power = 3439 MWt (104.4% of nuclear boiler rated power),

(2) vessel dome pressure ≤ 1020 psig, and

(3) steamflow = 14.153 x 10^6 lb/hr (105% of nuclear boiler rated steamflow)

These conditions are the most severe because maximum stored energy exists at these conditions. At lower power conditions the transients would be less severe.

5.2.2.2.2.2 Transients

The overpressure protection system must accommodate the most severe pressurization transient. There are two major transients, the closure of all main steamline isolation valves and a turbine/generator trip with a coincident failure of the turbine steam bypass system valves that represent the most severe abnormal operational transient resulting in a nuclear system pressure rise. The evaluation of transient behavior with final plant configuration has shown that the isolation valve closure is slightly more severe when credit is taken only for indirect derived scrams, therefore, it is used as the overpressure protection basis event and shown in Figure 5.2-1. Table 5.2-9 lists the sequence of events of the various systems assumed to operate during the main steam line isolation closure with flux scram event. The ODYN results for the same event are shown in Figure 5.2-1A; sequence of events in Table 5.2-9A.

5.2.2.2.2.3 Scram

(1) scram reactivity curve - Figure 5.2-3 for the REDY model and Figure 5.2-1A for the ODYN model.

(2) control rod drive scram motion - Figure 5.2-3

5.2.2.2.2.4 Safety/Relief Valve Transient Analysis Specifications

(1) valve groups:

spring-action safety mode - 5 groups

(2) pressure setpoint (maximum safety limit):

spring-action safety mode - 1177-1217 psig

The set points are assumed at a conservatively high level above the nominal set points. This is to account for initial set point errors and any instrument set point drift that might occur during operation. Typically the assumed setpoints in the analysis are 1 to 2% above the actual nominal set points. Highly conservative safety/relief valve response characteristics are also assumed.

5.2.2.2.2.5 Safety Valve Capacity

Sizing of the safety valve capacity is based on establishing an adequate margin from the peak vessel pressure to the vessel code limit (1375 psig) in response to the reference transients Subsection 5.2.2.2.2.2.

5.2.2.2.3 Evaluation of Results

5.2.2.2.3.1 Safety Valve Capacity

The required safety valve capacity is determined by analyzing the pressure rise from a MSIV closure with flux scram transient. The plant is assumed to be operating at the turbine-generator design conditions at a maximum vessel dome pressure of 1020 psig. The analysis hypothetically assumes the failure of the direct isolation valve position scram. The reactor is shut down by the backup, indirect, high neutron flux scram. For the analysis, the spring-action safety set points are assumed to be in the range of 1177 to 1217 psig. The analysis indicates that the design valve capacity is capable of maintaining adequate margin below the peak ASME code allowable pressure in the nuclear system (1375 psig). Figures 5.2-1 and 5.2-1A show curves produced by this analysis. The sequence of events assumed in these analyses was investigated to meet code requirements and to evaluate the pressure relief system exclusively.

Under the General Requirements for Protection Against Overpressure as given in Section III of the ASME Boiler and Pressure Vessel Code, credit can be allowed for a scram from the reactor protection system. In addition, credit is also taken for the protective circuits which are indirectly derived when determining the required safety valve capacity. The backup reactor high neutron flux scram is conservatively applied as a design basis in determining the required capacity of the pressure relieving safety valves. Application of the direct position scrams in the design basis could be used since they qualify as

acceptable pressure protection devices when determining the required safety valve capacity of nuclear vessels under the provisions of the ASME code.

The parametric relationship between peak vessel (bottom) pressure and safety valve capacity for the MSIV transient with high flux and position trip scram is described in Figure 5.2-4. Also shown in Figure 5.2-4 is the parametric relationship between peak vessel (bottom) pressure and safety valve capacity for the turbine trip with a coincident closure of the turbine bypass valves and direct scram, which is the most severe transient when direct scram is considered. Pressures shown for flux scram will result only with multiple failure in the redundant direct scram system.

The time response of the vessel pressure to the MSIV transient with flux scram and the turbine trip with a coincident closure of the turbine bypass valves and direct scram for 16 valves is illustrated in Figure 5.2-5. The results of the REDY analysis show that the pressure at the vessel bottom exceeds 1250 psig for less than 6 seconds which is not long enough to transfer any appreciable amount of heat into the vessel metal which was at a temperature well below 550°F at the start of the transient. The ODYN results are also shown on these figures. The more extensive model predicts even more safety margin to the allowable pressure limit of the reactor vessel system.

5.2.2.2.3.2 Pressure Drop in Inlet and Discharge

Pressure drop on the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressures. Pressure drop in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent backpressure on each safety/relief valve from exceeding 40% of the valve inlet pressure, thus assuring choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping. Each safety/relief valve has its own separate discharge line.

5.2.2.3 Piping & Instrument Diagrams

Figure 5.1-3 is the P&ID for the Nuclear Boiler System including pressure-relieving devices.

5.2.2.4 Equipment and Component Description

5.2.2.4.1 Description

The nuclear pressure relief system consists of safety/relief valves located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. These valves protect against overpressure of the nuclear system.

The safety/relief valves provide three main protection functions:

(1) Overpressure relief operation. The valves open automatically to limit a pressure rise.

(2) Overpressure safety operation. The valves function as safety valves and open (self-actuated operation if not already automatically opened for relief operation) to prevent nuclear system overpressurization.

(3) Depressurization operation. The ADS valves open automatically as part of the emergency core cooling system (ECCS) for events involving small breaks in the nuclear system process barrier. The location and number of the ADS valves can be determined from Figure 5.1-3.

Chapter 15 discusses the events which are expected to activate the primary system safety/relief valves. The chapter also summarizes the number of valves expected to operate during the initial blowdown of the valves and the expected duration of this first blowdown. For several of the events it is expected that the lowest set safety/relief valve will reopen and reclose as generated heat drops into the decay heat characteristics. The pressure increase and relief cycle will continue with lower frequency and shorter relief discharges as the decay heat drops off and until such time as the RHR system can dissipate this heat. The duration of each relief discharge should in most cases be less than 30 seconds. Remote manual actuation of the valves from the control room is recommended to minimize the total number of these discharges, with the intent of achieving extended valve seat life.

A schematic of the safety/relief valve is shown in Figure 5.2-7. It is opened by either of two modes of operation:

(1) The spring mode of operation which consists of direct action of the steam pressure against a spring-loaded disk that will pop open when the valve inlet pressure force exceeds the spring force. Figure 5.2-6 diagrams the valve lift vs time characteristic.

(2) The power actuated mode of operation which consists of using an auxiliary actuating device consisting of a pneumatic piston/cylinder and mechanical linkage assembly which opens the valve by overcoming the spring force, even with valve inlet pressure equal to zero psig.

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The pneumatic operator is so arranged that if it malfunctions it will not prevent the valve disk from lifting if steam inlet pressure reaches the spring lift set pressure.

For overpressure safety/relief valve operation (self-actuated or spring lift mode), the spring load establishes the safety valve opening setpoint pressure and is set to open at setpoints designated in Table 5.2-2. The ASME code requires that full lift of this mode of operation should be attained at a pressure no greater than 3% above the setpoint.

To prevent backpressure from affecting the spring lift setpoint, each valve is provided with a device to counteract the effects of backpressure which results in the discharge line when the valve is open and discharging steam.

The safety function of the safety/relief valve is a backup to the relief function described below. The spring-loaded valves are designedin accordance with ASME III, NB 7640 as safety valves with auxiliary actuating devices and manufactured in accordance with ASME Section III Class I component requirements.

For overpressure relief valve operation (power actuated mode), each valve is provided with a pressure sensing device which operates at the setpoints designated in Chapter 15. When the set pressure is reached, it operates a solenoid valve which in turn actuates the pneumatic piston/cylinder and linkage assembly to open the valve.

When the piston is actuated, the delay time, maximum elapsed time between receiving the overpressure signal at the valve actuator and the actual start of valve motion, will not exceed 0.1 seconds. The maximum full stroke opening time will not exceed 0.15 seconds.

The safety/relief valves can be operated in the power actuated mode by remote-manual controls from the main control room.

Each safety/relief valve is provided with its own pneumatic accumulator and inlet check valve. The accumulator capacity is sufficient to provide one safety/relief valve actuation, which is all that is required for overpressure protection. Subsequent actuations for an overpressure event can be spring actuations to limit reactor pressure to acceptable levels.

The safety/relief valves are designed to operate to the extent required for overpressure protection in the following accident environments:

(1) 340°F for 3 hours at drywell design pressure

(2) 320°F for an additional 3 hour period, at drywell design pressure

(3) 250°F for an additional 18 hour period, at 25 psig

(4) 200°F during the next 99 days at 20 psig. The duration of operability is two days following which the valves will remain fully open or closed for the remaining time period.

The Automatic Depressurization System (ADS) utilizes selected safety/relief valves for depressurization of the reactor (See Section 7.3) . Each of the safety/relief valves utilized for automatic depressurization is equipped with an air accumulator and check valve arrangement. These accumulators assure that the valves can be held open following failure of the air supply to the accumulators. They are sized to be capable of opening the valves and holding them open against a drywell pressure of 45 psig with the reactor completely depressurized. The accumulator capacity is sufficient for each ADS valve to provide two actuations against 31.5 psig drywell pressure.

Each safety/relief valve discharges steam through a discharge line to a point below the minimum water level in the suppression pool. Safety relief valve discharge line piping from the safety relief valve to the suppression pool consists of two parts. The first part is attached at one end to the safety relief valve and attached at its other end to the containment diaphragm slab through a pipe anchor. The main steam piping, including this portion of the safety relief valve discharge piping, is analyzed as a complete system. This portion of the safety relief valve discharge lines is therefore classified as quality group C and Seismic Category I.

The second part of the safety relief valve discharge piping extends from the upstream anchor to the suppression pool. Because of the upstream anchor on this part of the line, it is physically decoupled from the main steam header and is therefore analyzed as a separate piping system. In analyzing this part of the discharge piping in accordance with the requirements of quality Group C and Seismic Category I, the following load combination will be considered as a minimum:

Pressure and temperature
Dead weight
Fluid dynamic loads due to S/R valve operation
Anchor relative seismic (SSE) movement

Movement of the safety relief valve discharge line will be monitored as a part of the preoperational and startup testing of the main steam lines, in accordance with the requirements of Chapter 14.

The safety/relief valve discharge piping is designed to limit valve outlet pressure to 40% of maximum valve inlet pressure with the valve wide open. Water in the line more than a few feet above suppression pool water level would cause excessive pressure at the valve discharge when the valve is again opened. For this reason, a vacuum relief valve is provided on each safety/relief valve discharge line to prevent drawing an excessive amount of water up into the line as a result of steam condensation following termination of relief operation. The safety/relief valves are located on the main steam line piping, rather than on the reactor vessel top head, primarily to simplify the discharge piping to the pool and to avoid the necessity of having to remove sections of this piping when the reactor head is removed for refueling. In addition, valves located on the steam lines are more accessible during a shutdown for valve maintenance.

The nuclear pressure relief system automatically depressurizes the nuclear system sufficiently to permit the LPCI or CS systems to operate as a backup for the HPCI system. Further descriptions of the operation of the automatic depressurization feature are found in Section 6.3, and in Subsection 7.3.1.1.1.

5.2.2.4.2 Design Parameters

Table 5.2-3 lists design temperature, pressure, and maximum test pressure for the RCPB components. The specified operating transients for components within the RCPB are given in Table 3.9-15. Refer to Section 3.7 for discussion of the input criteria for design of Seismic Category I structures, systems, and components.

The design requirements established to protect the principal components of the reactor coolant system against environmental effects are discussed in Section 3.11.

5.2.2.4.2.1 Safety/Relief Valve

The discharge area of the valve is 16.117 square inches and the coefficient of discharge K is equal to 0.966. The diameter and length of the discharge pipe from each valve to the discharge device in the suppression pool is defined in the Design Assessment Report (DAR), Table 1.3-2. The discharge pipe routing within the suppression chamber is shown in the DAR, Figures 1.3-2 through 1.3-4. The design pressure and temperature of the valve inlet and outlet are 1250 psig @ 575°F and 550 psig @ 500°F, respectively.

Cyclic testing has demonstrated that the valves are capable of at least 60 actuation cycles between required maintenance.

See Figure 5.2-7 for a schematic cross section of the valve.

5.2.2.5 Mounting of Pressure Relief Devices

The pressure relief devices are located on the main steam piping header. The mounting consists of a special, contour nozzle and an over-sized flange connection. This provides a high integrity connection that accounts for the thrust, bending and torsional loadings which the main steam pipe and relief valve discharge pipe are subjected to. This includes:

(1) The thermal expansion effects of the connecting piping.

(2) The dynamic effects of the piping due to SSE.

(3) The reactions due to transient unbalanced wave forces exerted on the safety/relief valves during the first few seconds after the valve is opened and prior to the time steady-state flow has been established. (With steady-state flow, the dynamic flow reaction forces will be self-equilibrated by the valve discharge piping).

(4) The dynamic effects of the piping and branch connection due to the turbine stop valve closure.

In no case will allowable valve flange loads be exceeded nor will the stress at any point in the piping exceed code allowables for any specified combination of loads. The design criteria and analysis methods for considering loads due to SRV discharge is contained in Subsection 3.9.3.3.

5.2.2.6 Applicable Codes and Classification

The vessel overpressure protection system is designed to satisfy the requirements of Section III, Nuclear Vessels, of the ASME Boiler and Pressure Vessel Code. The general requirements for protection against overpressure as given in Article 9 of Section III of the Code recognize that reactor vessel overpressure protection is one function of the reactor protective systems and allows the integration of pressure relief devices with the protective systems of the nuclear reactor. Hence, credit is taken for the scram protective system as a complementary pressure protection device. The NRC has also adopted the ASME Codes as part of their requirements in the Code of Federal Regulations (10CFR50.55A).

5.2.2.7 Material Specification

Pressure retaining components of valves in Quality Group A are constructed only from ASME designated materials.

5.2.2.8 Process Instrumentation

Overpressure protection process instrumentation is shown on Figure 5.1-3.

5.2.2.9 System Reliability

This system is designed to satisfy the requirments of Section III of the ASME Boiler and Pressure Vessel code, therefore, it has high reliability. The consequences of failure are discussed in subsections 15.1.4 and 15.6.1.

5.2.2.10 Inspection and Testing

The safety/relief valves are tested at the vendor's shop in accordance with quality control procedures to detect defects and to prove operability prior to installation. The following tests are conducted:

(1) Hydrostatic test at specified test conditions.

(2) Pneumatic seat leakage test at 90% of set pressure, with maximum permitted leakage of 30 bubbles per minute emitting from a 0.250-in diameter hole submerged 1/2 in. below a water surface or an equivalent test using an approved test medium.

(3) Set pressure test: valve pressurized with saturated steam, with the pressure rising to the valve set pressure. Valve must open at nameplate set pressure ± 1%.

(4) Response time test: each safety/relief valve tested to demonstrate acceptable response time.

The valves are installed as received from the factory. The GE equipment specification requires certification from the valve manufacturer that design and performance requirements have been met. This includes capacity and blowdown requirements. The set

points are adjusted, verified, and indicated on the valves by the vendor. Specified manual and automatic actuation relief mode of each safety/relief valve is verified during the preoperational test program.

It is not feasible to test the safety/relief valve set points while the valves are in place. The valves are mounted on 1500-lb primary service rating flanges. They can be removed for maintenance or bench checks and reinstalled during normal plant shutdowns. The valves will be tested to check set pressure in accordance with the requirements of Chapter 16. The external surface and seating of all safety/relief valves are 100% visually inspected when the valves are removed for maintenance or bench checks. Valve operability is verified during the preoperational test program in accordance with the requirements of Chapter 14.

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

5.2.3.1 Material Specifications

Table 5.2-4 lists the principal pressure retaining materials and the appropriate material specifications for the reactor coolant pressure boundary components.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 PWR Chemistry of Reactor Coolant

Not applicable to BWRs.

5.2.3.2.2 BWR Chemistry of Reactor Coolant

The coolant chemistry requirements discussed in this subsection are consistent with the requirements of Regulatory Guide 1.56 (6/73).

Materials in the primary system are primarily austenitic stainless steel and Zircaloy cladding. The reactor water chemistry limits are established to provide an environment favorable to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it can be continuously and reliably measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress

corrosion cracking of stainless steel. For further information, see Reference 5.2-2.

Several investigations have shown that in neutral solutions some oxygen is required to cause stress corrosion cracking of stainless steel, while in the absence of oxygen no cracking occurs. One of these is the chloride-oxygen relationship of Williams (Reference 5.2-3), where it is shown that at high chloride concentration little oxygen is required to cause stress corrosion cracking of stainless steel, and at high oxygen concentration little chloride is required to cause cracking.

These measurements were determined in a wetting and drying situation using alkaline-phosphate-treated boiler water and, therefore, are of limited significance to BWR conditions. They are, however a qualitative indication of trends.

The water quality requirements are further supported by General Electric stress corrosion test data summarized as follows:

- Type 304 stainless steel specimens were exposed in a flowing loop operating at 537°F. The water contained 1.5 ppm chloride and 1.2 ppm oxygen at pH 7. Test specimens were bent beam strips stressed over their yield strength. After 2100 hours exposure, no cracking or failures occurred.

- Welded Type-304 stainless steel specimens were exposed in a refreshed autoclave operating at 550°F. The water contained 0.5 ppm chloride and 1.5 ppm oxygen at pH 7. Uniaxial tensile test specimens were stressed at 125% of their 550°F yield strength. No cracking or failures occurred at 15,000 hours exposure.

When conductivity is in its normal range, pH, chloride and other impurities affecting conductivity will also be within their normal range. When conductivity becomes abnormal, chloride measurements are made to determine whether or not they are also out of their normal operating values. Conductivity could be high due to the presence of a neutral salt which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are high because of the purposeful use of additives. In BWRs, however, where no additives are used and where near neutral pH is maintained, conductivity provides a good and prompt measure of the quality of the reactor water. Significant changes in conductivity provide the operator with a warning mechanism so he can investigate and remedy the condition before reactor water limits are reached. Methods available to the operator for correcting the off-standard condition included operation of the reactor water cleanup system, reducing the input of impurities, and placing the reactor in the cold shutdown

condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the cleanup system to reestablish the purity of the reactor coolant.

The following is a summary and description of BWR water chemistry for various plant conditions.

(1) Normal Plant Operation

The BWR system water chemistry is conveniently described by following the system cycle as shown on Figure 5.2-8. Reference to Table 5.2-5 has been made as numbered on the diagram and correspondingly in the table.

For normal operation starting with the condenser-hotwell, condensate water is processed through a condensate treatment system. This process consists of filtration and demineralization, resulting in effluent water quality represented in Table 5.2-5.

The effluent from the condensate treatment system is pumped through the feedwater heater train, and enters the reactor vessel at an elevated temperature and with a chemical composition typically as shown in Table 5.2-5.

During normal plant operation, boiling occurs in the reactor, decomposition of water takes place due to radiolysis, and oxygen and hydrogen gas is formed. Due to steam generation, stripping of these gases from the water phase takes place, and the gases are carried with the steam through the turbine to the condenser. The oxygen level in the steam, resulting from this stripping process, is typically observed to be about 20 ppm (see Table 5.2-5). At the condenser, deaeration takes place and the gases are removed from the process by means of steam jet air ejectors (SJAEs). The deaeration is completed to a level of approximately 20 ppb (0.02 ppm) oxygen in the condensate.

The dynamic equilibrium, in the reactor vessel water phase, established by the steam-gas stripping and the radiolytic formation (principally) rates, corresponds to a nominal value of approximately 200 ppb (0.2 ppm) of oxygen at rated operating conditions. Slight variations around this value have been observed as a result of differences in neutron flux density, core-flow and recirculation flow rate.

A reactor water cleanup system is provided for removal of impurities resulting from fission products formed in the primary system. The cleanup process consists of

filtration and ion exchange, and serves to maintain a high level of water purity in the reactor coolant.

Typical chemical parametric values for the reactor water are listed in Table 5.2-5 for various plant conditions.

Additional water input to the reactor vessel originates from the Control Rod Drive (CRD) cooling water. The CRD water is essentially feedwater quality. Separate filtration for purification and removal of insoluble corrosion products takes place within the CRD system prior to entering the drive mechanisms and reactor vessel.

No other inputs of water or sources of oxygen are present during normal plant operation. During plant conditions other than normal operation additional inputs and mechanisms are present as outlined in the following section.

(2) Plant Conditions Outside Normal Operation

During periods of plant conditions other than normal power production transients take place, particularly with regards to the oxygen levels in the primary coolant. Systems other than the reactor are not affected significantly to impact primary system components or subsequent operation. In essence, depending on what the plant condition is, i.e., hot standby with/without reactor vessel venting or plant shutdown, the hotwell condensate will absorb oxygen from the air when vacuum is broken on the condenser. Prior to startup and input of feedwater to the reactor, vacuum is established in the condenser and deaeration of the condensate takes place by means of mechanical vacuum pump and steam jet air ejector (SJAE) operation and condensate recirculation. During these plant conditions, continuous input of control rod drive (CRD) cooling water takes place as described previously.

a) Plant Depressurized and Reactor Vented

During certain periods such as during refueling and maintenance outages, the reactor is vented to the condenser or atmosphere. Under these circumstances the reactor cools and the oxygen concentration increases to a maximum value of 8 ppm. Equilibrium between the atmosphere above the reactor water surface, the CRD cooling water input, any residual radiolytic effects, and the bulk reactor water will

be established after some time. No other changes in water chemistry of significance take place during this plant condition because no appreciable inputs take place.

b) Plant Transient Conditions Plant Startup/Shutdown

During these conditions, no significant changes in water chemistry other than oxygen concentration take place.

(i) Plant Startup

Depending on the duration of the plant shutdown prior to startup and whether the reactor has been vented, the oxygen concentration could be that of air saturated water, i.e., ∿8 ppm oxygen.

Following nuclear heatup initiation, the oxygen level in the reactor water will decrease rapidly as a function of water temperature increase and corresponding oxygen solubility in water. The oxygen level will reach a minimum of about 20 ppb (0.02 ppm) at a coolant temperature of about 380°F, at which point an increase will take place due to significant radiolytic oxygen generation. For the elapsed process up to this point the oxygen is degassed from the water and is displaced to the steam dome above the water surface.

Further increase in power increases the oxygen generation as well as the temperature. The solubility of oxygen in the reactor water at the prevailing temperature controls the oxygen level in the coolant until rated temperature (∿ 540°F) is reached. Thus, a gradual increase from the minimum level of 20 ppb to a maximum value of about 200 ppb oxygen takes place. At, and after this point (540°F) steaming and the radiolytic process control the coolant oxygen concentration to a level of around 200 ppb.

(ii) Plant Shutdown

Upon plant shutdown following power operation, the radiolytic oxygen generation essentially ceases as the fission process is terminated.

Because oxygen is no longer generated, while some steaming still will take place due to residual energy, the oxygen concentration in the coolant will decrease to a minimum value determined by steaming rate temperature. If venting is performed, a gradual increase to essentially oxygen saturation at the coolant temperature will take place, reaching a maximum value of < 8 ppm oxygen.

(iii) Oxygen in Piping and Parts Other Than the Reactor Vessel Proper

As can be concluded from the preceding descriptions, the maximum possible oxygen concentraton in the reactor coolant and any other directly related or associated parts is that of air saturation at ambient temperature. At no time or location, in the water phase, will oxygen levels exceed the nominal value of 8 ppm. As temperature is increased and hence, oxygen solubility decreased accordingly, the oxygen concentration will be maintained at this maximum value, or reduced below it depending on available removal mechanisms, i.e., diffusion, steam stripping, flow transfer or degassing.

Depending on the location, confirguration, etc., such as dead legs or stagnant water, inbentories may contain < 8 ppm dissolved oxygen or some other value below this maximum limitation.

Conductivity is continuously monitored on the primary coolant with instruments connected to redundant sources, the reactor water recirculation loop and the reactor water cleanup system inlet. The effluent from the reactor water clenaup system is also monitored for conductivity on a continuous basis. These measurements provide reasonable assurance for adequate surveillance of the reactor coolant.

Grab samples are provided, for the locations shown on Table 5.2-7, for special and non-continous measurements such as pH, oxygen, chloride and radiochemical measurements.

The relationship of chloride concentration to specific conductance measured at 25°C for chloride compounds such as sodium chloride and hydrochloric

acid can be calculated, see Figure 5.2-9. Values for these compounds essentially bracket values of other common chloride salts or mixtures at the same chloride concentration. Surveillance requirements are based on these relationships.

In addition to this program, limits, monitoring and sampling requirements are imposed on the condensate, condensate treatment system and feedwater by warranty requirements and specifications. Thus, a total plant water quality surveillance program is established providing assurance that off specification conditions will quickly be detected and corrected.

The sampling frequency when reactor water has a low specific conductance is adequate for calibration and routine audit purposes. When specific conductance increases, and higher chloride concentrations are possible, or when continuous conductivity monitoring is unavailable, increased sampling is provided.

For the higher than normal limits of < 1 μ mho/cm , more frequent sampling and analyses are invoked by the coolant chemistry surveillance program, see Table 5.2-5.

The primary coolant conductivity monitoring instrumentation, ranges, accuracy sensor and indicator locations are shown in Table 5.2-7. The sampling is coordinated in a reactor sample station especially designed with constant temperature control and sample conditioning and flow control equipment.

c. Water Purity During a Condenser Leakage

The condensate cleanup system is designed to maintain the reactor water chloride concentration below 200 ppb during a condenser tube leak of 23 gallons per minute indefinitely. The condensate cleanup system will sustain an effluent conductivity of 0.15 micromho with a 46 gpm condenser leak when the circulating water contains 1000 ppm of TDS. Refer to Subsection 10.4.6.

To protect against a major condenser tube leak, sufficient instrumentation is provided to maintain a reserve of 50 percent of the theoretical ion exchange capacity during normal operation per Regulatory Guide 1.56.

5.2.3.2.3 Compatibility of Construction Materials with Reactor Coolant

The materials of construction exposed to the reactor coolant consist of the following:

(1) Solution annealed austenitic stainless steels (both wrought and cast) Types 304, 304L, 316 and 316L.

(2) Nickel base alloys - Inconel 600 and Inconel 750X.

(3) Carbon steel and low alloy steel.

(4) Some 400 series martensitic stainless steel (all tempered at a minimum of 1100°F).

(5) Colmonoy and Stellite hardfacing material.

All of these materials of construction are resistant to stress corrosion in the BWR coolant. General corrosion on all materials, except carbon and low alloy steel, is negligible. Conservative corrosion allowances are provided for all exposed surfaces of carbon and low alloy steels.

5.2.3.2.4 Compatibility of Construction Materials with External Insulation and Reactor Coolant

The materials of construction exposed to external insulation are:

(1) Solution annealed austentitic stainless steels. Types 304, 304L and 316.

(2) Carbon and low alloy steel.

Two types of external insulation are employed on BWRs. Reflective metal insulation used does not contribute to any surface contamination and has no effect on construction materials. Nonmetallic insulation used on stainless steel piping and components complies with the requirements of the following industry standards:

(1) ASTM C692-71, Standard Methods for Evaluating Stress Corrosion Effects of Wicking Type Thermal Insulation on Stainless Steel (Dana Test).

(2) RDT-M12-1T, Test Requirements for Thermal Insulating Materials for Use on Austenitic Stainless Steel, Section 5 (KAPL Test).

Chemical analyses are required to verify that the leachable sodium, silicate, and chloride are within acceptable levels. Insulation is packaged in waterproof containers to avoid damage or contamination during shipment and storage.

Since there are no additives in the BWR coolant, leakage would expose materials to high purity, demineralized water. Exposure to demineralized water would cause no detrimental effects.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

Fracture toughness requirements for the ferritic materials used for pumps piping and valves of the reactor coolant pressure boundary were as follows:

The pump components except for the bolting, are austenitic stainless steel. The bolting meets Section III of ASME B&PV Code, Summer 1971 Addenda which requires impact testing to be performed at 10°F.

Safety/Relief Valves were exempted from fracture toughness requirements because Section III of the 1971 ASME Boiler and Pressure Vessel Code did not require impact testing on valves with inlet connnections of 6 inches or less nominal pipe size.

Main Steam Isolation Valves were also exempted because the Code existing at the time of the purchase, ASME Section III Summer 1971 Addenda did not require brittle fracture testing on ferritic pressure boundary components when the system temperature was in excess of 250°F at 20% of the design pressure.

Main Steam Piping was tested in accordance with and met the fracture toughness requirements of paragraph NB-2300 of the 1972 Summer Addenda to ASME Code, Section III, the applicable code at the time of the purchase order.

5.2.3.3.1.1 Compliance with Code Requirements

The ferritic pressure boundary material of the reactor pressure vessel was qualified by impact testing in accordance with the 1968 Edition of Section III ASME Code and Addenda to and including the Summer 1970 Addenda. From an operational standpoint, this Code would require that for any significant pressurization (taken to be more than 20% of Code hydrostatic test pressure = 312 psig) the minimum metal temperature of all vessel shell and head material be 100°F (NDTT +60°F).

5.2.3.3.1.2 Acceptable Fracture Energy Levels

Operating limits on reactor vessel pressure and temperature during normal heatup and cooldown, and during inservice hydrostatic testing, were established using as a guide Appendix G, Summer 1972 Addenda, of Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition.

These operating limits will assure that a large postulated surface flaw, having a depth of one-quarter of the material thickness, can be safely accommodated in regions of the vessel shell remote from discontinuities. In addition the specific additional margins required by 10CFR50, Appendix G, paragraph IV.A.2.c are included in the operating limits for core operations.

For the purpose of setting these operating limits, the reference temperature, RT_{NDT}, was determined from the impact test data taken in accordance with requirements of the Code to which this vessel is designed and manufactured. The dropweight NDT temperature was used as the reference temperature.

The highest reference temperature of any part of the reactor pressure vessel pressure boundary material was used as the reference temperature for calculating one set of operating temperature and pressure limits for the shell remote from the core beltline region. A second set of temperature and pressure limits for the core beltline region was calculated based on the core beltline region material reference temperature.

The requirements of the Code to which the vessel was designed and manufactured results in a third set of vessel shell temperature pressure limits; namely, NDTT +60F or CVN +60F at pressure greater than 20% of preoperational system hydrostatic test pressure. The more conservative of the above three limits was used to set pressure and temperature limits for the vessel shell.

5.2.3.3.1.3 Operating Limits During Heatup, Cooldown, and Core Operation

Since 100°F/hour is the maximum average normal heatup or cooldown rate for which the reactor vessel is designed, a conservative fracture toughness analysis was done for this assumed rate.

The maximum temperature gradient through the wall corresponding to this rate was considered. The results of this analysis are a set of operating limits for non-nuclear heatup or cooldown following nuclear shutdown, and another set for operating limits for operation whenever the core is critical (except for low level physics tests).

5.2.3.3.1.4 Temperature Limits for ISI Hydrostatic or Leak Pressure Tests

The fracture toughness analysis for pressure tests resulted in the curves shown on Figure 5.3-4 of minimum vessel shell and head temperatures versus vessel pressure as measured in vessel top head. The dashed line curve, beltline region, is based on an assumed initial RT_{NDT} of +10°F, the predicted shift in the RT from Figure 5.3-5 based on neutron fluence at 1/4 of vessel wall thickness must be added to the beltline curve to account for the effect of fast neutrons on the beltline material properties. The curve for areas remote from the beltline (upper curve) is based on an assumed RT_{NDT} of +40°F. The controlling minimum temperature for a desired pressure is then selected as the greater of the solid curve or the dashed curve plus the shift.

5.2.3.3.1.5 Temperature Limits for Boltup

Minimum closure flange and closure stud temperatures of 70°F (NDTT + 60°F) are required whenever the closure studs are under preload or are being tensioned.

5.2.3.3.1.6 Reactor Vessel Annealing

In-place annealing of the reactor vessel because of radiation embrittlement is unnecessary since the predicted value in transition of adjusted reference temperature will not exceed 200°F - see 10CFR50, Appendix G, Paragraph IV.C.

5.2.3.3.2 Control of Welding

5.2.3.3.2.1 Control of Preheat Temperature Employed for Welding of Low Alloy Steel. Regulatory Guide 1.50 (Rev. 0)

The use of low alloy steel is restricted to the reactor pressure vessel. Other ferritic components in the reactor coolant pressure boundary are fabricated from carbon steel materials.

Preheat temperatures employed for welding of low alloy steel meet or exceed the recommendations of ASME Section III, Subsection NA. Components were either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat was maintained until post weld heat treatment. The minimum preheat and maximum interpass temperatures were specified and monitored.

All welds were nondestructively examined by radiographic methods. In addition, a supplemental ultrasonic examination was performed.

5.2.3.3.2.2 Control of Electroslag Weld Properties Regulatory Guide 1.34. (Rev. 0)

No electroslag welding was performed on BWR components.

5.2.3.3.2.3 Welder Qualification for Areas of Limited Accessibility. Regulatory Guide 1.71 (Rev. 0)

For non-NSSS items, refer to response to Regulatory Guide 1.71 in Section 3.13.

There are few restricted access welds involved in the fabrication of NSSS reactor coolant pressure boundary components. Welder qualification for welds with the most restricted access was accomplished by mock-up welding. Mock-ups were examined with radiography or sectioning.

5.2.3.3.3 Nondestructive Examination of Ferritic Tubular Products

For non-NSSS items, refer to response to Regulatory Guide 1.66 in Section 3.13.

Wrought tubular products were supplied in accordance with applicable ASTM/ASME material specifications. These specifications require a hydrostatic test on each length of tubing or pipe.

These components met the requirements of the ASME Codes existing at the time of placement of order which predate Regulatory Guide 1.66. (Rev.0)

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

For non-NSSS items, refer to response to Regulatory Guide 1.44 in Section 3.13

5.2.3.4.1 Avoidance of Stress Corrosion Cracking

5.2.3.4.1.1 Avoidance of Significant Sensitization

All austenitic stainless steel was purchased in the solution heat treated condition in accordance with applicable ASME and ASTM specifications. Carbon content was limited to 0.08% maximum, and cooling rates from solution heat treating temperatures were required to be rapid enough to prevent sensitization.

Welding heat input was restricted to 110,000 joules per inch maximum, and interpass temperature to 350°F. High heat welding processes such as block welding and electroslag welding were not permitted. All weld filler metal and castings were required by specification to have a minimum of 5% ferrite.

Whenever any wrought austenitic stainless steel was heated to temperatures over 800°F, by means other than welding or thermal cutting, the material was re-solution heat treated.

These controls were used to avoid severe sensitization. Compliance with Regulatory Guide 1.44 (5/73) is discussed in Section 3.13.

5.2.3.4.1.2 Process Controls to Minimize Exposure to Contaminants

Exposure to contaminants capable of causing stress corrosion cracking of austenitic stainless steel components was avoided by carefully controlling all cleaning and processing materials which contact the stainless steel during manufacture and construction.

Special care was exercised to insure removal of surface contaminants prior to any heating operations. Water quality for cleaning, rinsing, flushing, and testing was controlled and monitored. Suitable packaging and protection was provided for components to maintain cleanliness during shipping and storage.

The degree of surface cleanliness obtained by these procedures meets the requirements of Regulatory Guides 1.44 (5/73) and 1.37 (3/73).

5.2.3.4.1.3 Cold Worked Austenitic Stainless Steels

Austenitic stainless steels with a yield strength greater than 90,000 psi are not used.

5.2.3.4.2 Control of Welding

For non-NSSS items, refer to response to Regulatory Guide 1.31 in Section 3.13.

5.2.3.4.2.1 Avoidance of Hot Cracking

All austenitic stainless steel filler materials were required by specification to have a minimum of 5% ferrite. This amount of ferrite is considered adequate to prevent hot cracking in austenitic stainless steel welds.

An extensive test program performed by General Electric Company, with the concurrence of the Regulatory Staff, has demonstrated that controlling weld filler metal ferrite at 5% minimum produces production welds which meet the requirements of Regulatory Guide 1.31, (Rev. 1). A total of approximately 400 production welds in five BWR plants were measured and all welds met the requirements of the Interim Regulatory Position to Regulatory Guide 1.31.

5.2.3.4.2.2 Electroslag Welds

Electroslag welding was not employed for reactor coolant pressure boundary components.

5.2.3.4.2.3 Welder Qualification for Areas of Limited Accessibility. Regulatory Guide 1.71. (Rev. 0)

For non-NSSS items, refer to response to Regulatory Guide 1.71 in Section 3.13.

There are few restrictive welds involved in the fabrication of NSSS reactor coolant pressure boundary components. Welder qualification for welds with the most restrictive access was accomplished by mock-up welding. Mock-ups were examined with radiography or sectioning.

5.2.3.4.3 Nondestructive Examination of Tubular Products. Regulatory Guide 1.66. (Rev. 0)

For non-NSSS items, refer to response to REgulatory Guide 1.66 in Section 3.13.

Wrought tubular products were supplied in accordance with applicable ASTM/ASME material specifications. These

specifications require a hydrostatic test on each length of tubing. Additionally, the specification for the tubular product used for CRD housings specified ultrasonic examination to paragraph NB-2550 of ASME Code Section III.

These components met the requirements of ASME Codes existing at time of placement of order.

5.2.4 IN-SERVICE INSPECTION AND TESTING OF REACTOR COOLANT PRESSURE BOUNDARY

The construction permits for the Susquehanna SES were issued in November, 1973. Relating this date to the requirements of 10CFR50.55a(g), the preservice examination program with provisions for design and access should comply, as a minimum, with the 1971 Edition of the ASME B&PV Code Section XI including the Summer 1972 Addenda. The Susquehanna SES preservice examination program will not be conducted to the minimum requirements of 10CFR50.55a(g) but rather to the more current 1974 Edition of Section XI including the Winter 1975 Addenda for the RPV and the Summer 1975 addenda as modified by Appendix III from the Winter 1975 addenda and IWA-2232 from the Summer 1976 addenda for the piping systems to the extent practical within the limitations of design, geometry, and materials of construction of the component.

Throughout the service life of the Susquehanna SES, components and their supports classified as ASME Code Class 1, Class 2 or Class 3, except for components excluded under IWB-1220, IWC-1220, and IWD-2600(c), will meet the requirements, except design and access provisions, set forth in Editions of Section XI of the ASME B&PV Code and Addenda that became effective subsequent to the editions specified above and are incorporated by reference in 10 CFR 50.55 a(g), and to the extent practical within the limitations of design, geometry, and materials of construction of the component.

The initial in-service examinations conducted during the first 40 months will comply, to the extent practical, with the requirements of the ASME B&PV Code Section XI and Addenda incorporated by reference in 10CFR50.55a (g) and in effect no more than six months prior to the starting date of each unit of Susquehanna SES commercial operation.

The in-service examinations conducted during successive 40-month periods throughout the service life of the Susquehanna SES will comply, to the extent practical with the requirements of the ASME B&PV Code Section XI and Addenda incorporated by reference in 10CFR 50.55 a(g) and in effect no more than six months prior to the start of each 40-month period.

5.2.4.1 System Boundary Subject to Inspection

The inspection requirements of Section XI of the Code are met for all Class 1 pressure-containing components (and their supports) except for components excluded under IWB-1220 of Section XI. The system boundary includes all pressure vessels, piping, pumps, and valves that are part of the reactor coolant system, or connected to the reactor coolant system, up to and including:

a) The outermost containment isolation valve in system piping that penetrates the primary reactor containment

b) The second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment

c) The reactor coolant system safety and relief valves.

5.2.4.2 Accessibility

The design and arrangement of system components are in accordance with IWA-1500, "Accessibility", of the 1971 Edition of Section XI. Adequate clearances for general access are provided as follows:

a) Sufficient space is provided for personnel and equipment to perform inspections.

b) Provisions are made for the removal and storage of structural members, shielding components, and insulating materials, to permit access to the components being inspected.

c) Provisions are made for hoists and other handling machinery needed to handle items in (b), above.

d) Provisions are made for alternative examinations if structural defects or indications reveal that such examinations are required.

e) Provisions are made for the necessary operations associated with repair or replacement of system components and piping.

Piping systems requiring volumetric ultrasonic inspection are designed so that welds requiring inspection are physically accessible for inspection and ultrasonic equipment. Access is provided by leaving adequate space around pipes at these welds and by removing insulation and shielding as required.

The surfaces of welds requiring ultrasonic examination have been ground and contoured to permit effective use of ultrasonic transducers, and to minimize geometric reflectors that could be misinterpreted as flaws.

Piping systems requiring surface or visual examination are designed to allow access and visibility adequate for performance of such examinations.

Access is provided to reactor vessel components to meet, as a minimum, the examination requirements of ASME Section XI as outlined above.

Because high potential radiation levels in the vicinity of the reactor vessel limit access to the vessel, considerations for meeting ASME Section XI have been incorporated into the plant design as follows:

a) An annular space (8-in. minimum) sufficient to accommodate remotely operated inspection equipment is provided between the reactor vessel shell and the thermal insulation for areas behind the reactor shield wall.

b) Removable sections of thermal insulation and openings in the reactor shield with hinged shield plugs are provided to allow access for remote or manual examination of the reactor vessel nozzle-to-shell, nozzle-to-safe-end, and safe-end-to-pipe welds.

c) Access to full penetration vessel welds, nozzle welds above the reactor shield, and all top head welds is provided by removable, freestanding thermal insulation.

d) Openings in the reactor shield and removable insulation are provided to allow access to the reactor skirt-to-bottom head welds.

e) Openings in the reactor skirt, removable insulation panels, and walk-on grating are provided to allow access to the bottom head welds inside the support skirt.

f) Remote visual examination of the required number of patches on the interior clad surface of the reactor vessel will be performed in accordance with the requirements of ASME Section XI.

g) The reactor vessel closure head is stored dry in an accessible area to provide direct access for inspection.

h) Reactor vessel studs, nuts, and washers are removed to dry storage for inspection.

In-service inspection access to other major reactor coolant system components is provided as follows:

a) Working platforms are provided to facilitate access to inspection areas.

b) The insulation covering component and piping welds and adjacent base metal is designed for easy removal and reinstallation in areas where inspection is required.

c) The physical arrangement of pipe, pumps, valves, and other components allows personnel access to welds requiring in-service inspection in accordance with ASME Section XI.

5.2.4.3 Examination Techniques and Procedures

The methods, techniques, and procedures used in the Susquehanna SES in-service inspection program comply with the requirements of ASME Section XI, Subarticle IWA-2200.

The visual, surface, and volumetric examination techniques are in compliance with IWA-2210, 2220, and 2230, respectively. If any alternative examination methods, combination of methods, or newly developed techniques are substituted for the above-described methods, results will be provided that demonstrate that the alternative methods are equivalent to or superior to those methods specified in Section XI.

If, as a result of the preservice or in-service examinations, flaw indications are found to have developed and/or propagated beyond the acceptance standards of IWB-3000, then further examinations will be conducted, as needed, to determine the exact condition. Following evaluation of this evidence, a decision will be made regarding repair requirements related to plant safety. Any repairs, if needed, will be performed to the rules of IWB-4000.

5.2.4.4 Inspection Intervals

In-service inspections will be performed during plant outages such as refueling shutdowns or maintenance shutdowns. With the exception of the examinations that may be deferred until the end of the inspection interval, the required examinations will be completed in accordance with IWB-2412, (Inspection Program B) for the vessel (Winter 1975 addenda) and IWB-2411 (Regular Program) for piping (1974 edition).

Discussed below is the first inspection interval program that will be performed on the Susquehanna SES Units 1 and 2 Class I systems. Each system is discussed in categories corresponding to ASME Section XI Editions and Addenda specified in Subsection 5.2.4. This discussion includes the area and extent of examination of each category.

A combination of manual and mechanized techniques will be used for in-service examinations. Preservice (or baseline) data will be generated accordingly using the technique that will be used for in-service examinations.

The detailed preservice and in-service inspection programs are presented in the technical specifications, Section 3/4.4.8 of Chapter 16.

5.2.4.4.1 Reactor Vessel

Item 1.1, Category A - Pressure Retaining Welds in Reactor Vessel

In-service examination of pressure retaining welds in the reactor vessel includes manual ultrasonic examination of 100 percent of the accessible length of each of the following welds:

a) Meridional, circumferential, and radial welds in the vessel heads

b) Vessel shell-to-flange and closure head-to-flange welds.

c) Longitudinal and circumferential welds in the vessel located above the reactor shield wall up to, but excluding the vessel-to-flange weld (vessel shell course Nos. 4 and 5).

In-service examination of pressure retaining welds in the reactor vessel also includes remote mechanized ultrasonic examination of 100 percent of the accessible length of each of the following welds:

a) Longitudinal and circumferential welds in the core region (vessel shell course No. 2)

b) Longitudinal and circumferential welds in the vessel located behind the reactor shield wall (vessel shell course Nos. 1 and 3)

Item 1.4, Category D - Full Penetration Welds of Nozzles in Vessels

The vessel nozzle welds and nozzle to vessel inside radiused section will be ultrasonically examined by remote mechanized equipment for nozzle diameters 10 in. and larger, and by manual ultrasonic techniques for nozzles smaller than 10 in. that are not excluded by IWB-1220.

The examination of each nozzle will cover 100 percent of the nozzle-to-vessel welds and 100 percent of the inner radius section of the nozzle-to-vessel junctures.

Item 1.5, Category E - Pressure Retaining Partial Penetration Welds in Vessels

The 185 control rod drive penetrations, 55 in-core penetrations, one drain penetration, and one core differential pressure and liquid control penetration for each unit are included in this category. The area around each of these penetrations will be visually examined when the system boundary is subjected to a pressure test. The examination will include 25 percent of each group of penetrations of comparable size and function.

Item 1.6, Category F - Pressure Retaining Dissimilar Metal Welds

The vessel nozzle-to-safe-end welds will be ultrasonically examined by remote mechanized equipment for nozzle diameters 10 in. and larger and by manual ultrasonic techniques for nozzles smaller than 10 in. that are not excluded by IWB-1220. Nozzle-to-safe-end welds will also be examined for surface indications using dye penetrants.

The examination will cover 100 percent of the welds.

Items 1.7, 1.8, 1.9, 1.10, Category G-1 - Pressure Retaining Bolting, Larger than 2 in. in Diameter

The closure studs and nuts will be surface examined when they are disassembled for removal of the vessel head. The closure washers and threads in the flange stud holes will be visually examined. The vessel studs and flange ligaments between threaded stud holes in the vessel flange will be volumetrically examined using manual ultrasonic techniques. There are no bushings used in the threaded stud holes. The examinations performed during the inspection interval will cumulatively cover 100 percent of the studs, nuts, washers, threads in base material, and flange ligaments between threaded stud holes.

Item 1.11, Category G-2 - Pressure Retaining Bolting, 2 in. and Smaller in Diameter

There is no pressure retaining bolting, 2 in. and smaller, on the reactor vessel.

Item 1.12, Category H - Vessel Supports

The reactor support skirt-to-reactor vessel weld will be examined volumetrically using manual ultrasonic techniques. The examination will include the weld to the vessel and the base metal beneath the weld zone and along the support skirt for a distance of two support thicknesses. The examination performed during each inspection interval will cover at least 10 percent of the circumference of the weld to the vessel.

Item 1.13, Category I-1 - Closure Head Cladding

There is no cladding on the closure head.

Item 1.14, Category I-1 - Interior Clad Surfaces of Reactor Vessels

There are six cladding examination patches at least 36 sq in. each distributed on the cladding surface of the Nos. 3 and 4 shell ring and the shell flange. The patches are accessible for examination by remote visual methods. The examinations performed during each inspection interval will cover 100 percent of the patch areas.

Item 1.15, Category N-1 - Interior of Reactor Vessels

Surfaces in the space above and below the reactor core will be examined in selected areas by remote visual methods when those areas are made accessible by the removal of components during normal refueling outages. The examination will be conducted at the first refueling outage and subsequent refueling outages at approximately three-year intervals.

Item 1.16, Category N-2 - Integrally Welded Core-Support Structures and Interior Attachments to Reactor Vessels

All visually accessible attachment welds and visually accessible surfaces of the core support structure will undergo a remote visual examination during each inspection interval.

Item 1.17, Category N-3 - Removable Core-Support Structures

Not applicable to direct-cycle boiling water reactors.

Item 1.18, Category O - Pressure Retaining Welds in Control Rod Drive Housings

The control rod drive housing weld metal and base metal for one wall thickness beyond the edge of the weld will be examined manually. During each inspection interval, 100 percent of the welds in 10 percent of the installed peripheral control rod drive housings will be volumetrically examined. For the preservice examination, 100 percent of the welds in the installed peripheral control rod drive housings will be examined.

Item 1.19, Category P - Components Exempted from Examination by IWB-1220

All accessible components within this category will receive a visual examination as required by Articles IWA-5000 and IWB-5000.

5.2.4.4.2 Piping Pressure Boundary

Item 4.1, Category F - Pressure Retaining Dissimilar Metal Welds

Volumetric and surface examinations of safe end to piping and safe end in branch piping welds will be performed on 100 percent of each dissimilar metal weld. Examination will include the base material for one-half wall thickness or 1-inch, whichever is less beyond the edge of the weld.

Item 4.2 and 4.3, Category G-1 - Pressure Retaining Bolting, 2 in. and larger in Diameter

There is no pressure retaining bolting larger than 2 in. in diameter within the piping pressure boundary.

Item 4.4, Category G-2 - Pressure Retaining Bolting, Smaller than 2 in. and Smaller in Diameter

Safety valve to flange pipe connection bolting, top head nozzle flange-to-pipe flange bolting and flanged pipe connection bolting will under go visual examination. Bolting at flow meter orifices will undergo visual examination. The examinations will include the bolts, studs, and nuts when in place under tension, when the connection is disassembled, or when the bolting is removed.

The visual examinations will cumulatively cover 100 percent of the bolts, studs, and nuts.

Item 4.5, 4.6, 4.7 and 4.8, Category J - Pressure Retaining Welds in Piping

During each inspection interval, 100 percent of the circumferential welds and longitudinal welds and the base metal for one wall thickness beyond the edge of the weld will be volumetrically examined. Longitudinal welds will be volumetrically examined for at least 1-foot from the intersecton with the edge of the circumferential weld selected for examination.

For pipe branch connections exceeding 6-inch diameter, the weld metal, the base metal for one pipe wall thickness beyond the edge of the weld on the main pipe run, and at least 2-inches of base metal along the branch run will be volumetrically examined.

Branch pipe connection welds 6-inch diameter and smaller and socket welds will undergo surface examination.

The examinations performed during each inspeciton interval will cover all of the area of 25 percent of the circumferential joints including the adjoining 1-foot sections of longitudinal joints and 25 percent of the pipe branch connection joints.Item 4.4, 4.5, 4.6, 4.8, Category J - Pressure Retaining

Item 4.9, Category K-1 - Support Members for Piping

Integrally welded pipe supports will be volumetrically examined to the extent practicable on 25 percent of the supports during each inspection interval. The examination will cover the welds to the pressure retaining boundary and the base metal beneath the weld zone and along the support attachment member for a distance of two support thicknesses.

Integrally welded pipe supports that cannot be practically inspected by volumetric examination will be examined by the surface method.

Item 4.10, Category K-2 - Support Components for Piping

Piping supports and hangers in this category will be visually examined. The examination will cover the support components from the pipe to and including the attachment to the supporting structure. The examinations performed during each inspection interval will cumulatively cover all support members and structures. Also included will be verification of the settings of constant and variable spring type hangers, snubbers, and shock absorbers.

Item 4.11, Category P - Components Exempted from Examination by IWB-1220

All accessible components within this category will receive a visual examination during each system pressure test, as required by Articles IWA-5000 and IWB-5000.

5.2.4.4.3 Pump Pressure Boundary

Items 5.2 and 5.3, Category G-1 - Pressure Retaining Bolting, 2 in. and Larger in Diameter

The reactor coolant recirculation pump cover-to-case studs are included in this category.

When the bolting is removed for required maintenance or for the examinations specified under Item 5.7 at the end of the inspection interval, a volumetric and surface examination will be performed.

Item 5.4, Category K-1 - Support Members for Pumps

The reactor coolant recirculation pump hanger bracket assemblies, which are integrally welded to the pump casings, are included in this category. The configuration of these assemblies precludes meaningful volumetric examination. During the inspection interval, 25 percent of the integrally welded supports will be examined by a surface method to the extent practicable.

Item 5.5, Category K-2 - Support Components for Pumps

The reactor coolant recirculation pump supports will be visually examined during the inspection interval. The examinations performed during each inspection will cumulatively cover all accessible support members from the pump attachment to and including the attachment to the supporting structure. Also included will be verification of the settings of the constant support hangers and hydraulic snubbers.

Item 5.6, Category L-1 - Pressure Retaining Welds on Pump Casings

There are no pressure retaining welds in pump casings within the inspection boundary.

Item 5.7, Category L-2 - Pump Casings

The internal pressure boundary surfaces of the reactor coolant recirculation pumps are not normally accessible. If maintenance of the pump internals is required during the inspection interval, visual examination will be performed on the accessible surfaces. Otherwise, visual examination of one pump will be performed in each group of pumps performing similar functions in a system at the end of the inspection interval.

Item 5.8, Category P - Components Exempted from Examination by IWB-1220

All accessible components within this category will receive a visual examination during each system pressure test, as required by Articles IWA-5000.

Item 5.9, Category G-2 - Pressure Retaining Bolting, and Smaller 2 in. in Diameter

The reactor coolant recirculation pump seal assembly bolting under this category will be examined visually in place. The visual examination performed during each inspection interval will cumulatively cover the accessible bolts on all pumps.

5.2.4.4.4 Valve Pressure Boundary

Items 6.1, 6.2 and 6.3, Category G-1 - Pressure Retaining Bolting 2 in. and Larger in Diameter

There are no items in this category.

Item 6.4, Category K-1 - Support Members for Valves

Integrally welded valve supports will be examined to the extent practicable on 25 percent of the supports during each inspection interval. Integrally welded supports that cannot be practically examined by ultrasonic methods will be examined by a surface method.

Item 6.5, Category K-2 - Support Components for Valves

Supports and hangers of the valves in this category will be visually examined. During each inspection interval, all accessible support members from the valve attachment to and including the attachment to the supporting structure will be examined, and the settings of constant and variable spring type hangers, snubbers, and shock absorbers will be verified.

Item 6.6, Category M-1 - Pressure Retaining Welds in Valve Bodies

There are no pressure retaining welds in valve bodies within the inspection boundary.

Item 6.7, Category M-2 - Valve Bodies

The internal pressure boundary surface of one disassembled valve exceeding 4 in. nominal pipe size in each group of valves of the same constructional design will be visually examined in each inspection interval when the valve is disassembled for normal maintenance, or near the end of the inspection interval.

Item 6.8, Category P - Components Exempted from Examination by IWB-1220

All accessible components within this category will receive a visual examination during each system pressure test, as required by Articles IWA-5000.

Item 6.9, Category G-2 - Pressure Retaining Bolting, Smaller than 2 in. in Diameter

All valve bonnet bolting will be accessible for a visual examination that will cumulatively cover all accessible bolts, studs, and nuts.

5.2.4.5 Evaluation of Examination Results

a) The standards for examination evaluation are in agreement with the requirements of Section XI, IWB-3000, "Standards for Examination Evaluations". The program for flaw evaluation agrees with Table IWB-3410, "Evaluation Standards".

b) The program regarding repairs of unacceptable indications or replacement of components containing unacceptable indications is in agreement with the requirements of Section XI, IWB-4000, "Repair Procedures". The criteria that establish the need for repair or replacement are in accordance with Section XI, IWB-3000.

5.2.4.6 System Leakage and Hydrostatic Pressure Tests

The pressure retaining Code Class 1 component leakage and hydrostatic pressure test program agrees with the requirements of Section XI, IWB-5000, "System Leakage and Hydrostatic Pressure

Tests". IWB-5222, "System Hydrostatic Test Pressure", presents criteria and a table of equivalent test temperatures versus test pressures at which the system must be tested. The program is in agreement with IWB-5222 with regard to the temperature-pressure relationship of the system at test, and is in agreement with the technical specification's requirements for operating limitations during heatup, cooldown, and system hydrostatic pressure testing. In some cases, these limitations may be more severe than IWB-5222.

5.2.4.7 AUGMENTED INSERVICE INSPECTION TO PROTECT AGAINST POSTULATED PIPING FAILURES

The augmented inservice inspection program to provide 100 percent volumetric examination of circumferential and longitudinal pipe welds in high energy systems beween containment siolation valves will be reviewed and implemented on an as described in subsection 6.6-8. 3 11

5.2.5 DETECTION OF LEAKAGE THROUGH REACTOR COOLANT PRESSURE BOUNDARY

5.2.5.1 Leakage Detection Methods

The nuclear boiler leak detection system consists of temperature, pressure, and flow sensors with associated instrumentation and alarms. This system detects, annunciates, and isolates (in certain cases) leakages in the following systems:

(1) Main steam lines

(2) Reactor water cleanup (RWCU) system

(3) Residual heat removal (RHR) system

(4) Reactor core isolation cooling (RCIC) system

(5) Feedwater system

(6) High Pressure Coolant Injection (HPCI) System

Isolation and/or alarm of affected systems and the detection methods used are summarized in Table 5.2-8.

Small leaks (5 gpm and less) are detected by temperature and pressure changes and drain pump activities. Large leaks are also

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detected by changes in reactor water level and changes in flow rates in process lines.

The 5 gpm leakage rate is a technical specification limit on unidentified leakage. The leak detection system is fully capable of monitoring flowrates with an accuracy of one gpm and is, thus, in compliance with Paragraph C.2 of Regulatory Guide 1.45.

5.2.5.1.1 Detection of Abnormal Leakage Within the Primary Containment (NSS-Systems)

Normal leakage will result in a decrease of reactor water level and a pressure differential between the core spray line and the vessel shroud. A low reactor water level will cause isolation of the main steam lines.

5.2.5.1.2 Detection of Abnormal Leakage Within the Primary Containment (Non-NSSS)

Leakage through the reactor coolant pressure boundary within the primary containment is detected by monitoring temperatures, pressures, airborne particulate radioactivity, and changes of levels in drain sumps. These monitors and their respective locations are listed in Table 5.2-14.

The following systems are used to monitor these variables:

a) Primary containment and suppression pool temperature monitoring system.

b). Primary containment and suppression chamber pressure monitoring system

c) Primary containment atmosphere monitoring system (containment radiation detection)

d) Drywell floor drain sump monitoring and drywell equipment drain tank level monitoring system.

The above mentioned leak detection systems are designed in accordance with recommendations of Regulatory Guide 1.45.

The drywell leak detection system is not intended to be qualified as a post LOCA system; it is designed for use during power operation as implied by the Technical Specifications. There would be no practical way of recalibrating the system after the LOCA transient.

5.2.5.1.2.1 Primary Containment Temperature Monitoring System

Temperatures within the drywell are monitored at various elevations. A drywell ambient temperature rise will indicate the pressure of reactor coolant or steam leakage. Temperature monitoring of the containment provides an indirect indication of leakage as defined in the regulatory position (3) of Regulatory Guide 1.45.

A detailed description of the system, sensitivity and response time, and the system reliability is discussed in Subsection 7.6.1b.1.2.

Limiting leakage conditions are included in the technical specification of Chapter 16.

Provisions for testing and calibration are described in Section 7.6.2b.

5.2.5.1.2.2 Primary Containment Pressure Monitoring System

Pressure monitoring within the containment provides an indirect method of detecting leakage.

The drywell pressure fluctuates slightly during reactor operation as a result of pressure changes in the reactor building and out-leakage. A pressure increase above normal values indicates a leak in the primary containment.

The primary containment monitoring system and instrumentation is described in Section 7.6.1b.

Section 7.5.1b identifies safety related display instrumentation.

5.2.5.1.2.3 Primary Containment Atmosphere Monitoring - Airborne Particulate Radioactivity Monitoring

The primary containment is continuously monitored for airborne radioactivity. A sample is drawn from the primary containment and a sudden increase of activity indicates a steam or reactor water leakage.

5.2.5.1.2.3.1 Sensitivity and Response Time

The objective of the drywell leak detection monitors as indicated in R.G. 1.45 is to detect 1 gpm of unidentified primary coolant pressure boundary leakage in 1 hour. Several detection systems supplied to accomplish this are the drywell sump level monitor (see Subsection 5.2.5.1.2.4), a noble gas radiation monitor, a radioiodine monitor, and a particulates radiation monitor. The three radiation monitors sample drywell for the activity levels on the assumption that flashing coolant leakage will result in radioactivity in the atmosphere.

The reliability, sensitivity and response times of radiation
11 monitors to detect 1 gpm in 1 hour of Reactor Coolant Pressure Boundary leakage will depend on many complex factors. The major factors are discussed below:

17

A. Source of Leakage

1) Location of Leakage

The amount of activity which would become airborne following a 1 gpm leak from the RCPB will vary depending upon the leak location and the coolant temperature and pressure. For example, a feedwater pipe leak will have concentration factors of 100 to 1000 lower than a recirculation line leak. A steam line leak will be a factor of 50 to 100 lower in iodine and particulate concentrations than the recirculation line leak, but the noble gas concentrations may be comparable. A RWCU leak upstream of the demineralizers and heat exchangers will be a factor of 10 to 100 higher than downstream, except for noble gases. Differing coolant temperatures and pressures will affect the flashing fraction and partition factor for iodines and particulates. Thus, an airborne concentration cannot be correlated to a quantity of leakage without knowing the source of the leakage.

2) Coolant Concentrations

Variations in coolant concentrations during operation can be as much as several orders of magnitude within a time frame of several hours. These effects are mainly due to spiking during power transients or changes in the use of the RWCU system. Examples of these transients for I-131 can be found in NEDO-10585 (8/72), Behavior of Iodine in Reactor Water During Plant Shutdown and Startup. Thus, an increase in the coolant concentrations could give increased containment concentrations when no increase in unidentified leakage occurs.

3) Other Sources of Leakage

Since the unidentified leakage is not the sole source of activity in the containment, changes in other sources will result in changes in the containment airborne concentrations. For example, identified leakage is piped to the equipment drain tank in the drywell, but the tank is vented to the drywell atmosphere allowing the release of noble gases and some small quantities of iodines and particulates from the drain tank.

B. Drywell Conditions Affecting Monitor Performance 11

1) Equilibrium Activity Levels

During normal operation the activity release from acceptable quantities of identified and unidentified leakage will build up to significant amounts in the drywell air. Conversations with several operating plants indicate that levels as high as .1 to 10 times MPC are not uncommon for noble gases and iodines. (MPC refers to "maximum permissible concentration" as defined by 10CFR20, 17
MPC is used here only as a convenient reference). Due to these high equilibrium activity levels the small increases due to a 1 gpm increase in leakage may be difficult to see within an hour. Typical MPC ranges are:

17

	.1 MPC to 10 MPC
Noble Gases	$1x10^{-6}$ - $1x10^{-4}$ µCi/cc
Particulates	$1x10^{-6}$ - $1x10^{-4}$ µCi/cc
Iodines	$5x10^{-7}$ - $5x10^{-5}$ µCi/cc

Fresh fuel backgrounds were not considered because no fission products are available at that point in time. The numbers given above include amounts of

failed and/or irradiated fuel. These numbers also include normal expected leakage rates.

2) Purge and Pressure Release Effects

Changes in the detected activity levels have occurred during periodic drywell purges to lower the drywell pressure. These changes are of the same order of magnitude as approximately a 1 gpm leak, and are sufficient to invalidate the results from iodine and particulate monitors.

3) Plateout, Mixing, Fan Cooler Depletion

Plateout effects on iodines and particulates will vary with the distance from the coolant release point to the detector. Larger travel distances would result in more plateout. In addition the pathway of the leakage will influence the plateout effects. For example, a leak from a pipe with insulation will have greater plateout than a leak
11 from an uninsulated pipe. Although the drywell air will be mixed by the fan coolers, it may be possible for a leak to develop in the vicinity of the radiation detector sample lines. In addition, condensation in the coolers will remove iodines and particulates from the air. Variations in the flow, temperature and number of coolers will affect the plateout fractions. Plateout within the detector sample tube will also add to the reduction of the iodine and particulate activity levels. The uncertainties in any estimate of plateout effects could be as much as one or two orders of magnitude.

C. Physical Properties and Capabilities of the Detectors

1) Detector Ranges

The detectors were chosen to ensure tha tthe operating ranges covered the concentrations expected in the drywell. The operating ranges are:

Noble Gases	$1x10^{-6}$ to $1x10^{-2}$	μCi/cc
17 Particulates	$1x10^{-9}$ to $1x10^{-4}$	μCi/cc
Iodines	$1x10^{-9}$ to $1x10^{-4}$	μCi/cc

2) Sensitivity

In the absence of background radiation and equilibrium drywell activity levels, the detectors have the following minimum sensitivity.

Noble Gas	$1x10^{-6}$	μ Ci/cc
Particulates	$1x10^{-9}$	μ Ci/cc
Iodine	$1x10^{-9}$	μ Ci/cc

17

3) Counting Statistics and Monitor Uncertainties

In theory these radioactivity monitors are statistically able to detect increases in concentration as small as 2 or 3 times the square root of the count rate, i.e., at 10^6 cpm an increase of $2x10^3$, or 0.2%, is detectable; at 10^2 cpm an increase of 240, or 20% is detectable. In addition at high count rates the monitors have dead-time uncertainties and the potential for saturating the monitor or the electronics. Uncertainties in calibration (±5%) sample flow (±10%) and other instrument design parameters tend to make the uncertainty in a count rate closer to 20% to 40% of the equilibrium drywell activity.

11

4) Monitor Setpoints

Due to the uncertainty and extreme variability of the concentrations to be measured in the containment the use of alarm setpoints on the radioactivity monitors would not be practical or useful. As indicated in the following section the setpoints which would be required to alarm at 1 gpm would be well within the bounds of uncertainty of the measurements. The use of such setpoints would result in many unnecessary alarms and the frequent resetting of setpoints. A setpoint alarm on the sump level monitor alone is used; the radioactivity monitors are for supporting information to confirm that the leak is radioactive. The alarm setpoints for the radiation monitors will be set significantly above background to prevent nuisance alarms. The actual setpoint will be changed as background increases. At these levels, the radiation monitors will provide no warning of a 1 gpm leak in one hour.

13

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5) Estimated Monitor Responses

Table 5.2-13 estimates the expected monitor responses for several types of leaks and several types of monitors. As indicated in column 3, the added activity in containment from a 1 gpm leak for 1 hour is less than the nominal 20% increase which could be meaningfully detected. The final columns estimate the detectable leakage in 1 hour.

6) Operator Action

There is no direct correlation or known relationship between the detector count rate and the leakage rate, because the coolant activity leavels, source of leakage, and background radiation levels (from leakage alone) are not known and cannot be cost-effectively determined in existing reactors. There are also several other sources of containment airborne activity (e.g. safety relief valve leakage) which further complicate the correlation.

Thus, the recommended procedure for the control room operator is to set an alarm setpoint at 1 gpm in 1 hour on the sump level monitor (measuring water collected in the sump which may not exactly correspond to water leaking from an unidentified source). When the alarm is actuated, the operator will review all other monitors (e.g., noble gas, particulates, temperature, pressure, fan cooler drains, etc.) to determine if the leakage is from the primary coolant pressure boundary and not from an SRV or cooling water system, etc. Appropriate actions will then be taken in accordance with Technical Specification 3/4.4.3. The review of other monitors will consist of comparisons of the increases and rates of increase in the values previously recorded on the strip chart recorders. Increases in all parameters except sump level will not be correlated to a RCPB leakage rate. Instead, the increases will be compared to normal operating limits and limitations (e.g., 2 psi maximum pressure for ECCS initiation) and abnormal increases will be investigated.

Since the 5 gpm Technical Specification limit is allowed to be averaged over 24 hours, quick and accurate responses are not necessary unless the leakage is very large and indicative of a pipe break. In this case, the containment pressure and reactor vessel water level monitors will alarm within seconds, and the sump level monitor would alarm within minutes or tens of minutes.

The radiation monitor alarms will not be set to levels that correspond to RCPB leakage levels since the correlations can't be made. Also, since the containment airborne activity levels vary by orders of magnitude during operation due to power transients, spiking, steam leaks, and outgassing from sumps, etc., an appropraite alarm setpoint, if

one is used, should be determined by the operator based on experience with the specific plant. A setpoint level of 2 to 3 times the background level during full power steady state operation may be useful for alarming large leaks and pipe breaks, but it would not always alarm for 1 gpm in 1 hour. 17

7) Conclusion

Due to the sum total of the uncertainties identified in the previous paragraphs the iodine and particulate monitors will not be relied upon for leak detection purposes but only as supporting instrumentation. The noble gas monitor is used to give supporting information to that supplied by the sump level monitor and it would be able to give an early warning of a major leak especially if equilibrium containment activity levels are low. However, the uncertainties and variations in noble gas leaks and concentrations would preclude the setting of a meaningful set point on the monitors. 11

5.2.5.1.2.4 Drywell Floor Drain Sump Monitoring System

The drywell floor drain sump monitoring system is designed to permit leak detection in accordance with Regulatory Guide 1.45.

5.2.5.1.2.4.1 System Description

Two drywell floor drain sumps are located in the primary containment for collection of leakage from vent coolers, control rod drive flange leakage, chilled water drains, cooling water drains, and overflow from the equipment drain sump. 11

The drywell floor drain sump is located at the drywell diaphragm slab low point. Unidentified leakages will, by gravity, flow down the slab surface into the floor drain sump. No floor drain piping system is employed. Piped inputs to the drywell floor drain sump are from clean system drains. No surveillance program is planned to detect piped equipment drain system blockage. 12

Small, unidentified leakages of concern flowing into the drywell floor drain sump will not be masked by larger, acceptable, identified leakages overflowing from the drywell equipment drain tank. The drywell equipment drain tank drains by gravity. During conditions of acceptable identified leakage rates, the gravity flow from the drywell equipment drain tank will be capable of preventing the drywell equipment drain tank from 17

overflowing to the drywell floor drain sump. The operation and control of the drywell equipment drain tank drain is the same as discussed in Subsection 5.2.5.1.2.4.1 for the drywell floor drain sumps. 17

Water flow rate better than 0.5 gpm can be obtained by monitoring changes of level over a time period. The following method of flow rate measurement was selected to comply with the requirements of Regulatory Guide 1.45. The necessary sensitivity is obtained by measuring the changes of level during a fixed time interval. For this purpose a continuous level measurement system is installed in each of the sumps. An electronic signal directly proportional to the actual sump level is applied to one pen of a two-pen recorder, to an electronic sample and hold device, and to an electronic differential switch. The sample and hold device, upon command from a timer, applies its output signal to the second pen of the two-pen recorder and to the second input of the electronic differential switch. The sample and hold unit's output signal level is regularly updated to the reference sump level signal. 12

The actual level signal of the sump and the reference level signal are continuously displayed on the two-pen recorder. The same signals are being monitored by the electronic differential switch. When the level signals differ by ± 50 gallons or more during a 50 minute period (equal to 1 gpm) an alarm is actuated on the local panel and on the control board in the main control room. The change in sump level per unit of time determines the leak rate and is available from the recorder slope for confirmation.

There is no reliable quantitative relationship between the sump level and the leakage rate from any source. The quantity is dependent upon the temperature and pressure of the containment and the leak and the location of the leak. Part of the leak will flash to steam; it may be partially trapped between insulation layers. Presumably the leakage will get to an equalibrium level where most of it ends up in the sump, unless the drywell is vented to relieve the pressure buildup. Since the Technical Specification allows 24-hour averaged leak limits, short term variations in the ability to relate the sump quantity to the leaked quantity are ignored, and its is assumed that all leakage reaches the sump. The errors introduced will not impair the ability to detect larger leaks which could rapidly result in severe accidents. Some leakage will no doubt be trapped in insulation etc., but no large reservoirs for leakage have been found. 17

Each sump is equipped with two submerged pumps which operate in an alternating mode. High sump level starts the pump automatically. Remote manual control of the pump is provided in the control room. Both pumps will be operating as soon as an

abnormal, high level is detected. The capability of each pump is such that normal expected flow rates can be easily accomplished.

5.2.5.1.2.4.2 Instrumentation

Magnetic float type continuous level probes are used to measure the fluid level and provide the signal for the recording of the actual sump level and the rate of level change in the control room. Excessive leak rate is alarmed on the local system panel and with a group trouble alarm in the control room. The leak rate can be observed by the control room operator. 11

5.2.5.1.2.4.3 Drywell Equipment Drain Tank Level Monitoring System

The drywell equipment drain tank collects identified leakage within the primary containment from reactor head seal leak off, bulkhead drain, refueling bellows drain, RPV head vent, recirculation pump seals, reactor recirculation pump cooler drains, and RPV bottom drain. 12 11 17 12

All identified leakages which may have temperatures of 212° F or above are hard-piped directly to the drywell equipment drain tank. These leakages will tend to partially flash into steam and then condense in the drain pipe. This approach minimizes the possibility that leakage will escape as steam into the containment atmosphere prior to measurement in the equipment drain tank. 12

The drywell equipment drain tank drains by gravity. The drain tank's discharge valves automatically open when a predetermined high level in the tank is reached. The discharge valves close at a predetermined low level. 17

5.2.5.1.2.4.4 Sensitivity and Response Time of Measurement 11

The method for liquid leak detection in the primary containment is designed to meet the recommended water flow rate changes of 0.5 to 1.0 gpm as defined in Regulatory Guide 1.45.

The following assumptions and design considerations were incorporated:

a) Leak rate is directly proportional to the associated change in sump level.

b) Measurement of the drywell floor drain sump is discontinued during the pump operation and starts immediately after the pump stops. Measurement of the drywell equipment drain tank is discontinued when the tank's discharge valves are opened and starts immediately after the discharge valves close. 17

c) The selected measurement period T for the average change in level is 50 minutes 11

d) The drywell drain sumps have a capacity of 300 gal with a depth of 5 in.

The drywell equipment drain tank capacity is 1000 gal with a depth of 42 in.

e) The level instrumentation accuracy is ± 5 percent of full range 11

f) Recorder response is better than 1 second for full range

g) Recorder chart size/drive speed: 4 in./3/4 in./hr.

h) The electronic differential switch setpoint will alarm rates less than or equal to one gpm. 12

These design factors allow a detection of 1 gpm flow rate within a 50 minute time period. 11

The operator can verify this leak rate on the recorder in the control room by observation of the average change of level. 13

5.2.5.1.2.4.5 Signal Correlation and Calibration

Drywell Drain Sump

The sump depth of 0-5 in. is displayed on a 0-100 percent recorder chart, which relates to the total sump capacity of 0-300 gal.

The average flow rate (changes of level) during the measurement period T is calibrated to read 0-4 gpm over the full chart range.

Drywell Equipment Drain Tank

The tank depth of 42 in. is displayed on a 0-100 percent recorder chart. This relates directly to the tank capacity of 1000 gal.

The average flow rate is calibrated to record 0-4 gpm over the full chart range.

5.2.5.1.2.4.6 Seismic Qualifications

The drywell floor drain sump, all drywell drain piping, and all instrumentation used to monitor drywell floor drain sump and equipment drain tank level will be qualified to operate following an OBE. The drywell equipment drain tank, drywell equipment drain tank cooling coil, and drywell floor drain sump pumps are not qualified to operate following an OBE. 17

12

Credit will be taken for monitoring unidentified leakage following an OBE thru the use of the drywell floor drain sump level monitoring system. The proper functioning of at least one leakage detection system following an SSE is provided by the design of the air borne radioactivity monitoring system. Refer to Section 7.6.1b for description. 13

5.2.5.1.2.4.7 Testing and Calibration

Calibration of level sensors is possible by observing the change in level during the periodic pump down operations of the drywell floor drain sump, and periodic draining of the drywell equipment drain tank. 17

For the drywell floor drain sump, the pumps are automatically started and stopped by mechanical level sensing switches (high and low level set points), but can also be operated manually, at any time, to check the calibration of the level sensors. In the event that the high-high level is reached, two pumps will operate. The drain tank discharge valves are opened automatically on high level and can be operated manually at any time, to check the calibration of the level sensors. 12 17

5.2.5.1.3 Detection of Abnormal Leakage Outside the Primary Containment

Outside the drywell, the piping within each system monitored for leakage is in compartments or rooms, separate from other systems where feasible, so that leakage may be detected by area temperature indications. Each leakage detection system discussed below is designed to detect leak rates that are less than the technical specification leakage limits. The method used to monitor for leakage for each RCPB component may be seen in Table 5.2-8.

(1) Ambient and Differential Room Ventilation Temperature

A differential temperature sensing system is installed in each room containing equipment that interfaces with the reactor coolant pressure boundary. These are the HPCI, RCIC, RHR, and reactor water cleanup systems equipment rooms, and main steam line tunnel. Temperature sensors are placed in the inlet and outlet ventilation ducts. Other sensors are installed in the equipment areas to monitor ambient temperature. A differential temperature switch between each set of sensors and/or ambient temperature switch initiates an alarm and isolation when the temperature reaches a preset value. The HPCI, RCIC and RHR leak detection area ambient temperature switch set points are designed to initiate isolation signals at 167°F. This set point includes sufficient margin above the post LOCA maximum area temperature to preclude inadvertent isolation signals. Consideration has been given to keeping this set point low enough to allow a timely detection of a 5 GPM leak, with the room starting at the design minimum temperature.

The HPCI, RCIC and RHR ventilation inlet and exhaust differential temperature switch set points are designed to initiate isolation signals at a differential temperature of 89°F. This set point includes sufficient margin to prevent inadvertent isolation signals when the area ventilation exhaust is at the maximum post LOCA temperature, and the ventilation inlet corresponds to the minimum reactor building recirculating ventilation design temperature. This set point will allow wide fluctuations in outside air temperature without causing inadvertent isolation signals. This setting will also permit timely detection of a 5 GPM leak, with the area starting at minimum design temperatures.

Annunciator Accessible areas are inspected periodically and the temperature and flow indicators discussed above are monitored regularly as required by Chapter 16. Any instrument indication of abnormal leakage will be investigated.

(2) Visual and Audible Inspection

Accessible areas are inspected periodically and the temperature and flow indicators discussed above are monitored regularly as required by Chapter 16. Any instrument indication of abnormal leakage will be investigated.

(3) Differential Flow Measurement (Reactor Water

Cleanup System Only)

Because of the arrangement of the reactor water cleanup system, differential flow measurement provides an accurate leakage detection method. The flow from the reactor vessel is compared with the flow back to the vessel. An alarm in the control room and an isolation signal are initiated when higher flow out of the reactor vessel indicates that a leak may exist. Major leakage is also detected by excess flow monitoring in the cleanup system suction lines.

5.2.5.2 Leak Detection Devices for NSS-System

(1) Reactor Vessel Head Closure

The reactor vessel head closure is provided with double seals with a leak off connection between seals that is piped through the normally closed manual valves to the equipment drain tank. Leakage through the first seal is indicated locally in the reactor building. The second seal then operates to contain the vessel pressure.

(2) Reactor Water Recirculation Pump Seal

As disussed in Subsection 5.4.1.3, the reactor recirculation pump shaft is provided with two seals. Leakage past each seal is piped to the Drywell Equipment Drain Tank. Leakage past the first stage seal is designed to flow at approximately 0.75 gpm normally. The first stage seal leakoff line is provided with a high/low flow alarm which actuates at 0.9 gpm increasing or 0.5 gpm decreasing. The second stage pump seal is designed for zero leakage normally. The second stage seal leakoff line is provided with a high flow alarm which actuates at 0.1 gpm.

(3) Safety/Relief Valves

Temperature sensors connected to a multipoint recorder are provided to detect safety/relief valve leakage during reactor operation. Safety/relief valve temperature elements are mounted, using a thermowell, in the safety/relief valve discharge piping several feet from the valve body. Temperature rise above ambient is annunciated in the main control room. See the nuclear boiler system P&ID, Figure 5.1-3.

(4) Valve Packing Leakage

Power-operated valves in the nuclear boiler system and recirculation system are provided with valve stem

14 packing leakoff connections. The packing leakoff connection is provided with normally closed isolation valves, and is capped. These valve stem packing leakoff
11 isolation valves will be opened only during shutdown or hydrostatic test conditions to verify the inner valve
14 packing leak tightness. Keeping these leakoff connections isolated provides two sets of packings for limiting stem leakage.

5.2.5.3 Limits for Reactor Coolant Leakage

5.2.5.3.1 Total Leakage Rate

The total leakage rate consists of all leakage, identified and unidentified, that flows to the drywell floor drain and equipment drain sumps. The criterion for establishing the total leakage rate limit is based on the makeup capability of the RCIC system. The total leakage rate limit is established at 30 gpm, 25 identified and 5 unidentified. The total leakage rate limit is also set low enough to prevent overflow of the drywell sumps.

5.2.5.3.2 Normally Expected Leakage Rate

The pump packing glands, valve stems, and other seals in systems that are part of the reactor coolant pressure boundary and from which normal design leakage is expected are provided with drains or auxiliary sealing systems. Nuclear system valves and pumps inside the drywell are equipped with double seals. Leakage from the primary recirculation pump seals is piped to the drywell
17 11 equipment drain tank as described in Subsections 5.2.5.2(2) and 5.4.1.3. Leakage from the safety/relief valves is identified by temperature sensors in the discharge line that transmit to the control room. Any temperature increase above the drywell ambient temperature detected by these sensors indicates valve leakage.

Except for the leakoffs from the reactor recirculation pumps, all drains routed to the Drywell Equipment Drain Tank are normally
17 isolated by closed values. Therefore, any leakage measured during normal plant operation in the Equipment Drain Tank is attributable to the recirculation pumps.

17 The leakage rates from the recirculation pumps, plus any other leakage rates measured while the drywell is open, are defined as
11 identified leakage rates. Table 5.2-11 lists normal and maximum identified leakage rates directed into the Drywell Equipment Drain Tank, and the associated activity concentrations.

5.2.5.4 Unidentified Leakage Inside the Drywell

5.2.5.4.1 Unidentified Leakage Rate

The unidentified leakage rate is the portion of the total leakage rate received in the drywell sumps that is not identified as previously described. A threat of significant compromise to the nuclear system process barrier exists if the barrier contains a crack that is large enough to propagate rapidly (critical crack length). The unidentified leakage rate limit must be low because of the possibility that most of the unidentified leakage rate might be emitted from a single crack in the nuclear system process barrier.

An allowance for leakage that does not compromise barrier integrity and is not identifiable is made for normal plant operation.

The unidentified leakage rate limit is established at 5 gpm rate to allow time for corrective action before the process barrier could be significantly compromised. This 5 gpm unidentified leakage rate is a small fraction of the calculated flow from a critical crack in a primary system pipe (Figure 5.2-10). Safety limits and safety limit settings are discussed in Chapter 16. Table 5.2-12 lists unidentified leakage rates directed into the Drywell Floor Drain Sump, and the associated Activity Concentrations. 11

5.2.5.4.2 Sensitivity and Response Times

Sensitivity, including sensitivity tests and response time of the leak detection system are covered in Subsection 7.6.1.

5.2.5.4.3 Length of Through-Wall Flaw

Experiments conducted by GE and Battelle Memorial Institute, (BMI), permit an analysis of critical crack size and crack opening displacement (Reference 5.2-4). This analysis relates to axially oriented through-wall cracks.

(1) Critical Crack Length

Satisfactory empirical expressions have been developed to fit test results. A simple equation which fits the data in the range of normal design stresses (for carbon steel pipe) is

$$\ell_c = \frac{15000D}{\sigma_h}$$

where

ℓ_c = critical crack length (in.)

D = mean pipe diameter (in.)

σ_h = nominal hoop stress (psi).

(2) Crack Opening Displacement

The theory of elasticity predicts a crack opening displacement of

$$w = \frac{2l\sigma}{E} \qquad \text{(Eq. 5.2-1)}$$

where

l = crack length

σ = applied nominal stress

E = Young's Modulus

Measurements of crack opening displacement made by BMI show that local yielding greatly increases the crack opening displacement as the applied stress s approaches the failure stress sf. A suitable correction factor for plasticity effects is:

$$C = SEC \frac{\pi\sigma}{2\sigma_f} \qquad \text{(Eq. 5.2-2)}$$

The crack opening area is given by

$$A = C \frac{\pi}{4} w\ell = \frac{\pi\ell^2\sigma}{2E} \; SEC \frac{\pi\sigma}{2\sigma_f} \qquad \text{(Eq. 5.2-3)}$$

For a given crack length ℓ, σ_f = 15,000 D/l.

(3) Leakage Flow Rate

The maximum flow rate for blowdown of saturated water at 1000 psi is 55 lb/sec-in². and for saturated steam the rate is 14.6 lb/sec-in², (Reference 5.2-5). Friction in the flow passage reduces this rate, but for cracks leaking at 5 gpm (0.7 lb/sec) the effect of friction is small. The required leak size for 5 gpm flow is

A = 0.0126 in². (saturated water)
A = 0.0475 (saturated steam)

From this mathematical model, the critical crack length and the 5 gpm crack length have been calculated for representative BWR pipe size (Schedule 80) and pressure (1050 psi).

The lengths of through-wall cracks that would leak at the rate of 5 gpm given as a function of wall thickness and nominal pipe size are:

Nominal Pipe Size (Sch 80), in.	Average Wall Thickness, in.	Crack Length P, in.	
		Steam Line	Water Line
4	0.337	7.2	4.9
12	0.687	8.5	4.8
24	1.218	8.6	4.6

The ratios of crack length, ℓ, to the critical crack length, ℓ_c as a function of nominal pipe size are:

Nominal Pipe Size (Sch 80), in.	Ratio ℓ/ℓ_c Steam Line	Ratio ℓ/ℓ_c Water Line
4	0.745	0.510
12	0.432	0.243
24	0.247	0.132

It is important to recognize that the failure of ductile piping with a long, through-wall crack is characterized by large crack opening displacements which precede unstable rupture. Judging from observed crack behavior in the GE and BMI experimental programs, involving both circumferential and axial cracks, it is estimated that leak rates of hundreds of gpm will precede crack instability. Measured crack opening displacements for the BMI experiments were in the range of 0.1 to 0.2 in. at the time of incipient rupture, corresponding to leaks of the order of 1 sq in. in size for plain carbon steel piping. For austenitic stainless steel piping, even larger leaks are expected to precede crack instability, although there are insufficient data to permit quantitative prediction.

The results given are for a longitudinally oriented flaw at normal operating hoop stress. A circumferentially oriented flaw could be subjected to stress as high as the 550°F yield stress, assuming high thermal expansion stresses exist. It is assumed that the longitudinal crack, subject to a stress as high as 30,000 psi, constitutes a "worst case" with regard to leak rate versus critical size relationships. Given the same stress level, differences between the circumferential and longitudinal orientations are not expected to be significant in this comparison.

Figure 5.2-10 shows general relationships between crack length, leak rate, stress, and line size, using the mathematical model described previously. The asterisks denote conditions at which the crack opening displacement is 0.1 in., at which time instability is imminent as noted previously under "Leakage Flow Rate". This provides a realistic estimate of the leak rate to be expected from a crack of critical size. In every case, the leak rate from a crack of critical size is significantly greater than the 5 gpm criterion.

If either the total or unidentified leak rate limits are exceeded, an orderly shutdown would be initiated and the reactor would be placed in a cold shutdown condition within 24 hours.

5.2.5.4.4 Margins of Safety

The margins of safety for a detectable flaw to reach critical size are presented in Subsection 5.2.5.4.3. Figure 5.2-10 shows

general relationships between crack length, leak rate, stress and line size using the mathematical model.

5.2.5.4.5 Criteria to Evaluate the Adequacy and Margin of the Leak Detection System

For process lines that are normally open, there are at least two different methods of detecting abnormal leakage from each system within the nuclear system process barrier located in the drywell and reactor building as shown in Table 5.2-8. The instrumentation is designed so it can be set to provide alarms at established leakage rate limits and isolate the affected system, if necessary. The alarm points are determined analytically or based on measurements of appropriate parameters made during startup and preoperational tests.

The unidentified leakage rate limit is based, with an adequate margin for contingencies, on the crack size large enough to propagate rapidly. The established limit is sufficiently low so that, even if the entire unidentified leakage rate were coming from a single crack in the nuclear system process barrier, corrective action could be taken before the integrity of the barrier would be threatened with significant compromise.

The leak detection system will satisfactorily detect unidentified leakage of 5 gpm.

Sensitivity, including sensitivity testing and response time of the leak detection system, and the criteria for shutdown if leakage limits are exceeded, are covered in Section 7.6.1.

5.2.5.5 Differentiation Between Identified and Unidentified Leaks

Subsection 5.2.5.1 describes the systems that are monitored by the leak detection system. The ability of the leak detection system to differentiate between identified and unidentified leakage is discussed in Subsections 5.2.5.1, 5.2.5.4 and 7.6.1.

5.2.5.6 Sensitivity and Operability Tests

Testability of the leakage detection system is contained in Subsection 7.6.1.

5.2.5.7 Safety Interfaces

The Balance of Plant-GE Nuclear Steam Supply System safety interfaces for the Leak Detection system are the signals from the monitored balance of plant equipment and systems which are part of the nuclear system process barrier, and associated wiring and cable lying outside the Nuclear Steam Supply System Equipment. These balance of plant systems and equipment include the main steam line tunnel, the safety/relief valves, and the turbine building sumps.

5.2.5.8 Testing and Calibration

Provisions for Testing and Calibration of the leak detection system is covered in Chapter 14.

5.2.6 References

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5.2-3 W. L. Williams, Corrosion, Vol. 13, 1957, p. 539t

5.2-4 GEAP-5620, Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flows, by M. B. Reynolds, April, 1968.

5.2-5 "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," NUREG-76/067, NRC/PCSG, dated October, 1975.

5.2-6 F. Odar, "Safety Evaluation for General Electric Topical Report: Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154, NEDE-24154-P Vol. I, II, III" dated June 1980.

TABLE 5.2-11

IDENTIFIED LEAKAGES INTO THE DRYWELL EQUIPMENT DRAIN TANK

Leakage Source	Normal Activity Concentration μc/cc	Normal Leak Rate (GPM)	Maximum Expected Leak Rate (GPM)
Reactor Head Seal Leakoff	10^{-4}	0	5 (Note 1)
Bulkhead Drain	--	0	0 (Note 2)
Bellows Leakage Drain	--	0	0 (Note 2)
RPV Vent	10^{-3}	0	50 (Note 1)
Recirculation Pump Seals	10^{-1}	1.5	20
Recirculation Pump Cooler Drain	--	0	0
RPV Bottom Drain	10^{-1}	0	0 (Note 2)

NOTES:

1. During hydrotest or cooldown only, with isolation valves open.

2. Drains valved closed. Drain opened only during shutdown, refueling or maintenance.

TABLE 5.2-12

UNIDENTIFIED LEAKAGES INTO THE DRYWELL FLOOR DRAIN SUMP

Leakage Source	Normal Activity Concentration-μc/cc	Normal Leak Rate (GPM)	Maximum Expected Leak Rate (GPM)
Vent Cooler Drains	10^{-4}	.5	20
CRD Flange/Control Blade Backseat	10^{-3}	0	2 (Note 3)
Chilled Water Drains	--	0	0 (Note 1)
Cooling Water Drains	--	0	0 (Note 1)
Misc. Valves & Equipment	$<10^{-5}$	--	.5 (Note 2)

NOTES:

1. Drains valved closed and capped. Opened only for maintenance.

2. Assumed value.

3. 2 gpm/drive, maintenance only, one drive at a time.

6.4 HABITABILITY SYSTEMS

Habitability systems are designed to ensure habitability inside the Control room Technical Support Center (TSC), Operational Support Center (OSC), computer, relay, cable spreading, and battery rooms for both Units 1 and 2 during all normal and abnormal station operating conditions including the post LOCA requirements, in compliance with Design Criterion 19 of 10CFR50, Appendix A. The habitability systems cover all the equipment, supplies, and procedures related to the control and auxiliary electrical equipment so that control room operators are safe against postulated releases of radioactive materials, noxious gases, smoke, and steam. Adequate water, sanitary facilities, and medical supplies are provided to meet the requirements of operating personnel during and after the accident. In addition, the environment of the Control Structure Envelope rooms are maintained to ensure the integrity of the contained safety related controls and equipment, during all the station operating conditions.

6.4.1 DESIGN BASES

The design bases of the habitability systems, upon which the functional design is established, are summarized as follows:

a) The control structure envelope is occupied continuously on a year-round basis. The occupancy of the operating personnel is ensured for a minimum of 5 days, after a design-basis accident (DBA).

b) HVAC systems for radiological habitability are designed to support personnel during normal and abnormal station operating conditions in the Control Structure Envelope.

c) Kitchen, sanitary facilities, and medical supplies for minor injuries are provided for the use of five control room personnel for five days during normal and accident conditions.

d) The radiological effects on the Control Structure Envelope that could exist as a consequence of any accident described in Chapter 15 will not exceed the guide lines set by 10CFR50, Appendix A, General Design Criteria 19.

e) The design includes provisions to preclude the effects of chlorine and smoke from inside or outside the plant from inhibiting the habitability of the control room, TSC and OSC.

f) Eye washes and emergency showers are located on the battery room floor. Respiratory and skin protection for emergencies are provided within the control room.

g) The habitability systems are designed to operate effectively during and after the DBA with the simultaneous loss of offsite power, Safe Shutdown Earthquake, and failure of any one of the HVAC system active components.

h) Radiation monitors, and chlorine and smoke detectors continuously monitor the outside air at the control structure envelope outside air intakes. The detection of high radiation, chlorine, or smoke is alarmed in the control room and related protection functions are simultaneously initiated for high radiation and chlorine. The operator may isolate the control structure on smoke alarm at his discretion.

6.4.2 SYSTEM DESIGN

6.4.2.1 Control Structure Envelope

Habitability system boundaries for Susquehanna SES is the control structure envelope.

a) An independent HVAC system is provided for the control room area. This includes: control room, TSC, OSC, kitchen, toilet and locker, office, and storage space. A detailed description of this redundant system is provided in Subsection 9.4.1.

b) Two independent HVAC systems are provided for the remaining areas. One system serves the computer room, relay rooms, computer maintenance room, office, and UPS rooms. The other system serves the lower cable spreading room, upper relay rooms, upper cable spreading rooms, electrician's office, battery rooms, cold instrument repair shop, and HV equipment room. Each of these systems is described in Subsection 9.4.1.

There are eleven exterior doors in the control structure envelope. These doors are gasketted to minimize leakage and will be tested to 1/8" H_2O differential pressure to assure tightness.

The leakage path across the ventilation barrier between the control structure envelope and outside environment is thru the isolation damper blades. The isolation dampers are located in the smoke removal systems and the relief air duct system.

Tests on the isolation dampers indicate a leakage rate as shown on Table 6.4-1 at a 4 inch w.g. pressure differential. The analysis for control room habitability given in section 15.6-5 and Appendix 15B assumed a leakage of 10 cfm of outside air to the Control Structure Envelope to account for door openings etc. Makeup air to the envelope is also filtered, so the inleakage to the Control Structure Envelope would not be at outside air concentrations.

The environment of the Control Structure Envelope is maintained to ensure the integrity of the contained safety related controls and equipment during all operating conditions. Technical Specification 3/4.7.2 discusses leakage allowable and pressurization verification testing.

6.4.2.2 Ventilation System Design

The detailed HVAC system design is presented in Subsection 9.4.1. These systems are shown on Figures 9.4-1, 9.4-2, and 9.4-3. Design parameters are listed in Table 9.4-2. A list of isolation dampers with their leakage characteristics and closure times is shown in Table 6.4-1.

All the components are designed to function during and after a SSE except for the outside air intake electric heating controls and humidification equipment, control room relief fan, reheat coils and their controls, which are supported to stay in position even though they may not function.

Components are protected from internally and externally generated missiles. Layout diagrams of the control structure, showing doors, corridors, stairways, shield walls, equipment layout, and the Control Structure Envelope are shown on Figures 6.4-1a through 6.4-1e.

The description of controls, instruments, and radiation and chlorine monitors for the control structure HVAC system is included in Subsections 9.4.1 and 7.3.1. The locations of outside air intakes and potential sources of radioactive and toxic gas releases are indicated on Figure 6.4-2.

A detailed description of the emergency makeup air filter trains is presented in Subsection 6.5.1.

6.4.2.3 Leaktightness

The entire Control Structure Envelope is of leaktight construction. The free air space volume is approximately 110,000 cubic feet in the control room floor, 80,000 cubic feet in the battery room floor, and 320,000 cubic feet in the remaining

spaces of the envelope. All cable tray and duct penetrations are sealed. Approximately 5810 cfm of outside air is introduced through charcoal filters into the envelope, to maintain approximately 1/8 in. H_2O positive pressure over atmosphere; this includes 3500 cfm to the battery rooms as make-up air. The battery rooms are exhausted thru the SGTS exhaust vent. The air intake rates are the same for normal operation and for emergency modes, except high chlorine isolation mode.

When chlorine isolation of the Control Structure Envelope is initiated, all the isolation dampers will automatically close, and the control room toilet exhaust, kitchen exhaust fans, the battery room exhaust fan, and the control room relief air fan will shut off. When isolated, the only source of outside air into the Control Structure Envelope is by leakage through the dampers and around doors, approximately 0.25 air changes by volume per hour.

6.4.2.4 Interaction with Other Zones and Pressure-Containing Equipment

The Control Structure Envelope is surrounded by the turbine building, reactor building, and central access control area. Each of these areas is separated from the control structure by shield walls and floors and served by independent HVAC systems.

All penetrations for conduits, pipes and ductwork penetrating the Control Structure Envelope will be completely sealed; all air outlet openings which continue to areas outside of the envelope will be isolated by a set of redundant isolation dampers. The ductwork penetration is of welded construction.

The Control Structure Envelope is surrounded by the Turbine Building and Reactor Building. These areas are served by independent HVAC systems described in Section 9.4. The control structure is isolated by the ventilation barrier between the control structure and the other areas consisting of concrete wall and floor slab construction and leaktight doors.

Except for fire protection halon bottles, fire extinguishers and self-containing breathing apparatus, there are no pressure-containing tanks in the control room area. Steam piping is excluded from the control structure.

6.4.2.5 Shielding Design

The Control Structure radiation shielding design is discussed in Section 12.3 which describes control structure room shield wall thicknesses, the location of associated plant structures relative to the control structure, and provision to reduce radiation from external sources. A description of radiation sources used to design control structure shielding is presented in Section 12.2 and in reference 6.4-1 and includes source strength, geometry, and attenutation parameters.

6.4.3 SYSTEM OPERATIONAL PROCEDURES

During normal plant operation, the mixture of recirculated air and outside air for the control structure HVAC systems is filtered through UL Class 1 particulate filters with a rated efficiency of 90 percent by ASHRAE Standard 52-68 atmos heric dust spot method. The control structure HVAC systems are started through remote hand switches that are located in the control room HVAC control panel. The operation of the control room HVAC system is described in Subsection 9.4.1.2.1.

To remove any noxious gases and odors from the control room, the operator can manually isolate the control room HVAC system (high chlorine simulation) and place the emergency outside filter train in recirculating operation.

To remove smoke from the control room, the operator can manually operate the smoke exhaust fan and fire protection control damper from the fire protection control panel in the control room. Smoke will be exhausted by the fans, through the duct system to the turbine building exhaust vent.

In the event of high radiation at the outside air intake of the control structure HVAC systems, the radiation monitoring system automatically shuts off normal outside air supply to the systems. The outside air is automatically routed through the emergency outside air filter train before entering the HVAC system.

In the event of a reactor isolation signal, the control structure HVAC system will automatically transfer to the emergency outside air filter train as described in the high radiation mode.

In the event of high chlorine in the outside air intake of the control structure HVAC systems, the chlorine detection system automatically shuts off all isolation dampers and closes off the outside air supply to the control structure. The control structure HVAC systems are automatically put in the recirculation mode. After isolation, the emergency outside air filter train

system can be manually placed in operation. This operation cleans up the air in the control room.

Two emergency outside air filter trains and fans are provided. Each train consists of an electric heater, prefilter, upstream HEPA filter, charcoal adsorber, and downstream HEPA filter. The system is designed to handle the requirements of outside air for the HVAC systems. Each train is sized to process 6000 cfm ±10% of outside air, providing 500 cfm ±10% to the control room HVAC system, 400 cfm ±10% to the computer room HVAC system, and 5100 cfm ±10% to the control structure HVAC system. The emergency outside air filter train system is described in detail in Section 6.5.

6.4.4 DESIGN EVALUATIONS

The control structure HVAC systems designed to maintain a suitable environment for personnel and equipment in the control structure under all the station operating conditions. The systems are provided with redundant equipment to meet the single failure criteria. The redundant equipment is supplied with separate Class 1E power sources and is operable during loss of offsite power. The power supply and control and instrumentation meet IEEE-279 and IEEE-308 criteria. All the HVAC equipments, except the normal outside air intake heating, humidification, control room relief fan, reheat coils and their controls, and surrounding structure, are designed for Seismic Category I.

The likelihood of an equipment fire affecting control structure habitability is minimized because early ionization detection is anticipated, fire fighting apparatus is available, and filtration and purging capabilities are provided. Refer to Subsection 9.5.1 for further description of the Fire Protection System.

The following provisions are made to minimize fire and smoke hazards inside the control structure and damage to nuclear safety related circuits:

a) Most electrical wiring and equipment are surrounded by, or mounted in, metal enclosures.

b) The nuclear safety related circuits for redundant divisions (including wiring) are physically segregated.

c) Cables used throughout the control structure are flame retardant.

d) Structural floors and interior walls are of reinforced concrete. Interior partitions are constructed of metal, masonry, or gypsum dry walls on metal joists. The control room ceiling is suspended type with non-

combustible acoustic tile, the door frames, and doors are metallic. Wood trim is not used.

The control room raised floor consists of steel plates and supports covered with carpet with acceptable fire safety characteristics (see Section 3.2 of the Susquehanna Fire Protection Review Report).

A system is provided to detect high radiation at the outside air intake. These monitors alarm the control room upon detection of high radiation conditions. The emergency outside air filter trains, designed to remove radioactive particulates and adsorb radioactive iodine from the HVAC system outside air supply, are automatically started upon high radiation signals.

A chlorine detection system is provided to detect chlorine at the outside air intake. These detectors alarm in the control room when chlorine is detected and automatically trip all isolation dampers and shut off all related fans. Further description is provided in Section 9.4.

The emergency outside air filter trains and control room shielding are designed to limit the occupational dose levels required by Design Criterion 19 of 10CFR50, Appendix A.

The introduction of sufficient outside air to maintain the Control Structure Envelope at a positive pressure with respect to surroundings, precludes infiltration of unfiltered air into the control structure at all the station operating conditions except when the system is in the recirculation mode.

6.4.4.1 Radiological Protection

The control room air purification system and shielding designs are based on the most limiting design basis assumptions, those of Regulatory Guide 1.3.

The airborne fission product source term in the primary containment following the postulated LOCA is assumed to leak from the containment at a rate of 1.0% per day. Full mixing in the building wake, in which the control room and its ventilation intake are presumed to be immersed for the duration of the post accident phase, is also assumed.

The concentration of radioactivity, which is postulated to surround the control room after the postulated accident, is evaluated as a function of the fission product decay constants, the containment spray system effectiveness, the containment leak rate, and the meteorology for each period of interest. The assessment of the amount of radioactivity within the control room considers the flow rate through the control room outside air intake, the effectiveness of the control room air purification system, the radiological decay of fission products, and the exfiltration rate from the control room.

The control room emergency filtration train draws the incoming air through an electric heating coil, moderate efficiency filter, HEPA filters, and a carbon adsorber to minimize the exposure of control room personnel to airborne radioactivity. In order to increase the effectiveness of the carbon adsorbers, incoming air is warmed by the heating coil to decrease its relative humidity. Air within the control room, TSC and OSC is recirculated continuously through the air handling unit, which controls room temperature 75°F ± 5°F and humidity 50% ± 5%.

The resulting calculated doses for control structure ingress, egress, and occupancy (on a rotating shift basis) are less than 5 rem to the whole body or the equivalent to any organ. These doses are within the dose limits specified in General Design Criterion 19. A detailed discussion of the dose calculation model for control structure operators is discussed in Subsection 15.1.13.

Control structure shielding design, based on the most limiting design basis LOCA fission product release, is discussed in Section 12.3 and is evaluated in Subsection 15.1.13. The evaluations in Chapter 15 demonstrate that radiation exposures to control structure personnel originate from containment shine, external cloud shine, and containment airborne radioactivity sources. Total exposures resulting from design basis accidents are below the dose limits specified by General Design Criterion 19; the portion contributed by containment shine and external cloud shine is reduced to a small fraction of the total by means of shielding.

6.4.4.2 Toxic Gas Protection

The control structure HVAC systems are designed to satisfy the recommendation of the Regulatory Guide 1.78 and 1.95. The HVAC systems are described in Subsection 9.4.1.

A detailed discussion of the toxic gas protection is in Subsection 2.2.3.

6.4.5 TESTING AND INSPECTION

The control structure HVAC systems and their components are thoroughly tested in a program consisting of the following:

a) Factory and component qualification tests (see Subsection 9.4.1)

b) Onsite preoperational testing (see Chapter 14)

c) Onsite subsequent periodic testing (see Chapter 16).

6.4.6 INSTRUMENTATION REQUIREMENTS

All safety-related instruments and controls for the control structure HVAC systems are electric or electronic, except for isolation damper actuators which are pneumatically operated. These dampers are designed to fail safe on loss of compressed air. The compressed air system is not safety-related.

a) Each redundant HVAC system is provided with an independent local control panel and each system is separately controlled. Important operating functions are controlled and monitored from the control room HVAC panel.

b) Instrumentation is provided to monitor important variables associated with normal operations. Instruments are provided to alarm in the control room if abnormal conditions are detected.

c) A radiation detection system is provided to monitor the radiation levels at the system outside air intakes. A high radiation signal is alarmed on the main control board.

d) A chlorine detection system is also provided to monitor the chlorine concentration at the system outside air intakes. A chlorine present signal is alarmed on the main control board.

e) Fire detection capability is provided in the outside air intake plenum. Fire detection is annunciated on the main control board via the fire protection control panel.

f) The control room and control structure HVAC systems are designed for automatic environmental control with manual starting of the fans. The chilled water system has a manual/auto selector switch.

g) A fire protection water spray system is provided for each charcoal adsorber bed in the emergency outside air filter train.

h) The emergency outside air filter train air flow rate and upstream HEPA filter differential pressure are transmitted to the H&V control board in the control room, recorded, and alarmed.

6.4.7 REFERENCES

6.4-1. Susquehanna Steam Electric Station, "Updated Response to TMI Related Requirements", PLA-659, dated March 16, 1981 (N. W. Curtis to B. J. Youngblood), section X.1.20

6.6 INSERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS

The construction permit for Susquehanna SES was issued in November 1973.

Based on the conditions listed in 10CFR50.55a(g), the mandatory preservice inspection requirements, including provisions for design and access, are stipulated to be Section XI of the ASME B&PV Code effective six months prior to the date of issuance of the construction permit. For Susquehanna SES the code in effect would be the 1971 Edition including the Summer 1972 Addenda, which required inspection of the reactor coolant pressure boundary only.

The actual preservice inspection for Susquehanna SES will be conducted in accordance with the requirements of the 1974 Edition of the ASME Code, Section XI, including Addenda through Summer, 1975 as modified by Appendix III to Winter 1975 Addenda and IWA-2232 of the Summer 1976 Addenda to the extent practical within the limitations of design and access provisions and the geometry and materials of construction of the component. Subsequent inservice inspections will be conducted in accordance with the requirements of 10CFR50.55a(g), also, on an "as practical" basis.

6.6.1 COMPONENTS SUBJECT TO EXAMINATION

The inspection requirements of ASME Code Section XI, Articles IWC-2000 and IWD-2000, will be met within the limitations of design and access provisions and the geometry and materials of construction of the component, for all Class 2 and Class 3 pressure retaining components (and their supports) except for components excluded under IWC-1220.

The scope and specific components subject to the requirements of IWC-1000 and IWD-1000 of Section XI, Code Class 2 and 3, are contained in tables that will be provided in an amendment to the FSAR.

6.6.2 ACCESSIBILITY

Inservice inspection access to the ASME Code Class 2 and 3 components listed in the technical specifications is provided in the design of the plant on an "as practical" basis. There is, at this time, no mandatory requirement for preservice inspection, and 10CFR50.55a(g)(4) requires inservice inspection "to the extent practical" within the limitations of design and access provisions for Code Class 2 and 3 components. Aside from providing normal access to components for installation,

maintenance, and testing, the following provisions have been considered in the Susquehanna SES design:

6.6.2.1 Piping and Component Welds

Access envelopes have been considered for Class 2 components requiring volumetric and/or surface examinations. Weld contours and surfaces have been prepared for meaningful ultrasonic examination where required.

6.6.2.2 Insulation Removal

Class 2 piping or components requiring volumetric and/or surface examinations are equipped with removable, numbered, insulation panels. Class 2 and Class 3 piping requiring a visual examination during system pressure tests will not be equipped with removable insulation. The visual examinations will be performed by inspecting the exposed surfaces of and joints in component insulation to locate evidence of leakage and the floor areas (or equipment) directly underneath components for evidence of accumulated leakage that may drip from components.

6.6.2.3 Inaccessible Class 3 Piping

Access will be provided for buried Class 3 piping systems. Spray pond piping is embedded in concrete and is open-ended.

6.6.3 EXAMINATION TECHNIQUES AND PROCEDURES

Inservice examination techniques and procedures used for Code Class 2 and 3 components will conform to the requirements of Subsection IWA-2200 of the 1974 Edition of Section XI, as modified Addenda through Summer, 1975 with the addition of Appendix III to the Winter, 1975 Addend and Subparagraph IWA-2232 of the Summer, 1976 Addenda.

6.6.3.1 Visual Examination

Visual examination techniques will be in agreement with IWA-2210 of Section XI of the Code. Visual examination will be employed as a basis for reporting the general condition of the part, component, or surface.

6.6.3.2 Surface Examination

Surface examination techniques will be in agreement with IWA-2220 of Section XI of the Code. Surface examination will be used to verify the presence of surface or near surface cracks or discontinuities.

6.6.3.3 Volumetric Examination

Volumetric examination methods will be in agreement with IWA-2230 of Section XI of the Code. Volumetric examination will be used to determine the presence of subsurface discontinuities, their size, locations, and orientation.

6.6.4 INSPECTION INTERVALS

The ISI program schedule of required examinations to be completed in each inspection interval meets the requirements of IWC-2400 and IWD-2400 of Section XI. This schedule will be provided in an amendment to the FSAR.

6.6.5 EXAMINATION CATEGORIES AND REQUIREMENTS

ISI examination categories and requirements for ASME Class 2 and 3 components will be examined in accordance with the criteria in IWC-2500, IWC-2600, and IWD-2600 of Section XI. Areas subject to examination and the extent of examination for Class 2 components comply with the requirements of Table IWC-2520 of Section XI on an "as practical" basis. A list of these components will be provided in an amendment to the FSAR.

6.6.6 EVALUATION OF EXAMINATION RESULTS

Evaluation of examination results will be in accordance with IWA-3100 for ASME Code Class 2 and 3 components.

Repairs to, or replacement of components containing unacceptable indications will agree with the requirements of IWB-4000 of Section XI. Criteria to establish the need or replacement will conform to IWB-3000.

6.6.7 SYSTEM PRESSURE TESTS

System pressure tests will meet the requirements of IWC-5000 and IWD-5000 of Section XI.

6.6.8 AUGMENTED INSERVICE INSPECTION TO PROTECT AGAINST POSTULATED PIPING FAILURES

The augmented inservice inspection program to provide assurance against postulated piping failures of high energy systems between containment isolation valves will be reviewed and implemented as described below. There are no guard pipes used to enclose high energy piping on the Susquehanna SES.

The following augmented inspection program applies to piping between the containment isolation valves for which no breaks are postulated;

For ASME III, Class 1 and 2 piping, the requriements of the applicable Code applies as is, with the exception that the extent of examination will be augmented such that 100% of the circumferential welds within the containment isolation boundary will receive 100% volumetric examination during each inspection interval.

Volumetric examination of branch connections containing weldolets, half-couplings, and socket welds would not be meaningful due to the geometry of the branch connection and the small pipe sizes involved. Full coverage of the weld and required volume cannot be obtained. Therefore, surface examination will be performed on all branch to main run welds and all socket welds up to the first isolation valve on the branch line. All butt welds included in the branch piping up to the first isolation valve will receive full volumetric examination.

The inspection program will be performed, completely in accordance with ASME Section XI requirements, however, the extent of examination of ASME Section XI will be supplemented to comply with the augmented inspection program requirements outlined above.

CHAPTER 8.0

ELECTRIC POWER

TABLE OF CONTENTS

8.2 OFFSITE POWER SYSTEM

8.2.1 DESCRIPTION

8.2.1.1 Transmission System

The bulk power transmission system of PP&L operates at 230 KV and 500 KV. Unit #1 of the Susquehanna Steam Electric Station supplies power to the 230 KV system through a 230 KV switchyard and Unit #2 supplies power to the 500 KV system through a separate 500 KV switchyard. The offsite power system for the plant is supplied through the 230 KV portion of the bulk power system.

Figure 8.2-1 shows the Susquehanna 230 KV and 500 KV switchyards and the transmission lines associated with each yard and in the vicinity of the plant. The figure shows the line arrangement with both units in operation. The two switchyards are physically separate and are tied together by a 230 KV yard tie line with a 230-500 KV transformer in the 500 KV yard.

Two independent offsite power sources are supplied to the Susquehanna plant. One source is established by tapping the Montour-Mountain 230 KV line north of the plant and constructing 1300 ft. of 230 KV line on painted steel pole structures to startup transformer #10. The Montour-Mountain line shares double circuit steel pole structures with the Stanton-Susquehanna #2 230 KV line in the vicinity of the plant. The double circuit line extends to a point 1.5 miles east of the transformer #10 tap at which point the two circuits split as shown in figure 8.2-1. The Montour-Mountain line extends 16.8 miles north on double circuit lattice towers with the Stanton-Susquehanna #1 230 KV line and terminates in the Mountain Substation. The Stanton-Susquehanna #2 circuit extends southward on double circuit towers with the Stanton-Susquehanna #1 circuit and terminates in the Susquehanna 230 KV Switchyard.

To the west of the tap into the Susquehanna plant the Montour-Mountain 230 KV circuit extends 1500 feet on double circuit steel pole structures at which point the Stanton-Susquehanna #2 circuit separates and extends northward to Stanton Substation. The Montour-Mountain 230 KV circuit then joins the Montour-Susquehanna 230 KV circuit on double circuit steel lattice towers and extends 29.0 miles to the Montour Switchyard. The total distance to Mountain Substation from the tap into the plant is 18.7 miles. The distance from Montour to the tap is 29.7 miles.

Several lines feed the Montour Switchyard and Mountain Substation, as can be readily seen in figure 8.2-3. These lines

offer a multitude of possible supplies for the tap into Susquehanna startup transformer #10. Montour Switchyard is supplied directly by generation from the Montour Steam Electric Station. Other generating stations are indirectly linked by the bulk power grid system. The conductors for the transformer #10 tap and the Montour-Mountain line are 1590 kcmil 45/7 ACSR and are supported by single string insulator assemblies. Maximum conductor tension is limited to 16,000 pounds on steel pole line sections and 21,900 pounds on lattice tower sections under maximum anticipated loading conditions.

The second offsite power source is supplied at 230 KV from the yard tie circuit between the Susquehanna 500 kV and 230 kV Substations south of the Susquehanna Steam Electric Station. The source is provided by a single 400 ft. span tap from the 230 KV yard tie circuit to startup transformer #20.

The yard tie line consists of 230 KV double circuit tubular steel pole structures supporting two parallel circuits of 1590 kcmil 45/7 ACSR conductors on single string insulator assemblies. The circuits are tied together to form a two conductor per phase single circuit line. The 400 ft. tap to transformer #20 consists of one 1590 kcmil 45/7 ACSR conductor per phase. The distance from the tap point west to the 500 KV yard is 1500 ft. The distance from the tap point east to the 230 KV yard is 1.6 miles. Maximum conductor tension is limited to 16,000 pounds in the yard tie line under maximum loading conditions.

The second offsite power supply is furnished by the multiple sources throughout the bulk power grid system through the 230 KV and 500 KV lines emanating from the Susquehanna 230 KV and 500 KV switchyards. See figure 8.2-3.

All transmission lines meet or exceed design requirements set forth by the National Electric Safety Code. One or two overhead ground wires are employed on the transmission lines above the phase conductors to provide adequate lightning flashover protection. All lines meet the Army Corps of Engineers requirements for clearance over flood levels. All bulk power transmission lines are designed to withstand 100 mph hurricane wind loads on bare conductors.

The Montour-Mountain 230 KV line is crossed by the Stanton-Susquehanna #2 230 KV line. No transmission lines cross over the Susquehanna 500 KV to 230 KV yard tie line or the two tap lines supplying transformers #10 and #20.

No single disturbance in the bulk power grid system will cause complete loss of offsite power to the Susquehanna SES. This is a basic system design criteria.

8.2.1.2 Transmission Interconnection

PP&L is a member of the Pennsylvania, New Jersey, and Maryland Interconnection which permits economical exchanges of power with neighboring utilities and provides emergency assistance. Direct bulk power ties are between PP&L and Philadelphia Electric, Luzerne Electric Division of UGI, Metropolitan Edison, Pennsylvania Electric, Jersey Central Power and Light, Public Service Electric and Gas, and Baltimore Gas and Electric Companies.

8.2.1.3 Switchyards

8.2.1.3.1 Startup Transformers #10 and #20

The Montour-Mountain 230 KV line and the 230 KV yard tie line supply power to startup transformers #10 and #20, respectively, through motor operated air break switches. High speed positive ground switches are installed between the motor operated air break switches (MOABs) and the startup transformers. The startup transformers and low side bus connections are discussed in Section 8.3.1. The startup transformer yards are physically separated from each other, the Unit #1 and #2 main transformer yards and the 230 KV and 500 KV switchyards as can be seen on figure 8.2-1. 1590 kcmil 45/7 ACSR conductors connect the air switches to the startup transformers. 13.8 KV cables are installed in underground conduit between the startup tranformers and the turbine building. Non-segregated phase bus ducts establish the tie to the 13.8 KV startup buses within the turbine building. See figure 8.2-4 for a one line diagram of the offsite power system.

Line relay protection for the Montour-Mountain 230 KV line and the 230 KV yard tie circuit is provided by two independent directional comparison carrier blocking pilot relaying and two zone directional distance backup systems which ensures adequate line protection in the event of a malfunction. These relaying schemes detect faults on the transmission line and isolate the power sources to the transformers by tripping the power circuit breakers (PCBs) at the line terminals. Breaker failure relaying, applied at each line terminal, detects a failure to trip or failure to interrupt condition at the line terminal and trips all associated PCBs necessary to isolate the line. Power to the line relaying facilities is supplied from the local switchyard power sources.

Startup transformers #10 and #20 are protected by high speed percentage differential, sudden pressure and overcurrent

relaying. Direct transfer trip facilities are utilized as the primary relaying scheme to open the PCBs at the transmission line remote terminals in the event of transformer trouble. Backup protection is provided by the high speed ground switch on the 230 KV side of the startup transformer. This switch is closed to place a positive fault on the 230 KV transmission line which will
15 be detected by the remote line terminal relaying systems if the primary direct transfer trip scheme fails to function correctly. The motor operated air switch automatically opens after the 230 KV system is de-energized to isolate the startup transformer from the transmission system and permit reclosing of the transmission line terminal PCBs.

A time delay undervoltage relay monitors the 13.8 KV startup bus voltage. On loss of offsite power the relay trips the startup bus incoming feeder breaker and initiates transfer of the bus loads to the other startup transformer through closure of the startup bus tie breaker. The time delay undervoltage relay also prevents unnecessary automatic trip of the incoming feeder breaker for short duration disturbances on the transmission line.

Power to transformer #10 and #20 switchgear, motor operated air break switches, and high speed ground switches is supplied from the station 125 V DC power supplies.

8.2.1.3.2 Susquehanna Unit #1 230 KV Main Transformer Leads

Overhead 1590 kcmil 45/7 ACSR conductors, bundled two per phase, tie the Unit #1 main stepup transformers, through a high voltage Disconnect switch-Synchronizing PCB-Disconnect switch arrangement, to the 230 KV switchyard. The synchronizing breaker
7 and disconnect switch arrangement is provided at the Susquehanna SES site to improve reliability in synchronization and flexibility of operating Unit 1. Steel pole structures support the strain bus and the 2.2 mile 230 KV tie with single string insulator assemblies. The tie line is capable of transmitting the full 1280 MVA output of the Unit #1 generator.

Relay protection between the Unit #1 transformer and the synchronizing breaker is provided by high speed percentage differential relays which trip Unit #1 and the synchronizing breaker by the unit master trip lockout relays. A second protection scheme is provided by the Unit #1 overall differential relaying which also detects fault conditions between Unit #1 transformer and the synchronizing breaker. Two directional
15 comparison carrier blocking pilot and two zone directional distance backup relaying systems provide fault protection between the 230 KV synchronizing PCB and the Susquehanna 230 KV Switchyard. Breaker failure protection relaying is applied at

each terminal to detect a failure to trip or failure to interrupt condition and to electrically isolate the faulty component.

Control power to the synchronizing power circuit breaker and power to the onsite relaying equipment are provided by the plant 125 V DC power supplies.

8.2.1.3.3 Susquehanna 230 KV Switchyard

The 230 KV switchyard is an outdoor steel structure, comprised of 6 bay positions containing 14-230 KV power circuit breakers arranged in a breaker and one half scheme. Terminating positions are provided for seven lines, one generator lead, and a yard tie to the 500 KV switchyard. The switchyard breakers can be operated by remote supervisory control from the PP&L System Operating Offices.

Service power to the 230 KV switchyard is provided by a local 12 KV distribution line with a backup diesel generator in the 230 KV switchyard. An automatic throwover scheme is employed in the event of one source failure. Line protection equipment power is provided by a single 125 V DC switchyard service battery equipped with two full capacity chargers.

8.2.1.3.4 Susquehanna Unit #2 500 KV Main Transformer Leads

Unit #2 generator output is connected to the 500 KV switchyard by a 1400 ft. overhead 500 KV transmission line. 2493 kcmil 54/37 ACAR conductors bundled two per phase are supported by V-string insulator assemblies on steel pole H-frame structures. The tie is capable of transmitting the full 1280 MVA generator output of Unit #2 to the 500 KV switchyard.

Relay protection for the connection between the Unit #2 transformer and the Susquehanna 500 KV switchyard is provided by high speed bus differential relays which trip Unit #2 and the three 500 KV switchyard generator breakers by the master trip lockout relays for a fault in the connection. An overall differential protection scheme provides a second system to trip Unit #2 and the three PCBs connected to the generator in the 500 KV switchyard for a fault on the transformer leads. Breaker failure protection is applied at each terminal to detect a failure to trip or failure to interrupt condition and to electrically isolate the faulty component.

8.2.1.3.5 Susquehanna 500 KV Switchyard

The 500 KV switchyard is an outdoor steel structure, comprised of three bays containing five 500 KV power circuit breakers arranged in a modified ring bus configuration. The switchyard provides for ultimate future expansion to 5 bays in a breaker and one half scheme. Terminating positions are provided for two lines, one 500 KV generator lead circuit and a circuit to a bank of three single phase 500-230 KV autotransformers. Manual operation of the 500 KV generator lead synchronizing circuit breakers is by the plant control room operator. The remaining PCBs can be operated by PP&L's remote supervisory control or by the plant supervisory control.

Service power to the 500 KV switchyard is provided by two sources: one from the generating station, and the second from the tertiary winding of the yard tie autotransformers with an automatic low voltage throwover scheme in the event of one source failure. Line protection equipment is powered by a single 125 V DC switchyard service battery equipped with two full capacity battery chargers.

8.2.1.3.6 Montour and Mountain 230 kV Switchyards

Figure 8.2-5 shows a one line diagram of the off-site power system for Startup Transformer #10.

The Montour Switchyard is an outdoor steel structure comprised of four bay positions containing 11-230 kV power circuit breakers arranged in a breaker and one half scheme. Two generating leads from the Montour Steam Electric Station and five transmission lines are terminated in the yard. The switchyard breakers can be operated by remote control from the PP&L System Operating offices.

The Mountain Switchyard is owned and operated by UGI Corporation, Luzerne Electric Division. It is an outdoor steel structure with two bay positions each containing one 230 kV PCB. The two PCBs are arranged back to back between the Montour-Mountain and Mountain-Lackawanna Lines. Between the two PCBs is a normally open MOAB to the Susquehanna-Stanton #1 line. The PCBs and MOAB can be operated by remote supervisory control from the UGI Corporation System operator's office. PCB and MOAB status is monitored by PP&L's System Operating offices.

8.2.1.4 Offsite Power System Monitoring

PP&L's transmission lines are patrolled approximately three times throughout a year to ensure that the physical and electrical integrity of the transmission line supports, hardware,

insulators, and conductors is maintained for safe and reliable continuity of service.

The periodic transmission line patrol is conducted by helicopter. Less frequent foot patrols and selective structure inspections are performed depending on the age of the line.

Monitoring of the Unit #1 and Unit #2 offsite power sources in the plant control room is via a hardwired mimic bus arrangement which shows startup transformers #10 and #20, the transformer #10 and #20 motor operated air break switches, the 13.8 KV start-up buses, the 13.8 KV bus feeder breakers, and the 13.8 KV bus tie breaker. Annunciation signals abnormal tripping to the control room operator. Control and status indication are provided for the 230 KV MOAB switches and the 13.8 KV breakers. Potential indication for the PP&L grid and 13.8 KV bus and status indication of the 230 KV high speed ground switches are provided.

A cathode ray tube (CRT) display is provided by the plant computer system which provides the operator with additional information about the offsite power sources. The display is a mimic bus arrangement, similar to the hardwired mimic bus, and includes the status of the PCBs at the remote terminals of the transformer #10 and #20 supply lines.

Monitoring of the Unit #1 main generator output leads to the 230 KV switchyard is provided in the control room. A hardwired mimic bus arrangement provides control and status indication of the synchronizing PCB. Potential indication and monitoring of current, watts, vars, watt hours and voltage are provided. Annunciation signals an abnormal change in status of the synchronizing PCB. The computer CRT display system provides the operator with the status of all PCB's in the 230 KV switchyard and the synchronizing PCB via input from PP&L's supervisory control system. Annunciation accompanies a failure of the supervisory system. Manual control of the 230 KV switchyard is by a supervisory system from selected PP&L System Operating facilities.

Monitoring of the Unit #2 main generator output leads and the 500 KV switchyard is provided in the control room via a mimic bus arrangement. PCB open-close status indication and control are provided for all PCBs in the 500 KV switchyard. Except for the main generator synchronizing breakers which are hardwired directly to the control room along with potential indication, all 500 KV PCB control and status indication in the control room is provided through a supervisory system. Digital displays monitor output current, watts, vars, watt hours, and voltage. Annunciation accompanies uncommanded PCB status changes, loss of potential, transformer trouble, fire protection system actuation, carrier equipment failure, and fault recorder failure. Control of the 500 KV switchyard fault recorder and tap change control on

the 500-230 KV transformer are made available to the operator. Similar information is provided to the control room operator via the computer CRT mimic bus arrangement display through the supervisory system. Primary control of the 500 KV switchyard is via the System Operating supervisory control system except for the main generator synchronizing breakers which can be controlled only by the plant operator.

Preoperational and initial startup testing of all apparatus, equipment, relaying, and PCBs is conducted at transformers #10 and #20 and the 500 KV and 230 KV switchyards to ensure compliance with design criteria and standards.

PCB protective relay testing, maintenance, and calibration in the 230 KV and 500 KV switchyards, Montour switchyard and at transformers #10 and #20 will be conducted approximately once every two years. PCB protective relay testing, maintenance and calibration at Mountain switchyard is performed approximately every year.

8.2.1.5 Industry Standards

The requirements, criteria and recommended practices set forth in the following documents are implemented in the design of the transmission system:

A National Electric Safety Code, 7th Addition.

B. PJM Interconnection Protective Relaying Philosophy and Design Standards

C. MAAC Group Reliability Principles and Standards for Planning Bulk Power Electric Supply System of MAAC Group, July 18, 1968 (Appendix 8.2A)

D. In general, high voltage circuit breakers are manufactured and tested in accordance with the latest recommendations and rules of the ANSI, IEEE, NEMA, and AEIC.

E. Pennsylvania Power & Light Company Substation and Relay and Control Engineering Instruction Manuals, Engineering and Construction Standards, Operating Principles and Practices; Relay and Control Facilities 3/3/76 and sound engineering principles. The design criteria include consideration of aesthetics, reliability, economics, and safety.

8.2.2 Analysis

8.2.2.1 Grid Availability

The offsite power sources provide adequate capacity and capability to start and operate safety related equipment. In addition, the sources provide both redundancy and electrical and physical independence such that no single event is likely to cause a simultaneous outage of both sources in such a way that neither can be returned to service in time to prevent fuel design limits and design conditions of the reactor coolant pressure boundary from being exceeded. Each of the circuits from the off-site transmision network to the safety related distribution buses has the capacity and capability to supply the assigned loads during normal and abnormal operating conditions, accident conditions and plant shutdown conditions.

The PP&L bulk power system is planned in accordance with established PP&L bulk power planning criteria. These criteria are based on the Reliability Principles and Standards of the Mid-Atlantic Area Council (MAAC). MAAC is a regional reliability council of the National Electric Reliability Council (NERC). MAAC is comprised of the electric utility companies of the Pennsylvania-New Jersey-Maryland (PJM) Interconnection, of which PP&L is a member. The primary objective of MAAC is to augment reliability through a continuing review of all planning in connection with additions or revisions to generating plant or bulk power transmission facilities. The PP&L bulk power system is designed to meet the MAAC Reliability Principles and Standards, which are included in Appendix 8.2A.

Digital power flow and transient stability studies were conducted to demonstrate that bulk power system is in compliance with the MAAC reliability criteria. The digital power flow studies included an evaluation of all practical single contingencies, including double circuit tower line, outage conditions and several abnormal system disturbance conditions.

Based on historical operating data for the PP&L transmission network, the annual forced outage rate per 100 circuit miles for 500 KV and 230 KV lines is 0.46 and 6.04 outages, respectively. The number of permanent faults per year per 100 circuit miles for 500 KV and 230 KV lines is 0.23 and 1.79 respectively. The duration of the individual outages varies greatly (from 3 minutes to in excess of 8 hours) depending on the cause of the outage. The major causes of forced outages and permanent faults are lightning and weather related phenomena, tree contacts, equipment failure or malfunction, and emergency maintenance.

8.2.2.2 Stability Analysis

Transient stability studies were conducted using a digital computer program. These studies show that for various 230 KV and 500 KV bus and line faults, system stability and satisfactory

recovery voltages are maintained resulting in uninterrupted supply to the offsite power system. Specifically, the system is stable for any three phase fault cleared in normal clearing time and for any single phase to ground fault with delayed clearing. The system is also stable for any three phase fault applied to the 500 KV and 230 KV transmssion associated with the Susquehanna plant and cleared with delay. A transient stability case list is included in Table 8.2-1.

The loss of either Susquehanna Unit #1 or Unit #2 represents the loss of the largest single supply to the network. For the loss of either Susquehanna unit, grid stability and the integrity of supply to the offsite power system are maintained. Grid stability and the integrity of supply to the offsite power system are also maintained for the loss of any other single generating unit in the network. Supply to at least one of the offsite power sources is also maintained for the following abnormal disturbances:

1. The sudden loss of all lines emanating from the Susquehanna 230 KV Switchyard,

2. The sudden loss of all lines emanating from the Susquehanna 500 KV Switchyard.

No single occurance is likely to cause a simultaneous outage of all offsite sources during operating, accident,or adverse environmental conditions. While the loss of all offsite power is improbable, such an event would not prevent the safe shutdown of the station because the onsite batteries and standby diesel generators are able to supply the necessary power to systems required for safe shutdown.

TABLE 8.2-1

1982 50% OF SUMMER PEAK LOAD

SUSQUEHANNA UNIT #1 & #2
STABILITY CASE LIST

CASE	DESCRIPTION	Result
1	3 phase fault at Susquehanna 500 KV on the Sunbury 500 KV line. Fault cleared in normal 3.5 cycle clearing time.	Stable
2	3 phase fault at Susquehanna 500 KV on the Sunbury 500 KV line with one breaker pole stuck at Sunbury. Clear Susquehanna in 3.5 cycles. Clear remote terminal in 7.5 cycles.	Stable
3	3 phase fault at Susquehanna 500 KV on Wescosville 500 KV line with one Susquehanna 500/230 KV transformer breaker pole stuck. Clear remote terminal in 3.5 cycles. Clear Susquehanna in 7.5 cycles.	Stable
4	3 phase fault at Susquehanna 500 KV on Sunbury 500 KV line with one Susquehanna 500/230 KV breaker pole stuck. Clear remote terminal in 3.5 cycles. Clear Susquehanna in 7.5 cycles.	Stable
5	Phase-ground fault at Susquehanna 500 KV on Sunbury 500 KV line with Susquehanna 500/230 KV breaker stuck. Clear remote terminal in 3.5 cycles. Clear Susuqehanna in 12.0 cycles.	Stable
6	3 phase fault at Susquehanna 230 KV on the Susquehanna 500/230 KV transformer. Fault cleared in normal 4.0 cycle clearing time.	Stable
7	3 phase fault at Montour 230 KV on Susquehanna 230 KV line. Fault cleared in normal 4.0 cycle clearing time. (Reclosed after 10 seconds).	Stable
8	3 phase fault at Susquehanna 230 KV on Montour line with stuck west bus breaker. Clear remote terminal in 4.0 cycles, clear Susquehanna in 8.0 cycles (lose Stanton-Susquehanna #2 230 KV line).	Stable
9	3 phase fault at Susquehanna 230 KV on Jenkins line with stuck	Stable

TABLE 8.2-1 (Continued)

CASE	DESCRIPTION	RESULT
	east bus breaker. Clear remote terminal in 6.0 cycles, clear Susquehanna in 8.0 cycles.	
10	3 phase fault at Susquehanna 230 KV on the 500/230 KV transformer with one pole stuck on west bus breaker. Clear two poles in 4.0 cycles, clear fault in 8.0 cycles (lose Stanton-Susquehanna #2 230 KV line).	Stable
11	3 phase fault at Susquehanna 230 KV on Harwood line with stuck tie breaker pole. Clear two poles in 4.0 cycles. Clear stuck pole in 8.0 cycles (lose Sunbury-Susquehanna 230 KV line).	Stable
12	3 phase fault at Susquehanna 230 KV on E. Palmerton line with one pole stuck on west bus breaker. Clear two poles in 4.0 cycles. Clear stuck pole in 8.0 cycles (lose Stanton-Susquehanna #2 230 KV line).	Stable
13	Phase-ground fault at Susquehanna 500 KV on Wescosville 500 KV line with Susquehanna 500/230 KV breaker stuck. Clear remote terminal in 3.5 cycles. Clear Susquehanna in 12.0 cycles.	Stable
14	Susquehanna-Wescosville 500 KV and Susquehanna-Harwood (E. Palmerton) Double Circuit 230 KV crossing failure (3 phase fault on all circuits). Trip Susquehanna Unit #1 in 12 cycles. Clear Susquehanna-Wescosville 500 KV line in 3.5 cycles. Clear Susquehanna-Harwood and Susquehanna-E. Palmerton 230 KV lines in 4.0 cycles.	Stable
15	3 phase fault near E. Palmerton on all lines in E. Palmerton-Harwood R/W corridor. Clear Susquehanna-Wescosville 500 KV line in 3.5 cycles. Clear E. Palmerton-Susquehanna and Harwood -Siegfried 230 KV lines in 4.0 cycles.	Stable

TABLE 8.2-1 (Continued)

CASE	DESCRIPTION	RESULT
16	3 phase fault near Susquehanna on both lines in Sunbury-Susquehanna R/W corridor. Clear Sunbury-Susquehanna 500 KV line in 3.5 cycles. Clear Sunbury-Susquehanna 230 KV line in 4.0 cycles.	Stable
17	3 phase fault near Susquehanna 500 KV at Sunbury 230 KV line crossing. Trip Susquehanna-Wescosville 500 KV, Sunbury-Susquehanna 500 KV, and Unit #2 in 3.5 cycles. Trip Sunbury-Susquehanna 230 KV in 4.0 cycles.	Stable
18	3 phase fault at Susquehanna 230 KV on Harwood (E. Palmerton) Double Circuit. Trip Harwood and E. Palmerton breakers in 4.0 cycles.	Stable
19	3 phase fault at Columbia-Frackville 230 KV line crossing. Trip Sunbury-Susquehanna 500 KV line in 3.5 cycles. Trip Columbia-Frackville and Sunbury-Susquehanna 230 KV lines in 6.0 cycles.	Stable
20	3 phase fault on 230 KV side of Unit #1 main transformer. Trip Unit #1 main transformer. Trip Unit #1 and overtrip Unit #2 in 4.0 cycles (loss of entire station).	Stable
21	3 phase fault at Susquehanna 230 KV on Unit #1 generator leads with a stuck west bus breaker. Trip Unit #1 and Stanton #2 line in 12.0 cycles.	Stable

Appendix 9B

COMPLIANCE WITH
NRC BRANCH TECHNICAL POSITION ASB 9-1
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 1 REACTOR BUILDING CRANE

The attached table compares the design of the Unit 1 crane with Branch Technical Position ASB 9-1.

Compliance with each regulatory position in the BTP is classified into one of the following categories:

a) comply

b) complied with based on our interpretation of the intent of regulatory position

c) complied with by use of alternate means or methods

d) do not comply

Justification is provided for each item of noncompliance.

TABLE 9B-1

COMPARISON OF UNIT 1 REACTOR BUILDING CRANE DESIGN WITH BTP ASB 9-1

REGULATORY POSITION		Compliance	Compliance based on our interpretation of regulatory position	Use of alternative method to meet the intent of regulatory position	Non-compliance	Remarks
C.1.a	Separate Performance Spec.		X			Item #1
b	Environ. Oper. Contitions Structural mvt. selection	X				
c	Seismic Category I	X				
d	NDE - Lamellar Tearing		X			Item #2
e	Fatique Analysis		X			Item #3
f	Preheat-Postheat-Welding		X			Item #4
C.2.a	Controls-Devices-Safe Holding Position	X				
b	Aux. Syst., Dual Comp. Immob. Position	X				
c	Means for Repairing		X			Item #5

TABLE 9B-1 page 2

COMPARISON OF UNIT 1 REACTOR BUILDING CRANE DESIGN WITH BTP ASB 9-1

REGULATORY POSITION		Compliance	Compliance based on our interpretation of regulatory position	Use of alternative method to meet the intent of regulatory position	Non-compliance	Remarks
C.3.a	Dual Load Attach. Points	X				
b	Lifting Devices-Redundant Design	X				
c	Dual hoisting eq. 5 fpm lim.	X				
d	Head Load Block balance	X				
e	Dual Reeving System-Rope Std.	X				
f	Fleet Angles				X	Item #6
g	200-static design test				X	Item #7
h	Sensor overspeed overloading etc.	X				
i	Control system Motors-torque		X			Item #8
j	Two-blocking-precautions, etc.		X			Item #9
k	Drum protection	X				

TABLE 9B-1 page 3

COMPARISON OF UNIT 1 REACTOR BUILDING CRANE DESIGN WITH BTP ASB 9-1

REGULATORY POSITION		Compliance	Compliance based on our interpretation of regulatory position	Use of alternative method to meet the intent of regulatory position	Non-compliance	Remarks
l	Excessive breakdown torque		X			Item #10
m	Hoisting brakes holding brakes		X			Item #11
n	Dynamic-Static alignment	X				
o	Increment drives	X				
p	Trolley + Bridge					
	i. Motors		X			Item #12
	ii. Speeds				X	
q	Cab located controls	X				
r	Safety devices, limit devices	X				
s	Operating Manuals-MWL	X				Item #13
t	Change from Constr. to Operating	X				
u	Installation Instructions	X				
C.4.a	Mechanical Check	X				

TABLE 9B-1 page 4

COMPARISON OF UNIT 1 REACTOR BUILDING CRANE DESIGN WITH BTP ASB 9-1

REGULATORY POSITION	Compliance	Compliance based on our interpretation of regulatory position	Use of alternative method to meet the intent of regulatory position	Non-compliance	Remarks
b 125% Static test (2 - block)	X				
i. 125% static test	X				
ii. 2-block				X	Item #14
c Preventive Maintenance Program	X				Item #15

TABLE 9B-1 page 5

NOTES:

Item #1

The load lifts during construction are not greater than those for plant operation, therefore no separate specifications have been prepared.

Item #2

We consider that the Regulatory Position is complied with to the extend that all major structural load carrying welds are 100% magnetic particle (MT) tested. Volumetric examination, in our opinion, (RT or UT) of the welds used in the assembly of the crane will not produce meaningful results because of the joint geometries, therefore, they are not performed.

Item #3

The crane is specified and has been designated as Service Class C, per CMAA-70. This standard determines allowable stresses for the crane structural and mechanical components as a function of the specified crane service class. Service Class C allows for 100,000 to 500,000 loading cycles, which by far exceeds our conservatively estimated 4,000 cycle life. Therefore, no additional fatigue analysis have been performed.

Item #4

This regulatory position is complied with to the extent that the preheat and postheat treatment of the welds is in accordance with AWS D1.1.

Item #5

Provisions are made for manual operation of the main hoist holding brakes for lowering the load (Item #11). No special provisions are made for manually moving the immobilized bridge or trolley. However, there are options for moving the bridge or trolley, if the electric power cannot be restored.

Item #6

The fleet angle from drum to lead sheave and between sheaves does not exceed 3-1/2 degrees (3'7" actual design). The NRC position recommends limiting the fleet angles between individual sheaves to 1-1/2 degrees. The use of the 3-1/2 degrees limit is justified because:

1. The 3-1/2 degree limitation has been proven to be a reliable parameter for rope leads off of drums which are more critical

than rope leads from sheaves; the latter being more deeply grooved.

2. With redundant reeving, sheave spacings are double the normal spacings. Thus to maintain a 1-1/2 degree fleet angle, the distance from the hook to the top of the crane would have to be needlessly and excessively increased to such a degree that it would be inconsistent with a good crane design. This would have necessitated, at least, eight to nine feet increase in the building height.

   The design ratio of running sheaves pitch diameters to the rope diameter is 24:1 instead of the 30:1 or 26:1 recommended by the NRC. The 24:1 ratio is justified because: Due to the large diameter of the wore rope used, 30:1 and 26:1 diameter ratio sheave blanks are not readily available. Also 24:1 ration is recommended by ASME Standard Committee on the Design of Overhead and Gantry Handling Systems for Critical Loads at Nuclear Power Plants, in their comments to the NRC on RG 1.104 dated March 18, 1976, and is consistent with the recommendations of CMAA Specification #70.

Item #7

The 200% load test is in conflict with current safety standard codes, specifically ANSI B.30.2 which states that the entire crane is to be load-tested in the field at 125% of the rated load. If this requirement for the load test of 200% of the rated load is to meet the safety requirements and be within the allowable stress values for the crane design, it would require a large crane. Also, the test may not proof the wire rope at 200% load as permanent deformation can results, and the rope will have to be discarded after the test. We do not recommend testing any portion of the crane at 200% load, except each redundant hook, which is specified to be tested at twice the rated load. However, the hoisting system components are all designed to support a static load of 200% of the design rated load.

Item #8

The electric controls are set to limit the motor torque to 150% of rated motor torque, and are field adjustable between 125% and 200% of that torque. Note that the "rated", not "required" torque is limited. The "required" rating of the motor is not clearly defined and opens the possibility for its misapplication. Ratios of motor horsepower are given in Items 10 and 12.

Item #9

The mechanical and structural components of the hoisting system should be protected against the possibility of two blocking or load hangup occurrence during hoisting. This protection is

provided by a system of limit switches such that two blocking could not occur after a first order failure. First order protection for raising and lowering is provided by a geared limit switch coupled to a shaft on the drive gear case and wired to stop the hoist motion and set a hoist brake by opening a reversing switch control circuit. The second order protection in the raising direction is provided by a power circuit limit switch wired to positively interrupt motor raising and lowering circuits and set brakes. The interruption by the power circuit limit switch will require manual release of the hoist holding brakes to lower the upper block and reset the switch. With this arrangement the operator will be alerted to the fact that the geared type lower upper limit switch has failed. The second order protection in the lowering direction is provided by a second geared limit switch coupled directly to drum shaft and wired so as to open the control circuit of the line contactor.

The first order protection against load hangup is an overload device in the hoisting train that senses the overload and interrupts motor raising circuit and set brakes. The overload device can be set as low as 110% of the rated load. The second order of protection is provided with "over current" and "current rate of rise" set at a higher torque (load) level, than the overload device. This is necessary to allow for an additional torque required to accelerate the load and the hoist mechanisms from a standstill position.

Item #10

The hoist motor rating is limited to 105% of the combined calculated running and accelerating horsepower required to accelerate the rated load to the maximum design hoist speed. This regulatory position does not directly address the accelerating portion of the calculated design horsepower; however, the paragraph entitled, "Drivers and Controls" on pages 1.104-3&4 of Regulatory Guide 104, dated February 1976 calls for its consideration. Based on the above interpretation, this regulatory position is considered implemented.

Item #11

The crane design meets the requirements of this Regulatory Position except that holding brake heat dissapation will be accomplished by alternating the lowering and holding to provide time for cooling the braking mechanism. Also administrative control (hand held tachometer) will be used to limit the lowering speed to less than 3.5 fpm.

TABLE 9B-1 page 8

Item #12

The ratios of motor horsepower ratings to the combined calculated running and accelerating horsepowers required to accelerate the load to the maximum design speed are as follows:

trolley - 101%
bridge - 104%

Refer to Item 9 for a discussion of the inclusion of the accelerating horsepower to the motor horsepowers.

No special provisions are made for manual operation of the bridge and trolley holding brakes. If necessary, they can be released by using various methods not excluding brake partial disassembly.

The requirement that "opposite wheels on bridge and trolley have identical diameters," is not practical, since it has no tolerance allowance. Our specification calls for wheels ground true to .001 inch per inch of diameter. Trolley speed, using main hoist is 10 fpm, well below 30 fpm recommended by NRC. However, the trolley speed, using auxiliary hoist, is 50 fpm, therefore administrative controls must be maintained to prevent inadvertent running of the trolley with loaded main hoist at the higher (50 fpm) speed.

The bridge speed (50 fpm) exceeds slightly the NRC recommended speed of 40 fpm. The substantial runway length (323 ft.) stepless type bridge speed control, and minor (10 fpm) difference between the NRC recommended and the specified speeds do not justify the reduction of the bridge speed from 50 fpm to 40 fpm.

Item #13

The Unit 1 and 2 crane main hoist rated loads and design loads are the same and equal 125 tons.

Item #14

As stated in Item 9, protective means are provided to prevent the occurrence of two-blocking or load hang-ups. Therefore, there is no need to run the recommended tests. In addition, the recommended tests present a potential for injuring personnel and for causing an undetectable damage to the hoise components. These conditions will exist whether the tests are performed in the vendor shop or at the site. Also, it would be difficult, after the tests, to assess any potential damages that might have resulted from those tests. Verification testing of the upper limit switches and the overloads will be performed to assure their proper functioning.

TABLE 9B-1 page 9

Item #15

The Unit 1 and 2 cranes will be maintained at their rated capacities, i.e., 125 tons main hoists and 5 tons auxiliary hoists.

CHAPTER 10

STEAM AND POWER CONVERSION SYSTEM

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CHAPTER 10

TABLES

10.1 SUMMARY DESCRIPTION

The components of the steam and power conversion system are designed to produce electrical power from the steam coming from the reactor, condense the steam into water, and return the condensate to the reactor as heated feedwater, with a major portion of the gaseous, dissolved, and particulate impurities removed.

The power conversion system consists of the following component:

1) Turbine Generator with Auxiliaries

2) Main Condenser

3) Condensate Pumps

4) Air Ejector with Water Condenser

5) Gland Steam Condenser

6) Condensate Demineralizer

7) Five Stages of Feedwater Heaters

8) Reactor Feed Pumps with Turbine Drives and Auxiliaries

9) Interconnecting Piping and Valves

10) Drain Coolers

Steam generated in the reactor is supplied to the high pressure turbine through the main stop and control valves. The steam then passes through the high pressure (HP) turbine and exhausts through cross around lines to two moisture separators which remove moisture from the steam. The dried steam leaves the moisture separators and enters the low pressure (LP) turbines, which share a common shaft with the HP turbine, through combined intercept valves. After passing through the low pressure turbines the steam exhausts to the main condensers where it is condensed by the circulating water system (Subsection 10.4.5) deaerated and collected in the hotwell of the condenser. The condensate pumps remove the condensate from the hotwell and pump it through the air ejector intercondenser, the gland steam condenser, the condensate demineralizer, the drain coolers and the five stages of feedwater heaters to the suction of the reactor feed pumps which pump the condensate back into the reactor vessel.

Steam is extracted from the HP and LP turbines and used to heat the condensate as it passes through the various feedwater

heaters. The extraction steam is condensed in each heater and the condensed steam drained to the next lowest pressure heater. The total cascaded heater drains are collected in the drain cooler from which they drain back to the condenser. The moisture removed from the steam by the moisture separators is drained to Heater No. 4 where it mixes with the condensed extraction steam and is eventually drained back to the condenser.

Should the water level in any heater or moisture separator become too high, the drains will be dumped directly to the condenser to prevent water damage to the turbine.

If the reactor produces more steam than the turbine can use the excess, up to 25 percent of rated flow, is dumped to the condenser through the bypass valves (See Subsection 10.4.4).

The steam and power conversion systems are sized for the turbine valves wide open condition of 3439 MWt.

Biological shielding is provided around the main turbine, moisture separators, feedwater heaters, condensers and reactor feed pump turbines to protect operating personnel from exposure to high radiation levels. Section 12.3 provides additional discussion on radiation protection.

Figures 10.1-1 and 10.1-2 show the maximum guaranteed and maximum calculated heat balances respectively.

Typical parameters are summarized in Table 10.1-1.

Instrumentation is commercial quality, designed to meet the process requirements and the G.E. turbine generator requirements. These instruments are described further in Sections 10.2 through 10.4. The turbine instrumentation is discussed in Subsection 7.2.2.1.3: Control value fast closure and stop value closure which initiates scram in the RPS.

10.2 TURBINE-GENERATOR

10.2.1 DESIGN BASES

The turbine is an 1,800 rpm, tandem compound, six-flow, non-reheat steam turbine with 38 in. last-stage buckets. The capability of the turbine is 1,084,825 kW when operating with initial steam conditions of 965 psia, 1191.5 H, while exhausting to the multipressure condenser at 2.99, 3.56 and 4.43 in. HgAbs, back pressures, 0 percent makeup and extracting steam for normal five stage feedwater heating and feed pump turbine drives. The turbine is expected (not guaranteed) to produce 1,134,993 kW when operating at valves wide open (VWO) and with corresponding VWO steam and cycle conditions shown on the heat balance.

The generator is a 1,280,000 kVa, 1,800 rpm, direct connected, 4 pole, 60 Hz, 24,000 V, liquid cooled stator, hydrogen cooled rotor, synchronous generator rated at 0.90 power factor, 0.58 short circuit ratio at a maximum hydrogen pressure of 75 psig. The generator is sized to accept the gross output of the turbine.

The Alterrex excitation system consists of a 60 Hz, 1,800 rpm air cooled Alterrex generator and liquid cooled rectifiers with static thyristor automatic regulation equipment. The exciter is rated for a maximum output of 3210 kW at 530 V.

The turbine-generator control is accomplished by an electrohydraulic control (EHC) system capable of controlling speed, load, steam pressure and flow under startup, shutdown, transient and steady state conditions.

The turbine-generator is normally base loaded. However, the design allows for the units to operate on a load following basis.

The turbine-generator unit, a GE design, is built in accordance with GE standards and codes. The moisture separator vessels and steam seal evaporator vessels are built in accordance with ASME B&PV Code, Section VIII.

The steam generation rate has the ability to follow turbine load demand changes by as much as 35 percent without control rod movement merely by changing the recirculation flow rate through the core. If a load reduction of more than 10 percent occurs, the turbine bypass valves will open momentarily until the recirculation rate is sufficiently reduced. Bypass valves have the ability to bypass 25 percent of the flow.

The turbine control valves are capable of changing turbine steam flow at a rate of at least 10 percent nuclear boiler warranted

flow per second in both the opening and closing directions for adequate pressure control performance.

During any event resulting in turbine stop valve closure, turbine inlet steam flow is not reduced faster than permitted by Figure 10.2-1.

During any event resulting in turbine control valve fast closure, turbine inlet steam flow is not reduced faster than permitted by Figure 10.2-2.

10.2.2 DESCRIPTION

10.2.2.1 Turbine

The turbine unit consists of one double flow high pressure turbine and three double exhaust flow low pressure turbines. The unit includes two horizontal moisture separator vessels located on the operating floor, one on each side of the turbine. The moisture separator vessels are of the non-reheat type.

Steam from the reactor enters the power conversion system through four main steamlines. Each of the four main steamlines to the high pressure turbine is connected to a main steam stop valve and a main steam control valve. The four stop valves and four control valves are combined to form a single valve chest. A pressure equalizing line connects the stop valves together just below the valve seats. Six combined intermediate valves (CIV) (each composed of an intercept valve and an intermediate stop valve) are located in each line between the moisture separator vessels and the low pressure turbines. A five valve bypass valve chest is connected to each of the main steamline between the main steam isolation valves and main steam stop valves to remove excess flow to the condenser.

There is one stage of extraction from the high pressure turbine and four stages of extraction from each low pressure turbine. The extraction steam is used to heat the five stages of feedwater heating.

A portion of the cross-around steam is used to drive the reactor feed pump turbines (RFPTs) during normal operation.

The turbine-generator is provided with an emergency trip system that closed the main stop valves, control valves and combined intermediate valves, thus shutting down the turbine, on the following signals:

1. Turbine approximately 10% above rated speed.

2. Turbine approximately 12% above rated speed.
3. Vacuum decreases to less than 21.7 Hg.
4. Excessive thrust bearing wear.
5. Exhaust hood temperature in excess of 225°.
6. Prolonged loss of generator stator coolant.
7. Electrical trip, via master trip solenoids.
8. Loss of hydraulic fluid supply pressure. Loss of emergency trip system fluid pressure automatically closes the turbine valves and then energizes the master trip relay to prevent a false restart at 1100 psig decreasing.

9. Signal from High turbine vibration.
10. Loss of both speed signals above 100 rpm.
11. Loss of both the primary and secondary 24 VDC power supplies.
12. Mechanical trip via manual trip handle of mechanical trip solenoid.
13. High level in a moisture separator drain system.
14. Main shaft lube oil pump low pressure trip above 1300 rpm.
15. Primary and backup unit protection lockout relay trip.
16. High reactor water level trip at 54".
17. Loss of ETS pressure trip at 800 psig decreasing.

Tripping the turbine will automatically cause the reactor to scram for reactor power greater than 30 percent.

10.2.2.2 Generator and Exciter

The generator stator is water cooled and the rotor is hydrogen cooled. The generator hydrogen system includes all necessary controls and regulators for hydrogen cooling (See Figure 10.2-11). The hydrogen purity inside the generator is monitored on a continual basis. The pipe from the Generator Hydrogen system is routed below grade to the generator and does not enter any safety related areas. A seal oil system is provided to prevent hydrogen leakage through the generator shaft seals. The Bulk Hydrogen system is located between the cooling towers at grade level. A hydrogen makeup supply is provided outside the turbine building to replace any hydrogen leakage from the generator. To avoid having an explosive hydrogen-air mixture in the generator at any time, either when the generator is being filled with hydrogen prior to being placed into service, or when hydrogen is being removed from the generator prior to opening the generator for inspection or repairs, carbon dioxide is used for purging out the air or hydrogen in the generator casing. The generator is designed to withstand a hydrogen detonation.

Automatic water type fire protection systems are provided to protect the turbine and generator bearings, the area below the generator, the hydrogen seal oil system, the permanent bulk hydrogen storage area and the hydrogen truck unloading area. In addition, portable fire extinguishers and fire hose will be provided.

10.2.2.3 Protective Valves Functions

The primary function of the turbine stop valves is to quickly shut off steam to the turbine under emergency conditions. The stop valve disks are totally unbalanced and cannot open against full pressure drop. An internal bypass valve is provided in one of the four stop valves to permit slow warming of all stop and control valves and to pressurize the stop valve below seat area to allow valve opening.

The function of the turbine control valves is to throttle steam flow to the turbine. The valves are of sufficient size, relative to their cracking pressure, to require that they be partially balanced. A small internal valve is opened first to decrease the pressure in a balance chamber. The valves are opened by individual hydraulic cylinders.

The function of the bypass valves is to pass steam directly from the reactor to the condenser without the steam going through the turbine. The bypass valve chest is connected directly to the steam leads from the reactor. This chest is composed of five valves operated by individual hydraulic cylinders. When the valves are open steam flows from the chest, through the valve seat, out the discharge casing, and through connecting piping to the pressure breakdown assemblies where a series of baffle plates and orifices is used to further reduce the steam pressure before the steam enters the condenser. (See Subsection 10.4.4)

The function of the combined intermediate valves (CIV's) is to protect the turbine against overspeed from stored steam in the cross-around piping and moisture separator vessels and to throttle and balance steam flow to the LP turbines. Each valve is composed of an intercept valve and an intermediate stop valve incorporated into a single casing. The two valves have separate operating mechanisms and controls. The valves are located as close to the turbine as possible to limit the amount of uncontrolled steam available as an overspeed source.

During normal plant operation the intercept valves will be open. The intercept valves are capable of opening against maximum cross-around pressure and of controlling turbine speed during blowdown following a load rejection. The intermediate stop valves also remain open for normal operation and trip closed by actuation of the emergency governor or operation of the master trip. They provide backup protection if the intercept valves or the normal control devices fail.

10.2.2.4 Extraction System Check Valves

The energy contained in the extraction and feedwater heater system can be of sufficient magnitude to cause overspeed of the turbine-generator following an electrical load rejection or turbine trip. To prevent this the energy must be contained in the piping and feedwater heaters. This is done by installing positive closing non return valves (PCNRV) and antiflash baffles in the heaters. GE steam turbine design rules and code requirements specify that the turbine controls will be capable of preventing the turbine speed from rising above a certain maximum value after a full load rejection or trip. The PCNR valves and antiflash baffles limit the amount of energy flashing back into the turbine so that the turbine speed increase is held below the maximum value. Antiflash baffles are used in feedwater heaters 1 and 2 extraction steamline since the distance to the turbine is short and internal energy is low. PCNRVs are installed in the extraction lines, to feedwater heaters 3, 4, and 5.

10.2.2.5 Control System

The turbine generator control system is a GE Mark I electrohydraulic control (EHC) system. The speed control unit produces the speed/acceleration error signal that is determined by comparing the desired speed from the reference speed circuit, with the actual speed of the turbine for steady state conditions. For step changes in speed, an acceleration reference circuit takes over to either accelerate or decelerate the turbine at a selected rate to the new speed. There is no limit to the deceleration. The speed/acceleration error signal is combined with the load requirements on the load control unit to provide the flow signal to the control valves.

Because of the importance of overspeed protection the speed control signal has two independent redundant channels. Two independent pulse signals are obtained from magnetic pick-ups located over a gear-toothed wheel on the turbine shaft. Loss of both speed signals will trip the turbine.

10.2.2.6 Overspeed Protection

To protect the turbine generator against overspeed two trip devices are provided either of which when initiated will close the main stop valves, control valves, and combined intercept valves thus shutting down the turbine.

These two trip devices are as follows:

1. A mechanical overspeed trip which is initiated if the turbine speed reaches approximately 10% above rated speed, and

2. An electrical overspeed trip which serves as a backup to the mechanical trip and is initiated at approximately 12% above rated speed.

The mechanical overspeed trip device is an unbalanced ring mounted on the turbine shaft and held concentric with it by a
11 spring (See Figure 10.2-3). When the turbine speed reaches the trip speed (10% above rated) the centrifugal force acting on the ring overcomes the tension of the spring and the ring snaps to an eccentric position. In doing this it strikes the trip finger which operates the mechanical trip valve, MTV. This is a three way valve that feeds hydraulic fluid (1600 psi) to the lockout valve, and when tripped blocks the hydraulic fluid supply system and removes the emergency trip system pressure which causes the main stop valves, control valves and combined intercept valves to
1 close. Failure of the hydraulic portion of this trip will result in a stop valve closure.

The electrical overspeed trip receives its signal from a 112% speed trip relay (VCS840) that is operated by a speed signal sensed by a magnetic pickup from a toothed wheel on the turbine shaft and fed to a power amplifier and megacycles circuit whose output is a dc voltage proportional to speed (See Figure 10.2-4).

The signal from the speed trip relay energizes the master trip relay XKT1000 (Figure 10.2-5) which then energizes the mechanical trip solenoid MTS and deenergizes the master trip solenoid valves MTSV-A & MTSV-B.

Either one of these actions will trip the turbine, that is close stop, control and combined intercept valves.

When the overspeed trip system is under test, the lockout valve, LV, is actuated which bypasses the mechanical trip valve.
11 However, under this condition, system protection is provided by the backup overspeed trip acting on the master trip solenoid valve, MTSV, by deenergizing MTSV-A & MTSV-B.

An additional feature of the protective system which will minimize the likelihood of an overspeed condition is the power/load unbalance circuitry (Figure 10.2-6)). Generator load is sensed by means of three current transformers and is compared with the turbine power input which is sensed by the turbine intermediate pressure sensor. If the difference between the steam power input and the generator output rises to at least 40% i 35 msec, auxiliary relays will be actuated which will energize the control valves fast closing solenoids, remove the load reference at the load control unit and automatically drive the load reerence motor to zero set point.

Table 10.2-1 summarizes the overall turbine overspeed protection assurance that a stable operation follows a turbine trip can be obtained from the requirement that both the stop valves and the combined intercept valves close in a turbine trip thereby accomplishing two things: a) Preventing steam from the main steam line from entering the turbine and b) preventing the expansion of steam already in the high-pressure stage and in the moisture separator. An addiitonal provision is also made to isolate the major steam extraction lines from the turbine.

There are four steam lines at the high pressure stage, each line is provided with one stop valve and one control valve in series. Steam from the high pressure stage flows to the moisture separators and then to the three low pressure stages. Each of the six low pressure lines has a combined intercept valve which is actually made up of a stop valve in series with a control valve in one housing. All of the above valves close on turbine trip. Assuming a single failure to the above system of 20 valves in case of a turbine overspeed trip signal, the turbine will be successfully tripped. Furthermore, each of the major steam extraction lines have an isolation valve and a bleeder trip valve which are independently closed in case of a turbine trip.

10.2.2.7 Turbine Shell Diaphragms

For overpressure protection of the turbine exhaust hoods and the condenser shells, two diaphragms are provided in each low pressure turbine exhaust hood, which rupture at approximately 5 psig. An exhaust hood spray system is provided to spray condensate into the hoods for overtemperature protection.

10.2.2.8 TURBINE-GENERATOR LOAD FOLLOWING CAPABILITY

The load following feature of the turbine-generator system may be discussed with the aid of Figs. 10.2-7 through 10.2-10. This discussion starts with the generator running at rated speed and at any given load, with bypass valves fully cleared and with control and intercept valve positions corresponding to the generator load. The load follower responds to the increase or decrease in generator output due to system frequency changes by controlling the reactor recirculation flow and hence the reactor steam generation rate. Fast response to a step load change is taken care of by the fast adjustment of the control and intercept valves positions. This increases or decreases the steam pressure drop at the valves when they are throttled closed or open and adjusts steam pressure to the turbine. After a time delay the reactor will have generated enough steam to cover the additional load requirement. The load follower also adjusts the reactor recirculation flow in response to an operator adjustment of the

load reference at the load control unit. The load following capability is limited to ±35% of reactor power level only.

Assume an electrical load increase due to a system frequency dip with steady steam input to the turbine, the turbine speed decreases. In Figure 10.2-7, region C-3, two speed sensors SSPU-1 & SSPU-2 pick up the change in speed. This signal is compared with the speed reference at the speed summing amplifiers and its derivative is compared with the acceleration reference at the acceleration summing amplifier in each of two low value gates (A23 & A26). The resulting outputs are gated together and the controlling signal (SCU) feeds into the load control unit for the control and intercept valves.

In Figure 10.2-8, the signal SCU from the speed control unit (Figure 10.2-7) is compared with the load reference, which is remote manually set, and inputed to the control and intercept valve amplifiers (A48 & A50). These amplifiers provide the position signals for the flow control units of the respective valves which are the final controlling units of the control and intercept valves. A typical valve flow control unit is shown in Figure 10.2-9.

The recirculation flow control signal is the speed signal minus the speed reference (Figure 10.2-8). This is amplified in the auto-load following unit Figure 10.2-10 and fed into the reactor recirculation system flow controller. This is the end point of the turbine-generator system control of load following.

The above analysis is based on a generator load increase. A similar discussion also applies to load decrease. The same effect can be obtained if, instead of a change in electrical load, a load reference change is made by the operator since a recirculation flow error signal is achievable in either case.

10.2.3 TURBINE DISK INTEGRITY

The turbine assembly is designed to withstand normal conditions and anticipated transients including those resulting in turbine trip without loss of structural integrity. The design of the turbine assembly meets the following criteria:

a) Turbine shaft bearings are designed to retain their structural integrity under normal operating loads and anticipated transients, including those leading to turbine trips.

b) The multitude of natural critical frequencies of the turbine shaft assemblies existing between zero speed and 20 percent overspeed is controlled in the design and operation so as to cause no distress to the unit during operation.

c) The maximum tangential stress in wheels and rotors resulting from centrifugal forces, interference fit and thermal gradients will not exceed 0.75 of the yield strength of the materials at 115 percent of rated speed.

10.2.3.1 Material Selection

Turbine wheels and rotors for turbines operating with light water reactors are forged from vacuum degassed Ni-Cr-Mo-V alloy steel by processes which minimize flaw occurrence and provide adequate fracture toughness. Tramp elements are controlled to the lowest practical concentrations consistent with good scrap selection and melting practices, and consistent with obtaining adequate initial and long life fracture toughness for the environment in which the parts operate. The turbine wheel and rotor materials have the lowest fracture appearance transition temperatures (FATT) and highest Charpy V-notch energies obtainable, on a consistent basis from water quenched Ni-Cr-Mo-V material at the sizes and strength levels used. Since actual levels of FATT and Charpy V-notch energy vary depending upon the size of the part and the location within the part, etc., these variations are taken into account in accepting specific forgings for use in turbines for nuclear application. Charpy tests essentially in accordance with Specification ASTM A-370 are included.

10.2.3.2 Fracture Toughness

Suitable material toughness is obtained through the use of materials described in Subsection 10.2.3.1 to produce a balance of adequate material strength and toughness to ensure safety while simultaneously providing high reliability, availability, efficiency, etc., during operation. Bore stress calculations include components due to centrifugal loads, interference fit, and thermal gradients where applicable. The ratio of material fracture toughness, K_{IC} (as derived from material tests on each wheel or rotor) to the maximum tangential stress for wheels and rotors at speeds from normal to 115 percent of rated speed is at least 2 $\sqrt{in.}$ (The highest anticipated speed resulting from a loss of load is 110 percent of rated speed). Adequate material fracture toughness needed to maintain this ratio is assured by destructive tests on material taken from the wheel or rotor using correlation methods which are more conservative than that presented by J.A.Begley and W.A.Logsdon in Westinghouse Scientific Paper 71-1E7-MSLRF-P1.

Turbine operating procedures are employed to preclude brittle fracture at startup by ensuring that the metal temperature of wheels and rotors (a) is adequately above the FATT and (b) as defined above is sufficient to maintain the fracture toughness to tangential stress ratio at or above 2 $\sqrt{in.}$ Details of these start-up procedures are contained in Reference 10.2-1.

10.2.3.3 High-Temperature Properties

The operating temperatures of the high pressure rotor in turbines operating with light water reactors are below the creep rupture range. Therefore, creep rupture is not considered to be a significant factor in assuring rotor integrity over the lifetime of the turbine. Basic data is obtained from laboratory creep rupture tests.

10.2.3.4 Turbine Design

The turbine assembly is designed to withstand normal conditions and anticipated transients including those resulting in turbine trip without loss of structural integrity. The design of the turbine assembly meets the following criteria:

a) The maximum tangential stress in wheels and rotors resulting from centrifugal forces, interference fit, and thermal gradients does not exceed 0.75 of the yield strength of the materials at 115% of rated speed.

b) Turbine shaft bearings are designed to retain their structural integrity under normal operating loads anticipated transients, including those leading to turbine trips.

c) The multiple of natural criteria frequencies of the turbine shaft assemblies existing between zero speed and 20% overspeed are controlled in the design and operation so as to cause no distress to the unit during operation.

10.2.3.5. Pre-service Inspection

The pre-service inspection program is as follows:

a) Wheel and rotor forgings are rough machined with minimum stock allowance prior to heat treatment.

b) Each rotor and wheel forging is subjected to a 100% volumetric (ultrasonic) examination. Each finish-machined rotor and wheel is subjected to a surface magnetic particle and visual examination. Results of the above examination are evaluated by use of General Electric acceptance criteria. These criteria are more restrictive than those specified for Class 1 components in the ASME Boiler and Pressure Vessel Code, Sections III and V and include the requirement that subsurface sonic indications are either removed or evaluated to assure that they will not grow to a size which will comprise the integrity of the unit during the service life of the unit.

c) All finish-machined surfaces are subjected to a magnetic particle examination. No magnetic particle flaw indications are permissible in bores, holes, keyways, and other highly stressed regions.

d) Each fully bucketed turbine rotor assembly is spin tested at or above the maximum speed anticipated following a load rejection from full load.

10.2.3.6 Inservice Inspection

The in-service inspection program for the turbine assembly and valves include the following:

a) Disassembly of the turbine is conducted during plant shutdown coinciding with the in-service inspection schedule. Inspection of all parts that are normally inaccessible when the turbine is assembled for operation, such as couplings,

coupling bolts, turbine shafts, low pressure turbine buckets, low pressure wheels, and high pressure rotors is conducted.

This inspection consists of visual, surface, and volumetric examinations, as indicated below.

1. The bore and keyway region of each wheel receives an ultrasonic examination. In addition, each wheel is inspected visually and by magnetic particle testing on all accessible surfaces. Also, ultrasonic inspection of the tangential entry dovetails and pins of the finger dovetails are conducted. This inspection is conducted at intervals of about 6 years.

2. A thorough volumetric ultrasonic examination of the high pressure rotor is conducted. In addition, all accessible rotor surfaces are inspected visually and by magnetic particle testing. This inspection is conducted at intervals of about 10 years.

3. Visual and surface examination of all pressure buckets.

4. 100% surface examination of couplings and coupling bolts.

b) Dismantle at least one main steam stop valve, one main steam control valve, one reheat stop valve, and one reheat intercept valve, at approximately 3-1/3 year intervals during refueling or maintenance shutdowns coinciding with the in-service inspection schedule and conduct a visual and surface examination of valve seats, wheels, and stems. If unacceptable flaws or excessive corrosion are found in a valve, all valves of its type are inspected. Valve bushings are inspected and cleaned, and bore diameters are checked for proper clearance.

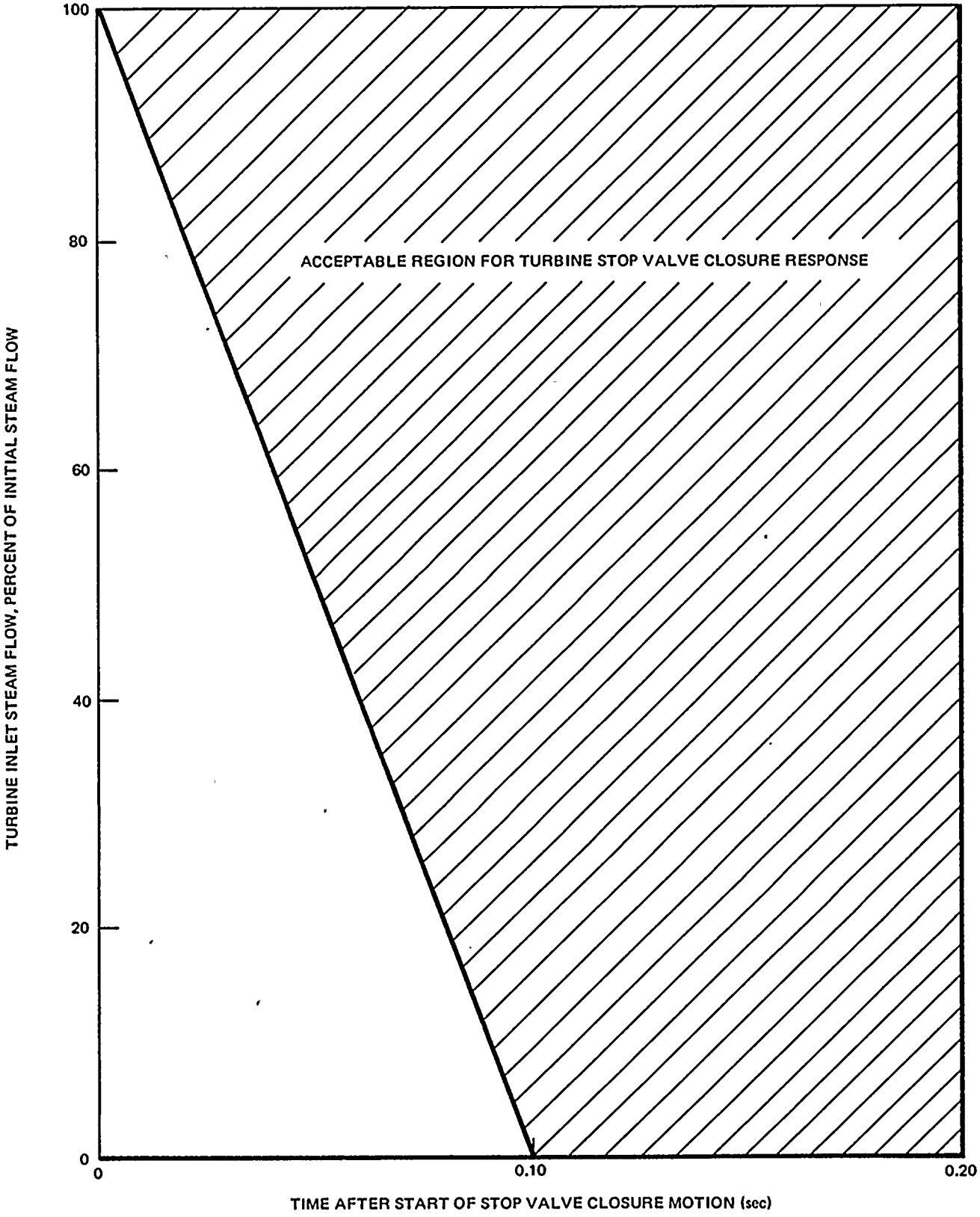
c) Main steam stop and control, reheat stop and intercept valves are exercised at least once a week by closing each valve and observing by the valve position that it moves smoothly to a fully closed position. At least once a month this observation is made by actually watching the valve motion.

10.2.4 EVALUATION

The turbine generator and the related steam system have been radiologically evaluated and the results are described in Chapter 12.

10.2.5 REFERENCES

10.1-1 Spencer, R.C.; and Timo, D.P.; Starting and Loading of Turbines, General Electric Company, Presented at the 36th Annual Meeting of the American Power Conference, Chicago, Illinois, April 29-May 1, 1974.



SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

TURBINE STOP VALVE CLOSURE CHARACTERISTIC

FIGURE 10.2-1

CHAPTER 17

QUALITY ASSURANCE

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17.2 QUALITY ASSURANCE DURING THE OPERATIONS PHASE

17.2.0 INTRODUCTION

PP&L is fully responsible for testing, operating, maintaining, refueling and modifying the Susquehanna SES in compliance with Federal, State, and local laws and the plant operating license requirements. These activities are also performed in response to required codes and specified QA related NRC regulatory guides. These regulatory guides and associated ANSI standards are listed in Table 17.2-1.

To assure compliance with 10CFR50, Appendix B requirements, PP&L has established and implemented a management control plan for assuring the quality of safety-related activities during the operations phase. The plan consists of 1) this Operational Quality Assurance (OQA) Program which contains PP&L's quality assurance commitments to the Nuclear Regulatory Commission; 2) the OQA Manual which contains Operational Policy Statements (OPS) and defines PP&L's policies for meeting these commitments; and 3) functional unit procedures which contain the detailed steps necessary for a functional unit to comply with the OQA Program requirements. The relationships between these documents are shown in Figure 17.2-1.

In implementing the OQA Program, PP&L assures that its activities comply with Federal Regulations which are designed to protect the health and safety of the public.

The OQA policies, goals and objectives of PP&L are stated in the following Nuclear Quality Philosophy and Intent statement.

For the Susquehanna Steam Electric Station, Pennsylvania Power & Light Company will comply with the requirements of 10CFR50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants and other applicable federal regulations with respect to all safety-related activities which include engineering, design, procurement, construction, preoperational testing, power testing, operation, maintenance, refueling, repairing, modification and in-service inspection. PP&L is also committed to be responsive to the applicable Regulatory Guides, Industrial Codes and Standards, or parts thereof, as specifically noted in controlling documents. The applicability of these Guides, Codes, and Standards, or parts thereof, and their effectiveness shall be interpreted by the responsible managers. If Guides, Codes, or Standards are nonexistent or inadequate, PP&L shall develop the required practices and procedures with the controls necessary for their implementation.

17.2.1 ORGANIZATION

PP&L has established the Nuclear Department in order to provide a cohesive management team with the primary objective of providing long term technical and management support for Susquehanna SES. In addition to the resources within the Nuclear Department, auxiliary support is provided by the Construction Manager and the Manager-Procurement. The key management positions responsible for the performance of safety related activities are described in the following subsections. Figure 17.2-2 shows the organizational structure and lines of responsibility for the groups that provide technical and management support for Susquehanna SES.

The positions listed below are described in the following subsections:

- Senior Vice President-Nuclear
- Vice President-Engineering and Construction (E&C)-Nuclear (Project Director)
- Vice President-Nuclear Operations
- Assistant Project Directors
- Manager-Nuclear Plant Engineering
- Project Construction Manager
- Manager-Nuclear Quality Assurance (NQA)
- Superintendent of Plant
- Assistant Superintendent of Plant
- Integrated Startup Group Supervisor
- Manager-Nuclear Support
- Manager-Nuclear Training
- Manager-Nuclear Safety Assessment
- Manager-Nuclear Fuels
- Manager-Nuclear Licensing
- Manager-Nuclear Administration
- Construction Manager
- Manager-Procurement

In addition to the above individuals, the Susquehanna Review Committee (SRC) is established as a review, audit and advisory group, comprised of at least five key Nuclear Department managers, whose function is to independently verify that the Susquehanna SES is being tested, operated and maintained in accordance with all safety related, ALARA and environmental requirements. The SRC will perform the independent review mandated by ANSI N18.7.

17.2.1.1 Senior Vice President - Nuclear

The Senior Vice President - Nuclear has overall authority and responsibility for the Susquehanna OQA Program and, as a result, he:

- Requires the performance of an annual, preplanned and documented assessment of the OQA Program in which corrective action is identified and tracked.
- Sets OQA Policies, goals and objectives for safe operation of Susquehanna SES.
- Commits PP&L to an OQA Program designed to assure compliance with regulatory requirements.
- Requires compliance with the provisions of the OQA Program and causes periodic assessments of PP&L commitments and established practices for safe plant operation.

In order to maintain a continuing involvement in QA matters, the Senior VP-Nuclear receives monthly written reports on the status and adequacy of the OQA Program issued by the Manager-NQA and reviews and approves the Operational Policy Statements contained in the OQA Manual.

The Senior VP - Nuclear delegates to the VP - E&C - Nuclear and the VP-Nuclear Operations those responsibilities for attaining specified quality levels and to the Manager-Nuclear Quality Assurance those responsibilities for verifying that those quality levels have been met.

The Senior VP - Nuclear delegates to the Manager-Nuclear Safety Assessment the responsibility for performing the on-site Independent Safety Engineering Group (ISEG) function mandated by NUREG-0731.

In addition, the Senior Vice President-Nuclear has overall corporate responsibility for Susquehanna SES activities related to engineering, construction, startup and operations. The Senior VP-Nuclear delegates these responsibilities to the Vice President-E&C-Nuclear, and the Vice President-Nuclear Operations. The reporting relationships are shown in Figure 17.2-2.

17.2.1.1.1 Vice President - Engineering & Construction - Nuclear

The VP - E&C (also identified as the Project Director on Figure 17.2-2) has overall corporate responsibility for the Susquehanna engineering, construction and licensing activities as delegated by the Senior VP-Nuclear. In addition, as Project Director, he directs and is accountable for all facets of project performance through project completion.

17.2.1.1.1.1 Assistant Project Directors

The Assistant Project Directors at the site (APD-S) and Allentown (APD-A) are responsible to the Project Director for the engineering and construction aspects of the project. Their responsibilities encompass the day-to-day decision-making process, conduct of project activities, and contract administration. They also coordinate the support functions of other company departments as they interface with the project.

The Assistant Project Director at SSES (APD-S) has a direct coordination and integration relationship with the NQA Resident Nuclear Quality Assurance Engineer (RNQAE). The RNQAE, in turn, has the responsibility to support the APDS objectives by alerting the APDS to quality related matters which have the potential for adversely affecting construction activities.

17.2.1.1.1.2 Manager - Nuclear Plant Engineering

The Manager - NPE is responsible for engineering activities and their quality management. These activities include a) design and design verification related to plant modifications, b) the technical evaluation and approval of acceptable suppliers (excluding nuclear fuel) of parts, components, equipment, and systems, c) specifying technical requirements for the procurement of spare parts, d) modifications to the "as-built" plant, and e) engineering outage support.

17.2.1.1.1.3 Project Construction Manager

The Project Construction Manager is responsible for the performance of construction activities at Susquehanna SES, including that of prime contractors, and for the preparation of equipment and systems for turnover to the Integrated Startup Group for testing. The Project Construction Manager receives administrative and project technical direction from the Project Director through the Assistant Project Director-SSES.

17.2.1.1.1.4 Manager - Nuclear Licensing

The Manager-Nuclear Licensing is responsible for directing the licensing aspects for Susquehanna SES. This includes interfacing with the Licensing Branch of the NRC, updating and changing the FSAR to reflect as-built conditions or modifications, and coordinating responses to the NRC relative to IE Bulletins.

17.2.1.1.2 Vice President - Nuclear Operations

The Vice President-Nuclear Operations is responsible for the Initial Test Program and operation of Susquehanna SES. This includes formulating and establishing the necessary technical and administrative staff and planning and coordinating the activities of these personnel.

The Vice President-Nuclear operations delegates responsibilities to the Superintendent of Plant, Manager-Nuclear Support, Manager-Nuclear Training, and Manager-Nuclear Fuel.

17.2.1.1.2.1 Superintendent of Plant

The Superintendent of Plant is responsible for Susquehanna SES during plant testing, startup and operation and has overall responsibility for the Initial Test Program conducted by the Integrated Startup Group.

The Superintendent of Plant is responsible for the safe operation of Susquehanna SES and has overall responsibility for the execution of the administrative controls at the plant to assure safety. The Superintendent of Plant ensures that plant operations are conducted in accordance with the plant operating license, technical specifications, the FSAR, and the QQA Program with its implementing documents. The Superintendent of Plant delegates his authority for performing activities related to operation of the plant to the Assistant Superintendent of Plant, Supervisor of Operations, Supervisor of Maintenance, Technical Supervisor, and other personnel assigned to the staff organization.

The Superintendent of Plant has direct accountability to and reports to the Vice President-Nuclear Operations for activities directly related to plant support of preoperational testing.

17.2.1.1.2.1.1 Assistant Superintendent of Plant

The Assistant Superintendent of Plant assists the Superintendent of Plant in all matters and assumes the responsibilities of the Superintendent of Plant in his absence.

17.2.1.1.2.1.1.1 Integrated Startup Group Supervisor

The Integrated Startup Group Supervisor has the responsibility for supervising the conduct of the Integrated Startup Group (ISG). The ISG Supervisor reports to the Superintendent of Plant on matters pertaining to the Initial Test Program (ITP). The qualifications for this position are listed in Chapter 14.2.

17.2.1.1.2.2 Manager - Nuclear Support

The Manager - Nuclear Support is responsible for coordinating both Nuclear Department activities and selected outside service organization activities in support of Susquehanna SES startup and operation.

The Manager - Nuclear Support provides technical assistance to the Susquehanna SES Plant Staff in the areas of operation and maintenance.

The Manager - Nuclear Support advises the VP-Nuclear Operations of activities within or affecting the Nuclear Department and advises the Susquehanna SES Plant Staff of potential changes to plant operating and maintenance requirements by reviewing changes to Regulatory Guides, Industry Standards and other industry literature.

17.2.1.1.2.3 Manager - Nuclear Training

The Manager-Nuclear Training is responsible for assessing the long term training needs regarding Susquehanna SES and developing training programs commensurate with those needs.

17.2.1.1.2.5 Manager - Nuclear Fuel

The Manager-Nuclear Fuel is responsible for fuel management activities off-site, such as procurement (including the technical evaluation and approval of acceptable fuel suppliers), design data verification, core transient analyses, and core analysis for the purposes of in-core fuel management and operations support. The Manager-Nuclear Fuels interfaces with the onsite operations group regarding nuclear fuel shipping and receiving, fuel and core performance monitoring, and spent fuel shipping.

17.2.1.1.2.5 Manager - Nuclear Administration

The Manager-Nuclear Administration is responsible for developing and implementing a nuclear records management system and directing all interfacing organizations toward the implementation of the system. The Manager-Nuclear Administration is also responsible for establishing and maintaining a document control system for SSES. 18

17.2.1.1.2.6 Manager - Nuclear Safety Assessment

The Manager-Nuclear Safety Assessment is responsible for independently reviewing and monitoring all nuclear activities to ensure that they are performed in a manner which results in safe reliable operation.

17.2.1.1.4 Manager - Nuclear Quality Assurance

The Manager-NQA is responsible for:

- Directing and coordinating the development and updating of PP&L's QQA Program.
- Assuring overall implementation of the QQA Program.
- Interpreting the QQA Program, subject to the approval of the Senior Vice President - Nuclear. 18
- Auditing, monitoring, inspecting and witnessing, as necessary, contractor, vendor and plant safety-related activities to assess compliance with the requirements of the QQA Program and/or procurement documents, and reporting the results of these activities to responsible management. 18
- Reviewing functional unit procedures to assure compliance with the QQA Program.
- Providing training assistance in QQA Program requirements.
- Implementing the QA and site QC activities identified in the QQA Program. 22
- Reviewing and auditing the QQA Program provisions that are applied to the fire protection program and

reporting the results of these activities to responsible management.

- Implementing the nondestructive examination training, qualification and certification program.

- Evaluating potential suppliers of material equipment and services to determine their capabilities for providing quality products or services.

- Administrative integration of the OQA and Environmental auditing programs for Susquehanna SES.

- Reviewing and approving quality assurance requirements in procurement documents.

The Manager - NQA is responsible for taking action (including work stoppage), except for plant operation, as necessary to correct conditions adverse to quality. At Susquehanna SES, the Manager - NQA is responsible for informing the Superintendent of Plant when it is determined that safety-related components or the activities performed on these components fail to comply with approved specifications, plans, or procedures. The Superintendent of Plant retains the responsibility for the evaluation of conditions adverse to quality with regard to plant operation and is responsible for determining when an operating unit(s) is to be shut down.

PP&L requires that the Manager-NQA shall have qualifications that are commensurate with the responsibilities of that position. As a minimum, these shall include a B.S. in Engineering and ten years experience in Engineering and/or Construction. At least one year of this ten years experience shall be nuclear power plant experience in the overall implementation of the quality assurance program.

The Manager - NQA and the NQA Staff are independent of organizations responsible for performing safety-related activities. The NQA Section has sufficient authority and organizational freedom to identify quality problems, to initiate, recommend or provide solutions through designated channels, and to verify implementation of solutions.

The PP&L Nuclear Quality Assurance functional structure is shown in Figure 17.2-3. The Manager - NQA delegates functional responsibilities for accomplishing quality assurance activities as follows:

- <u>Quality Engineering & Procurement</u>

  1. Quality Engineering

(a) Interface with engineering organizations to accomplish the incorporation of quality requirements in design, test, & procurement documents via the specification, review and approval process.

(b) Interface to provide QA coverage of nuclear fuel.

(c) Review and maintain cognizance of applicable codes and standards.

(d) Review and support responses to NRC Bulletins, Circulars, and Information Notices.

(e) Review and support for reporting items per 10CFR50.55(e) and 10CFR21.

(f) Provide technical support for auditing.

2. Procurement

(a) Vendor QA program evaluations, surveys and performance trending and rating, vendor audits.

(b) Technical review/acceptance of Vendor QA records.

(c) Post Award Vendor meetings (review P.O. provisions).

(d) Source surveillance/verification.

o <u>Construction</u>

1. (a) Interface with QA and QC organizations for plant construction support.

(b) Interface with NRC construction inspectors.

(c) Direct support of the preservice inspection activities.

(d) Direct responsibility for the review of NDE procedures.

(e) QA support for specified major modifications during plant operations.

(f) Interface with the Authorized Nuclear Inspector.

(g) Audits of construction activities.

(h) Review of construction procedures and instructions.

(i) Field checks and verification of responses to NRC citations, bulletins, circulars and reportable conditions related to construction activities.

(j) Completion of N-3 Forms.

(k) Quality trending of construction related activities.

o <u>Operations</u>

1. Quality Assurance

(a) Interface with the Plant Staff and ISG for the QA support of preoperational, startup testing and plant operations.

(b) Review administrative preoperational and startup test procedures.

(c) Interface with NRC operations inspectors.

(d) Field checks and verification of responses to NRC citations, bulletins, circulars and reportable conditions related to operations activities.

(e) Audits of operations.

(f) Quality trending of plant operations related activities.

2. Quality Control

(a) Inspection of maintenance, modification, repair, testing and PP&L Construction activities.

(b) Performing and interpreting the results of NDE.

(c) Receipt inspection and acceptance of material, equipment and consumables.

(d) Evaluating NCRs for trends.

(e) Procedure review for insertion of witness/hold points.

(f) Inspection planning.

o <u>Quality Systems & Training</u>

1. (a) Auditor training, qualification and certification.

   (b) Inspector training, qualification and certification.

   (c) QA indoctrination and training for the NQA section and other PL organizations.

   (d) Maintaining the construction QA program.

   (e) Developing and maintaining the operational QA program.

   (f) Developing and maintaining NQA Section procedures.

   (g) Coordination of responses to NRC inspections.

   (h) Administrative support functions.

o <u>Auditing</u>

1. (a) Scheduling and scoping programmatic audits of PP&L and other organizations.

   (b) Coordinating the implementation of programmatic audits and the allocation of auditor resources through the other NQA supervisors.

   (c) Performing audits of other NQA subsections.

   (d) Evaluating and trending the results of the auditing effort.

   (e) Audit follow-up and verification/close-out.

The Manager-NQA is responsible for initiating correspondence such that the NRC is notified of changes to (1) the accepted FSAR QA program description prior to their implementation, and (2) organizational elements within thirty (30) days after their announcement. (Note--editorial changes or personnel reassignments of a non-substantive nature do not require NRC notification.)

<u>17.2.1.2 Construction Manager</u>

The Construction Manager is responsible for providing the necessary organization and trained resources and equipment for the performance of maintenance tasks during normal operations and for outages. These same resources will also be responsible for completion of plant modifications repairs and/or additions to the operating plant. These operations will encompass projects/tasks

assigned by the Superintendent of Plant either directly or through his on-site organization.

Activities will be defined in functional unit procedures developed in accordance with OQA Program requirements.

17.2.1.3 Manager - Procurement

The Manager - Procurement is responsible for the purchase of equipment, materials, supplies and related services that conform to all applicable purchasing specifications and for placing orders for equipment, materials, supplies, and services with approved suppliers (except for nuclear fuel as specified in Subsection 17.2.1.1.2.5). Functional unit procedures shall define how the procurement process is controlled in accordance with OQA Program requirements.

17.2.2 QUALITY ASSURANCE PROGRAM

The Operational Quality Assurance (OQA) Program is applied to all safety-related Susquehanna SES structures, systems, components, and activities.

SAFETY RELATED is a generic term applied to:

1. Those systems, structures, and components that meet one or more of the following requirements:

   a. Maintain the integrity of the Reactor Coolant System pressure boundary;

   b. Assure their capability to prevent or mitigate the consequences of accidents that could cause the release of radioactivity in excess of 10CFR100 limits;

   c. Preclude failures which could cause or increase the severity of postulated accidents or could cause undue risk to the health and safety of the public due to the release of radioactive material;

   d. Provide for safe reactor shutdown and immediate or long term post accident control.

2. Those activities that affect the systems, structures and components discussed in Item 1 above such as their design, procurement, construction, operation, refueling, maintenance, modification and testing.

The Manager - NPE is responsible for maintaining a list designating those structures, systems, components, parts, and procured services which are safety-related. (See Table 3.2-1). 18

The OQA Program will be implemented at least 90 days prior to fuel load. Safety-related activities occurring prior to the implementation of the OQA Program will be controlled by the Susquehanna QA Program. The Susquehanna QA Program will be modified through amendments to the PP&L QA Manual, as necessary, to cover new activities occurring during the preoperational testing phase. 22 18

The Senior Vice President - Nuclear has assigned to the Manager - Nuclear Safety Assessment the responsibility for regularly assessing the scope, status, implementation, and effectiveness of the OQA Program. This will assure that the Program is adequate and complies with 10CFR50, Appendix B. 18

The OQA Program requires that safety-related activities be performed using specified equipment under suitable environmental conditions and that prerequisites have been satisfied prior to inspection and test.

The Manager - NQA is responsible for establishing and maintaining the OQA Program and for insuring that it provides adequate control of all activities. The Manager - NQA is responsible for assuring that functions delegated to principal contractors are being properly accomplished. Supplier QA programs are evaluated to determine that the requirements of 10CFR50 Appendix B will be implemented and this evaluation is documented.

The corporate OQA policies, goals, and objectives are transmitted to the persons performing activities which are required by the OQA Program and supporting documents. The commitments of the OQA Program are described in Chapter 17 which also assigns responsibilities for implementing OQA Program commitments. The OQA Manual contains Operational Policy Statements (OPS) which stipulate PP&L QA policies, goals and objectives for implementing the OQA Program commitments. These policies give generic direction for the performance of activities. A synopsis of the OPS and a matrix which cross-references them to each criterion of Appendix B to 10 CFR Part 50, is contained in Table 17.2-2.

The OQA Program is patterned after and fully complies with ANSI N18.7-1976 as modified by NRC Regulatory Guide 1.33, Revision 2. The degree of compliance with other regulatory guides and associated ANSI Standards is listed in Table 17.2-1. Where guides, codes or standards are nonexistent or inadequate, PP&L will develop methods to provide the necessary control.

The OQA Program requirements are mandatory for all safety-related activities. Each functional unit manager is responsible for assuring that safety-related activities performed by that functional unit, meet the requirements of the OQA Program. The Manager - NQA is responsible for the audit, review, inspection
18 and verification of activities both onsite and offsite to assure that they are accomplished according to the OQA Program requirements. QC activities shall be performed in compliance with the OQA Program requirements.

18 Disagreements between NQA and other department personnel (such as
18 Engineering, Construction, Fuels, Engineering Services, and Procurement) concerning the OQA Program and related activities will be resolved between the Manager-NQA and the affected department's supervisor or manager. Disagreements not resolved at these levels will be referred to the Senior Vice President -
18 22 Nuclear for resolution.

The OQA Manual, which contains OPS, is controlled and distributed by the NQA Section. All managers responsible for the performance of safety-related activities will be issued controlled copies of the OQA Manual.

The Manager - NQA is responsible for obtaining appropriate review and approval of the content and changes to the OQA Program and Manual. Any group performing activities governed by the OQA Program and Manual may propose changes to these documents. All OQA Program (FSAR Section 17.2) changes require review by the
18 Manager - NQA, the Vice President - E&C - Nuclear and the V.P. - Nuclear Operations, and approval by the Senior Vice President - Nuclear. All OQA Manual changes shall be reviewed by functional unit managers affected by the change and reviewed and approved by the Manager - NQA, Vice President - E&C - Nuclear, V.P. - Nuclear
18 Operations and Senior VP - Nuclear. Functional unit procedures shall be reviewed by the Manager - NQA and reviewed and approved by the appropriate functional unit manager. Control of QA programs other than the applicant's is addressed in Subsection 17.2.7.

Individuals performing inspection, examination and testing functions associated with normal operations of the plant, such as surveillance testing, routine maintenance and certain technical reviews normally assigned to the on-site operation organization shall be qualified to ANSI N.18.1-1971. Personnel whose qualifications are not required to meet those specified in ANSI N18.1 and who are performing inspection, examination and testing activities during the operational phase of the plant shall be qualified to ANSI N45.2.6-1973, except that the QA experience cited for Levels I, II and III shall be interpreted to mean actual experience in carrying out the types of inspection, examination and testing activity being performed.

Managers are responsible for assuring that their personnel receive the indoctrination and training necessary to properly perform their activities. The indoctrination and training program shall be such that personnel performing activities are knowledgeable in procedures and requirements and proficient in implementing those procedures. The program assures that:

- Personnel responsible for performing activities are instructed as to the purpose, scope, and implementation of the safety-related manuals, instructions, and procedures which control their activities.
- Personnel performing activities are trained and qualified in the principles and techniques of the activity being performed.
- The scope, the objective, and the method of implementing the indoctrination and training program are documented.
- Proficiency of personnel performing activities is maintained by retraining. Re-examination and/or recertification will be utilized as applicable.
- Methods are provided for documenting training sessions; including a description of the content and results and a record of attendance.

The Management and technical interfaces between Bechtel, General Electric and PP&L during the Initial Test Program are described in the Start-up Administrative Manual. The Susquehanna SES QA Program as modified by amendments to the QA Manual will describe the receipt and processing of QA records by PP&L.

In addition to safety-related structures, systems, components and activities, certain provisions of the OQA Program are applied to fire protection. These provisions apply to those items within the scope of the fire protection program such as fire protection systems, emergency lighting communication and breathing apparatus, as well as the fire protection requirements of applicable safety-related equipment. Specifically, the OQA Program applies to the 10 criteria listed in Regulatory Position C.3 in the U.S. NRC Regulatory Guide 1.120, Revision 1, Fire Protection Guidelines for Nuclear Power Plants.

17.2.3 DESIGN CONTROL

The OQA Program documents identify those managers responsible for performing design activities and describe their responsibilities and methods for meeting the OQA Program requirements.

The functional unit's procedures detail the steps necessary for its compliance with the requirements for its associated design activities. These procedures assure that design activities including changes in the design are carried out in a planned, controlled, and orderly manner.

Applicable design inputs such as regulatory requirements, codes and standards, and design bases shall be reflected in design output documents such as specifications, drawings, written procedures, and instructions. These design output documents shall specify the appropriate quality standards and any deviations from these quality standards will be accomplished in accordance with QQA Program requirements.

The design control process shall include, but not be limited to, the following, where applicable:

- Reactor physics
- Seismic, stress, thermal, hydraulic, radiation, and accident analyses
- Material compatibility
- Accessibility of items for in-service inspection, maintenance, and repair
- Verification that the design characteristics can be controlled, inspected and tested
- Identification of inspection and test criteria

The design engineer shall evaluate and select suitable materials, parts, equipment, and processes for safety-related structures, systems, and components. This evaluation and selection shall include the use of appropriate industry standards and specifications. Materials, parts, and equipment which are standard, commercial (off the shelf), or which have been previously approved for a different application, shall be reviewed for suitability in the intended application prior to use.

Internal and external interfaces between organizations performing work affecting quality of design shall be identified. Procedures shall be established to control the flow of design information between organizations. These procedures shall include the review, approval, release, distribution, and revision of documents involving design interfaces with other organizations.

Designs shall be reviewed to assure that design characteristics can be verified and acceptance criteria are identified.

Designs shall be verified by reviewing, alternate calculations, or qualification testing. Design verification shall be performed by a qualified person or group other than the original designer or the designer's immediate supervisor. However, supervisors may perform design verification subject to the restrictions of Paragraph C.2 of Regulatory Guide 1.64, Revision 2. Procedures for design verification shall identify the responsibility and authority of persons or groups performing design verifications. When a test program is used to verify the adequacy of a design, the test will be performed on a prototype unit or initial production unit and shall demonstrate adequacy of performance under the most adverse design conditions.

Changes to design output documents, including field changes, shall be subjected to design control measures the same as, or equivalent to, the original measures.

Responsible plant personnel are made aware of design changes/modifications which may affect the performance of their duties by:

- Plant Operations Review Committee review of all modification packages prior to installation.
- Installation of modifications are controlled by the plant work authorization system.
- Nuclear Plant Engineering notifies plant supervisors of design changes to allow updating of procedures.
- Effects of modifications are incorporated into the plant training program.

Errors and deficiencies in the design or the design process that could adversely affect safety-related structures, systems, and components will be documented and corrective action will be taken in accordance with Subsection 17.2.16. Design documents, including changes are filed as described in Subsection 17.2.17.

17.2.4 PROCUREMENT DOCUMENT CONTROL

OQA Program documents identify those managers responsible for activities related to the control of procurement documents and describe their responsibilities and methods for meeting the OQA Program requirements. Functional unit procedures detail the steps to be accomplished in the preparation, review, approval and control of procurement documents. Each manager is responsible for establishing, maintaining and implementing the functional unit's procedures in compliance with OQA Program requirements.

Procurement documents shall contain or reference as applicable:

- Design basis technical requirements including the applicable regulatory requirements.
- Component and material identification requirements.
- Drawings.
- Specifications.
- Codes and industry standards.
- Manufacturers' test and inspection requirements.
- Special process instructions.

Procurement documents shall identify a) the applicable quality requirements which must be met and described in the supplier's QA program, b) the documentation (such as drawings, specifications, procedures, inspection and fabrication plans, inspection and test records, personnel and procedure qualifications and material, chemical and physical test results) to be prepared, maintained and submitted to PP&L for review and approval, and c) those records which shall be retained, controlled, maintained or delivered to PP&L prior to use or installation of the purchased items. Procurement documents shall also contain provisions for PP&L or its agent, as applicable, to have the right of access to suppliers' and subtier suppliers' facilities and records for source inspection and audits. Procurement documents shall also require that the supplier submit, when required, its QA Program or portions thereof to PP&L for review and approval by qualified QA personnel prior to initiation of activities controlled by the Program.

Procurement documents shall be reviewed by qualified personnel for adequacy of quality requirements (such as acceptance and rejection criteria). Quality requirements shall be correctly stated, inspectable and controllable. Prior to their release, procurement documents shall have been prepared, reviewed and approved in accordance with OQA Program requirements. The procurement document review and approval is documented and filed as described in Subsection 17.2.17.

When procurement documents are revised, they are subject to the same or equivalent review and approval as the original document.

Procurement documents for safety-related spare or replacement parts for structures, systems and components are subject to controls the same as or equivalent to those used for the original equipment. All activities described in this subsection are to be performed by personnel qualified to perform the activity.

17.2.5 INSTRUCTIONS, PROCEDURES AND DRAWINGS

Activities shall be accomplished in accordance with documented instructions, procedures or drawings. This subsection applies to internal PP&L instructions, procedures and drawings. Such requirements for contractors and vendors are included in procurement documents as discussed in Subsection 17.2.4.

There are two general levels of QQA Program documents which are used to implement the QQA Program. The first document level is comprised of Operational Policy Statements (OPS) which describe PP&L's policies for complying with 10CFR50, Appendix B and QQA Program requirements. These OPS delineate the requirements for preparing, reviewing, approving, and controlling instructions, procedures, and drawings.

The second document level used to implement the QQA Program consists of functional unit procedures which describe how each functional unit performs its activities. These activities include specifying in instructions, procedures or drawings the methods for complying with OPS requirements. Instructions, procedures and drawings controlled by the QQA Program shall include quantitative (such as dimensions, tolerances, and operating limits) and qualitative (such as workmanship samples) acceptance criteria for use in determining that important activities have been satisfactorily accomplished.

The functional unit manager shall prepare, obtain the appropriate review, approve, issue, and revise the functional unit procedures which control the activities of that group. These procedures are reviewed by cognizant functional unit personnel for accuracy and workability and by QA personnel for compliance with QQA Program requirements.

Inspection plans; test, calibration, special process, maintenance, modification and repair procedures; drawings and specifications; and changes thereto are subject to audit for their compliance with QQA Program requirements.

17.2.6 DOCUMENT CONTROL

The document control system described in QQA Program documents requires that, prior to their release, documents and changes thereto are reviewed for their adequacy and approved and released by authorized personnel and distributed for use at the location where the prescribed activity is to be performed. The documents controlled under this subsection as a minimum include:

- Design Specifications

- Procurement Documents
- Test Procedures
- Design, Manufacturing, Construction and Installation Drawings
- Manufacturing, inspection, and testing instructions
- Final Safety Analysis Report
- QQA Program documents
- Maintenance, modification and operating procedures
- Non-conformance Reports

The NQA Section or other qualified individuals delegated by NQA, but other than the person who generated the document, shall review and concur with the document and changes thereto, with regard to QA-related aspects prior to implementation.

Each manager who is responsible for issuing a document, is also responsible for obtaining the proper review and approval of that document. Changes to documents are reviewed and approved by the same organizations that performed the original review and approval unless specifically delegated to other qualified organizations. This review will be completed prior to issuing the document except for temporary procedures/instructions issued by the Susquehanna SES Plant Staff. This special case is described in Section 6 of the Technical Specifications and the Susquehanna Plant Administrative Procedures.

Each functional unit manager is responsible for preparing and periodically issuing distribution lists and revision status lists for the control of quality documents issued by that functional unit. These lists identify the additions and changes made to documents since the previous report period and assist recipients in maintaining up-to-date files. Each recipient is responsible for reviewing the latest distribution lists to confirm that the current revision of each document is available. Prior to implementation, approved changes are included in instructions, procedures, drawings, or other documents by procedurally controlled change mechanisms.

It is the responsibility of each functional unit supervisor/manager to assure that the proper documents such as instructions, procedures, and drawings are available at the location where the prescribed activities are performed.

The issuing department is responsible for describing and implementing measures which provide controls to prevent the inadvertent use of obsolete or superceded documents.

Individuals or groups responsible for preparation, review, approval, issue and distribution of quality documents and their revisions are identified in the OQA Program documents.

17.2.7 CONTROL OF PURCHASED MATERIAL, EQUIPMENT & SERVICES

PP&L OQA Program documents list those managers responsible for performing activities related to the control of purchased material, equipment and services; describe their responsibilities; and specify their methods for meeting the OQA requirements. Each functional unit's procedures detail the steps necessary for complying with these requirements for their activities.

PP&L's system for control is comprised of supplier evaluation, surveillance of the supplier during production, receipt inspection of items or services, and evaluation of supplier records. The extent and methods of control used assure compliance with applicable technical, manufacturing, and quality requirements.

Prior to the award of a purchase order or contract, PP&L evaluates the prospective supplier's ability to provide material, equipment, and services which comply with the technical, design, manufacturing and quality requirements. The suppliers judged capable of meeting the requirements are considered approved suppliers for the specific article. The results of supplier evaluations are documented and the records maintained in accordance with Subsection 17.2.17.

The evaluation includes, as necessary, reviews of the records and performance of suppliers who have previously supplied similar articles, surveys of their facilities and evaluations of their quality assurance programs to determine their ability to meet the design, manufacturing and quality requirements of the purchase order. These quality requirements include the applicable elements of 10CFR50 Appendix B.

Suppliers' activities during the design, fabrication, inspection, testing, and preparation for shipment of material, equipment and components are under surveillance to assure their compliance with the purchase order requirements.

The surveillance of suppliers is planned and performed in accordance with written procedures as described in Subsection 17.2.18. These procedures specify the characteristics or processes to be witnessed, inspected or verified and accepted;

the method of surveillance; the extent of documentation required; and those responsible for implementing these procedures. These procedures also specify the audits and surveillances required to assure that the supplier complies with the quality requirements where compliance cannot be determined by receipt inspection.

As applicable, qualified personnel perform receipt inspection of material, equipment and services to assure that:

- The material, component or equipment is properly identified and corresponds with the receiving documentation.
- The material, component or equipment and its acceptance records are judged acceptable in accordance with predetermined inspection instructions prior to installation or use.
- Inspection records or certificates of conformance attesting to the acceptability of material, components, and equipment are available at Susquehanna SES prior to its installation or use.

Upon completion of the receipt inspection, items accepted and released are identified as to their inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work.

Supplier furnished records shall be reviewed and accepted by a qualified individual knowledgeable in quality assurance. These records shall, as a minimum, contain:

- Documentation that specifically identifies by purchase order number the purchased material or equipment and the specific procurement requirements, such as codes, standards, and specifications met by the items.
- Documentation that identifies any procurement requirements which have not been met together with a description of those nonconformances dispositioned "accept as is" or "repair".

The requirements of this subsection shall also be applied to the purchase of spare and replacement parts and shall assure that these parts have a level of quality consistent with their importance, complexity, and quantity.

Supplier certificates of conformance are periodically evaluated to verify their validity.

The effectiveness of the control of quality by suppliers is assessed by PP&L at intervals consistent with the importance, complexity, and quantity of an item.

17.2.8 IDENTIFICATION AND CONTROL OF MATERIALS, PARTS & COMPONENTS

QQA Program documents list those managers responsible for performing activities related to the identification and control of materials, parts and components, including partially fabricated subassemblies, describe their responsibilities, and specify the methods for meeting the QQA program requirements.

Each functional unit's procedures detail the steps necessary to comply with these requirements.

Procurement documents specify the requirements that PP&L suppliers must comply with for the identification of material, parts, and components (including partially fabricated subassemblies).

Item identification is maintained either on the item or on records traceable to the item to prevent the use of incorrect or defective items throughout fabrication, erection, installation and use. The location, type, and application method of the identification shall not affect the fit, function, or quality of the item being identified.

Materials and parts, as required by their importance to plant safety and applicable Codes, Standards and Regulatory requirements, shall be traceable to appropriate documentation such as drawings, specifications, purchase orders, manufacturing and inspection documents, deviation reports and physical and chemical mill test reports.

The correct identification of materials, parts, and components is verified and documented prior to release for fabrication, assembly, installation or shipping.

17.2.9 CONTROL OF SPECIAL PROCESSES

Special processes are those that require interim in-process controls in addition to final inspection to assure quality. QQA Program documents identify those managers responsible for the writing, qualifying, approving and issuing of procedures for special processes. Procedures for special processes are prepared in accordance with applicable codes, standards, specifications, criteria, and other special requirements to control processes such as welding, heat treating, nondestructive examination (NDE),

and chemical cleaning. Personnel performing special processes and the procedures and equipment used for this activity are qualified in accordance with applicable codes, standards and specifications. The procedures for special processes specify the requirements for their control, the parameters to be considered, the methods of documentation, and applicable codes, standards, specifications or supplementary requirements which govern their qualification. The special processes are accomplished in accordance with written process sheets, or equivalent, with recorded evidence of verification. When special processes are not covered by existing codes and standards, or when item quality requirements exceed the requirements of established codes or standards, the necessary qualifications for personnel, procedures or equipment are defined.

Records verifying the qualification of personnel to perform special processes are maintained in a current status.

Procurement documents specify contractor responsibility for controlling special processes and for maintaining records to verify that special processes are performed in accordance with established requirements.

17.2.10 INSPECTION

QQA Program documents identify those managers responsible for the preparation, approval, and issuance of inspection procedures. The documents also identify those managers responsible for the performance of inspections. Onsite and offsite activities affecting quality are inspected in accordance with written controlled procedures to verify conformance with applicable procedures, design documents, codes and specifications for accomplishing the activity. Activities affecting quality are subject to inspections in areas such as:

- Special processes as identified in Subsection 17.2.9.
- Modifications to the plant.
- Receipt of materials, parts or components.
- Plant operation
- Repairs or replacement of equipment.
- Inservice inspection.

Inspection activities conform to the following requirements:

- Inspection personnel are qualified individuals other than those who performed or directly supervised the activity being inspected.

- Mandatory inspection hold points are identified in inspection procedures.

- Modifications, repairs and replacements are inspected in accordance with the original design and inspection requirements or approved alternatives.

- Maintenance and modification procedures are reviewed by qualified personnel knowledgeable in quality assurance requirements to determine the need for (a) inspection, (b) identification of inspection personnel, and (c) documenting inspection results. The criteria for performing inspections are based upon an activity's complexity, uniqueness and impact on safety.

- If direct inspection of processed material or products is impossible or disadvantageous, indirect control by monitoring processing methods, equipment, and personnel is provided.

- Inspectors are trained and qualified in accordance with appropriate codes, standards, and company training programs and their qualifications and certifications are kept current.

- Inspection instrumentation is calibrated and has an uncertainty (error) equal to or less than the tolerance stated in the acceptance criteria.

- Inspection of activities is accomplished according to approved procedures, instructions, and check lists. These inspection documents contain the following:

  (a) Identification of the items or activities to be inspected.

  (b) Identification of the characteristics of the items or activities inspected.

  (c) Identification of the individuals or groups responsible for performing the inspection.

  (d) Identification of acceptance and rejection criteria.

(e) A description of the method of inspection including necessary measuring and test equipment.

(f) Evidence of completion and verification of a manufacturing inspection, or test.

(g) A record of the inspector, or data recorder, the date and results of the inspection.

- Inspection procedures or instructions contain or reference necessary procedures, drawings and specifications to be used when performing inspection operations.

- Provisions for inspection results to be documented, evaluated and accepted by the supervisor responsible for the inspection function.

17.2.11 TEST CONTROL

The OQA Program documents identify those managers responsible for testing structures, systems and components during the preoperational testing, power testing and operations phases of Susquehanna SES. (Prior to implementation of this OQA Program preoperational testing will be performed under the control of the Susquehanna Quality Assurance Program as supplemented by interim procedures to the PP&L Quality Assurance Manual.) The test program described herein and further detailed in Operations Policy Statements is designed to assure that structures, systems and components will perform satisfactorily in service. Modifications, repairs and replacements are tested in accordance with the original design and testing requirements or by approved alternates.

Testing is established, documented and accomplished in accordance with written controlled procedures. These procedures contain or reference:

- The requirements and acceptance limits specified in the applicable design and procurement documents.

- The instructions for performing the test.

- The test prerequisites such as:

  - That test instrumentation is calibrated and has an uncertainty (error) equal to or less than the tolerance stated in the acceptance criteria.

- That testing equipment is adequate and appropriate for the test.
- That personnel performing the test are properly trained and qualified and licensed or certified as required.
- That the item is sufficiently complete to be tested.
- That environmental conditions are suitable and controlled.
- That provisions are made for data collection and storage.

- The mandatory inspection hold points for witness by PP&L, their contractor or agent.
- The test acceptance and rejection criteria.
- The methods of documenting or recording the test data and test results.

Tests are required to be performed:

- Periodically to provide assurance that failures or substandard performance do not remain undetected and that the required reliability of safety-related systems is maintained.
- Following maintenance, modification or procedural changes to demonstrate satisfactory performance.

The test results are documented and evaluated to determine the acceptability of the test. The individuals or groups responsible for evaluating the test results shall be qualified to perform this evaluation.

When by evaluation of the test results, the structure, system or component is determined to be nonconforming, it shall be controlled in accordance with Subsection 17.2.15.

17.2.12 CONTROL OF MEASURING AND TEST EQUIPMENT

PP&L's QQA Program documents provide measures to assure that tools, gauges, instruments and other measuring and testing devices are controlled. Calibrations are scheduled with sufficient frequency to maintain required accuracy. The measuring and test equipment controls assure that:

- Procedures are used to control measuring and test equipment. These procedures describe the calibration technique and frequency, maintenance and method of control of measuring and test equipment (such as instruments, tools, gauges, fixtures, reference and transfer standards, and nondestructive examination equipment) which are used in the measurement, inspection, and monitoring of components, systems and structures.

- Measuring and test equipment is identified and traceable to the calibration test data.

- Measuring and test instruments are calibrated at specific intervals based on the required accuracy, purpose, degree of usage, stability characteristics and other conditions affecting the measurement.

- Measuring and test equipment is labeled or tagged to indicate the date of the calibration and the due date of the next calibration.

- When measuring or test equipment is found to be out of calibration, measures are taken and documented to determine the validity of previous inspections performed since the last valid calibration.

- Calibration standards have an uncertainty (error) of no more than 1/4 of the tolerance of the equipment being calibrated, unless limited by the "state-of-the-art".

- A complete status of all items under the calibration system is recorded and maintained.

- Reference and transfer standards are traceable to nationally recognized standards; or where national standards do not exist, provisions are established to document the basis for calibration.

17.2.13 HANDLING, STORAGE, AND SHIPPING

QQA Program documents list those managers responsible for the handling, preservation, storage, cleaning, packaging and shipping of materials, parts and components; and describe their authorities and methods for meeting the quality requirements.

Each functional unit's procedures control their activities and assure compliance with the quality requirements contained in drawings, specifications and procurement documents. These requirements include those necessitated by the design, as

outlined in the design output documents, and those submitted by the supplier. These procedures provide control to prevent damage and loss or deterioration by environmental conditions, such as temperature or humidity, and specify the personnel qualifications required to satisfactorily accomplish the activity.

Consumables such as chemicals, reagents, weld rod, lubricants, etc. shall be stored in accordance with manufacturer's instructions or other approved methods to prevent harmful deterioration of the item. Materials with an identified shelf life shall be controlled such that they are used or discarded prior to expiration date.

17.2.14 INSPECTION, TEST, AND OPERATING STATUS

QQA Program documents list those managers responsible for the development and implementation of procedures to assure that the inspection, test, and operating status of structures, systems, and components is properly identified and controlled.

These procedures incorporate the following provisions:

- The inspection, test, and operating status of structures, systems, and components is identified to the affected parties.
- Application and removal of inspection and welding stamps and status indicators, such as tags, markings, labels, and stamps are procedurally controlled.
- Methods for bypassing of required inspection, tests, and other critical operations are controlled through documented Functional Unit Procedures.
- The status of nonconforming, inoperative, or malfunctioning structures, systems or components is identified to prevent their inadvertent use.

17.2.15 NONCONFORMING MATERIALS, PARTS OR COMPONENTS

QQA Program documents list those managers responsible and their methods for handling nonconforming materials, parts, components, or services. Procedures control the identification, documentation, segregation, review, disposition and notification to affected organizations of nonconforming materials, parts, components, or services.

Materials, parts, components or services which do not meet established drawing, specification, or workmanship requirements, are identified as nonconforming and documented on a nonconformance report. Nonconforming items are identified as discrepant and segregated from acceptable items until they are properly dispositioned.

The manager of each functional unit is responsible for the review and disposition of nonconforming items which fall under the scope of responsibility of that manager. When requested, or identified as having responsibility for the dispositioning, other departments will be notified of the nonconformance.

Nonconformance reports and associated records are maintained in accordance with Subsection 17.2.17.

Information pertaining to nonconforming items is recorded on the nonconformance report which identifies the nonconforming item or service, details of the nonconformance, the disposition and the approval signature(s).

Acceptability of rework or repair of materials, parts, components, systems, and structures is verified by re-inspecting and re-testing the item by a method which is the same as or comparable to the original inspection and test and in accordance with written procedures.

Inspection, testing, rework, and repair procedures are documented. Vendor nonconformance reports dispositioned "accept as is" or "repair" are made part of the inspection records and forwarded with the hardware to PP&L for review and assessment.

Nonconformance reports are periodically analyzed for quality trends, and the results are reported to management for review and assessment.

17.2.16 CORRECTIVE ACTION

PP&L's OQA Program establishes the requirements for controlling conditions adverse to quality (such as nonconformances, failures, malfunctions, deficiencies, deviations, and defective material and equipment).

Conditions adverse to quality are promptly identified, reported, evaluated, corrected and documented. OQA Program documents identify the methods used and personnel responsible for these activites.

Conditions adverse to quality are identified and reported to the appropriate levels of management of the affected organizations.

The responsible organization evaluates the conditions to determine if they are significant conditions adverse to quality and to determine the corrective action required.

If significant conditions adverse to quality are detected, the cause of the condition and the corrective action taken are reported to the appropriate management levels of affected home office organizations, plant staff and quality assurance for review and assessment.

The corrective action for conditions adverse to quality shall correct the specific conditions. For conditions determined to be significant, the corrective action provides measures to correct specific conditions and preclude recurrence.

The responsible organization shall implement the corrective action and document the details of the conditions including their resolution. Follow-up action is conducted to determine that the required corrective action has been completed and that the corrective action documentation has been closed out.

17.2.17 QUALITY ASSURANCE RECORDS

A QA record system, detailed in OQA Program documents, has been established by PP&L which assures that records are identifiable, retrievable and that sufficient records are maintained to provide documentary evidence of the quality of items and services. The system assures that requirements and responsibilities for record transmittal, retention (such as duration, location, fire protection and assigned responsibilities) and maintenance, subsequent to completion of work, are consistant with applicable codes, standards and procurement documents. QA records include:

- Plant historical records
- Operating logs
- Principal maintenance and modification activities
- Reportable occurrences
- Results of reviews, inspections, tests, audits and material analyses
- Monitoring of work performance
- Qualification of personnel, procedures and equipment

These records also include other documentation such as drawings, specifications, procurement documents, calibration procedures and reports, nonconformance reports, and corrective action reports.

Each manager is responsible for developing procedures which control the origination and transmittal of QA records within that functional unit. Each manager is responsible for transmitting QA records to the storage location designated for that record.

PP&L record storage facilities are constructed, located, and secured to prevent destruction of the records by fire, flooding, theft, and deterioration by environmental conditions such as temperature or humidity.

17.2.18 AUDITS

The PP&L audit program requires the planning, performing, documenting, and evaluating of audits. It assures compliance with license commitments, OQA Program requirements, technical specifications, and other applicable requirements. It also assures that corrective measures are taken in response to audit findings to resolve the original problem and minimize the probability of its recurrence.

Audits of selected operational phase activities are performed by NQA. These audits include areas which require implementation of 10CFR50, Appendix B. These areas include activities associated with:

- Plant operation, maintenance and modification.
- The preparation, review, approval and control of designs, specifications, procurement documents, instructions, procedures and drawings.
- Receiving and plant inspections.
- Indoctrination and training programs.
- The implementation of operating and test procedures.
- Calibration of measuring and testing equipment.

Audits are regularly scheduled based on the status and safety importance of the activity. Audits are also scheduled according to the requirements of Section 6 of the Technical Specification. The audit schedule assures proper coverage of all applicable activities. Additionally, the audit program provides for scheduling audits which can be conducted on short notice to respond to specific quality problems.

Audits are structured with a sufficiently defined scope to permit objective evaluation of the activity observed. Quality-related practices, procedures, and instructions are audited to measure both the effectiveness of their implementation and their conformance to OQA Program requirements.

The audit process is conducted according to procedures which require that a written audit plan be prepared. The audit plan ensures the proper scope, team preparation, and depth of coverage. The audit process includes, as applicable, an evaluation of work areas, activities, processes, and items. Audits include a review of associated documents and records.

Audit teams consist of trained personnel, not directly responsible for the areas audited. Each team shall have a designated leader who is responsible for the planning, conduct, and reporting of the audit.

The auditor qualification program ensures that audit team members are qualified to perform their assigned tasks.

Audit results are documented in a formal audit report which is transmitted to the responsible levels of management.

Audit team leaders, through their supervisors, ensure that responsible management takes necessary action to correct deficiencies noted, and provide a basis for preventing their recurrence. Team leaders verify, either through review of documentation resulting from corrective action, or if necessary, reaudit, that deficiencies have been properly corrected.

Formal audit reports are reviewed by NQA Management to determine the effectiveness of the OQA program, and indications of quality trends. If additional management action is required, the results of these reviews are formally reported to the appropriate manager of the responsible organization.

TABLE 17.2-1

OPERATIONAL QUALITY ASSURANCE PROGRAM

COMPLIANCE MATRIX

NRC Reg. Guide	ANSI Standard	Subject	Clarifications & Exceptions	
1.8 Rev. 1	N18.1 - 1978	Personnel Selection & Training	See Chapter 13	22
1.17 Rev. 1	N18.17 - 1973	Security	Not included in the OQA Program	6
1.28 Rev. 1	N45.2 - 1977	QA Requirements	Full compliance	
1.30 8/72	N45.2.4 - 1972	Electrical Installation, Inspection & Testing	Commitment to the extent required by ANSI N18.7-1976*	6
1.33 Rev. 2	N18.7 - 1976	Administrative Controls & Operational QA	Full compliance	
1.37 3/73	N45.2.1 - 1973	Cleaning Fluid Systems & Components	Commitment to the extent required by ANSI N18.7-1976*	
1.38 Rev. 2	N45.2.2 - 1972	Packaging, Shipping, Receiving, Storage & Handling	Commitment to the extent required by ANSI N18.7-1976*	6
1.39 Rev. 2	N45.2.3-1973	Housekeeping	Commitment to the extent required by ANSI N18.7-1976*	6
1.54 6/73	N101.4 - 1972	QA for Protective Coatings	Commitment to the extent required by ANSI N18.7-1976*	
1.58 Rev. 1	N45.2.6 - 1978	Qualifications of Inspection, Examination, & Testing Personnel	Commitment to the extent required by ANSI N18.7-1976*; personnel who only handle test results or perform document control activities will not be certified.	22
1.64 Rev. 2	N45.2.11 - 1974	QA for Design	Full compliance	
1.74 2/74	N45.2.10 - 1973	QA Terms & Definitions	Full compliance	
1.88 Rev. 2	N45.2.9 - 1979	Collection, Storage & Maintenance of Records	Full compliance	20
1.94 Rev. 1	N45.2.5 - 1974	Concrete & Structural Steel Installation, Inspection, & Testing	Commitment to the extent required by ANSI N18.7-1976*	
1.116 Rev. 0-R	N45.2.8 - 1975	Mechanical Installation, Inspection & Testing	Commitment to the extent required by ANSI N18.7-1976*	6
1.123 Rev. 1	N45.2.13 - 1976	QA for Procurement of Items & Services	Full compliance	6

TABLE 17.2-1 (Continued)

1.120 Rev. 1	N/A	Fire Protection Guidelines for Nuclear Power Plants	Full compliance limited to Regulatory Position C.3, Quality Assurance Program	6	12
1.144 1/79	N45.2.12-1977	Auditing of QA Programs	Full compliance	6	
1.146 8/80	N45.223-1978	Qualification of QA Program Audit Personnel	Full compliance	20	

*These standards will be applied as directed by the Manager-NQA for activities which "...are comparable in nature and extent to related activities occurring during construction" as required by ANSI N18.7-1976.

Table 17.2-2
OPERATIONAL POLICY STATEMENT CROSS REFERENCE MATRIX
WITH 10CFR50 APPENDIX B CRITERIA

OPS	TITLE	SYNOPSIS	CRITERIA 1, 2
1	Operational Quality Assurance Program Definition	Defines the scope and applicability of the OQA Program. Establishes requirements for the OQA Manual and defines the tiers of documents comprising the OQA Program.	II, V, VI
2	Terms and Definitions	Defines those terms having particular meaning within the context of the OQA Program.	II
3	Control and Issuance of Documents	Establishes controls for the issuance and use of documents. Defines those documents controlled by the OQA Program and requires review, approval, and use of documents at required locations.	V, VI
4	Document Reviews	Establishes the requirements for performing and documenting document reviews.	III, IV, VI
5	Deficiency Control	Delineates those activities associated with the control and correction of nonconforming material, parts or components; other conditions adverse to quality; and significant conditions adverse to quality.	VIII, XV, XVI
6	Personnel Qualification and Training	Establishes the requirements for the training and qualification of personnel performing activities affecting quality to assure that they achieve and maintain suitable proficiency.	II, IX, XVIII
7	Auditing/Quality Verification Activities	Establishes the requirements for the development of programs for auditing and monitoring quality related activities and includes performance, qualifications, reporting, and follow-up action.	II, XVIII
8	Records	Establishes the requirements for the collection, storage, and maintenance of quality assurance records.	XVII
9	Control of Modifications & Design Activities	Establishes the requirements for ensuring that the quality of modified structures, systems or components is at least equivalent to that specified in the original design bases, material specifications, and inspection requirements.	III
10	Procurement	Establishes the requirements for the procurement of material, parts, components, services and spare parts.	IV, VII
11	Procurement of Nuclear Fuels	Establishes the requirements for the procurement of reload nuclear fuel.	IV, VII
12	Administrative Control of Plant Operations	Establishes the requirements for the administrative and procedural controls that ensure the plant is operated in a safe and efficient manner.	V
13	Control of Maintenance	Establishes the requirements for ensuring that structures, systems, and components are maintained in a condition to perform their intended function. The field activities associated with modifications are also included.	IX, XIV
14	Control of Testing and Inspection Activities	Establishes the requirements for testing and inspection activities.	X, XI, XIV
15	Inservice Inspection	Establishes the requirements for the quality-related Inservice Inspection activities.	X, XI
16	Instrument and Calibration Control	Establishes the requirements for the calibration and control of calibration standards, installed plant instrumentation, and measuring and test equipment.	XII
17	Control of Plant Material	Establishes the requirements for the control of plant material and includes receipt inspection, handling, storage, and shipping.	VII, VIII, XIII

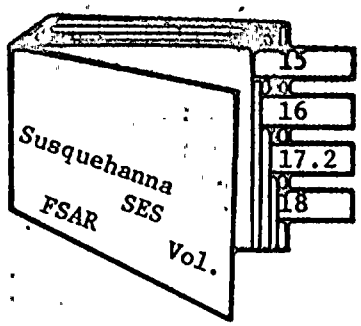
Table 17.2-2 (Cont.)

18	ASME Supplement	Establishes the requirements for PP&L to perform engineering, fabrication, and repair activities in accordance with Section XI of the ASME Code.	N/A
19	Reporting of Substantial Safety Hazards and Licensee Events	Establishes the requirements for reporting substantial safety hazards (10 CFR 21) and reportable occurrences.	N/A

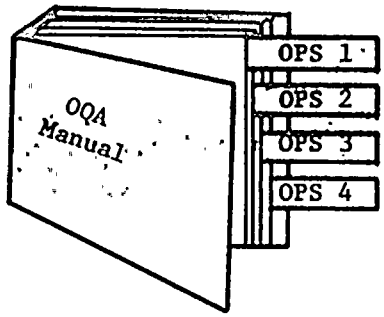
Footnotes:

(1) Criterion I, Organization, is covered extensively in Section 17.2.1 and is not repeated in a separate OPS. However, the "Responsibility" section in each OPS identifies the managers responsible for implementation and verification of the OPS' requirements.

(2) Criteria such as V, Instructions, Procedures, and Drawings, and XVII, Records, could be cross referenced with the majority of OPS identified. A deliberate effort was made to cross reference the Criteria only to those OPS which have a direct relationship.



Operational Quality Assurance (OQA) Program — 17.2



Operational Policy Statements (OPS) — OPS 3

OQA PROGRAM DOCUMENTS

PPE Functional Unit Procedures	NQA Functional Unit Procedures	Susquehanna SES Plant Functional Unit Procedures	Procurement Functional Unit Procedures	Other Department Functional Unit Procedures
1. Engineering design 2. Preparation of Engineering output documents	1. Activities of NQA personnel such as audit and document reviews	1. Preparation of procedures controlling activities such as operation, maintenance & testing 2. Preparation, control & coordination of requisitions for spare Parts	1. Preparation of purchase orders	1. Nuclear Department Instructions 2. Safety-related activities identified in OQA Program Documents as requiring control

TYPICAL ACTIVITIES CONTROLLED

Rev. 18, 11/80

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

Operational Quality Assurance Documents Relationships

FIGURE 17.2-1

Superseded per Amdt 43 FSAR dtd 3-22-82 50-387

5

CHAPTER 2

TABLES

2.5 GEOLOGY, SEISMOLOGY, AND GEOTECHNICAL ENGINEERING

2.5.1 BASIC GEOLOGIC AND SEISMIC INFORMATION

2.5.1.1 Regional Geology

2.5.1.1.1 Physiography and Geomorphology

The site is located (Figure 2.5-1) in the Valley and Ridge Physiographic Province which is bordered on the southeast by the Reading Prong and on the northwest by the Appalachian Plateau Physiograghic Province (Figure 2.5-2).

The Valley and Ridge Province is characterized by folded Paleozcic sedimentary rocks of varying erosional resistance. These strata form a series of level ridges and intervening valleys which trend generally northeast to southwest. Higher ridges are formed on the more resistant, inclined sandstone whereas lower ridges are underlain by other competent formations. The valleys occur in less resistant limestone and shale.

The Valley and Ridge Province attains a maximum width of about 80 miles along a line drawn northwest through Harrisburg, Pennsylvania. In Pennsylvania, folds generally plunge away from this line to the northeast and to the southwest. Because of the folding, resistant strata form broad, zig-zag outcrop patterns across the province. These strata form steep slopes, that flank anticlinal valleys, and cance-shaped synclinal valleys. Lithology and structure control the drainage pattern, the principal direction of which is to the southeast. Major drainage generally follows the strike of less competent strata and crosses the strike at water gaps where transverse structures, such as a high concentration of fractures, exist. Minor drainage trends normal to the regional strike or along major fracture sets, and usually intersects major streams at right angles to form a trellis pattern.

The Great Valley Section, in the southeastern third of the province, consists of broad, rolling valleys of low relief formed in Paleozoic soft limestones and calcareous shales. To the southeast, the Reading Prong exposes the oldest rocks (Precambrian) within 50 miles of the site (Figures 2.5-2 and 2.5-3). In general, the Reading Prong consists of high grade metasedimentary and metavolcanic rocks along with dominantly acidic plutonic rocks. These rocks experienced deformation commencing with the Grevillian orogeny about 1 billion years ago which imparted the dominant structural fabric, i.e., foliations, lineations and polyphase folds, present today. Succeeding

tectonism during the Paleozoic and Mesozoic eras, and possibly even more recently, have also affected these rocks. According to Drake (Ref 2.5-1) the rocks of the Reading Prong are allochthonous. The Triassic Lowlands of the Piedmont Province lie to the east and southeast of the Reading Prong, and contain the youngest rocks in eastern Pennsylvania (Figures 2.5-2 and 2.5-3). The rocks in the lowlands are dominantly red clastic sediments with associated basic intrusives and flows. Diabase dikes near Pottstown, Pennsylvania yielded K/Ar whole rock ages of 151 to 198 million years (Ref. 2.5-2, p. 3-25).

Northeast of the site, folds are broader and more open and give way to the gentle synclinal Pocono Plateau Section which is underlain by Devonian sandstone and shale. To the northwest, the Valley and Ridge Province terminates abruptly at the Allegheny Structural Front. Beyond the front lies the Appalachian Plateaus (Figure 2.5-2), a gently rolling highland formed on broad folds of low structural relief that plunge gently to the southwest. The strata consist predominantly of an upper Paleozoic cyclic sequence of sandstone, shale, limestone and coal.

The Susquehanna River, which flows past the site, has two important features associated with it. First the river makes several sharp bends along its length with the closest being adjacent to the site. East of the site the river maintains a west-southwest course which parallels the regional tectonic fabric. However at Shickshinny, Pennsylvania, about 5 miles north of the site, it makes a sharp right-angle bend and flows in a south-southeast direction for about 5 miles. Just below the site it again swings sharply and resumes its west-southwest flow direction. This phenomenon has been cogently explained by Itter (Ref. 2.5-3). He noted that this area was submerged during the Cretaceous and coastal plain sedimentation ensued. Sedimentation completely covered the pre-existing drainage pattern. Following coastal plain sedimentation, the area underwent broad uplift while the North Branch of the Susquehanna River apparently flowed southeastward, across the Pocono Plateau to Trenton, New Jersey (Ref. 2.5-3, p. 12-13). Tributaries must also have developed on this coastal plain. Following downcutting of the coastal plain sediments the streams encountered bedrock. Of these downcutting streams the present-day Susquehanna River, south of the confluence of the North and West Branches, apparently was able to incise more rapidly than other major streams. This resulted in stream capture and the pronounced bends seen along the river today.

The second feature of import is the buried valley of the Susquehanna. This buried valley occurs in bedrock overlain by a broad, flat plain across which the present-day Susquehanna flows (Ref. 2.5-3, p. 26). It extends upstream as a series of elongate basins, for about 15 miles, from near Nanticoke, Pennsylvania (approximately 10 miles northeast of the site) to just above West

Pittston (Ref. 2.5-4, p. 8). This valley is filled with alternating layers of water laid gravel, sand and clay. The development of this valley is attributed to the erosive action of the Wisconsinan ice sheet which must have flowed diagonally across the valley (Ref. 2.5-3, p. 27 and 2.5-4, p. 7). Subsequently, this valley was filled with sediment deposited by streams which emanated from this melting ice.

Most of the region, north and east of the site, has been scoured by at least three periods of glaciation in the last 150,000 years (Ref. 2.5-5, p. 15 and 2.5-6). The three major directions of ice advance were postulated as follows: 1) south and southeast from central Ontario, 2) south and southwest from approximately the Adirondack region, and 3) south and southwest from the Hudson Valley by way of the Catskills (Ref. 2.5-5, p. 18). Effects of glacial scouring are most notable on the Pocono Plateau.

At the present time, there is no positive evidence that any pre-Illinoian glaciation occurred in northeastern Pennsylvania although elsewhere in the eastern United States, such evidence does exist for pre-Illinoian glaciation. It is thus suggested that it should also have occurred here (Ref. 2.5-6). The recorded glacial events include the Illinoian and two stages of the Wisconsinan, the Altonian which spans the interval from about 70,000 years to about 28,000 years B.P., and the Woodfordian which lasted from about 21,000 years to approximately 13,000 years B.P.

In earlier literature (Ref. 2.5-5) the terms Altonian and Woodfordian were not utilized. Instead the Wisconsinan was divided into the Binghamton, Olean, Valley Heads and Mankato substages. However, it is not precisely known how the Altonian and Woodfordian subdivisions relate to the older terms except that the Mankato is somewhat younger than the Woodfordian (Ref. 2.5-6).

In the site region a lobe of Illinoian ice extended down the valley of the Susquehanna River from just above Berwick to the West Branch of the Susquehanna at Northumberland, Pennsylvania. The exposed length of deposits left by this lobe is about 40 miles whereas the maximum width does not exceed 8 miles. Illinoian drift is present on the slopes to within 60 feet of the present river level suggesting that only moderate deepening of the Susquehanna Valley has occurred since deposition of the Illinoian drift (Ref. 2.5-7, p. 24-25).

Valley trains of Wisconsinan gravel were examined along the Susquehanna River and its tributaries by Leverett (Ref. 2.5-7). He noted that from just east of Berwick downstream (westward) to the West Branch at Northumberland, the surface of a Wisconsinan gravel train is well defined and generally occurs at about 40 to 60 feet above the river.

Moderately eroded terraces underlain by freshly appearing gravels mark the upper level attained by waters derived from the Wisconsinan ice sheet along the Susquehanna River and its tributaries. These terraces occur at lower elevations than those exposing Illinoian gravel and have also been much less extensively eroded than Illinoian gravel (Ref. 2.5-7, p. 16). In general the relative heights of the terrace levels representing each of the four Wisconsinan sub-stages are fairly constant. For example, in the site vicinity relative heights for the Mankato, Valley Heads, Binghamton and Olean are 9, 15, 30 and 45 feet respectively (Ref. 2.5-5, p. 77). The surfaces of most of the terraces have been eroded subjecting the observed height of any terrace to an error of as much as 30 percent. However, no evidence is presented indicating differential vertical offest of these terraces.

At Berwick several well developed kame terraces and terrace remnants of frontal kames which formed at the end of marginal kames occur (Ref. 2.5-5, p. 91). The four lowest marginal kames were identified by Peltier as the First Olean, Second Olean, Third Olean, and Fourth Olean kame terraces which are respectively 86, 98, 110, and 158 feet above the river.

2.5.1.1.2 Stratigraphy and Lithology

2.5.1.1.2.1 The Appalachian Basin

The Valley and Ridge Province, in which the site is located, is part of a structural entity known as the Appalachian Basin. As defined by Colton (Ref. 2.5-8, p. 6-7), the Appalachian Basin is not a physiographic province. Rather, it is an elongate feature extending from the Canadian Shield in southern Quebec and Ontario, southwestward to central Alabama (Figure 2.5-4). It is bounded on the west by the Findlay Arch and on the south by the boundary between Paleozoic and Cretaceous strata. The eastern edge is marked by the surface contact between slightly-to-unmetamorphosed Paleozoic rocks on the west and more intensely metamorphosed Paleozoic and Precambrian rock on the east. In Pennsylvania this boundary coincides with the boundary between the Valley and Ridge and Piedmont Physiographic Provinces.

Isopach maps and stratigraphic columns show the respective thicknesses and relationships of the Cambrian through Pennsylvanian sequences in the Appalachian Basin (Figures 2.5-5 and 2.5-6).

In the Appalachian Basin, as outlined by Colton (Ref. 2.5-8), the Lower Cambrian clastic sequence is a wedge-shaped mass which is thickest along the eastern margin of the basin and thinnest along

the northern and western margins. The rocks along the eastern margin are dominantly Early Cambrian whereas the rocks along the northern and western margins are mainly Late Cambrian. The Lower Cambrian sequence is conformably overlain in most of the Appalachian Basin by a suite of dominantly carbonate rocks with lesser amounts of quartz sandstone. This overlying suite consists mainly of rocks ranging in age from Middle and Late Cambrian to Early and Middle Ordovician and was designated by Colton (Ref. 2.5-8, p. 19) as the Cambrian-Ordovician carbonate sequence. This sequence ranges in thickness from about 600 ft. in northern New York State to a little more than 10,000 ft. in southeastern Tennessee. A belt of maximum thickness extends along, and approximately parallel to, the eastern edge of the Appalachian Basin from southeastern New York State to northern Alabama (Ref. 2.5-8, p. 23).

This carbonate sequence is conformably overlain in most of the basin by dominantly non-calcareous clastic rocks, the majority of which are Late Ordovician in age. These Upper Ordovician clastic rocks are thickest along the northeastern margin of the basin in Pennsylvania and show a generally uniform thinning to the north, west and southwest. An exception to this generally uniform pattern of thinning occurs in Pennsylvania and adjacent New York State where the sequence thins more abruptly against the southwestern extension of the Adirondack axis (Ref. 2.5-8, p. 23).

Although the boundary between these rocks and the older Cambrian-Ordovician sequence is conformable, the boundary with the overlying Silurian clastics is marked by an unconformity in the northeast and southwest portions of the Appalachian Basin. "Volumetrically the unconfirmity is greatest in eastern Pennsylvania and contiguous parts of New Jersey" (Ref. 2.5-8, p. 23). This unconformity was considered by Colton as evidence of Late Ordovician or Early Silurian diastrophism.

The Early Silurian rocks are mainly clastic and extend across most of the Appalachian Basin. These rocks are thickest (2,600 ft.) and coarsest in the northeastern part of the basin where they are composed mainly of sandstone and conglomerate.

Carbonate rocks, classified by Colton (Ref. 2.5-8, p. 31) as the Silurian-Devonian carbonate sequence, range in age from Middle Silurian to early Middle Devonian and occur throughout much of the basin. They, like the older sequences, are thickest in the east and thinnest in the west. The thickest section is found in northeastern Pennsylvania where it is about 3300 ft. thick. In northeastern Pennsylvania the lower half of this sequence actually consists of a thick wedge of red clastic rocks comprising, among others, the Bloomsburg Red Beds. West, northwest and southwest of this area the red beds grade into an alternating suite of variegated shale and siltstone, carbonates

and evaporites. The margins of this suite are predominatly dolomite and limestone.

Rocks of Middle and Late Devonian age consist of a moderately thick sequence of shale, mudrock, siltstone and sandstone and extend throughout most of the basin. In most areas these Middle to Upper Devonian clastic rocks rest conformably on the strata of the Silurian-Devonian carbonate sequence. Like the underlying rocks this Devonian suite is wedge-shaped with the thickest part near the eastern margin of the basin and the thinnest part near the western periphery. The northeastern part of the basin, which includes east-central Pennsylvania, contains the thickest accumulation (more than 10,000 ft.) dominated by coarse-grained sedimentary rocks (including red beds). As the thickness decreases the average grain size of the rocks shows a corresponding diminution, being medium-grained where the rocks are of intermediate thickness and fine-grained where the section is thinnest (Ref. 2.5-8, p. 34).

The Middle to Upper Devonian clastic suite is conformably overlain by Mississippian rocks in most of the basin, but the contact is slightly disconformable along much of the eastern margin. However, in parts of northeastern Pennsylvania the entire Mississippian is missing. This anomalous unconformity is probably due to erosion prior to Pennsylvanian sedimentation (Ref. 2.5-9, p. 35). Generally the Mississippian sequence defines a crudely wedge-shapped mass. The greatest accumulation occurs in southeastern Virginia (6800 ft.), but Wood (Ref. 2.5-10, p. C39) reported a thickness exceeding 6,000 ft. in eastern Pennsylvania.

Pennsylvanian rocks overlie those of Mississippian age with the basal boundary, in much of the basin, marked by a sudden change from older, thinly-bedded, relatively fine-grained rocks to younger, massively-bedded, conglomeratic quartz sandstone. The Pennsylvanian sequence is commonly thickest and coarsest-grained along the eastern periphery. In eastern Pennsylvania, where only the lower half of the Pennsylvanian is preserved, a thickness of 4600 ft. of principally sandstone, conglomeratic sandstone and conglomerate was recorded (Ref. 2.5-10).

2.5.1.1.2.2 The Valley and Ridge Province

The Valley and Ridge is a physiographic province which is situated within the Appalachian Basin and consists of a nearly continuous sequence of rocks extending from the Cambrian to the Pennsylvanian. Within this sequence are two major clastic intervals, the Cambrian-Silurian Taconic cycle and the Devonian-Pennsylvanian Appalachian cycle (Ref. 2.5-11, p. 231). Each cycle consists of pre-orogenic carbonates and orthoquartzites

overlain by turbidite flysch deposits which are, in turn, succeeded by molasse. The first phase of the older cycle is represented by Cambrian to Middle Ordovician carbonates (Ref. 2.5-12, p. 4). The flysch phase is represented by siltstone, silty-shale and gray sandstone of the Upper Ordovician Reedsville. The molasse phase comprises the Upper Ordovician Bald Eagle and Juniata Formations and the Lower Silurian Tuscarora Formation. The transition from the molasse phase to the renewal of marine conditions is delineated by the successively younger Rose Hill, Keefer, Mifflintown and Bloomsburg Formations of Middle Silurian age. Upper Silurian to Lower Devonian carbonates (Wills Creek to Onondaga) identify the first phase of the Appalachian cycle (Ref. 2.5-12, p. 4). Within this younger cycle direct passage from the carbonate phase to the turbidite phase was interrupted by deposition of a local sub-aqueous delta identified in central Pennsylvania as the Mahantango Formation. Following sedimentation of the Mahantango, the turbidite beds of the Upper Devonian Trimmers Rock Formation, constituting the second phase of the Appalachian cycle, were laid down. The molasse phase was initiated by the Upper Devonian Catskill Formation which, at an outcrop along the Lehigh River (Figure 2.5-7), is in gradational contact with the underlying Trimmers Rock (Ref. 2.5-13, p. 8). The molasse phase culminated twice, first in the Misissippian Pocono Formation and later in Pennsylvanian rocks (Ref. 2.5-12, p. 4).

2.5.1.1.2.3 Stratigraphic Units Within the Site Vicinity

Stratigraphic nomenclature used throughout this FSAR follows the recent usage of the Pennsylvania Geologic Survey who have not recently used the terms Susquehanna Group, Hamilton Group or Fort Littleton Formation in the site vicinity (See for example Ref. 2.5-12 and 2.5-17).

Middle Silurian to Pennsylvanian rocks within 10 miles of the site have been folded on the Berwick Anticlinorium. The units exposed across the fold are:

- The Middle Silurian Bloomsburg
- Upper Silurian Wills Creek
- Upper Silurian Tonoloway
- Middle Devonian Marcellus
- Middle Devonian Mahantango
- Upper Devonian Trimmers Rock
- Upper Devonian Catskill
- Upper Devonian-Lower Mississippian Pocono
- Middle Mississippia-Pennsylvanian Mauch Chunk
- Pennsylvanian Pottsville and "Post-Pottsville" (Llewellyn) Formations

In central Pennsylvania the Bloomsburg Formation was deposited in a brackish, shallow water, marine environment which is transitional between fluvial, continental sediment to the east and marine carbonates, shale and marl of the interfingering Wills Creek Formation to the west (Ref. 2.5-14, p. 119). It is a thick-to massive-bedded, dominantly grayish-red silty claystone with two sandstone intervals which occur both at the base and near the top (Ref. 2.5-14, p. 119). The sandstone intervals are medium-to-thin-bedded, poorly sorted hematitic sub-graywacke. The Bloomsburg is highly calcareous in the vicinity of Lewisburg, Pennsylvania, approximately 40 miles southwest of the site.

In central Pennsylvania the Bloomsburg is separated from the Marcellus Formation by about 1770 ft. of dominantly limestone and calcareous shale. These lithologies belong, in stratigraphically higher order, to the Wills Creek (Upper Silurian), Tonoloway (Upper Silurian), Keyser (Upper Silurian to Lower Devonian), Old Port (Lower Devonian) and Onondaga (Lower to Middle Devonian) formations (Ref. 2.5-12, Table 1).

The Wills Creek Formation gradationally overlies the Bloomsburg and consists of interlayered dark gray to greenish shale, red siltstone, light gray-green to olive siltstone and silty shale (all calcareous) and light gray dolomite to argillaceous dolomite. Medium gray limestone may be present.

The Tonoloway Formation gradationally overlies the Wills Creek and is composed of medium to dark gray, thinly laminated to thinly bedded limestone with some thin beds of medium gray calcareous shale. The Tonoloway is dolomitic at several locations.

The Upper Silurian to Middle Devonian Keyser, Old Port and Onondaga Formations were not mapped north of, nor east of Bloomsburg, Pennsylvania. The Keyser, therefore, does not appear to occur within ten miles of the site but was mapped further southwest (Subsection 2.5.1.1.3.3).

The lower Keyser is dominantly medium gray, fossiliferous, "pseudo-nodular" limestone which is cobbly when weathered. The upper Keyser contains laminated to thin bedded limestone similar to the underlying Tonoloway.

The Old Port and Onondaga Formations do not occur in the site vicinity but do crop out north of Bloomsburg (Subsection 2.5.1.1.3.3). The Old Port consists of dark gray, whitish weathering chert, underlain by calcareous shale and thin gray limestone. The chert is locally overlain by gray to buff, medium to coarse grained fossiliferous sandstone.

The lower Onondaga is medium gray, highly fissile shale which is calcareous toward the top. The upper Onondaga is medium to dark gray, dense, fossiliferous argillaceous, locally carbonaceous, microcrystalline limestone.

The Marcellus, in central New York State, where it was defined, is about 350 ft. thick and consists predominantly of black shale with lsser amounts of black limestone (Ref. 2.5-15, p. 103). In east-central Pennsylvania, between Harrisburg and Williamsport, the Marcellus is a uniformly massive black, carbonaceous shale with several thin to thick bedded fine grained sandstone units (Ref. 2.5-16, p. 156). According to Faill (Ref. 2.5-17) the Marcellus refers exclusively to black shale overlying the Onondaga Formation.

The Mahantango Formation, which underlies most of the site, consists primarily of silty mudrock, shale, siltstone and sandstone with local occurrences of conglomerate, limestone and ironstone (Ref. 2.5-18, p. 13-14). In eastern Pennsylvania the Mahantango overlies the Marcellus shale and is, in turn, overlain by the Harrel Shale (Ref. 2.5-19, p. 18), a feature corroborated by Kaiser (Ref. 2.5-18, p. 6) who indicated that the Mahantango is defined by black shale, both at its base and its top.

Kaiser (Ref. 2.5-18, p. 18) informally divided the Mahantango into lower, middle and upper members. The basal member consists of an olive-gray shale with a basal sandstone and the middle member contains siltstone and shale, but where it is sandy it is identified as the Montebello. The upper member comprises an olive-colored shale, siltstone and sandstone with the sandstone locally highly ferruginous, finer grained, darker colored and more argillaceous than the underlying Montebello. Faill (Ref. 2.5-17, p. 23-24) divided the Mahantango into five members which are, in stratigraphically higher order, the Turkey Ridge, Dalmatia, Fisher Ridge, Montebello and Sherman Creek. According to Wells and Faill (Ref. 2.5-12, Table 1) the Turkey Ridge is a light to olive-gray, fine to coarse-grained sandstone and the Fisher Ridge is predominantly a laminated olive gray to medium-gray silty shale. The Montebello is an olive-gray, medium-light gray to dusky yellow, fine to medium-grained, locally conglomeratic, fossiliferous sandstone with interbedded siliceous siltstone and silty claystone which display cycles of reverse graded bedding. The Sherman Creek (Ref. 2.5-17) or Sherman Ridge (Ref. 2.5-12) comprises olive-gray, fossiliferous, silty claystone with two interbedded siltstone and fine sandstone units which coarsen upward.

Faill (Ref. 2.5-17, p. 23-24) noted that the Middle Devonian rocks are cyclic with each cycle marked by black to dark gray, silty claystone at the base and displaying an upward increase in grain size to conglomeratic sandstone. Immediately overlying the

coarsest rock units there is a marked decrease in grain size with claystone or siltstone marking the base of the overlying cycle.

The cyclic nature of the Middle Devonian strata is reflected within the Mahantango. These internal cycles are asymmetric and are smaller scale reflections of the cyclicity recorded throughout the entire Middle Devonian. That is, they commence with black to dark or olive-gray silty claystone which grades upward into argillaceous siltstone, silty sandstone and fine to medium-grained, locally conglomeratic, silty sandstone. The cycles within the Mahantango are repetitive and range, in thickness, from approximately 7 to 250 ft. The thicker cycles can usually be traced over distances of from 5 to 35 miles. (Ref. 2.5-20, p. 113).

The average thickness of the Mahantango is about 1650 ft. with respective maximum and minimum thicknesses of about 2900 ft. at McCullochs Mills, Pennsylvania and 840 ft. at Riverside which is northeast of McCullochs Mills and about 20-22 miles west of the site (Figure 2.5-7). Overall the Mahantango shows a general thinning to the north, a feature reflected in the Montebello sandstone member which is thickest just west-northwest of Harrisburg and thins to the west, north and east. North of the 41st parallel, which lies just south of the site, the Montebello has totally disappeared (Ref. 2.5-18, p. 13-14).

In the Anthracite region of Pennsylvania the Mahantango and the Marcellus were combined to form a lithotectonic unit. As defined by Wood and Bergin (Ref. 2.5-21, p. 151) this lithotectonic unit actually includes the Marcellus, Harrell and Brallier Shales and the Tully Limestone. However, the Brallier overlies the Mahantango at about the Tully horizon (Ref. 2.5-19, p. 18). The Tully, in this area, has been incorpoated into the Mahantango. Thus, in this report, the lithctectonic unit of Wood and Bergin is considered as containing the Marcellus, Mahantango, Tully and Harrell. In the southwestern and western parts of the Anthracite region this unit is 1100 ± ft. thick whereas in the central and eastern parts it is about 3000 ± ft. thick; the average thickness is 2000 ± ft. (Ref. 2.5-21, p. 148). However, a greatly thickened section of this unit occurs in the P. Good No. 1 Well (Figure 2.5-7) on the crest of the Berwick Anticlinorium, east of the site. This excess thickness is believed to be due to faulting and disharmonic folding (Ref. 2.5-21, p. 148).

At the site the Mahantango consists of a lower, gray, calcareous siltstone (120-150 ft. thick) overlain by a dark gray, locally fossiliferous siltstone which is intermittently calcareous. These two members are lithologically similar to and occur within the same stratigraphic interval as the Harrell Shale and the underlying Tully Limestone; thus, the latter two units were incorporated into the Mahantango.

In the site vicinity, the Mahantango is represented only by the uppermost member, the Sherman Creek which is dominantly a dark gray to blue gray, olive gray to brown weathering mudrock. Siltstone and fine grained sandstone units crop out locally. Both calcareous and non-calcareous strata occur at several localities.

In the site vicinity an interval of light, medium-gray argillaceous limestone, near or at the top of the Mahantango, was recognized as a Tully Limestone equivalent and was included within the Mahantango. Calcarous silty mudrocks which may be Tully equivalents also occur (Subsection 2.5.1.2.2). Faill (Ref. 2.5-17, p. 24) also included the Tully as part of the Mahantango because of its lithologic similarity to the Sherman Creek Member. The overlying Harrell Formation, a poorly exposed, dark silty shale which appears to be in gradational contact with the Mahantango was incorporated into this map unit (Subsection 2.5.1.2.2).

Fossils are relatively abundant within the Sherman Creek member of the Mahantango Formation and include various genera of brachiopods, bryozoa, pelecypods, coral, trilobites and crinoid fragments. Fossil casts are abundant with occasional molds and rare preservation of internal structure and original shell material.

Concretions (commonly rusty weathering), spheroidal weathering and prominent closely spaced steeply dipping cleavage, which may quite easily be mistaken for primary bedding fissility, are other features characteristic of the Mahantango. Due to its predominantly argillaceous nature and cleavage, the Mahantango is fairly easily eroded and is thus topographically expressed as a relatively low area.

In the general area marked by the confluence of the Susquehanna and Juniata Rivers, the Trimmers Rock Formation comprises an interlayered assemblage (about 2000 ft. thick) of thin to medium-bedded, medium gray siltstone and medium gray, slightly silty and somewhat fissile shale. Thin layers of fine-grained sandstone occur in the upper part (Ref. 2.5-12, Table 1). Graded bedding, along with groove and flute casts, occur in some of the siltstone beds indicating deposition by turbidity currents. Load casts or ball and pillow structures are also present in some siltstone layers. These lithologies and sedimentary structures are also present in the Trimmers Rock at an outcrop along the Lehigh River about 16 miles southeast of the site (Ref. 2.5-13, p. 8) (Figure 2.5-7). At this exposure both bedding thickness and grain size increase upward in this formation which is about 1165 ft. thick.

In the site area the Mahantango and Harrell grade upward into the Trimmers Rock (Subsection 2.5.1.2.2). The Trimmers Rock is dominantly interbedded, medium to olive gray, thinly laminated

siltstone, silty shale and fine grained, laminated to massive sandstone. These rocks weather to brownish gray color. Sedimentary structures include fining-upward sequences, groove casts, current lineations, load casts, ball and pillow and flow roll structures. Ripple marks were also locally indentified. These structures indicate deposition by turbidity currents in a marine environment. Fossils are often restricted to relativley thin layers of brachiopods. Other fossils include pelecypods and crinoid fragments.

The upper Trimmers Rock Formation consists of light to medium grayish green silty shale and micaceous, dark greenish gray siltstone (both of which weather to a dark reddish brown color), a reddish brown silty fine to medium grained sandstone to siltstone and an olive green vitreous, fine grained sandstone to siltstone. The uppermost units of the Trimmers Rock Formation grade upward into the basal Catskill Formation. This gradation between the Trimmers Rock and the Catskill has apparently caused problems concerning the placement of the contact between them. However, Faill and Wells (Ref. 2.5-22) have placed this contact at the base of a thick sandstone unit which is the lowest occurrence of upward fining cycles, a feature characteristic of the Catskill. This unit was selected by Faill and Wells because it is easily mappable. Glaeser (Ref. 2.5-13, p. 4) evidently concurred with Faill and Wells for he considered the base of the Catskill to lie at the first occurrence of distinctive sandstone units which are found above or near the top of the turbidite suite which constitutes the bulk of the Trimmers Rock Formation.

The Catskill Formation at the Lehigh River outcrop is about 7675 ft. thick and consists mainly of siltstone and sandstone with some conglomerate and shale. Upward fining cycles were recognized at various intervals throughout the entire formation both at Lehigh River (Ref. 2.5-13, Figure 2) and near Halifax, Pennsylvania (Figure 2.5-7), (Ref. 2.5-22, p. 107).

Complete or nearly complete sections of Upper Devonian rocks are preserved, both in outcrop and in the subsurface, at the Lehigh River outcrop, the Richards Well and the Hudson Realty Well (Figure 2.5-7). Based on these occurrences Glaeser (Ref. 2.5-13, p. 35) estimated the original thickness of the Upper Devonian section at various locations in northeastern Pennsylvania. He then compared these estimates to the amount of section preserved today and ultimately estimated the amount of section lost. The amount of missing section ranges from 0 ft. to about 6125 ft. (Ref. 2.5-13, p. 38) a variation due mainly to the location of the sections with respect to structure. For example, in the P. Good Well, which occurs on the nose of the Berwick Anticlinorium, about 8 miles east of the site (Figure 2.5-7), about 5203 ft. are missing. Glaeser (Ref. 2.5-13, p. 36) assumed that these sections were lost due to erosion.

In the site area, the contact between the Catskill and the underlying Trimmers Rock was drawn at the base of the first relatively thick reddish brown to maroon sandstone or brownish red siltstone and more massive reddish brown (maroon) micaceous fine grained sandstone. This mappable contact appears to occur at or near the lowest fining upward sequences. The lower Catskill Formation contains sedimentary structures such as intraformational clasts of green shale within light gray fine sandstone, oscillation ripple marks and roots which are indicative of the marine to non-marine transition zone.

The Pocono Formation which overlies the Catskill, consists typically of medium and coarse grained light gray to white, rusty weathering quartz sandstone with thin layers of quartz pebble conglomerate. Olive gray, fine grained sandstone, reddish gray medium to fine grained sandstone and siltstone and greenish gray medium grained, cross bedded sandstone also occur within this formation. Cross bedding is common.

Grayish red sandstone layers occur near the base of the Pocono. These were recognized along the east side of the Susquehanna River South of Mocanaqua and along the road between Aldean and Folstown. North of the site, the Pocono consists of an interlayered sequence of predominantly medium gray, thick, well laminated, gray weathering quartz sandstone and subordinate, red, flaggy quartz sandstone. Near Folstown, well laminated red sandstone is interlayered with, but decidedly subordinate to, well laminated, rusty weathering, light gray, coarse grained sandstone and fine grained gray sandstone. Coarse to medium grained, mainly grayish to greenish gray sandstone featuring rather subtle cross bedding dominate the upper portion of the exposure. These strata, along with an underlying thin zone of light greenish gray sandstone, in turn, underlain by green shale and mudrock, has been selected as marking the basal Pocono. Beneath all of these units, is a red, well laminated, argillaceous siltstone which is interpreted as marking the top of the Catskill. Red shale, which marks the base of the outcrop underlies cross bedded medium, light gray, olive gray weathering quartz sandstone. The lower (topographically and stratigraphically) portion of the outcrop is dominated by red lithologies in contrast to the upper part in which no red lithologies were exposed. Besides the obvious color change the sandstone above the inferred contact is coarser grained and more subtly cross bedded than sandstone which occurs between the red units near and at the base of the outcrop. Thus, contrary to other interpretations the Catskill-Pocono contact appears to be gradational in the site area rather than unconformable.

The upper Pocono Formation in the vicinity of Shickshinny consists of medium to light gray conglomeratic sandstone with rounded to sub-rounded quartz pebbles and shale fragments and rusty weathering, fine to medium grayish green, micaceous

siliceous sandstone and finely laminated greenish gray, rusty weathering, siliceous quartz sandstone. Rusty weathering, medium light gray, medium to coarse grained quartz sandstone is interbedded with thin layers of dark gray silt. Shale and medium gray quartz lithic sandstone fills channels.

The overlying Mauch Chunk Formation is generally bright red in color and consists of mudrock, silty shale, siltstone and fine to medium grained cross bedded, well laminated sandstone. In the southern part of the Anthracite region, this sequence of red beds is 2,400 ± feet thick and is overlain by a sequence of alternating red sandstone and shale beds and gray conglomerate and sandstone beds 300 to 600 feet thick. This upper sequence represents a transition zone in which red beds typical of the underlying Mauch Chunk are interbedded with gray beds typical of the overlying Pottsville Formation (Ref. 2.5-10). Detailed stratigraphic studies indicate that the beds of the transition zone (upper Mauch Chunk) intertongue with and laterally replace the lower beds of the Pottsville Formation from south to north. The upper Mauch Chunk is, therefore, late Mississippian and Early Pennsylvanian in age (Ref. 2.5-10).

Both lower and upper members of the Mauch Chunk Formation are exposed in the site area. The lower member is exposed immediately above the conformable contact with the underlying Pocono Formation at several locations. The upper part of the formation along the south limb of the Lackawanna Synclinorium is marked by interlayered red and olive gray sandstone, siltstone, and silty shale. Locally the siltstone contains layers of rounded, circular to elliptical calcite filled voids. Elsewhere the Mauch Chunk contains greenish gray to grayish green medium to coarse grained, locally micaceous sandstone, thinly laminated gray, fine grained sandstone and siltstone and massive medium grained sandstone.

The Pottsville and Llewellyn formations represent the coal bearing zones of the Anthracite Region and have, for the purpose of this report, been combined and treated as a single formation. The Pottsville is composed of coarse pebble conglomerate, quartzose sandstone, subgraywacke, siltstone, shale and anthracite. This formation ranges in thickness from about 1,400 feet in the southern Anthracite field to about 600 feet in the Western Middle Anthracite field (Ref. 2.5-10).

Strata overlying the Early to Middle Pennsylvanian Pottsville have been informally termed "Post-Pottsville" rocks (Ref. 2.5-24). "Post-Pottsville rocks" in the Southern and Western Middle Anthracite fields were named the Llewellyn Formation by Wood (Ref. 2.5-10). This name is informally used for the grayish and brownish conglomeratic sandstones, quartz sandstones, subgraywackes, and siltstones overlying, but not subdivided from, the Pottsville Formation in the site area. Usage of the name

Llewellyn for "Post-Pottsville" rocks in the Northern Anthracite field is consistent with Bergin (Ref. 2.5-23).

Within five miles of the site, the Pottsville and Llewellyn, collectively consist of quartz pebble conglomerate in a quartz sandstone matrix, quartz pebble conglomerate in a carbonaceous quartz sandstone matrix, coarse grained dark to medium gray, massive and flaggy, carbonaceous sandstone and shale, dark gray to black siltstone and coal. The non-carbonaceous quartz pebble conglomerate displays cross beds.

2.5.1.1.3 Regional Tectonics

2.5.1.1.3.1 Tectonic Provinces

The Appalachian orogen in the northeastern United States was divided into two parts, the mobile belt and the craton (Ref. 2.5-25 and Subsection 2.5.2.2). The mobile belt in this area lies along the east coast with its western edge parallel to, and west of, the eastern limit of North America (Figure 2.5-8). In general the mobile belt is underlain partly by Precambrian crustal rocks and partly by presumably mafic crust. However, in the Maritime Provinces of Canada as well as in southeastern Massachusetts it is underlain by a volcanic-sedimentary sequence which formed less than 600 million years ago. These three grossly-grouped lithologies, i.e. Precambrian crustal rocks, mafic crust and volcanic-sedimentary rocks of the Avalon Platform provided the basis for dividing the mobile belt into the 1) eastern cratonic margin, 2) the Central New England tectonic province and 3) the Avalon Platform tectonic province respectively (Ref. 2.5-25). In the eastern cratonic margin the Precambrian basement is overlain by 1) Late Precambrian clastic rocks and associated mafic dikes and volcanics, 2) a miogeosynclinal assemblage and 3) a eugeosynclinal assemblage (Ref. 2.5-25). The eastern cratonic margin is marked by a zone of faulting, contrasting structural styles and contrasting metamorphic facies. The Central New England province is bounded on the east by the Avalon platform and on the west by the Inner Piedmont. It features a thick, dense, presumably mafic crust overlain by eugeosynclinal sediments. It is also marked by intense deformation and Lower Paleozoic metamorphism. The Avalon Platform province is characterized by crystalline, continental crust (Late Precambrian) intruded by Ordovician to Devonian age plutons. In juxtaposition with, and to the west of the mobile belt, lies the craton which is underlain by Precambrian crystalline rocks that were deformed during the Grenvillian orogeny, about 1 billion years ago. Based on gross geologic structure the craton was divided into an eastern belt and a western basin. The eastern belt is coincidental with the

Highlands Tectonic Province (Figure 2.5-8) which is characterized by Grenvillian (Precambrian) rocks deformed during Paleozoic crustal convergence. The western portion, which is subdivided into the Fold and Thrust Belt and the stable interior is characterized by the absence of basement involvement during Paleozoic crustal convergence (Ref. 2.5-25). The Fold and Thrust Belt, in which the site is located, is in contact with the Highlands Province of the eastern craton and exposes tightly folded and faulted Paleozoic sedimentary rocks. To the west of, and in sharp contact with, the Fold and Thrust Belt lies the Stable Interior which is underlain by very gently folded, shelf delta type deposits. The Fold and Thrust Belt and the Stable Interior coincide approximately with the Valley and Ridge (including the Great Valley) and Appalachian plateaus Physiographic provinces respectively.

2.5.1.1.3.2 Structural Elements within the Craton

The site is situated upon the Scranton gravity high (Figure 2.5-9) which extends southwestward from Albany, New York to Harrisburg, Pennsylvania where it abruptly terminates (Ref. 2.5-26, p. 198; Ref. 2.5-27, p. 711). To the west of this termination, regional gravity patterns suggest a northwest trending Precambrian fault with left lateral displacement of several tens of miles (Ref. 2.5-27, p. 711).

The high, itself, is located in both the Fold and Thrust Belt and the Stable Interior. The maximum Bouguer anomaly values associated with this feature occur in northeastern Pennsylvania and adjacent New York State where the overlying sedimentary section is at least 39,370 ft. thick (Ref. 2.5-28, p. 52 and 2.5-26, p. 201). Of all the models (which include tensionally induced rifting) proposed to explain this feature (see Ref. 2.5-26, p. 203-209) the favored one involves warping of the mantle with the anomaly due to an extensively broad mass occurring deep within or at the base of the crust. This structure is apparently related to the tectonic evolution of the Appalachian system (Ref. 2.5-26, p. 213 and 218-219).

Principally the Fold and Thrust Belt contains deformational features indicative of regional crustal compression (Figure 2.5-8). In the area northeast of Roanoke, Virginia the structural style, at the surface, is dominated by folding with faulting subordinate, whereas southwest of Roanoke reverse faults predominate over folding at the surface (Ref. 2.5-29, p. 125). Another feature present in this belt, and indeed in the entire Appalachian Orogen, is an arcuate configuration which is especially well expressed in central Pennsylvania.

The largest folds in the Fold and Thrust Belt exceed 125 miles in length but folds of microscopic to hand specimen scale have also been recognized. The largest folds, with wave lengths ranging from 6 to 11 miles, were classified by Nickelsen (Ref. 2.5-30, p. 16) as first order folds, whereas the hand specimen and microscopic size folds were classified as fifth order folds. Second through fourth order folds are intermediate in size. The largest folds are not restricted to the Fold and Thrust Belt for they also occur in the adjacent Stable Interior (Ref. 2.5-30).

Generally these folds do not display an ideally parallel form, rather their hinges are usually narrow relative to their wave lengths. They are somewhat akin to similar folds yet they lack the characteristic features of similar folds which include attenuated limbs, with correspondingly thickened axial regions, and a sinusoidal form (Ref. 2.5-31, p. 10). Thus, according to Faill, the fold geometry is neither parallel, similar nor intermediate for it shows features that are not associated with either geometric type, i.e., the bedding in the limbs is planar and the hinges are narrow (Ref. 2.5-30, p. 19).

Although the folds lack the characteristic geometry of parallel folds, they are flexural slip in nature for they display wedge faults, uniform bed normal thickness across the fold and slickensides on bedding surfaces (Ref. 2.5-32, p. 1289 and 2.5-31, p. 11). In addition to these flexural slip folds, kink bands, a few inches to hundreds of feet wide, are visible in outcrop. Kinematically and geometrically the kink bands and the flexural slip folds are congruent and, therefore, related (Ref. 2.5-32, p. 1289).

Kink bands are usually considered to be small scale structures, however they occur on a much larger scale in the Fold and Thrust Belt with smaller kink bands and folds present in the limbs of the larger-scale structures. Faill attributed the existence of large scale kink bands to the wide spacing between bedding surfaces.

From southeast to northwest across the Fold and Thrust Belt, and westward into the Stable Interior, the folds become progressively less tightly appressed. This gradual change from tight to more open folds is illustrated by changes in the inter-limb angle which is about 50°-70° on the east side of the Great Valley and approximately 80° on the west side. In the central Valley and Ridge (Fold and Thrust Belt) the limbs subtend an angle of about 100° and in the Appalachian Plateaus (Stable Interior) this angle is nearly 180° (Ref. 2.5-32, p. 348). This change is also expressed by differences in structural relief which diminishes from possibly 35,000 ft. on the South Mountain Anticlinorium on the southeast (Ref. 2.5-33, p. 348) to 7,000-9,000 ft. in the central Valley and Ridge to about 4,500 ft. in the western Valley and Ridge. In the eastern Plateaus area 2,500

to 3,000 feet of structural relief occur. Across the Plateaus area this relief continues its progressive decrease with the most westerly folds showing less than 300 ft. (Ref. 2.5-33, p. 349). In fact in the Plateaus region the folds are so broad and gentle that structural contour maps are required in order to analyze them. Between the Valley and Ridge (Fold and Thrust Belt) and the Appalachian Plateaus (Stable Interior) there is an abrupt decrease in structural relief. This area has been termed the Appalachian Structural Front (Ref. 2.5-34).

LANDSAT images of an area along the west branch of the Susquehanna River at Lewisburg, Pennsylvania suggested the presence of a nearly northwest trending cross or tear fault. This structure shows about one mile of left lateral separation of a prominent ridge underlain by the Tuscarora Formation. However, geological reconnaissance mapping confirms that this left lateral topographic offset is caused by a kink fold as shown on the Geologic Map of Pennsylvania (Ref. 2.5-24).

Faults, like the folds, occur on all scales within the Fold and Thrust Belt and show displacements ranging from inches to hundreds of feet. The largest faults range in length from about 7 miles to 200 miles (Ref. 2.5-33, p. 349).

According to Faill and Nickelsen (Ref. 2.5-31, 20) most faults seen at the surface can be classified as wedge faults or cross faults. Root (Ref. 2.5-33, p. 349) stated that all major faults in the Fold and Thrust Belt and the Stable Interior are moderate to steep thrusts with dips ranging from 40°-70° to the southeast. However, in the southern Great Valley (affecting part of the Fold and Thrust Belt), he also identified steeply dipping, west facing thrust faults and tear or cross faults (Ref. 2.5-34). Wood and Bergin (Ref. 2.5-21) noted that in the southeastern part of the Anthracite region (of the Valley and Ridge Province) there are hundreds of reverse, tear and bedding faults, whereas in the northern part faults are far more scarce with only reverse faults, showing minor displacements, having been recognized. Glass (Ref. 2.5-36, p. 9) identified at least eighty major, essentially vertical faults in the Appalachian Plateaus Province. Generally these faults are normal to the regional tectonic grain although variations between N05W and N89W were observed. Wedge, splay and reverse (thrust) faults are categorized together because they all result in crustal shortening and duplication of strata. Faill (Ref. 2.5-32, p. 1298) indicated that splays off decollements and wedge faults are identical. However, Root (Ref. 2.5-35) distinguished between thrusts dipping steeply to the west and those dipping steeply to the east with the former equated to wedge faults and the latter to splay faults off decollements. The Hunting Valley-Cream Valley Faults and the Sweet Arrow Thrust (Figure 2.5-7) appear to fit into this category.

Wedge faults intersect bedding at a low angle (10°-30°) and terminate in bedding planes (Ref. 2.5-32, p. 1295), although a number of them terminate in folds (Ref. 2.5-32, p. 1298). Commonly they occur as isolated structures on the limbs of folds but they have also been observed in fold hinges. In outcrop wedge faults generally occur in interlayered sequences displaying contrasting mechanical properties and only cut across beds of sandstone or siltstone that are surrounded by shale (Ref. 2.5-32, p. 1297 and 2.5-31, p. 23). This same relationship also holds on a much larger scale, since most of the large, mappable faults occur in interlayered sequences of contrasting lithologies.

Root (Ref. 2.5-35, p. 105-106) identified faults in the southern Great Valley which presently dip steeply to both the east and west. The steeply inclined, east dipping reverse faults generally parallel the trace of anticlinal hinges and cut the vertical to overturned, west facing anticlinal limbs. These faults developed as steeply dipping schuppen structures which splayed off subhorizontal reverse faults (decollements) (Ref. 2.5-37, Figure 5, 2.5-33, P. 350 and 2.5-35, Figure 5). Reverse faults, which dip steeply to the west, are also present in the west-facing subvertical to overturned limbs and parallel the structural fabric. However, some have been rotated during folding and now display the geometry of east-dipping normal faults.

In the Anthracite region, faults (akin to the steeply east-dipping splays described by Root) are inferred to occur in the cores of anticlines (Ref. 2.5-21). Wood and Bergin interpreted that many of these faults were folded along with the rock units, however folded splay faults have not been depicted in other publications (e.g. Ref. 2.5-37 and 2.5-33).

Cross (transverse) faults are commonly vertical to nearly vertical structures which are approximately perpendicular to the regional tectonic grain. They are less common than wedge faults although they have been mapped in the southeastern part of the Anthracite region, in the Great Valley and in the Appalachian Plateaus region (Ref. 2.5-21, p. 149, 2.5-35 and 2.5-36). They are also evident in the Cambro-Ordovician rocks of the Conestoga Valley near York and Lancaster, Pennsylvania (Ref. 2.5-24). Commonly cross faults display strike-separation and, in fact, have been described in the Appalachian Plateaus Province as wrench faults (Ref. 2.5-36). According to the verbal description provided by Glass (Ref. 2.5-36, p. 6) at least some of these faults could be identified as paired conjugate wrench faults, for those that strike nearly north show left lateral separation whereas those which strike in a more westerly direction display right lateral separation. Despite his verbal description the map pattern shows a general pattern of anastomosing fault traces which are more or less subparallel to each other and normal to the trend of the Allegheny Structural Front (Ref. 2.5-36, p. 8).

It is conceivable that both wrench faults and cross faults occur in the Appalachian Plateau but attempting to distinguish them is of little import for both are predictable cogenetic products of regional horizontal compression. Two cross faults in the southern Great Valley have lateral movements associated with them (Ref. 2.5-35, p. 104), and the map pattern in the Conestoga Valley shows lateral separations of lithologic units along the cross faults there (Ref. 2.5-24). Another feature common to most cross faults is that reverse or wedge faults frequently terminate against them. According to Root (Ref. 2.5-35, p. 103), however, their most distinctive feature is that the rocks on either side of these faults have experienced different amounts of horizontal shortening.

2.5.1.1.3.3 Structural Elements in the Site Vicinity

The site rests on the easterly plunging nose of the Berwick Anticlinoirum. Across this folded structure, at depths of 17,500-25,000 ft ± 3,500 ft, the strata (based on seismic reflections) are nearly horizontal to slightly north dipping (Ref. 2.5-38, p. 134). According to Wood and Bergin (Ref. 2.5-21) either a major decollement or a series of decollements probably exists within the Marcellus Shale throughout most of the Anthracite region. Although not seen at the surface the presence of decollements is inferred based upon the existence of disturbed outcrops "differing styles, wavelengths and amplitudes of folds above and below the Marcellus, and by wells that penetrated either duplicated sections or greatly thickened sections" (Ref. 2.5-21, p. 151).

It is worthy to note, however, that neither this fault nor any other fault appears as a surface feature associated with the Berwick Anticlinorium in either Wood and Bergin's paper (Ref. 2.5-21, Figure 2) or Gwinn's paper (Ref. 2.5-38, Figure 1). The 1960 edition of the Pennsylvania Geologic Map (Ref. 2.5-24) does indicate a fault along part of the north limb of the Berwick Anticlinorium just north of Bloomsburg. This fault approximately 8 miles long, separates the undifferentiated Wills Creek, Tonoloway and Keyser Formations from undifferentiated Onondaga, Marcellus and Mahantango Formations (Ref. 2.5-24, map unit Skw from map unit Dho, respectively). According to the state map, the missing interval between these two sets of units includes the Mandata and Oriskany Formations. Recent mapping indicates that the Tonoloway Formation is juxtaposed to the Marcellus Formation (Figure 2.5-10, Stations DJ-4B, -33 and -34). The fault on the geologic map of Pennsylvania was probably postulated to explain the missing stratigraphic interval on the north limb of the Berwick anticlinorium, which includes dominantly carbonate rocks of the Keyser, Old Port and Onondaga Formations; these units are present west of the indicated fault. The south limb of the

Berwick anticlinorium shows the same relationships occur on the north limb; that is, that the Keyser, Old Port, and Onondaga Formations are missing along the more western portion. However, no fault has been indicated to account for the missing section along the south limb (Ref. 2.5-24).

Mapping on a scale of 1:24,000 (Figure 2.5-10) indicates that west of the confluence of Fishing Creek and Little Fishing Creek (and also west of the interpreted fault), the Tonoloway, Old Port, Onondaga and Marcellus Formations were recognized but no limestone of the Keyser Formation was identified. Thus, one of the units (the Keyser Formation) which is interpreted as missing due to faulting is absent from the stratigraphic section about 4,000 feet west of the postulated fault, despite the fact that the Old Port and Onondaga are present there.

Field investigations at two locations along the trace of the proposed fault (Stations DJ-4A, DJ-4B, DJ-32, DJ-33, and DJ-34; Figure 2.5-10) failed to reveal any evidence of cataclasis except for dip slip slickensides associated with a small flexural slip fold in the Marcellus Formation at DJ-4B. In both cases the Tonoloway limestone and Marcellus shale were exposed within 100 feet of the hypothesized fault. In the former case, however, at DJ-4A and DJ-4B the Tonoloway and Marcellus are separated by less than 20 feet. Consideration was given to the difficulty of detecting faulting within argillaceous units; however, where faulting was recognized within the study area, all lithologies (i.e., limestone, shale, mudrock, siltstone, and sandstone) showed some evidence of cataclasis. While continuous exposure across the postulated trace was not available, no positive evidence for faulting was observed and thus, at best, the fault can only be inferred. The only location along or near the trace of the postulated fault where significant cataclasis occurs is at Station DJ-31 (Figure 2.5-10) about 1,500 feet north of the "fault trace" where shale of the Mahantango Formation is exposed. Here the strata show noticeable variations in both strike and dip directions. For example, at the west end of the exposure strata swing from an attitude of N45°E: 35°SE to N40°W: 15°SW. In the center the attitude changes from N20°E: 30°SE to N10°W: 35°E back to N15°E: 45°SE, and at the eastern end, strata are oriented about N60°E: 40°NW and N35°E: 45°SE. At the western end of the outcrop the change in orientation represents a fairly smooth continuum whereas at the eastern end both continuous and discontinuous changes in orientation occur. The discontinuous changes are marked by oblique slip faults with slickensides displaying rakes of about 60° to 70°. Three such faults trend N55°E: 35°NW, N80°E: 35°NW, and N88°E: 29°NW. No unequivocal movement plan was identified, but structures recognized elsewhere in the Valley and Ridge Province, as well as the area within five miles of the site, show that reverse faults form in response to the release of stored strain energy in tightly appressed kink bands; thus, these faults are interpreted as relatively small

scale accommodating reverse faults. This exposure (DJ-31) occurs within an area in which there is map scale folding producing deflections in the lithologies which is not unlike patterns seen elsewhere in the Valley and Ridge Province.

In the P. Good Well, located on the Berwick Anticlinorium east of the site (Figure 2.5-7), faulting has been interpreted at an approximate depth of 5,800 feet which is well above the depth range (17,000 to 25,000 ft) at which a major decollement is implied (Ref. 2.5-38). Thus, the decollement alluded to by Wood and Bergin (Ref. 2.5-21) as well as the fault seen on the Geologic Map of Pennsylvania (Ref. 2.5-24) 12 miles west of the site may actually be a splay fault(s) off a more deeply buried decollement. This suggested splay fault may also be, in part, responsible for the excess thickness seen in the P. Good Well of lithotectonic Unit 2, as defined by Wood and Bergin (Ref. 2.5-21).

In summary, no direct unequivocal evidence exists to either postulate or refute the existence of the fault which has been mapped on the north limb of the Berwick Anticlinorium west of the site (Ref. 2.5-24). However (1) the absence of the Keyser Formation on the north limb, west of the western limit of the inferred fault, and (2) the map pattern in which the units missing from the north limb of the anticlinorium are also absent from the south limb, yet the fault restricted to the north limb suggests that the missing section of the north limb is perhaps better explained by an unconformity than by faulting. Regardless of whether or not a fault is interpreted, data from the P. Good Well show that the limestone sequence missing from both the north and south limbs of the Berwick Anticlinorium is absent from the nose of the fold as well. Thus, if a fault is postulated, it pre-dates the formation of the Berwick Anticlinorium and does not pose a safety-related problem to the site.

Flanking the Berwick Anticlinorium on the north and south respectively, are the Lackawanna and Eastern Middle synclinoria. The Lackawanna Synclinorium, the axis of which passes about 3-1/2 miles north of the site, is about 120 miles long and displays a wavelength of 8 to 9 miles and an average amplitude of 4,000-5,000 ft (Ref. 2.5-21, Table 2). Near the axis of this fold the Mocanaqua decollement is exposed. The axis of the Eastern Middle Synclinorium is approximately 3-4 miles south of the site. This fold is about the same size as the Lackawanna Synclinorium (Ref. 2.5-21, Table 2).

The Nittany Anticlinorium is a large structure which is located just to the west of the Berwick Anticlinorium about 50 miles west of the site. Based on seismic records garnered from longitudinal and traverse profiles, there exists a series of well defined, subhorizontal velocity interfaces which are correlative with similar velocity contrasts in the Cambrian sequence and at the

top of the Precambrian basement (Ref. 2.5-38, p. 134). The deepest interface was estimated to occur at a depth of about 25,000 ± 3,000 ft.

The Birmingham Fault is located in the core of the Nittany Anticlinorium and is the only fault in Pennsylvania associated with a major, well-exposed decollement known as the Sinking Valley Fault (Ref. 2.5-33, p. 350). This fault is a steeply, east dipping splay (thrust) which is about 33 miles long.

2.5.1.1.3.4 Relationship Among Structural Elements

In the opinion of all previous workers all of the structural elements encountered in the Valley and Ridge Province are genetically related, a feature which is clearly demonstrated in several instances. First, cross faults and reverse faults are generally spatially related with the reverse (or wedge) faults commonly terminating against the cross faults. In fact, there are instances where one passes into the other. A prime example of this is in the Carbaugh-Marsh Creek Fault in the southern Great Valley. This fault is composed of two segments, an east dipping reverse fault which parallels the regional grain and passes continuously into a nearly east trending, subvertical cross fault across which right lateral separation has occurred (Ref. 2.5-37, p. 8, 9 and 822 and 2.5-35, p. 102-103). Rock units across this cross, or transverse, fault segment have shortened independently of one another. Because of this independent behavior of rock units across the transverse portion of this fault, Root (Ref. 2.5-35, p. 103) concluded that this segment existed prior to most of the regional deformation.

Faill and Nickelsen (Ref. 2.5-31) and Faill (Ref. 2.5-32) pointed out that slickenside orientations on: 1) bedding surfaces associated with flexural slip folding, 2) wedge faults, and 3) cross faults are similar indicating kinematic compatibility among these three structural elements. Furthermore, the lines of intersection of the wedge faults and bedding are subparallel to the fold axes. Splay faults with large displacements occur in the hinges of anticlines which posses a kink band geometry suggesting that these folds were produced by the splays which originate at depth along unexposed decollements (Ref. 2.5-38, 2.5-37 and 2.5-33, p. 349). Faill (Ref 2.5-32, p. 1298) also conceded that possibly all major anticlinoria are underlain by splay faults. Prior to Gwinns work, the existence of major decollements and thin skinned tectonics in the Appalachian orogen was somewhat debatable but the results of his latest study (Ref. 2.5-38) indicate clearly that these subsurface structures exist.

In the southern Great Valley, Root (Ref. 2.5-35) deduced the sequential development of folding and faulting. He suggested

that the cross faults existed prior to, or at the initiation of, most of the regional deformation, although he indicated that their origin is not understood. Nonetheless, he concluded (Ref. 2.5-35, p. 109) that the cross faults which are subvertical and normal to the regional grain were, during folding, equivalent to ac fractures. This is logical since the orientation of cross faults with respect to the other structural elements is not consonant with having originated as shear fractures; rather it is likely that they formed early in the tectonic history of this area as extension (ac) fractures which were subsequently utilized as strain discontinuities (tear faults) during the same protracted period of stress application. This interpretation is supported by Nickelsen and Hough (Ref. 2.5-39) who stated that systematic extension fractures in shales are grossly transverse to northeast-trending fold axes, and formed early and independently of folding and faulting. Root continued (Ref. 2.5-35, p. 111) by indicating that the earliest folds were broad and open and as horizontal shortening continued and the folds became more appressed; west dipping steep thrusts (wedges) formed in the upper strata. Continued shortening resulted in the development of subsidiary folds, east dipping thrusts (splay faults) and the rotation of the earlier formed, west dipping thrust to a steeper inclination. In the case of the Carbaugh-Marsh Creek Fault the east dipping splay fault linked up with a portion of the cross faults to form the Carbaugh-Marsh Creek Fault system.

Faill and Nickelsen (Ref. 2.5-31, p. 37) noted that deformation was initiated by vertical compaction during sedimentation. Regional horizontal compression followed, while the strata were still horizontal, with the earliest stage marked by microfolding in shales and limestones. Major decollements probably also occurred during this early stage and were followed by buckling and kink band folding. As the folds tightened, faults in the hinge along with some wedge and cross faults, developed. These faults were accompanied by tightly spaced fractures (fracture cleavage) which formed parallel to the axial planes of the folds. Faill and Nickelsen further concluded that with time the deformed materials passed progressively from a ductile stage to a more brittle stage.

2.5.1.1.3.5 Age of Deformation

Root (Ref. 2.5-35) concluded that all the structural elements in the craton developed during a single orogenic event (Alleghenian) about 230 million years ago. This age is based on an inferred episodic lead loss about 230 million years ago recorded by Rankin (Ref. 2.5-40) in zircons from the Catoctin Formation. However, Faill and Nickelsen (Ref. 2.5-36, p. 19) implied that the deformation occupied a greater time span, possibly commencing prior to complete lithification of Silurian age sediment (e.g.,

the Lower Silurian Tuscarora Formation). Despite this apparent difference concerning the age of the onset of deformation, there is general agreement that the major structural elements in the Fold and Thrust Belt and the Stable Interior are no younger than Late Permian to Middle Triassic in age (Ref. 2.5-37, 2.5-33, 2.5-35, 2.5-32, 2.5-31, 2.5-38, 2.5-41, 2.5-34, and 2.5-21).

There is considerable disagreement as to the age, nature and method of formation of the curvature that is so prominent along the entire Appalachian Chain. Drake and Woodward (Ref. 2.5-42, p. 49) conlcuded that this arcuation in central Pennsylvania is truly a rounded structure which formed in response to right lateral slip along the east trending Cornwall-Kelvin Fault, perhaps around Late Devonian time (Ref. 2.5-42, p. 59). Faill (Ref. 2.5-32, p. 1305-1306) noted that it is not smooth and continuous, as it appears, but instead is composed of straight segments, the joining of which marks the boundary between the northern and southern Appalachians Furthermore, he concluded that this "curvature" and the major folds were contemporaneous. Root (Ref. 2.5-37, p. 825) noted the same observations as Faill and indicated that these rectilinear elements oriented N17°E south of the Carbaugh-Marsh Creek Fault and about N40°-50°E north of the fault, assume their present position by rotation across the fault. He also added that all structural elements seen in the Piedmont to the southeast would be more compatible if the Piedmont were arcuate by the Early Paleozoic. This appears to be a contradiction since on the one hand, Root suggested that rotation across the Carbaugh-Marsh Creek Fault (presumably during the Late Paleozoic Alleghenian event) was responsible for this curvature, whereas on the other hand, he suggested that the curvature already existed by Early Paleozoic time. Fleming and Sumner (Ref. 2.5-42, p. 58) suggested that an embayment associated with the Late Precambrian-Early Paleozoic proto-Atlantic was responsible for this arcuration in central Pennsylvania. Rankin (Ref. 2.4-40) and Rodgers (Ref. 2.5-44) stated that Appalachian salients and recessess formed during the initial breakup of a continental mass which commenced about 820 million years ago. Thus, at present, although there is no unique hypothesis concerning the age and origin of this curvature there is general agreement that it is pre-Mesozoic in age.

Evidence of younger tectonics is confined to the mobile belt. The southern border of the Newark-Gettysburg basin is obscured in New Jersey by the overlap of Coastal Plain sediments; however, in Pennsylvania and Maryland, the Triassic sedimentary rocks lie unconformably upon lower Paleozoic quartzites and carbonates and, in a few areas, upon Precambrian gneisses, granites and metabasalts. Residual gravity anomalies indicate that southern "border faults" are covered by the younger Triassic sediments (Ref. 2.5-45).

The northern edge of the basin, in the area east of the Schuylkill River, borders on the granitic-gneissic complex of the New Jersey Highlands and its southwest extension, the Reading Prong. West of the Schuylkill River, rocks north of the border are Cambrian and Ordovician carbonates. Triassic rocks unconformably overlie adjacent older rocks along much of the northern border suggesting that this margin is not continuously faulted (Ref. 2.5-2). In Pennsylvania, only 35 percent of the margin is known to be faulted (Ref. 2.5-46). The northern border faults are characterized as en echelon fault zones that gives a crenulated appearance to the northern margin. Where overlap has occurred the contact dips approximatley 20° to the south (Ref. 2.5-46)

The Ramapo Fault System, a continuation of the northeast trending system of border faults, crosses the New York-New Jersey State boundary north of New York City. No evidence of surface rupture, warping or offset of geomorphic features has been observed along its member fault zones (Ref. 2.5-47).

Several faults of apparently large displacement occur within the center of the Newark-Gettysburg basin (Figure 2.5-11). These are the Chalfont and Furlong Faults in Pennsylvania and the Flemington and Hopewell Faults in New Jersey. Orientation and direction of movement of these faults are not known. Although generally considered to be steeply south dipping normal faults (Ref. 2.5-48 and 2.5-49), Sanders (Ref. 2.5-50) has suggested predominant strike slip movement and Faill (Ref. 2.5-46) indicates they may be high angle reverse faults resulting from intersection of two different axes of monoclinal folding within the basin.

Smaller Triassic faults cross cut the basin margins and extend well into the surrounding rocks, but usually show less than 3,000 feet of displacement. Associated with these faults are local concentrations of small faults of constant attitude and sense of displacement (Ref. 2.5-2).

The Triassic basins and associated faulting are located in the mobile belt, whereas the site is situated on the craton which includes the Fold and Thrust Belt. No tectonic structures of Mesozoic or younger age have been recognized in the Fold and Thrust Belt.

Analysis of lineaments observable on LANDSAT imagery yielded data consistent with these observations. The greatest number of linears plotted in the Valley and Ridge Province within the Appalachian salient strike N10-25°W (Figure 2.5-11). This is roughly normal to fold axes in the area and thus the cluster of lineaments parallels the direction of extension fractures and cross faults. The fold axes and bedding are well expressed as a cluster of east-northeast trending lineaments.

Secondary trends oriented north-northeast and northeast may indicate jointing of Mesozoic age. These are the dominant lineament trends expressed in the Newark-Gettsburg Basin.

2.5.1.1.4 Regional Uplift and Subsidence

Several investigators have presented evidence for present day crustal movement in the Atlantic Coastal Plain, the Fold and Thrust Belt and exposed shield areas (Ref. 2.5-51, 2.5-52, 2.5-53 and 2.5-54). Based on the accumulation of an eastward facing clastic wedge of sediments along the coastal plain, Owens (Ref. 2.5-52) concluded that post-Triassic diastrophism has affected the entire central and southern Appalachians with the latest recorded upwarping having occurred in the Pliocene to Quaternary. Brown and Oliver (Ref. 2.5-54) concluded that the "Appalachian Highlands are presently rising relative to the Atlantic Coast at rates of up to 6 mm/yr" with the elongate zones of relative movement paralleling either the major Appalachian Structural trend or the Appalachian drainage divide. Superimposed on this broader uplift are local zones marked by a slightly greater rate of vertical crustal movement. One of these, known as the Harrisburg feature, occurs near the eastern limit of the Valley and Ridge Province along a line which extends northward from the eastern edge of the Blue Ridge Province (Ref. 2.5-54, p. 26). They further suggested that the entire thickness of the lithosphere is involved in these movements.

No instances of faulting due to tectonism have been associated with this regional scale activity. The only known instance of surficial displacements in the Fold and Thrust Belt occurs in New York (about 120 miles east of the site) and New England where small scale (less than one inch of vertical separation), high angle reverse faults that parallel the regional tectonic fabric offset glacial striations. Oliver and others (Ref. 2.5-51) have considered glacial rebound and surficial effects such as thermal changes, hydration or a chemical process in the shales as well as tectonic stresses as possible causes of these faults. While admitting the available data are inconclusive, they appear to favor the hypothesis that the faults are the result of expansion due to hydration or the release of continuing pressure by melting of overlying ice or other causes. They further suggest that if the faults are of tectonic origin, an apparently poor correlation between fault locations and modern seismicity indicates that that episode of deformation is already completed (Ref. 2.5-51, p. 587). No structures of this nature have been found in Pennsylvania.

As discussed further in Subsection 2.5.1.2.3, the available data do not indicate that regional uplift is of significance to the Susquehanna SES.

2.5.1.1.5 Natural Hazards

A natural hazard has been defined by Burton and Kates as "those elements of the physical environment that are potentially harmful to man and his works". Thus, geological natural hazards would be potentially harmful geological elements of the physical environment. The geologic hazards to be considered are: subsidence due to coal mine collapse, subsidence due to karst collapse, and landslides.

The coal (anthracite) of northeastern Pennsylvania is located in the Northern, Middle, and Southern Anthracite fields. The southwest end of the Northern Field is the closest to the site being about 4 miles to the northeast (Figure 2.5-7). Within this Northern Field there are many well documented incidences of subsidence, particularly in the cities of Scranton, Wilkes-Barre, Nanticoke, and Pittstown (Ref. 2.5-55). There have also been incidences of subsidence in the Middle and Southern Fields, which at their closest points, are about 10 miles southeast of the site (Fig. 2.5-7). Thus, the site will not be affected directly by subsidence due to coal mine collapse.

The nearest major carbonate units are the Silurian Keyser and Tonoloway Formations which are composed of gray to dark gray, thick to thin bedded, crystalline to argillaceous limestones (Ref. 2.5-24). These formations are not major cavern producers (Ref. 2.5-56) especially in this portion of Pennsylvania, and thus do not pose a hazard of collapse. As mapped, these two formations occur as relatively thin beds on both limbs of the Berwick Anticlinorium which come together in the town of Berwick, Pennsylvania. This location is about 5 miles west of the site; thus, these units would pose no subsidence problems at the site. Miller (Ref. 2.5-57) stated that Onondaga limestone crops out along road and railroad cuts near Beach Haven, Pennsylvania. Examination of these outcrops indicates that the rock is dark gray, brownish weathering calcareous silty mudrock interbedded with thin layers of silt to clay shale or with siltstone. Lithology and fossil fauna indicates that these rocks belong to the Mahantango Formation (Subsection 2.5.1.2.2) which does not pose a subsidence problem at the site.

Radbruch-Hall (Ref. 2.5-58) placed the site in a region of moderate landslide incidence with a high susceptibility to landsliding. Moderate incidence means that generally less than 15 percent, but more than 1.5 percent, of the underlying rock or earth material is estimated to be involved in landsliding. A high susceptibility means that natural or artificial cutting, loading of slopes or anomalously high precipitation may cause landsliding involving more than 15 percent of the rock or soil. On this regional basis no specific statement can be made on local susceptibility to landsliding. However, some general statements

can be made. Although moderate to steep, natural slopes of the local formations (Marcellus, Manhatango, which is stratigraphically equivalent to the Hamilton, and Trimmers Rock) are stable, cut slopes generally have only poor to fair stability due to rapid disintegration of the shales upon exposure to weathering. Much of the surface area in the vicinity of the site is covered with glacial till and outwash. The stability of this material in cut slopes needs to be carefully analyzed Slope stability and landslide potential at the site are discussed in greater detail in Subsection 2.5.1.2.5.

2.5.1.2 Site Geology

2.5.1.2.1 Site Physiography

The Susquehanna Steam Electric Station is located in the Valley and Ridge Physiographic Province which is described in Subsection 2.5.1.1.1. The site is situated within a broad undulating valley developed in mudrock, shale and siltstone of the Devonian Mahantango Formation along the axis of the Berwick Anticlinorium (Figure 2.5-12).

Lee Mountain (about 2-1/2 miles north) and Nescopeck Mountain (about 4-1/2 miles south of the site) are held up by the more resistant sandstone and conglomerate of the Mississippian Pocono Group. Lesser ridges formed by sandstone of the Trimmers Rock Formation occur at the north end of the site and along the south bank of the Susquehanna River (about 2 miles south of the site).

Topography and drainage of the area is controlled to a large degree by the lithologic and structural characteristics of the bedrock. Ridges and valleys generally trend east-northeast parallel to the strike of the Paleozoic strata. The site fronts on the south flowing Susquehanna River which here flows perpendicular to the east-northeast trending axis of the fold, resuming its west-southwest flow about 1-1/2 miles south of the site, to follow the strike of the shale valley. The north-northwest segment of the Susquehanna River which flows normal to the strike appears to have been inherited from the course of the Ancient Little Schuylkill River (Ref. 2.5-3) (Refer to Subsection 2.5.1.2.4).

Topographic elevations in the site vicinity range from 500 to 1,100 feet above sea level. Higher elevations occur in the more rugged terrain further north and west of the site. The site itself contains generally gentle to moderately sloping hills and well developed drainage patterns. Existing surface elevations vary from about +750 feet in the western portion to about +500 feet in the east. Portions of the area were formerly cultivated.

In those areas not cultivated, heavy to moderate woodlands and scrub brush are found. A steep sandstone ridge borders the north side of the site. A narrow east-west trending interior bedrock ridge, rising some 60 feet above the surrounding ground surface, is located just north of the center of the site. A rounded bedrock knoll about 80 feet high occurs at the western edge of the site in the southwest quandrant.

The site is well-drained by eastward trending depressions near the north and south edges of the site. Plant grade at 650 feet above sea level (about 150 feet above the flood plain of the Susquehanna River) is at about the level of the Fourth Clean Kame Terrace (Ref. 2.5-5) which is well preserved southward between the site and the Susquehanna River.

The irregular bedrock surface underlying the site is the result of a combination of preglacial weathering and stream erosion, glacial scour, later erosion by glacial melt waters, and the varying resistance of the lithologic units to erosion. The maximum thickness of the overburden is on the order of 40 feet in the southern half of the site, with bedrock occasionally cropping out at the surface. North of the east-northeast bedrock ridge that is located near the center of the site just north of the reactor and turbine buildings, glacial deposits fill a bedrock valley to a depth exceeding 100 feet.

During excavation at the site, abundant evidence of glacial and glacio-fluvial scour of the bedrock surface was found in the form of channels, potholes, grooves, striations and fluted rock. A large, buried pothole over 30 feet wide and more than 30 feet deep was exposed in the Unit 1 turbine building excavation (Figure 2.5-13). Similar large buried potholes have been documented farther north along the Susquehanna River Valley (Ref. 2.5-3, p. 23-27 and 2.5-59, p. 195). At the site, numerous other smaller potholes and rounded pits and channels in unweathered bedrock were observed in the excavations. Smooth, east-northeast trending linear channels about 6 to 8 feet deep eroded in unweathered bedrock were observed north of the radwaste building. Similarly, somewhat larger features were excavated in the northeast and west rims of the Unit 1 cooling tower. This fluting of the rock surface observed in a number of places at the site was either gouged by ice, eroded by water or both, and apparently served as flumes for torrential glacial meltwater runoff which evidently at one time cascaded across much of the site area. Undoubtedly many steep or even undercut surfaces of the bedrock at the site are attributable to ice scour and intense fluvial erosion that was associated with the Olean and earlier glaciations. These features are discussed in Subsection 2.5.1.2.3.3.

As indicated in Subsection 2.5.1.2.5, landslide potential, surface or subsurface subsidence, uplift or collapse are not of concern at the site.

2.5.1.2.2 Site Lithology and Stratigraphy

2.5.1.2.2.1 Lithology and Stratigraphy in the Site Vicinity

Figure 2.5-12 illustrates the distribution of the geologic units within at least 5 miles of the site. The stratigraphic relationships of the various formations are shown on the site geologic column (Figure 2.5-14). Silurian and Devonian formations occur throughout the Valley and Ridge Province. Silurian and lower and middle Devonian strata consist of marine shale, mudrock, siltstone, sandstone and limestone. The upper Devonian strata are generally non-marine sandstone and shale.

A northeast trending fold, referred to as the Berwick Anticlinorium, completely encompasses the site area. This feature has been breached by erosion, exposing rocks of Silurian and Devonian age along the core and at the flanks of the anticlinorium. As it plunges to the east, progressively younger formations are exposed. Silurian formations present west of the site include, from oldest to youngest: the Tuscarora sandstone, the Clinton ferruginous sandstone, the McKenzie greenish shale with limestone, the Bloomsburg red shale, the Wills Creek shale and the Tonoloway limestone. The Tuscarora sandstone caps Montour Ridge along the axis of the Berwick Anticlinorium in the vicinity of the West Branch of the Susquehanna River. Here as elsewhere in the Valley and Ridge Province, the Tuscarora is a prominent ridge former. The Clinton Formation contains a fossil iron ore which was formerly mined along Montour Ridge. The Bloomsburg Formation supports the eastern extension of Montour Ridge. The Wills Creek and Tonoloway Formations occur in the flanks of the Anticlinorium west of Berwick (Figure 2.5-10).

The basal Devonian formations are the Keyser limestone and Old Port sandstone. These formations crop out along the flanks of the Berwick anticlinorium. East of Bloomsburg they are no longer exposed having presumably been removed from the section by erosion or faulting (Subsections 2.5.1.1.3 and 1.5.1.2.3).

Stratigraphic units exposed in the map area are from oldest to youngest: the Devonian Mahantango (which includes the Marcellus Shale, Trimmers Rock and Catskill Formations), the Mississippian Pocono Formation, the Mississippian to Pennsylvania Mauch Chunk Formation, and the Pennsylvania Pottsville and post-Pottsville Formations.

Above the Marcellus, the Mahantango Formation is represented only by the uppermost member, the Sherman Creek which is dominantly a dark gray to blue gray, olive gray to brown weathering mudrock. Siltstone and fine grained sandstone units crop out locally, including at the site and at certain outcrop locations both calcareous and noncalcareous strata coexist (Figure 2.5-12, Stations DF-2, DF-3, and DF-6).

An interval of light, medium gray argillaceous limestone near the top of the Mahantango, was recognized as correlative with the Tully Limestone and was mapped as part of the Mahantango (Figure 2.5-12, Stations DF-53 and DF-45). The overlying Harrel Shale, a poorly exposed, dark silty shale which appears to gradationally overlie the Mahantango (Figure 2.5-12, Stations DF-30, DF-31, DF-43, and DF-44b) was also mapped as part of the Mahantango Shale.

Fossils are relatively abundant within the Sherman Creek member of the Mahantango Formation and include various genera of brachiopods, bryozoa, pelecypods, coral, trilobites, and crinoid fragments. Fossil casts are abundant with occasional molds and rare preservation of internal structure and original shell material.

Concretions (commonly rusty weathering), spheroidal weathering, and prominent, closely spaced steeply dipping cleavage, which may quite easily be mistaken for primary bedding fissility, are other features characteristic of the Mahantango.

The Mahantango grades upward into the Trimmers Rock (Figure 2.5-12, Stations DF-30 and DF-33). The Trimmers Rock Formation is dominantly interbeded, medium to olive gray, thinly laminated siltstone, silty shale and fine grained, laminated to massive sandstone. These rocks weather to a brownish gray color. Sedimentary structures include fining-upward sequences (Figure 2.5-12, DF-7, DF-8, and JW-10); groove casts, current lineations, load casts, ball and pillow structure, and flow rolls (Figure 2.5-12, JW-7B, JW-10, and JW-11). Ripple marks were also locally identified. These structures indicate deposition by turbidity currents in a marine environment. Fossils are often restricted to relativley thin layers of brachiopods (spirifers DF-9). Other fossils include pelecypods and crinoid fragments (DF-17b). Channels were observed at DF-25.

The upper Trimmers Rock Formation consists of light to medium grayish green silty shale and micaceous, dark greenish gray silstone, both of which weather to a dark reddish brown color, a reddish brown silty fine to medium grained sandstone to siltstone, and olive green vitreous fine grained sandstone to siltstone (DF-26, DF-27, DF-28, DF-32, DF-36, DF-37 and DF-68).

The Catskill Formation rests conformably upon lithologically similar interlayered rocks of the upper Trimmers Rock but is

predominantly red colored. It is this dominantly red color as well as sedimentary structures (roots, oscillation ripple marks; Station DF-68) which distinquishes the Catskill from the Trimmers Rock. The contact between the Catskill and the underlying Trimmers Rock was mapped therefore, at the base of the first relatively thick reddish brown to maroon sandstone (Figure 2.5-12, Station DF-68) or brownish red silstone and more massive reddish brown (maroon) micaceous, fine grained sandstone (Figure 2.5-12, Station DF-37c). At Station JW-65 (Figure 2.5-12), the upper Trimmers Rock consists of fine grained, medium gray sandstone overlain by a thin band of green mudrock which is, in turn, overlain by a fine grained, green, well laminated sandstone. The green sandstone grades upward into a fine grained, red, well laminated sandstone which marks the basal unit of the Catskill.

The basal red unit at Station DF-37c (Figure 2.5-12) is overlain by greenish gray, fine grained sandstone and olive green shale and at DF-68 it is overlain by thinly laminated, light green, silty shale and siltstone. These units are succeeded, upward, by interbedded maroon and light olive gray to greenish gray sandstone, siltstone, and shale (Figure 2.5-12, Stations DF-37c and DF-68). Stratigraphically younger units within the Catskill include: (a) reddish brown mudrock and silty mudrock, (b) brownish red medium grained sandstone, (c) reddish brown to marcon micaceous, fine grained sandstone, and (d) greenish gray micaceous, fine to medium grained sandstone. Sedimentary structures include intraformational clasts of green shale, oscillation ripple marks, roots, and prominent cross bedding.

The Pocono Formation, which overlies the Catskill, consists typically of medium and coarse grained light gray to white, rusty weathering quartz sandstone with thin layers of quartz pebble conglomerate. Olive gray, fine grained sandstone, reddish gray medium to fine grained sandstone and siltstone (Figure 2.5-12, Station DF-14) and greenish gray medium grained, cross bedded sandstone (Figure 2.5-12, Station DF-67) also occur within this formation. Cross bedding is common.

Near the base of the Pocono, grayish red sandstone layers occur. These were recognized along the east side of the Susquehanna River south of Mocanaqua (Stations JW-1 and JW-64) and along the road between Alden and Folstown (Stations JW-28 and JW-29). At JW-1 and JW-64, the Pocono consists of an interlayered sequence of predominantly medium gray, thick, well laminated, gray weathering quartz sandstone and subordinate, red, flaggy quartz sandstone. At JW-28 well laminated red sandstone is interlayered with, but decidedly subordinate to, well laminated, rusty weathering, light gray, coarse grained sandstone and finer grained gray sandstone. Coarse to medium grained, mainly grayish to greenish gray sandstone featuring rather subtle cross bedding dominate the upper portion of the exposure at JW-29. This, along

with an underlying thin zone of light greenish gray sandstone, in turn, underlain by green shale and mudrock, has been selected as marking the basal Pocono. Beneath all of these units at JW-29, is a red, well laminated, argillaceous siltstone which we have interpreted as marking the top of the Catskill. This red siltstone rests upon a green shale which overlies a cross bedded, medium light gray, olive gray weathering quartz sandstone. Red shale, which marks the base of the outcrop, underlies the quartz sandstone. The lower (topographically and stratigraphically) portion of the outcrop is dominated by red lithologies in contrast to the upper part in which no red lithologies were exposed. Besides the obvious color change the sandstone, above the inferred contact, is coarser grained and more subtly cross bedded than the sandstone which occurs between the red units near and at the base of the outcrop. Thus, contrary to other interpretations (e.g. Ref. 2.5-16) we suggest that the Pocono-Catskill contact is gradational in this area, rather than unconformable.

The upper Pocono Formation in the vicinity of Shickshinny consists of medium to light gray conglomeratic sandstone with rounded to sub-rounded quartz pebbles and shale fragments, and rusty weathering, fine to medium grayish green micaceous siliceous sandstone (DF-56) and finely laminated greenish gray, rusty weathering, siliceous quartz sandstone (DF-57, DF-58). Rusty weathering, medium light gray, medium to coarse grained quartz sandstone is interbedded with thin layers of dark gray silty shale and medium gray quartz-lithic sandstone fills channels at DF-59.

The Mauch Chunk Formation is generally bright red in color and consists of mudrock, silty shale, siltstone and fine to medium grained, cross bedded, well laminated sandstone. The upper part of the formation along the south limb of the Lackawanna Synclinorium (Figure 2.5-12, Stations JW-22 to JW-24) is marked by interlayered red and olive gray sandstone, siltstone, and silty shale. Locally the siltstone contains layers of rounded, circular to elliptical calcite filled voids. Elsewhere the Mauch Chunk consists of greenish gray to grayish green medium to coarse grained, locally micaceous sandstone, thinly laminated gray, fine grained sandstone and siltstone (Figure 2.5-12, Station DF-54) and massive medium grained sandstone (Station DF-55).

The Pottsville and Llewellyn Formations represent the coal bearing zones of the Anthracite Region and have, for the purpose of this report, been combined and treated as a single formation. Collectively, the Pottsville and Llewellyn (formerly post-Pottsville) consist of quartz pebble conglomerate in a quartz sandstone matrix, quartz pebble conglomerate in a carbonaceous quartz sandstone matrix, coarse grained dark to medium gray, massive and flaggy, carbonaceous sandstone and shale, dark gray

to black siltstone and coal. The non-carbonaceous quartz pebble conglomerate displays cross beds.

Pleistocene unconsolidated deposits of glacial drift blanket most of the region north of the site. They extend approximately 10 miles to the south and 50 miles to the west of the site. Deposits of various glacial advances are recognized in the region. The drift materials include glacial till and stratified waterlain deposits consisting of poorly sorted mixtures of clay, silt, sand, gravel and boulders. The youngest and best preserved deposits are those of the Wisconsinan glacial stage.

The site lies just behind the Pleistocene terminal moraine of Olean drift, deposited between 55,000 and 60,000 years ago (Ref. 2.5-60, Figure 4; 2.5-61, plate 3; and 2.5-5, p. 25). The Olean drift represents an early glacial substage of late Pleistocene, or Wisconsinan time and has been correlated with the Altonian substage by Sevon (Ref. 2.5-62) to distinguish it from the later Wisconsinan, or Woodfordian, drift farther north. The Olean drift is an assemblage of contemporaneous drifts deposited by several ice lobes that occurred from New Jersey westward to Indiana, believed to represent a regional glacial advance in early Wisconsinan time. A correlation chart of deposits of early and middle Wisconsinan age by Dreimanis and Goldthwait (Ref. 2.5-60, Figure 4) utilizing available geomorphic, lithologic, paleontologic and radiocarbon data, shows the Olean drift to be between about 55,000 and 60,000 years old; conservatively, the Olean drift may therefore be considered to be in excess of 50,000 years old according to this correlation. Drifts recording later ice advances in Wisconsinan time are not present in northeastern Pennsylvania (Ref. 2.5-61, plate 4), so in this area evidence of the earlier Wisconsinan drift is preserved (Ref. 2.5-62 and 2.5-63).

The leading edge of the Olean terminal moraine is depicted by Denny and Lyford (Ref. 2.5-61, plate 4) as occurring in the Susquehanna Valley about three miles southwest of the site, just west of the village of Beach Haven. Ahead of (downstream from) this moraine are deposits left by an earlier Illinoian glaciation (Ref. 2.5-7, p. 24); however, no Illinoian deposits have been recognized north of the Wisconsinan terminal moraine in Pennsylvania (Ref. 2.5-5, p. 26), indicating that Olean ice overrode and reworked apparently all of the pre-existing Illinoian drift. Nevertheless, it is possible that some buried drift at the site and elsewhere, particularly that located in bedrock depressions, may represent unrecognized remnants of overridden Illinoian or earlier deposits.

The glacial deposits near the Susquehanna site have been studied in some detail by Peltier (Ref. 2.5-5). He describes (Ref. 2.5-5, p. 25) the various features and processes associated with the terminal moraine near Beach Haven. He characterizes the morainic

material as "a gravel moraine... composed largely of poorly sorted, coarse kame gravel, medium-grained valley train gravel, and sand... During the early stages of kame terrace development, the marginal channels flowed at a level which was high above the valley... Continued ablation of the ice in the valley probably caused the marginal streams to flow at successively lower levels. These streams, where they flowed along the ice, both eroded the earlier deposits and filled in their channels... In this manner any till deposited at the ice front became buried or eroded." This description of erosion and deposition near the site by ice-margin streams at elevations above the valley floor is consistent with the development of large potholes and steep or even undercut erosional contacts at the site. Evidently waterfalls and large-volume torrential streams occurred at the site during retreat of the early Wisconsinan ice. (Additional discussion of the origin and features of glacial deposits at the site is presented in Subsection 2.5.1.2.3.3).

Peltier (Ref. 2.5-5, Figure 33) profiles discontinuous kame terraces along a 25-mile stretch of the Susquehanna River including the site. The highest such terrace formed by a stream marginal to Olean ice is indicated to occur at about 650 ft. msl at the site (about mile 165), or about 160 ft. above the river. Glacial deposits at elevations higher than this, which would include the glacial deposits in most of the site area, would be part of either the Olean terminal moraine or the ground moraine behind it.

The moraine in the Berwick-Beach Haven area is noncalcareous (Ref. 2.5-5, p. 24). Sedimentary rocks, mostly gray and red sandstone and siltstone, constitute over four-fifths of the material in the moraine (Ref. 2.5-5, Table 3). Peltier (Ref. 2.5-5, p. 24) considers that the remaining igneous and metamorphic types in the moraine indicate it was derived from the Mohawk tongue of an Olean ice lobe originating from the Adirondack area or east of it (Ref. 2.5-60, p. 83).

Unconsolidated sediments mantle most of the Susquehanna River Valley within 5 miles of the site. The valley deposits consist of glaciofluvial deposits (outwash alluvial terraces, kame terraces), alluvium and colluvium. Unconsolidated deposits were examined at Stations DF-4, DF-15, DF-16, DF-37, DF-42, DF-44a, DF-44b, DF-52, DF-64, and DF-66 at all DJU stations. Thin deposits were noted at various other localities (Figure 2.5-12).

Station DJU-1 is located at a currently (Spring, 1977) operating gravel quarry exhibiting excellent exposures. This quarry contains well layered, brownish gray, very coarse sand to medium gravel interlayered with gray, medium, well sorted gravel with medium to coarse gravel and cobble layers. This is overlain by pebble to cobble gravel with coal and rare boulders interlayered with coarse sand to medium gravel with little coal. The dip of

bedding tends to decrease or flatten toward the south. The middle level contains medium to coarse, well rounded, gravel with coarse sand containing lenses of cobbles and gravel below fine to medium gravel and coarse sand with some coal rich laminae. The overlying unit is generally finer grained and dominantly fine to medium sand, some silt with fine laminae of coal. This unit contains layers of cobbly gravel, silt plus fine sand, and coarse to medium finely laminated gravel and coarse sand. Local coarse sand to medium gravel plus cobble layers have steeper dipping beds which appear to flatten southward. On the uppermost level, tan cobbly gravel with rare boulders under tan fine grained sand with coal exhibiting possible load casts is exposed above slumped material. This is overlain by medium yellow brown silt with little fine sand. This silt contains rare sub-angular cobbles. Feint layering is visible in the thickly bedded silt.

The deposit described above is the largest good exposure of unconsolidated sediments observed during this mapping program. Based on sedimentology (well sorted, rounded gravels in contact with well sorted sands or well sorted silts which appear to indicate rapidly changing hydraulic regimes; gravels with intersticial silt) sedimentary structure (steeply dipping bedding whose dip flattens southward or downstream) and geographical location (against the valley wall); these sediments are interpreted as a kame terrace deposit. No faults were observed cutting this layered sequence.

Kame terrace deposits were observed at the other DJU stations. Ice contact deformation was obsrved at several locations (refer to Subsection 2.5.1.2.3).

A yellow-brown silt with some fine sand, occasional to rare pebbles or cobbles was observed at several locations (DJU-1 at Elevation ∿665 feet; DJU-2 at Elevation ∿600 feet; DJU-3 at Elevation ∿ 580 feet; DJU-4 at Elevation∿ 595 feet; possibly at DJU-7 at Elevation∿ 640 feet overlain by cobbly gravel; and DF-15 at Elevation∿ 1040 feet). These deposits have been interpreted as loess (Ref. 2.5-5) but may represent relatively quiet fluvial conditions.

Well rounded cobbly gravels observed at DF-52 and DJU-5 may represent either valley train or kame terrace deposits.

2.5.1.2.2.2 Lithology and Stratigraphy at the Site

At the site, the thickness of the surficial materials occurring south of coordinate line N342,000, which includes all of the principal plant structures except the spray pond facilities, ranges from zero to about 40 feet. These materials consist of till and kame outwash, typically grading upward from a basal

gravelly boulder zone to a surface layer of silty fine sand and sandy silt. The surface layer may represent reworked loess. Rock fragments in the gravelly outwash are well rounded and are composed mainly of hard, well cemented, white to brown or red sandstones of various textures. No calcareous fragments were noted. In places, the sands and gravels contain minor amounts of anthracite grains and rounded anthracite pebbles up to a foot in diameter. These anthracite fragments cannot have been transported less than 3-1/2 miles from Shickshinny, the nearest occurrence of coal beds.

In the spray pond area in the northern part of the site, permeable, gravelly outwash and alluvial material fill an east-west bedrock valley to depths in excess of 100 feet. Cobble and boulder pockets were encountered at various depths in most of the boreholes drilled in this locality. The deposit is glacial in origin, possible in part pre-glacial and overridden by ice, and reworked by water derived from ablation of the ice mass in the manner described by Peltier. It consists of sequences of sand, gravel and boulders, overlain by sand and gravel, overlain in turn by sand and silty sand. A geologic map of the surficial materials excavated in the spray pond area is presented on Figure 2.5-15.

Bedrock at the site is in the upper part of the Middle Devonian Mahantango Formation, except for a strip along the northern margin of the site. The uppermost member of the formation, which forms the top of rock in the east-west bedrock valley north of about N342,000, is a dark gray, noncalcareous siltstone in which bedding is generally delineated by thin, inconsistent, light-gray, fine-grained sandstone stringers. Upward and with increasing sand content the Mahantango Formation grades into the Trimmers Rock Formation, which occurs north of about N342,500 at the northern edge of the site. The Trimmers Rock, a gray fine-grained sandstone which caps the high, northeast-trending ridge north of the site, is massive to flaggy and exhibits well-developed joint systems.

Beneath its uppermost member, the Mahantango is comprised of 120 to 150 feet of hard, dark gray calcareous siltstone. It is harder and more resistant to erosion than the uppermost member, forming the east-west trending bedrock ridge just north of the reactor location and underlying the site to past the southern limit of the site area. The principal plant structures are founded on it.

These two upper members of the Mahantango Formation are similar in lithology and occur at the same stratigraphic position as the Harrell shale and underlying Tully limestone. However, the characteristic fossils of the Tully are not present in the site area which require these members to be assigned to the Mahantango Formation.

As exposed in the foundations, the unweathered bedrock is a dark gray, massive to thick bedded slaty siltstone, homogeneous in appearance and lacking the bedding plane fissility that is normally associated with less well indurated shaly rocks. The rock also exhibits a variably developed slaty cleavage or fracture cleavage, further indication of its indurated nature. Typically the rock is slightly calcareous and has intermittent fossiliferous zones which display impressions of brachipods, crinoids, corals, bryozoa and trilobites. Scattered veinlets and joint fillings consist of white, crystalline calcite or a mixture of calcite and quartz.

The rock weathers to a brown color, with iron oxide stains on joint and cleavage surfaces. Weathering progresses initially by dissolution of calcite from joint and fracture fillings, followed by more pervasive weathering of the rock mass and refilling of joints and veinlets with clay and other weathered material. Advanced weathering on exposed, natural surfaces evidently proceeded mainly along cleavage planes, so that on well weathered outcrops the platy cleavage fabric dominates greatly over jointing or bedding. Lack of significant weathering of the rock surface is often associated with areas where there is evidence of considerable glacial scour or fluvial erosion of the rock. Additional information on the engineering characteristics of the bedrock at the site is given in Subsection 2.5.1.2.5.

2.5.1.2.3 Structural Geology

2.5.1.2.3.1 Major Geologic Structures in Site Vicinity

The major structural features in the vicinity of the site are the Berwick Anticlinorium and the Lackawanna and Eastern Middle synclinoria which are discussed in Subsection 2.5.1.1.3. Structurally the site is situated slightly north of the axis of the Berwick Anticlinorium. The term "anticlinorium" as used herein is defined as a series of minor, intermittent anticlinal structures so arranged that they form a general arch or anticline.

Virtually all structural elements in the site area are related to Paleozoic crustal compression. These elements include kink bands which occur on all scales (Ref. 2.5-31 and 2.5-32) and most likely account for the Berwick and Lackawanna folds, contraction (reverse and bedding-plane) faults, and small scale flexural slip folds. These structures occur on all scales (e.g., Ref. 2.5-30 and 2.5-31). Where exposed it can generally be inferred that the small scale kink bands, contraction faults, and flexural slip folds are cogenetic, developed early in the tectonic history and were rotated by later, larger scale, genetically related folds.

For example, at Station JW-3 (Figure 2.5-12) bedding strikes N70°-75°E and dips 70°-75°NW. A reverse fault strikes N80°E, dips 70°NNW, and displays slickensides which rake 85° in the direction S80°W. The axis of the associated drag fold plunges 15° in the direction N85°E. The enveloping bedding on a small kink fold at this same exposure is oriented N70°E: 70°NW and the kink band is oriented N68°E: 60°SE. Similarly, at Stations DF-34 and DF-55 the geometric relations among cleavage, faulting and folding strongly suggest these structures are all coeval.

Like the folds, contraction faults also occur at different scales. At least some of the larger faults appear to have developed in response to a space problem created by the development of tightly appressed folds. An example of this is noted at Station JW-30, where the strain energy associated with a tight kink fold was released along one fairly large reverse fault (which parallels bedding on the hanging wall and cross cuts bedding on the foot wall) and several smaller faults which strike parallel to the larger one yet dip in the opposite direction. At JW-30 bedding, the kink fold and the associated faults all show nearly parallel trends; slickensides on the faults and bedding surfaces deformed by the kink band rake approximately 90°. Similar observations have been made elsewhere in the Fold and Thrust Belt (Valley and Ridge Province) and, as described by Faill and Nickelsen (Ref. 2.5-31), all of these structures are kinematically congruent, i.e., cogenetic.

Besides the aforementioned structures, local evidence of lateral movement was recognized along the north-south segment of the Susquehanna River at Stations JW-3 and JW-60. At JW-3, slickensides rake 20° in the direction S05°E on a surface striking N05°W and dipping 70°W. At JW-60, slickensides rake 20° in the direction S10°W on a surface striking N10°E and dipping 50°E. This movement appears to be related to cross faulting in which case it, too, would be cogenetic with the other structures (Subsection 2.5.1.1.3). In any case lateral movement along this segment of the river was too small to produce any perceptible displacement on the map scale of 1:24,000.

As indicated in Subsection 2.5.1.1.3, all of these structural elements developed during the Late Paleozoic. No evidence was observed in outcrops within five miles of the site which would suggest that they have been active since that time.

Minor structural features were observed in Pleistocene sediment at a few locations in the site vicinity. Two small faults (exhibiting 18 inches and 2.5 inches of vertical separation) were observed at the margin of an apparent ice melt collapse feature in kame or kame terrace deposits at station DF-47. Asymmetric, reclined folds in unconsolidated sand and silt were observed at stations DJU-3, DJU-7, DF-7 and DF-47 (Figure 2.5-12). A small scale fault oriented N30-35E: 73-75SE with approximately 2 mm of

dip slip separation occurs at DJU-9. This apparent reverse fault dies out upward. A coal bearing sand which lies about 6 cm above the observed displacement is not disturbed. Similar features at the site have been related to syndepostional slump, differential compaction and ice contact phenomena (Subsection 2.5.1.2.3.3).

2.5.1.2.3.2 Geologic Structures at the Site

During preconstruction exploration at the site, geologic structures in the bedrock at the site were defined and evaluated. Since bedrock exposures at the site were scarce (see Figure 2.5-17), most of this information was obtained from borehole cores, supplemented by geophysical logging of boreholes, seismic refraction surveys, cross-hole and down-hole measurements, test pits and trenches, and geologic mapping of the surface. Presented herein is a summary discussion of the geologic structures at the site as defined from the preconstruction exploration, followed by a description and discussion of the geologic structures that were observed in the excavations for the principal plant facilities.

The principal structural features in bedrock beneath the site are shown on Figure 2.5-18. The axis of a minor anticline crosses the site generally along the east-west base line (approximately N341,700). To the north of this base line, the strata dip to the north at between 20 degrees and 35 degrees. South of the base line, dips are to the south at between 5 degrees and 15 degrees. The predominant strike of the strata is N75E.

The prominent joint directions are parallel and perpendicular to the strike of the strata. The major joints strike parallel to bedding. This joint set is nearly perpendicular and dips opposite in direction to the dip of the bedding. A more open but less frequent series of vertical joints strikes parallel to the direction of dip of the strata. High angle joints healed by secondary calcite and quartz mineralization are present in the vicinity of minor shear zones.

The most prevalent type of rock displacements occurring in the region generally are low angle thrust faults. It has been indicated (Ref. 2.5-21) that many low angle thrusts shear upward through competent rocks utilizing incompetent strata as glide zones. Small shear planes that step stratigraphically from one shale-siltstone layer to another by shearing across intervening sandstone or conglomerate strata have been reported exposed in numerous road cuts and strip pits (Ref. 2.5-21).

Based on interpretation of initial data obtained for the PSAR from 100 and 200 series borings, particularly those located near coordinate line E2,442,400 (location of Section A-B, Figure 2.5-

22), two areas of minor shearing were recognized at the site; namely, one in the vicinity of N341,200, slightly east of the reactor facilities and the other in the vicinity of N342,700. The evidence of shearing is manifested by the presence of slickensides, calcite-healed gash fractures, and breccia zones. The shears are of the low-angle type generally parallel to the bedding and are mechanically associated with the forces that acted to produce the folding of the strata.

The shear zone which occurs to the north of N342,600 is characterized by a series of bedding plane slips associated with breccia, slickensides, thin clay seams, and numerous fractures. The shear zone is contained within the less competent upper member of the Mahantango Formation and the lower portion of the Trimmers Rock Formation. The shear zone probably terminates at depth in the more competent calcareous member of the Mahantango.

No evidence of displacement was encountered in the main body of the more competent strata of the calcareous member of the Mahantango Formation between N341,950 and N342,600. The stresses that acted on these strata were taken up by the development of joints and fracture cleavage. Detailed inspection of the rock cores extracted from this area reveals microshear offsets along the cleavage planes. The net effect of this mechanism is to thicken the strata as revealed by the stacking and shortening of sandy stringers. The cleavage planes are generally healed by secondary lithification of the rock matrix.

The contact between the top of the Mahantango Formation and the base of the Trimmers Rock Formation was encountered in borings 117, 108, 122 and 126, all located north of N342,550. Detailed examination of bedding planes observed in the rock cores from these borings indicated that the dip of the strata increases with increasing depth. This is confirmed by borehole geophysical data. The numerous breccia, slickensides, thin clay seams and fractures encountered in borings 122 and 126, and to a lesser extent in boring 108, represent a zone of en echelon shear planes, both parallel and subparallel to the bedding. These shears are related to the original tectonic stresses which produced the regional folding.

A petrographic examination of the clay and rock encountered in some of these borings in the northern part of the site was conducted by Dr. Charles Thornton of Pennsylvania State University. The examination indicated that the rock and clay in the broken zones were mineralogically similar to the intact rock obtained from the core above and below the broken zones. Since no secondary mineralization was encountered in association with the clay and broken rock, it appears that this condition was mechanically induced and is not a result of chemical alteration and/or weathering.

In the second area of minor shearing identified above, evidence of structural adjustment which may be called a shear zone is present as slickensides and healed breccia at various depths in borings 125, 127, 132 and 103 as indicated on the subsurface section (Figure 2.5-21), and in borings 100, 217 and to a minor extent in 105 perpendicular to the section. These borings are in the area adjacent to and immediately east of the reactor facilities. Based on this evidence, this zone of structural adjustment strikes east-notheast and dips southerly at approximately 10 degrees. If this zone of structural adjustment extended northward beyond boring 102, it has been subsequently removed by erosion.

Detailed inspection of the microstructue in the rock core extracted from the borings at the site reveal shear-fold structural relationships similar to those encountered on a larger scale across the site. The displacements observed in the rock core are completely healed by secondary calcite and quartz mineralization.

It is probable that the bedrock at the site served as an intervening buffer or adjustment zone during the regional folding of the strata.

Stresses that formed the Berwick Anticlinorium and synclinal structures appear to have been absorbed within the rocks of Mahantango and underlying Marcellus formations, as flexural slip, disharmonic folding and glide thrusting. The stresses that were necessary to produce these structural features were compressional from the southeast. These structural features were formed no later than the close of the Paleozoic Era, approximately 200 million years ago.

Based on thorough consideration of all the information provided by the pre-construction foundation exploration, it was concluded that the minor structural conditions observed at the site are not of significance with respect to siting or design for the use of the site for its intended purpose. An evaluation of subsequent data assembled from additional boring exploration and from geologic mapping of the foundations, confirms the initial conclusion.

During excavation and clean-up of the rock at Unit 1 reactor and turbine foundations, at the circulating water pumphouse, and along the trench for the hot water intake pipeline to Unit 1 cooling tower, a bedding plane shear showing strong slickensides was uncovered. This bedding plane shear is the same shear plane that was identified in the early phases of the site exploration and is herein referred to as "bedding plane shear A" (refer to Figures 2.5-18 and 2.5-19).

In the northeast corner of the Unit 1 reactor foundation, bedding plane shear "A" strikes N85°E and dips 7°SE. The surfaces of the bedding plane contain 1/4-inch to 3/4-inch thick laminae of calcite, siltstone and some quartz. The calcite laminae are approximately 1/16-inch thick, alternating with thinner siltstone laminae. The entire exposed area of this bedding plane contains prominent slickensides trending N30° to 40°W, with a 6° to 7°SE plunge. Up-dip and closer to top of rock, the bedding plane contains a 1/2 to 1-inch wide, iron-stained zone, and it also shows extensive leaching of the minerals filling the shear. In places, the adjacent rock is weathered to a granular sandy soil. The calcite which fills the bedding plane shows no sign of crushing. The weathering and staining on the bedding plane shear occurs only near top of rock where surface water and groundwater could penetrate along the plane; at foundation grade which is well below the weathered zone, the unweathered laminae have the properties of firm rock. In places the bedding plane shear is apparently not a prominent feature in the unweathered rock. For example, it was identified only as a slickenside surface with associated jointing in boring 105 and as horizontal jointing planes in boring 351 (geologic section E-E', on Figure 2.5-19).

A second essentially parallel bedding plane shear striking N75°E and dipping 7°SE was exposed in the trench for the circulation pipe, at the interesection of column lines 19 and G. Slickensides trending N30°W with a 7°SE plunge are also exposed on this plane. The surface is coated with a 1/8 to 1/4-inch-thick layer of unweathered calcite. This shear plane is designated "bedding plane shear B" on Figure 2.5-18. Although similar in appearance to bedding plane shear A at this location, apparently this shear is more restricted in areal extent, because it was not recorded on the logs of nearby bore holes nor was it mapped in the radwaste foundation area where it should have been exposed if it had continued that far north.

It proved possible to collect intact samples from the sheared portion of bedding plane shear "A" for more detailed analysis, including petrographic thin sectioning. The mineralization along the bedding plane consists of thin, parallel bands of intergrown calcite and quartz. The bands, 0.5 to 5.0 mm wide, are separated by thin films of dark shaly material on which slickensided striations caused by shearing have formed. Within the bands, the majority of quartz grains shows recrystallization into interlocking, strain-free grains up to 5 mm long, but becoming cryptocrystalline in the thinner bands. These relationships suggest that the quartz-calcite mineralization was not a late, post-tectonic occurrence, but rather was probably introduced in association with shearing, which is known to have taken place at the end of the Paleozoic (refer to discussion at the end of this Subsection 2.5.1.2.3.2). Undeformed microscopic veinlets of calcite can be observed to cut across the bands at nearly right angles. These veinlets are not themselves offset, and therefore

constitute mineralization that has not been crushed or deformed since its deposition. Similar instances of undeformed calcite veinlets crossing slickensided bedding planes are observed on a megascopic scale in the site excavation. Figures 2.5-20a through 2.5-20q illustrate such occurrences.

Bedding plane shear "A" was mapped in the excavations westward from the northeast corner of the Unit 1 reactor foundation to the west slope of the circulating water pumphouse excavation (Figure 2.5-18). It was also exposed in the trench for the Unit 1 cooling tower hot water intake piping and in two pedestal (No. 6 and No. 7) excavations for the tower itself. Although it displays minor undulations, the average strike of the bedding plane shear is close to N85°E eastward from the turbine and reactor foundations, approximately parallel to the axis of the minor anticline at the site and to the regional structural trend. Near the Unit 1 cooling tower, the bedding plane shear strikes about N70°E, consistent with measured bedding attitudes in that area. Representative dip measurements on the shear plane in the foundations were between 5° and 8°S, which is parallel to the dip of bedding. The trend of the slickenside lineation on this bedding plane shear across the foundation area ranges between S10°E and S40°E, most between S20°E and S30°E, a direction consistent with regional north-northwest compression during folding.

Drill hole data were utilized to project bedding plane shear "A" down-dip. Geologic sections E-E' and F-F' on Figure 2.5-19 show profiles of the shear through the reactor and turbine foundations. The source of data for these profiles is from foundation geologic mapping and elevation surveys, supplemented by subsurface data from the boring logs. It is evident that the foundation mapping and boring log data are in very good agreement, and that the minor shear zone originally identified in this area from exploratory borings is identical to the bedding plane shear "A" identified during construction (see Figure 2.5-21, which was prepared before excavation for the plant structures began). Although this figure suggests that bedding plane shear "A" may not be completely parallel to bedding, no evidence was found during later exploration and excavation to indicate that the shear plane transects bedding.

Bedding plane shear "A" can be traced updip along the Unit 1 hot water intake pipeline trench to the excavation for Unit 1 cooling tower pedestals 6 and 7, where the shear plane crosses the axis of the minor anticline that trends through the site. At pedestal 6 which was excavated to Elevation 667 feet, the weathered bedding plane shear was exposed and dips gently south, conformable to bedding (Figure 2.5-18). At the adjacent pedestal 7 which was excavated to elevation 668 feet, the same weathered bedding plane shear was again exposed, but here it dips gently north, again conformable to bedding. At these locations the

weathered shear is two to three inches thick. Where unweathered, the shear is tightly healed with calcite and quartz mineralization; where weathered, these minerals have been partially removed and replaced with claylike material. A roller-bit probe made during the Unit 1 cooling tower foundation exploration recorded a thin seam of soft rock in the vicinity of pedestals 8 and 9 at about elevation 662 feet, which was probably a penetration of bedding plane shear "A", and, together with measured bedding attitudes, reveals a continuation of the northward dip of the shear plane. West and south of the circulating water pumphouse, undulations in the bedding are evidenced by local northward dips of 5 to 10 degrees. Elevations at which shears were intersected by boreholes 318, 321 and B-5 suggest that bedding plane shear "A" closely parallels the undulations of the strata in this area. These structural relationships are shown in profile in geologic section G-G' on Figure 2.5-19. The fact that the shear plane is folded in conformance to local structure demonstrates that the shear plane originated before or during the time of folding and effectively dates its formation at 200 million years ago or earlier, which is the minimum age of Appalachian deformation in the region (Refer to Subsection 2.5.1.1.3 and the discussion at the end of this Subsection 2.5.1.2.3.2).

Other slickensides were recorded on many joint planes at the site, particularly on low-angle joint planes. Most of these slickensides plunge southeast. The geologic map (Figure 2.5-18) shows these measurements. Numerous slickensided joint planes had been recorded in bedrock cores in the early stages of the site exploration see boring logs, holes 100-132 and 210-219, Figures 2.5-23a through 2.5-23t); they were also observed in rock removed during foundation excavation. Many of these low-angle slickensided joint planes are calcite-coated, and some are undulatory in form rather than planar. They were noted in some instances to splay out from the more prominent bedding-plane shears described above. Evidently, differential movement which occurred principally along bedding planes was transmitted laterally to the encompassing bedrock mass along these bifurcating slickensided joints or shear planes. Such slickensides and shears should be expected in view of the tectonic history and the nature of deformation which the region has undergone.

Significantly, regardless of the orientation of the planes on which slickensides occur (whether they dip north or south), the trend of the slickenside lineation is almost invariably in the northwest-southeast quadrant, clustering N20-35°W (or S20-35°E). This direction is completely consistent with the northwesterly-directed tectonic compressive stress that produced the regional folding and thrust faulting during the Appalachian orogeny, and is further evidence that the slickensides that occur at the site are geologically old; that is, they originated over 200 million

years ago. Their consistent orientation suggests deformation during a single tectonic episode, rather than recurrent deformation at different times in geologic history.

Bedding plane shear "A" intersects the top of bedrock surface in the diesel generator and Unit 1 turbine and reactor area. During excavation, two exposures of this intersection were examined to determine the nature of this contact (exposures at intersections of grid line N341,400 with column line G and with column line N (Figure 2.5-18), and photographs (Figures 2.5-20b through 2.5-20e) were taken. Glacial deposits overlay the rock at these points. In each case the eroded rock surface was continuous across the trace of the bedding plane without displacement or offset. If displacement had occurred subsequent to erosion of the rock surface, this would be apparent as an angular, sharp projection of rock into the overlying glacial deposits; instead, the rock surface across the trace of the bedding plane is smoothed by erosion. Figures 2.5-20b through 2.5-20d show this relationship. In the area north of this intersection, the bedding plane shear had been eroded away, thus confirming the original evaluation based on exploratory borings (compare geologic section E-E', Figure 2.5-19 with geologic section B-C, Figure 2.5-21). The erosion of the rock surface would necessarily have occurred prior to the deposition of the overlying glacial deposits, which have been established as being more than 50,000 years old (refer to Subsection 2.5.1.2.2.1). Consequently, this relationship shows that any displacement along bedding plane shear "A" occurred more than 50,000 years ago. Actually, regional relationships plus the fact the plane is folded indicate that any displacements are a result of the tectonic forces which occurred prior to the late Triassic, over 200 million years ago.

Thus, the original preconstruction appraisal of shears which occur at the site as reported in the PSAR remains the same. These minor shears and structural conditions are consistent with the mode of deformation which occurred during the Appalachian orogeny, over 200 million years ago. They are not significant to the plant site or to the operation of the plant.

Cleavage. Secondary cleavage is variably as developed in the rock exposed at the site; in some places, such in the slopes of the ESSW pipe trench north of the circulation water pumphouse and in parts of the cooling tower areas, it forms the dominant structural feature of the rock, both on fresh and on weathered exposures. The strike of the cleavage is oriented east-northeast, approximately parallel to the trend of the major fold axes, and dips with variable steepness to the south, but generally in the range of 40-80°. Where the dip of the cleavage locally becomes fairly shallow, such as along the eastern perimeter of the south cooling tower, it is sometimes difficult to distinguish cleavage planes from bedding planes. The fact

that the cleavage is oblique to bedding demonstrates its secondary origin, apparently during the episode of regional tectonic deformation, 200 million or more years ago.

Joints and Fractures. Jointing in the rock excavated for foundations is fairly well developed. Figure 2.5-18 maps the principal joints encountered at foundation grade, which is at a sufficient depth below top of rock to be in essentially unweathered material. Here joints are tight and either uncoated or coated with calcite or a mixture of quartz and calcite. Relatively few joints at foundation level contained significant iron staining; some iron-stained joints are mapped in the radwaste foundation area. Toward the surface these joints generally become more heavily iron-stained with greater degree of weathering, and calcite coatings tend to be leached out, resulting in open joints, in joints partly coated with quartz or in clay-filled joints in the zone of weathering.

The most abundant joint set in the principal foundations area (Figure 2.5-18) strikes east-northeast (N60°-85°E), roughly parallel to the major regional fold axes and to the secondary fold axis at the site. North of the anticlinal axis at N341,300, these joints strike N70-85°E and dip, with some skatter about the vertical, 75°S-75°N, most 85°S85°N. South of N341,100 similar but more numerous joints, shown diagramatically on Figure 2.5-18, strike N50°-60°E, dip uniformly 50°-60°SE, and appear to comprise a distinguishable set. Less numerous but quite prominent joints with a similar east to east-northeast trend dip gently northward at 10°-18° and are best represented along the vicinity of coordinate line N341,200.

Other dominant joint sets are steeply dipping to vertical north-northwest to northwest joints, and north-south joints. Dips in both sets are usually greater than 70° with both east and west dips represented although the majority of those measured dip toward the west.

Many joints are filled with white calcite or a mixture of calcite and quartz, but there appears to be no preferential orientation for these filled joints. The low-angle joints are commonly slickensided (discussed above). In the turbine building excavation, two vertical, calcite-filled joints cut across bedding plane "B". The calcite in these vertical joints is continuous across the bedding plane with no offset, showing that the joints were formed and the calcite was deposited in the joints subsequent to the development of the slickensides on the bedding plane. Photographs were taken of this exposure (Figures 2.5-20e through 2.5-20f).

In addition to these principal joints, high-angle, discontinuous, white calcite and quartz-calcite veinlets are typically exposed locally throughout principal plant foundations. These veinlets

probably represent fractures that originated during Late Paleozoic tectonic deformation. They tend to occur most abundantly in the vicinity of bedding plane shears (discussed below) and as such may have arisen as gash fractures, as for example the veinlets mapped in the vicinity of N341,350-E2, 441,550 (Figure 2.5-18). At this same location is a singular occurrence of numerous west-dipping open vugs and seams up to several inches wide containing undeformed, euhedral quartz crystals up to 2 inches long. These seams were here exposed several feet above a bedding plane shear (see description above). Chunks of loose, coarsely crystalline white calcite also occur in the vugs. It is evident that these vugs had originally been relatively wide (up to 5 inches) gash fractures containing a coarsely crystalline quartz-calcite mineral filling; later the rock weathered and the calcite was selectively dissolved by circulating ground water (Refer to Subsection 2.5.1.2.5.1).

Bedrock Configuration at the Site. Figure 2.5-17, a map showing top of rock contours at the site, illustrates the general original configuration of the bedrock surface. It is evident that the major erosional feature of this surface is a buried, east-west bedrock valley in the northern part of the site, including the spray pond location. Here glacial or pre-glacial erosion has incised the bedrock surface approximately 100 feet below the general top of rock elevations to the south. In detail, the bedrock surface is very irregular due to the action of glacial plucking and subsequent glacio-fluvial erosion. The large pothole over 30 feet deep and 30 feet wide was found in the Unit 1 turbine building excavation; other smaller ones also occur at the site. Additional discussion of erosional features in bedrock at the site is presented in Subsections 2.5.1.2.1 and 2.5.1.2.3.3.

Relation of Site Geologic Structure to Regional Structure. Geologic mapping at the foundation excavations for the plant structures, together with subsurface borehole data, shows that bedding plane shear "A", the only shear plane traceable across a significant part of the foundations area is, within the accuracy of the data, parallel to bedding and follows the folds which the bedding defines, indicating that the bedding plane shear was either formed prior to folding, or, more likely, developed in conjunction with folding (refer to geologic section G-G' on Figure 2.5-19). Therefore, knowledge of the age of folding would provide a minimum date of origin of the bedding plane shears exposed at the site. With that objective in mind, the literature was examined first to determine whether or not the structures of the site are consistent with the regional structure, and second to date as accurately as possible the age of deformation.

The attitude of the sheared bedding planes and the trend of the slickensides on the planes may be compared to the nearest major tectonic structure (The Berwick Anticlinorium) to the site that

is an obvious and consistent member of the pattern of regional deformation in the Valley and Ridge province. The strike of the sheared bedding planes (N75°-85°E) are essentially parallel to the axis of the Berwick Anticlinorium (N75°-80°E) immediately south with compressive forces from the southeast which caused the folding in the region, and of the Berwick Anticlinorium in particular. The Berwick Anticlinorium is one of a series of folds in the Pennsylvania Valley and Ridge Province. It is located in the northwestern part of the province near the Allegheny plateau. Rocks involved in this deformation within the Valley and Ridge province range as recent as Permian in age, and the intensity of deformation increases toward the southeast -- from broad, gentle open folding at the Allegheny front to overturned, recumbent folds and nappes complicated by thrust faulting at the Blue Ridge.

Arndt and Wood (Ref. 2.5-64) have classified this progressive deformation resulting from compressive stresses originating to the southeast into a number of stages, each stage being categorized by effects of successively more intense deformation. Thus, the effects of deformation were transmitted with time northwestward over an increasingly greater distance, and deformation acted at any one locality with increasingly greater intensity with time. It follows that "the areas of most complex structures to the southeast underwent each of the first four stages of deformation, whereas the least intensively deformed area to the northwest was subjected only to the last orogenic force and contains features characteristic of only the first stage of deformation" (Ref. 2.5-64).

The first stage of deformation is characterized by horizontal strata cast into broad, open folds without significant thrust faulting. The second stage, which characterizes the area in which the Berwick Anticlinorium is located, exhibits low-angle thrusting and imbricate faulting followed by formation of subsidiary folds on the larger folds to develop anticlinoria and synclinoria. Structures in the vicinity of the site are consistent with this categorization. Subsidiary flexures at the site are broad, open features (refer to geologic section G-G', Figure 2.5-19), and low-angle thrust faulting is represented by the decollement in the site vicinity as discussed in Subsection 2.5.1.1.3. Subsequent stages, in which the folds are overturned and then additionally folded and faulted, are absent from the Berwick anticline area. Arndt and Wood (Ref. 2.5-64, p. B134) state, "the process of structural evolution appears to have been continuous and the result of a single orogeny that was not necessarily punctuated by pulsations.... The orogeny began after rocks of Pennsylvanian age were consolidated and prior to deposition of rocks of Late Triassic age." It is obvious from this model of deformation that thrust faulting was a logical and integral accompaniment to folding, rather than being part of some separate tectonic episode subsequent to folding.

In this process of deformation, "rocks of the more competent units characteristically folded into generally concentric, symmetric to asymmetric anticlines and synclines broken variably by faults. The rocks of the less competent units developed disharmonic folds broken by decollements, low angle thrust and bedding faults, and commonly separate discordant folds in the more competent rocks" (Ref. 2.5-21, p. 160). As a result, it is expected that rocks least able to withstand great shear pressures, such as shales, would display evidence not only of large magnitude differential movement as is found near the major thrust zones, but also of lesser but more prevalent minor structural adjustments, such as shears, incipient bedding plane faults, zones of closely spaced joints or fractures, slickensides, slaty cleavage, and so on. Thus, it would be surprising if the Mahantango Formation which occurs at the site did not show at least some of these features produced during Appalachian deformation.

"Northwestward-directed stresses of the late Paleozoic Appalachian orogeny were largely responsible for the development of the tectonic framework of the Anthracite region and the remainder of the Valley and Ridge province in Pennsylvania" (Ref. 2.5-21). There is general agreement that the time of the Valley and Ridge deformation, which is equated with the "Appalachian Revolution" (Ref. 2.5-28, p. 645), also termed the "Alleghany" or "Allegheny orogeny" (Ref. 2.5-65), ended before late Triassic time over 200 million years ago, but there is surprisingly little evidence to indicate a more exact dating of the events. As Rodgers (Ref. 2.5-34, p. 34) states, "traditionally...the deformation has been dated at the end of the Paleozoic, and in fact for generations American students were taught that it was the event that marked the end of the era." The youngest known deformed strata are lower Permian in age (in the Georges Creek syncline just west of the province boundary in Maryland) (Ref. 2.5-34, p. 64). Therefore, typical Valley and Ridge folding and faulting occurred in the Permian and perhaps continued into the early Triassic; apparently it formed most if not all the major structural features of the province (Ref. 2.5-34, p. 64).

According to Woodward (Ref. 2.5-65, p. 2320), "there is no tangible evidence regarding the time of this deformation save that part of it must have occurred after the Pennsylvanian (or after the early Permian) and all of it before the late Triassic...Nothing fixes its appearance specifically at the end of the Permian; even its latest movements could have ceased by Middle Permian. They could also have continued through the Middle Triassic for any evidence to the contrary." The Upper Triassic shale and red bed deposits in their tilted and downfaulted basins provide an upper age limit for the Allegheny orogeny (Ref. 2.5-34, p. 115) because it is thought that the pervasive northwest-southeast compressive force field required for the northwest-directed thrust faulting and folding during the

Allegheny orogeny could not have been present during the formation of the Upper Triassic basins, which required essentially extensional or tensional stress acting in the east-west or northwest-southeast direction.

In several places undeformed Triassic features are directly superimposed on Valley and Ridge structures, establishing an upper limit for Valley and Ridge deformation, of which the Berwick anticline is a part. Between the Schuylkill and Susquehanna Rivers, Triassic basin sediments rest directly on and truncate the recumbent folds and nappes (Ref. 2.5-66) developed in the southeast part of the Valley and Ridge province. These upper Triassic sediments were deposited on a peneplained surface; thus, the Valley and Ridge structures had become inactive and were exposed and eroded to near base level before upper Triassic time over 200 million years ago. Late Triassic diabase dikes are shown on the Tectonic Map of the United States (Ref. 2.5-67) crossing Appalachian fold structures about 20 miles northwest of Harrisburg near the mouth of the Juniata River. Since these dikes are neither deformed nor offset by Valley and Ridge faults, they also establish a pre-late Triassic age for Valley and Ridge tectonism.

According to Dr. Gordon H. Wood of the U. S. Geological Survey (verbal communication, 1974) there are no local specific field relationships in the Anthracite basin which could be used to supply a definite date for faulting and folding in the Anthracite basin. The only known date for Appalachian structures is supplied by regional relationships such as the Triassic events. However, Dr. Wood stated that all faulting related to Appalachian structures, except possibly for some very minor Triassic faulting, is Paleozoic in age.

On the basis of the foregoing discussion, it is concluded that thrust faulting, shearing, bedding plane faults and other similar features in the area near the site, and the slickensides and striations in the foundation rock underlying the site, were formed during the "Allegheny orogeny" or "Appalachian Revolution" which produced the folds and thrusts of the Valley and Ridge province, of which the Berwick Anticline is a part; thus, these events became tectonically inactive before upper Triassic time or over 200 million years ago. The slickensides and shearing which are evident on various bedding planes and joint planes in the foundation rock at the Susquehanna Site are therefore of no significance to the plant structures.

2.5.1.2.3.3 Geologic Features in Surficial Materials at the Site

Surficial material in the site vicinity consists of glacial drift deposited near the Olean terminal moraine (refer to Subsection 2.5.1.2.2.1). The glacial deposits near the Susquehanna site have been studied in some detail by Peltier (Ref. 2.5-5, p. 25). He describes the various features and processes associated with the terminal moraine near Beach Haven (3 miles southwest of the site) as follows:

> (The moraine) is a gravel moraine and is composed largely of poorly sorted, coarse kame gravel, medium-grained valley train gravel, and sand. These gravels were deposited in marginal channels between a stagnant tongue of ice, which lay in the center of the valley, and the valley walls... During the early stages of kame terrace development, the marginal channels flowed at a level which was high above the valley, and, at the front of the ice, fell sharply to the valley floor. At the ice front a steep alluvial deposit, composed largely of coarse gravel, was formed. Continued ablation of the ice in the valley probably caused the marginal streams to flow at successively lower levels. These streams, where they flowed along the ice, both eroded the earlier deposits and filled in their channels; where they crossed the "terminal moraine" they cut channels in the previously deposited alluvium and laid down sand and gravel on more gently sloping gradients toward the river valley beyond it. In this manner any till deposited at the ice front became buried or eroded.

At the site, little till was exposed in the excavations for the principal plant structures, in conformance with Peltier's nearby observations. Essentially all of the glacial material excavated consist of stratified drift in the form of kame delta and terrace deposits, alluvial outwash and stream gravels, much of it probably reworked in the manner described by Peltier. Indeed, the scoured and fluted bedrock surface, large potholes, and steep and even undercut contacts between bedrock and glacial drift attest to the torrential flow of water which at one time evidently cascaded across the site; and the coarse boulder gravels and erosion channels within the outwash indicate energetic reworking of the materials. In keeping with this glacio-fluvial mode of deposition, contemporaneous sedimentary features, such as those resulting from slumps at undercut or eversteepened stream banks, from differential compaction of materials deposited on irregular surfaces, and from other adjustments during deposition, may be expected (see for example (Ref. 2.5-68, p. 184-185). A few minor features such as sedimentary creep or small slumps were observed in the stratified drift, as for example north of the radwaste building, where they

are associated with an undulating, fluted rock surface. Apparently these features, which terminate above the rock surface, arose through differential compaction across the irregular bedrock surface.

None of the sedimentary features exposed in the glacial materials were observed to extend downward to intersect the bedrock surface. It is concluded that all such features observed in the surficial materials at the site are consistent with their known mode of origin by glacio-fluvial process that occurred near the terminal moraine of the Olean glaciation.

2.5.1.2.4 Site Geologic History

The geologic history of this region can be traced from Precambrian times. Rocks beneath the Paleozoic strata at the site form the Grenvillian cratonic basement, approximately 1 billion years old. The sediments that were deposited to form the Precambrian rocks in the region were subjected to magmatic intrusion, metamorphism and erosion before the onset of Cambrian time.

Paleozoic sedimentary strata in the site vicinity are estimated to be on the order of 30,000 feet thick (Ref. 2.5-28 and 2.5-38). The deposition and deformation of these strata is related to the opening and closing of the Proto-Atlantic Ocean (Ref. 2.5-69). Although deformation in the Appalachian Orogen culminated three times in the Paleozoic -- the Taconic, the Acadian, and the Alleghenian (Appalachian) orogenies -- the effects of the first two orogenies in the folded Appalachians in which the site occurs were mainly sedimentologic rather than structural, being evidenced as unconformities and as changes in provinence, lithology and in sedimentation characteristics, in contrast to the intense folding, faulting, volcanism and metamorphism which occurred at these times on the Mobile Belt during the Taconic and Acadian events. The Alleghenian orogeny, on the other hand, resulted in the structural configuration at the site today. The structural evolution of the Fold and Thrust Belt is described in Subsection 2.5.1.1.3.

Crustal divergence in Late Precambrian, Cambrian and Early Ordovician time allowed the accumulation of a thick sequence of miogeosynclinal sediment in the Appalachian Basin (Subsection 2.5.1.1.2). The Taconic Orogeny beginning in Middle Ordovician time signifies convergence and uplift in the Mobile Belt.

The highly deformed early Paleozoic strata are unconformably overlain by less deformed, coarser grained clastic sediment which is in turn overlain by the Siluro-Devonian carbonate sequence. This sequence is thickest in the east and thins westward.

Northeastward from the site the carbonate strata interfinger with clastic detritus. Late Paleozoic strata are clastic through most of the Appalachian Basin (Subsection 2.5.1.1.2).

These strata reflect the closing of the Proto-Atlantic Ocean. At the peak of the Taconic Orogeny along the cratonic margin to the east, ophiolitic rocks (presumably oceanic crust) were cbducted from the eugeosyncline, and the miogeosynclinal strata (carbonate and detrital alike) were thrust onto the craton. The geologic setting at the site is the result of this activity. Folding and thrust faulting occurred through mechanical detachment from rigid basement rocks along decollements in shaly strata near the base of the Paleozoic section (Ref. 2.5-38). The site rests on the northern limb of one such fold, the Berwick Anticlinorium. The deformation progressively increased in intensity toward the southeast, from broad, gentle open folding northwest of the Allegheny front to overturned, recumbent folds and nappes complicated by thrust faulting at the Blue Ridge. The effects of final convergence and translation during the Late Paleozoic appear to be limited to the Mobile Belt (Subsection 2.5.1.1.3 and 2.5.2.2).

The Appalachians appear to have undergone erosion through most of the Mesozoic Era. Tectonic activity related to the opening of the Atlantic Ocean appears to have had no significant structural effect in the Fold and Thrust Belt and Stable Interior (Subsections 2.5.1.1.3 and 2.5.2.2) until the Cretaceous Period. At that time, subsidence of the Atlantic continental margin allowed transgression of the sea well inland of the site vicinity.

During Cenozoic uplift, major drainage in the area followed relatively straight southeastward courses through the Cretaceous sedimentary strata to the Atlantic. The Ancient Little Schuylkill River flowed past the site toward the present day Delaware Bay. The ancient north branch of the Susquehanna River flowed through Wilkes-Barre, Pennsylvania toward Trenton, New Jersey. As the Appalachians were exhumed, the east-northeast structural fabric began to exhibit control of the drainage pattern. The present course of the north branch of the Susquehanna River resulted from stream capture of the Ancient Little Schuylkill and ancient north branch through their east-northeast tributaries by the main branch of the Susquehanna River (Ref. 2.5-3).

Northeastern Pennsylvania has undergone at least three glaciations during the last 150,000 years and possibly one or more prior to that date. Till at the site was deposited during the Olean substage about 55,000 to 60,000 years ago (Ref. 2.5-5 and 2.5-6). Older Illinoian drift occurs in the valley of the Susquehanna River between the Olean terminal morain at Beach Haven (about 3 miles southwest of the site) and the confluence of

the north and west branches of the Susquehanna River. Post-Olean advances did not reach the site vicinity (Ref. 2.5-5 and 2.5-6).

Peltier (Ref. 2.5-5) mapped discontinuous kame terraces along the Susquehanna River in the site vicinity. The highest such terrace formed by ice marginal streams occurs at about 650 feet above sea level at the site. Refer to Subsections 2.5.1.2.2 and 2.5.1.2.3.3 for further discussion of Pleistocene erosion and deposition at the site.

Since the retreat of the Wisconsinan ice sheets from the region, broad regional uplift appears to have occurred, probably at least in part as a result of crustal rebound subsequent to the removal of ice load. Erosion has continued and soil profiles have formed.

2.5.1.2.5 Engineering Geology Evaluation

Site subsurface exploration is described and discussed in Subsection 2.5.4.3. Laboratory tests of foundation materials, and in situ geophysical tests of the foundation materials are discussed in Subsections 2.5.4.2 and 2.5.5. Geologic mapping of the final foundations is described in Subsections 2.5.1.2.2, 2.5.1.2.3 and 2.5.4.1.3. It was concluded from these studies and evaluations that the site geologic and foundation conditions are entirely suitable for the construction and operation of the plant.

2.5.1.2.5.1 Geologic Conditions Under Category 1 Structures

All Seismic Category 1 plant facilities, except the spray pond and the Engineered Safeguard Service Water (ESSW) pumphouse and pipeline, are founded on bedrock. The ESSW pipeline trench is excavated partly in soil and partly in rock. The location of these facilities is shown on Figure 2.5-24.

The foundation rock is a hard, indurated siltstone, a member of the Devonian Mahantango Formation. In the foundations area it is quite massive and lithologically homogeneous, with bedding generally not well defined, and lacking the bedding plane fissility usually associated with less well indurated shaly siltstones and silty shales. In places the rock exhibits a slaty cleavage, further evidence of its indurated nature. All Category 1 rock foundations were excavated to unweathered bedrock. Geologic maps and sections of the Category 1 excavations in rock are shown in Figures 2.5-18 and 2.5-19. More detailed discussion of the foundation geologic conditions is contained in Subsections

2.5.1.2.2 and 2.5.1.2.3. Engineering properties of the foundation rock are described in Subsection 2.5.4.

The spray pond is situated over a glacial or preglacial, east-west trending bedrock valley as outlined by contours on top of bedrock (Figure 2.5-17). The valley is filled with dense gravelly and sandy glacial outwash and till deposits which attain a maximum thickness of about 110 feet adjacent to the spray pond area. They were deposited no later than the Olean substage (early Wisconsinan) of the Wisconsinan glaciation which occurred over 50,000 years ago. In general, the deposits are permeable and consist of a sequence of sand, gravel, and boulders overlain by sand and gravel, overlain in turn by silty sand. The entire sequence is highly variable in grain size distribution and sorting, and contains discontinuous pockets of similar materials. As a rule, grain size decreases and sorting increases toward the top of the sequence.

The southwestern tip of the spray pond is cut into bedrock while the remainder was excavated in these permeable glacial materials. The thickness of the glacial deposits beneath the bottom of the spray pond ranges from zero at the rock contact to 93 feet at the eastern end of the pond. The spray pond is lined to minimize seepage losses to the underlying permeable glacial deposits. The foundation of the pumphouse structure located at the southeastern corner of the pond is underlain by 35 to 60 feet of glacial material. The ESSW circulation pipelines between the pumphouse and the plant intersect bedrock at an elevation of 668 feet, approximately 260 feet southeast of the pumphouse (refer to Figure 2.5-17A). A geologic map of the spray pond area is presented on Figure 2.5-15. 2

Further discussion of conditions at the ESSW pumphouse and spray pond are contained in Subsections 2.5.1.2.2, 2.5.3 and 2.5.5. 3

2.5.1.2.5.2 Landslide Potential

Natural slopes adjacent or close to the principal plant structures are relatively flat. Most of these slopes are composed of soil; few rock slopes occur (Figure 2.5-17 shows areas of rock outcrops).

North of the spray pond the Trimmers Rock Formation forms a relatively steep ridge rising approximately 380 ft. above the pond. The south-facing slope of this ridge is essentially a rock slope underlain by flaggy, resistant sandstone thinly mantled with soil and rock fragments. The closest approach of this slope to the spray pond is along the northern perimeter of the pond; the toe of the slope, at elevation 710-720 feet, is 250 feet or more from the edge of the pond (at elevation 679 feet). The maximum slope along the ridge is about 2 horizontal to 1 vertical, with an overall slope of 3-1/2 horizontal to 1 vertical, a relatively flat slope for competent rock. Figure 2.5-56 shows a typical profile along the steepest portion of this 2

2 slope north of the spray pond area. Bedding in the rock dips approximately 30° to the north into the slope; thus, it is favorably oriented for slope stability. Data of McGlade, et al. (Ref. 2.5-56, p. 108) indicate that natural slopes eroded on Trimmers Rock strata are "steep and stable".

No old landslides, rock slips, or landslide scars have been noted near the plant structures. It is concluded that the natural slopes present no significant hazards to plant structures.

Stability of excavated and fill slopes is discussed in Subsection 2.5.5.

2.5.1.2.5.3 Areas of Potential Subsidence, Uplift, or Collapse

Potential sources of subsidence or collapse in the Pennsylvania Valley and Ridge region include coal mines and karst terrain; however, neither of these conditions are known to occur within several miles of the site and therefore they are not significant to the site.

No coal beds are present beneath the site; the nearest coal measures are about 3-1/2 miles north of the site near Shickshinny, which is at the extreme western end of the anthracite producing belt in the Lackawanna syncline (Figures 2.5-25 and 2.5-26). The nearest underground coal workings are about 2 miles east of Shickshinny (Ref. 2.5-70); the nearest that have been associated with surface settlement are near Nanticoke, Pennsylvania, approximately 10 miles northeast of the site. Rocks in the site area have no known potential for oil or gas production. The nearest oil or gas field is located 25 miles northeast of the site.

The shallowest carbonate rock that may be present beneath the site occurs below the Marcellus-Manhantango sequence as limy beds within the Upper Silurian, Tonoloway and Wills Creek Formations and the Lower Devonian, Keyser, Old Port, and Onondaga Formations (refer to Subsection 2.5.1.1.2.3 and Figure 2.5-14). Some of these units crop out on the flank of the Berwick anticlinorium north of Bloomsburg about 10 miles west-southwest of the site, but most are absent nearer than this to the site having been removed by erosion or faulting (Subsection 2.5.1.1.2.3). None have been mapped within five miles of the site (Figure 2.5-12). The Onondaga may occur in the subsurface near Berwick, five miles west-southwest of the site, inasmuch as mud-filled caves have been encountered during well drilling operations at Berwick; however, since the secondary porosity along joints and bedding

planes in the Onondaga has been characterized as of only "medium" magnitude (Ref. 2.5-56), such cavities would be expected to be neither large in size nor extensively developed. If the Onondaga does extend eastward beneath the site, it would be at a depth probably exceeding 1,000 feet and beneath the Marcellus-Mahantango shale and siltstone sequence (Figure 2.5-14). At this depth of burial beneath the site, carbonate beds possibly present would have no significant potential for subsidence or collapse at ground surface.

The site is not known to be in an area experiencing localized doming or subsidence. Relative rates of regional uplift or subsidence are not well defined for this area, but some recent studies have been presented in the literature. Brown and Oliver (Ref. 2.5-54, Figures 5 and 7) show a releveling profile across the Appalachians at the latitude of Harrisburg about 60 miles south of the site. This profile suggests that the Valley and Ridge province at the latitude of Harrisburg is rising uniformly at the rate of about 5 mm per year. They find in general that "the Appalachian Highlands are being uplifted with respect to the Atlantic Coast at rates up to 6 mm per year" (Ref. 2.5-54, p. 31). Because the measurements are referenced to a given station, it is not possible to determine absolute vertical crustal velocities. Since Brown and Oliver (Ref. 2.5-54. p. 31) also state the "Atlantic Coastal Plain... is tilting away from the continental interior" the data may indicate simply that the Appalachian Highlands are nearly stationary, or that they are subsiding more slowly than the coastal region. Inasmuch as differential rates of this magnitude are greater than the geologic record suggests could be sustained over geologic time, the authors presume an oscillatory mode of continental interior uplift or coastal depression with time on the order of 10^6 years per oscillation.

Superimposed on these broad, regional differentials are smaller, secondary variations in the rate of vertical movement within the Appalachian Highland on the order of 1 to 3 mm per year total amplitude (Ref. 2.5-54, Figures 7 and 8). The location of some of these secondary maxima or minima appear to correlate spatially with seismicity; others do not. The wave length between such secondary maxima is on the order of 300 km, a distance suggestive of origin in "large scale movements of the earth's crust" (Ref. 2.5-54, p. 27). Although the authors speculate there may be, in some areas, a possible association of these secondary peaks in the vertical velocity profiles with seismicity, they acknowledge (Ref. 2.5-54, p. 30) that "without further data it is impossible to demonstrate that the relationship is more than coincidental." In any event, flexure of a very few millimeters over hundreds of kilometers is very broad indeed and is not significant to structures.

In eastern New York, post-glacial offsets in shales and slates are documented by Oliver, et al. (Ref. 2.5-51). The nearest locality listed by Oliver, et al. (Ref. 2.5-51) is near Hyde Park, New York, about 120 miles east of the site. The authors wonder whether the cause of these offsets "might stem from crustal rebound following removal of the ice load or from still more broadly based tectonic or orogenic forces" (Ref. 2.5-51, p. 586). However, it is significant that these high-angle reverse offsets of the glacial striations in bedrock are on the order of only one inch or less of displacement. The authors also mention other possible mechanisms such as "...thermal changes, hydration, or a chemical process in the shales," and conclude the data "do not seem adequate to resolve this point" (Ref. 2.5-51, p. 569). Another assessment of the same data concluded the offsets arise from either frost heave or glacial rebound. If related in some way to frost action or the severity of frost, then one might expect the effects of heave, such as on precise leveling monuments, to be more pronounced with altitude. Precisely such a correlation of secondary velocity maxima with elevation was noted and discussed by Brown and Oliver (Ref. 2.5-54, p. 27).

Additional independent evidence on the magnitude of general Appalachian uplift, or lack thereof, in the Pliocene and Quaternary is provided by Owens (Ref. 2.5-52), who based his assessment on the assumption that uplift in the source area is accompanied by erosional transport of clastic material to adjacent basins. He found that Pliocene and Quaternary sediments of the Atlantic coastal plain from New Jersey southward are only 50 feet or less in thickness, indicating no great intensity of uplift through this period; and moreover that there are no marked accumulations of sediment in centers of deposition, suggesting a general, uniform uplift, or even static conditions, of the entire Appalachians.

Therefore, the total geologic record strongly suggests that unusual regional crustal instability has not recently occurred in the Appalachians. Brown and Oliver (Ref. 2.5-54, p. 31) conclude, "Although the rates of relative vertical movements determined from leveling seem large by comparison with rates deduced from some forms of geological evidence...these velocities do not seem unreasonable in terms of other types of geological information." Further, "the rates of vertical crustal movement presented...compare very favorably with those found in other portions of the world". Relative rates of vertical uplift observed for the site region therefore appear to be quite typical compared to observations elsewhere, and accordingly do not seem to be reflective of abnormal conditions.

The balance of evidence favors Appalachian crustal activity restricted to generally uniform uplift, probably differing slightly in local areas, and perhaps having an oscillatory character with time. Little if any evidence has been presented

to demonstrate that these may be significant to engineering planning or seismic risk evaluation. No faulting has been shown to be involved in this recent activity, and correlation with seismicity is likewise not established. In areas adjacent to the Appalachians, small-scale post-glacial offsets have not been correlated with seismicity and tectonic origin of the offsets has not been established.

It is concluded that the available data do not indicate that existing or future uplift or subsidence, as from man's activities or from geologic conditions such as regional warping, will be significant to the site.

2.5.1.2.5.4 Behavior of Site During Prior Earthquakes

There is no evidence at the site of any effects, such as landslides, fissuring or subsidence, which could be attributed to the occurrence of prior earthquakes.

No Pennsylvania earthquakes have been reported as felt in the site vicinity. Within historical times, approximately fourteen earthquakes originating outside Pennsylvania could have been felt at the site, with the probable maximum intensity of IV on the Modified Mercalli Scale. The nearest of these earthquakes occurred 90 to 100 miles from the site (Wilmington, Delaware, 1871, epicentral Intensity VII). Ground motion at this intensity (IV) would have had no effect on the site.

2.5.1.2.5.5 Zones of Deformation or Structural Weakness

As reported in the PSAR, the preconstruction investigation defined a number of structural features at the site such as folds, joints, fractures, cleavage, slickensided joint planes, and bedding plane shears. The PSAR stated (p. 2.5-16),

> The prominent joint directions are parallel and perpendicular to the strike of the strata. The major joints...(strike) parallel to bedding. This joint set dips nearly perpendicular and opposite in direction to the dip of the bedding. A more open but less frequent series of vertical joints strikes parallel to the dip of the strata. High angle joints healed by secondary calcite and quartz mineralization are present in the vicinity of minor shear zones. The observed fractures, while relatively numerous in the upper 20 feet of rock, decrease with depth. At a depth of about 20 feet into rock, the fractures are tight (generally healed with calcite filling) and would not adversely affect foundation performance.

Minor bedding plane slips at depth have been observed in the site area, both north and south of the interior ridge. Those slips have not experienced movements in more than 200 million years. A minor slip of this nature could be exposed in any large excavation anywhere in the area; however, it would not affect the structural design of the facilities.

Excavation during construction confirmed the PSAR evaluation and supplied additional data. During excavation, numerous slickensided joint planes were exposed and mapped (Figure 2.5-18). Thin, slickensided bedding plane shears , well healed with laminar quartz and calcite mineralization, were also exposed in the foundations. The field data indicate these shear planes are folded in the same manner as the bedding (Figure 2.5-19). They are, moreover, cut by vertical, calcite-filled joints which are continuous across the shears with no offset. In addition, where the shears can be traced to an intersection with a smooth, glacially eroded surface forming the top of rock, the eroded surface displays no displacement or offset across the shear plane. Since the erosion of the rock surface would necessarily have occurred prior to the deposition of the overlying glacial deposits, which have been established as being more than 50,000 years old (refer to Subsection 2.5.1.2.2.1), this relationship shows that any displacement along the shearing occurred more than 50,000 years ago. In reality, the most probable age of the shearing is pre-Triassic, or over 200 million years ago. This is indicated by regional relationships plus the fact that the shear plane is folded (A detailed presentation and analysis of the relationship between site and regional structure is presented at the end of Subsection 2.5.1.2.3.2).

All features arising from tectonic deformation at the site are geologically old. In the foundation rock, all shears and joints are tight or are fully healed with calcite and quartz mineralization; accordingly, they do not adversely affect the strength, bearing capacity or compressibility of the foundation rock. They are therefore not significant to the plant structures. The conclusion stated in the PSAR (P. 2.5-18) that "the minor structural conditions observed at the site are not of significance with respect to siting or design for the use of the site for its intended purpose," has been confirmed.

2.5.1.2.5.6 Zones of Alteration or Irregular Weathering

Bedrock beneath the seismic Category I structures is a dark gray, indurated, massive siltstone. It is not susceptible to rapid weathering; no appreciable slaking of fresh rock exposures was observed in the foundations. Depth of weathering below original top of rock varies from zero to about 20 feet. The rocks occupying depressions in the bedrock surface are generally

unweathered. However, open fractures were encountered to a depth of about 20 feet. Below that depth, fractures are not common but where they do occur, they are generally "healed" with calcite filling.

Weathering appears to have progressed initially downward by dissolution of calcite from calcite-coated joints, seams and shear planes, and refilling by clay or other weathered material. In one exceptional case, weathering along joints and fractures, as distinguished from weathering of the rock itself, was observed nearly 70 ft. below original top of rock. In this instance, which was between the circulating water pumphouse and the Unit 1 cooling tower, the bedrock originally formed a knoll and contained numerous low angle, slickensided, calcite and quartz-filled joint planes and abundant vertical, calcite-filled fractures, forming an intersecting, permeable network of channels through which water readily percolated downward, dissolving the soluble carbonate joint and fracture fillings. Notwithstanding the depth of weathering here, the design elevation of the bottom of the circulating water pumphouse is below the depth to which this zone is weathered, and this structure was founded on sound unweathered bedrock.

2.5.1.2.5.7 Potential for Unstable or Hazardous Rock or Soil Conditions

Foundation rock at the site is hard, indurated, unweathered siltstone, a member of the Middle Devonian Mahantango Formation. Similar materials underlie the site to a depth of over 1,000 feet. This rock does not contain unstable minerals; it provides highly stable foundation conditions and does not constitute a source of potential hazard to the plant.

Soils at the site are, except for the uppermost few feet, glacial in origin and consist of resistant fragments of rock that were transported from the region north of the site. Most were deposited by flowing melt water from the glaciers under torrential flow conditions, and some of the soils probably have been overridden by ice sheets. These glacial soils are noncalcareous and over four-fifths of the rock in them consists of sandstone (Ref. 2.5-5).

The origin and mineralogy of the soils is such that they present no hazardous conditions. Detailed engineering characteristics of soils in regard to slope stability, bearing capacity, stability under dynamic loads and consolidation characteristics under structural loads are discussed in Subsections 2.5.4 and 2.5.5.

All foundations for the Category 1 plant structures that are founded on rock are excavated to or into essentially unweathered material. No significant irregular alteration or weathering, or zones of structural weakness due to weathering, shearing or

fracturing were encountered at the bearing elevation for those structures designed to rest on unweathered rock.

2.5.1.2.5.8 Unrelieved Residual Stress on Bedrock

No indications were found during excavation and construction at the site of the presence of significant stress in bedrock. No popping or spalling rock was observed. There were no indications of heave at the base of rock slopes or in the bottom of excavations. Some vertical joints close to and subparallel to vertical excavated slopes opened very slightly, but this was attributed to gravity forces, not in situ stresses. If significant in situ stresses did in fact exist in the rock, evidence of this should have been noted in the excavation; no such indications were noted. For example, a thin mudmat was routinely placed on the rock after evacuation to foundation grade. If significant heave had occurred, it would have been readily detected by cracking of the mudmat. No such cracking was observed.

2.5.1.2.5.9 Conclusions and Summary

Consideration of all engineering geologic factors at the Susquehanna site leads to the conclusion that the site is well suited for the construction and operation of the plant. There is no geologic feature or condition at the site or in the surrounding area which precludes the use of the site for a nuclear generating facility. The bedrock in the construction area is competent and provides satisfactory foundation support for all major structures.

2.5.1.2.6 Site Groundwater Conditions

The groundwater table beneath the site generally occurs within 35 feet of the ground surface. The notable exception is in the deep, easterly sloping, buried bedrock valley present about 1000 feet north of the center of the plant, where a water table depth of as much as 79 feet was recorded. Generally, the lower part of the overburden deposits is saturated, although over portions of the upland area on the site, the groundwater level is found only in the underlying bedrock. Depth to bedrock is variable and ranges from zero to over 100 feet. The groundwater level contours shown in Figure 2.4-31 appear to be controlled to a large extent by the top of bedrock contours (Figure 2.5-17).

Groundwater movement on the site is generally in an easterly direction. With the exception of a few springs on site. Most of the groundwater is believed to discharge ultimately to the Susquehanna River. The average groundwater velocity is estimated to be between 1.5 and 2.0 feet per day along flow paths from the station to the Susquehanna River.

The site is not located in a recharge zone for any aquifer. However, groundwater recharge to the uncolsolidated sand and gravel dces occur over the site area. The predominantly siltstone strata of the Mahantango Formation beneath the site constitutes a source of limited domestic water supply. Because of its relatively low transmissibility characteristics, the Mahantango cannot be considered to be an aquifer.

From a hydrologic standpoint, there are two general types of aquifers in the region. The first type consists of sandstone and occasional limestone strata which occur within the predominantly shale sections of the Paleozoic age bedrock. The second type is unconsolidated Quaternary deposits which consist for the most part of Pleistocene stratified drift, till, or kames which are often overlain by a thin recent soil cover. From a survey of domestic water supplies within two miles of the station, it was found that nearly all of the wells are completed in shale bedrock.

Detailed information on groundwater conditions and movement on the site and in the region is given in Subsection 2.4.13.

2.5.2 VIBRATORY GROUND MOTION

A discussion and evaluation of the seismotectonic characteristics of the Susquehanna SES site and the surrounding region is presented in this section. The purpose of this investigation is to present the seismic design criteria for major structures at the station in conformance with guidelines as outlined in Standard Format and Content of the Safety Analysis Reports for Nuclear Power Plants, and Appendix A of 10 CFR, Part 100.

A description and results of the field investigation and laboratory testing program, which provided background information for this investigation, is presented in detail in Subsection 2.5.1.

2.5.2.1 Seismicity

The station is situated in a region which has experienced only a minor amount of moderate earthquake activity in historic time. The record of earthquake occurrences in the region dates back to the middle 16th century. Many earthquakes have been reported since that time and minor structural damage has been associated with several of the events; however, none of these earthquakes were considered to be of major or catastrophic proportion. Because this region has been fairly heavily populated since the early 18th century, it is quite certain that any significant earthquake activity (MM Intensity VII or greater as defined by the Modified Mercalli Intensity Scale, 1931, see Table 2.5-1) would have been reported in local newspapers, private journals or diaries. The lack of any such documentation is indicative of the absence of significant major earthquake activity in the region during this period.

Structural damage is the primary rating criterion for larger shocks. The effects of earthquakes on the rather large variety of construction materials used in older structures such as chimneys, rock walls, etc., are highly variable, making intensity evaluations based on such reports imprecise. The rather long period of record, however, and the evenly distributed population provide a reasonable basis for estimates of future activity.

Table 2.5-2 lists all events within 200 miles of the station with magnitudes (Richter) greater than 3.0 or MM intensities greater than III, and all seismic events within 50 miles of the station. Figure 2.5-8 displays these events on the regional structure of the area around the station, along with the significant earthquakes (MM Intensity V and greater) which have occurred outside the 200-mile radius.

The largest events to have occurred in the immediate station vicinity were the Wilkes-Barre disturbances of February 21 and 23, 1954 (local dates), with assigned intensities of VII and VI, respectively. These intensities were based on the damage inflicted upon a very small area, about 0.15 square miles of the city. These disturbances occurred only 16 miles from the station; however, they are considered to have resulted from mine collapse as discussed in detail in Appendices 2.5A and 2.5B. Thus, these shocks need not be considered in the analysis of earthquake risk as regards the station.

The largest event to have occurred within 200 miles of the station was the Intensity VII-VIII shock at Attica, New York, on August 12, 1929, some 150 miles northwest of the station. This earthquake imposed some moderate damage at Attica and villages in the immediate vicinity of the epicenter, but was not reported as felt in the Berwick area (Ref. 2.5-71).

Four Intensity VII earthquakes have occurred at a distance of about 100 miles from the station: two (1737 and 1884) were near New York City, one (1927) near Asbury Park, New Jersey and one (1871) near Wilmington, Delaware.

The closest of these Intensity VII events was the shock which occurred in the vicinity of Wilmington, Delaware on October 9, 1871, approximately 100 miles to the south of the station. Based on damage reports and intensities felt, the epicenter has been located near Wilmington, whereas the shock was felt from near Chester, Pennsylvania on the north to Middletown, Delaware, on the south and from Salem, New Jersey on the east to Oxford, Pennsylvania on the west. The initial shock was followed by a much smaller shock just after midnight on October 10. A contemporary newspaper account indicates that the initial shock was felt at Wilmington "with greater distinctness." Buildings were shaken severely and a number of chimneys were damaged in the surrounding towns of Oxford, Pennsylvania, and New Castle and Newport, Delaware. An interesting aspect of this earthquake is the fact that it was accompanied by a very loud sound, as of an explosion. This loud noise, in fact, led to the belief that the shock was caused by an explosion, probably at the powder mill of the E.I. DuPont deNemours Company, near Wilmington. This possibility was carefully investigated at the time and it was concluded that the shock was a legitimate earthquake. Existing reports, however, do not report the shock being felt in the Berwick area.

The two events near New York City, about 118 miles from Berwick, may have been felt at the station, but with an intensity certainly no greater than III. The 1884 shock affected an area extending from Portsmouth, N.H., to Burlington, Vt., southwest to Binghamton, N.Y., Williamsport, Pa., southeast to Baltimore, Md., and Atlantic City, N.J. At Bradford, Pennsylvania, reports were made of panes of glass broken in a large hotel, and other moderate damage was sustained. All hotels in Brooklyn, New York were shaken violently. In 1927, a similar shock listed as Intensity VII was centered near Asbury Park, New Jersey. Several successive waves, described as seeming to travel from west to east, caused homes near Asbury Park to shake and oscillate perceptibly.

Several Intensity VI earthquakes have occurred less than 100 miles from the station. On May 31, 1908, 48 miles from the site, Allentown, Pennsylvania was shaken by what was believed to be a mild earthquake. The shock lasted about a second and was described as a rumbling sound followed quickly by a report which "sounded like the falling of chimneys of a building" (The Lehigh Register June 3, 1908). The Philadelphia Inquirer adds, "The only other place where the shock was felt was Catasauqua, three miles away. At first it was thought that a battery of boilers in some local industry might have exploded, but no such accident was

reported. The quake was accompanied by a low rumbling noise and lasted about two seconds but was only felt over an area of some 50 square miles. On January 7, 1954, an Intensity VI shock occurred approximately 54 miles from the station. This was the first of a series of minor shocks in Berks County. The Reading Times reported on January 8, 1954, that a "minor" earthquake shook Berks County communities, and succeeding tremors of lesser intensity were experienced mainly by residents in West Reading, Wyomissing, West Louwe, West Wyomissing, Wyomissing Hills and Sinking Spring. Continued mild aftershocks shook the downtown area of Sinking Spring which appeared to be the epicentral area for these disturbances. The damage was described (Ref. 2.5-72) as minor: broken chimneys, breaks in brick and plaster walls, and broken dishware. Similar damage, including broken windows, was reported in other communities west of the Schuylkill River. The main shock was recorded by seismograph stations at Fordham, Palisades (Columbia University) and the U.S. Coast & Geodetic Survey at Washington, D.C. Dr. Jack Oliver of Columbia University described the initial tremor as "a typical east coast earthquake."

According to the Philadelphia Inquirer, as of January 8, 1954, the earth tremors which had been recorded in Berks County since 1900 were:

- August 30, 1902
- June 6, 1915
- February 28, 1925
- November 1, 1935
- June 8, 1937
- February 18, 1938
- September 4, 1944

The events of 1902, 1915 and 1938 are not reported in the standard cataloques and are, therefore, considered as very small, localized disturbances for which there is only local record. The shocks of 1925 and 1944 were large events in the St. Lawrence River near Quebec, Canada and Massena, New York which were felt with intensities of less than III at the station.

Shocks on January 24, 1954 and on August 11, 1954 affected Sinking Hole, Pennsylvania, according to the Reading Times (August 11, 1954). These shocks were attributed by the U.S.G.S. to the caving of solution cavities. Similar conclusions about the "sinking of the earth in general" were reached by Penn State University scientists after the event on September 24, 1954 which was assigned an Intensity II at Sinking Spring Borough, 5 miles west of Reading, Pennsylvania.

Another shock of Intensity VI occurred 63 miles southeast of the station, near Cornwall, Pennsylvania, on May 12, 1964. Coffman and Von Hake (Ref. 2.5-72) report a cracked wall, fallen plaster,

and small landslides. The Lebanon Daily News reported: "...the tremor was so mild that many persons slept right through it." However, Dr. B.F. Howell, Jr., Chairman of the Geophysics Department at Penn State University, reported that the quake was the most intensive to hit the state in 10 years.

On September 1, 1895, an event of Intensity VI near Philadelphia, 76 miles from the station, was felt from Sandy Hook, New Jersey to Brooklyn, New York to Darby, Pennsylvania, and Wilmington, Delaware. Another shock of Intensity VI occurred on March 23, 1957 in the same general vicinity, 79 miles southeast of the station. These shocks were not reported felt in the Berwick area.

Five shocks (Intensity VI, III, V, III, and IV) occurred in central and southern New Jersey on August 23, 1938, 116 to 128 miles from the station. The largest shock was felt from northern New Jersey to Wilmington, Delaware.

Although it is indicated that several of the large, distant shocks listed above were probably felt at the station, no damaging effects were experienced. No Pennsylvania earthquakes have been reported as felt in the area of the Susquehanna SES.

In summary, there are no reports from the Berwick area of Pennsylvania which would indicate that ground motions from any historical earthquake in the east have exceeded (or even equalled) an intensity as great as IV on the competent rock on which the station is located.

2.5.2.2 Geologic Structures and Tectonic Activity

2.5.2.2.1 Tectonic Provinces

The area within a 200 mile radius of the Susquehanna SES includes parts of six tectonic provinces (Figure 2.5-8). The provinces are, from west to east, Stable Interior, Fold and Thrust Belt, Blue Ridge-Highlands, Conestoga Valley, Inner Piedmont, and Coastal Plain.

The tectonic province concept used to define these provinces is based on an evolutionary model of the Appalachian orogen (Ref. 2.5-73) and derived from early studies in the region (Ref. 2.5-74). This concept was used in this study to provide the province boundaries of significance to the station.

2.5.2.2.2 Tectonic Differentiation of the Appalachian Orogen

The outline and discussion presented below summarizes the relationships and derivations of tectonic provinces of the Appalachian orogen, as displayed in Figure 2.5-8 any parts of which occur within 200 miles of the station.

1. Craton

    a. Eastern Belt

        (1) Blue Ridge-Highlands

    b. Western Basin

        (1) Stable Interior
        (2) Fold and Thrust Belt

2. Mobile Belt

    a. Eastern Cratonic Margin

        (1) Conestoga Valley
        (2) Inner Piedmont
        (3) Coastal Plain

Considering the tectonic evolution of the Appalachian orogen, it is subdivided as above into two fundamental areas; the part affected only by convergent diastrophism (craton), and the part affected by initial divergent, convergent, translational, and final divergent diastrophisms (mobile belt). The mobile belt, as defined in this report, is situated east of the great anticlinoria cored by Grenvillian rocks; i.e., east of the Long Range (Nova Scotia), the Green Mountains, the Berkshire Highlands, the Hudson-New Jersey Highlands-Reading Prong, and the Blue Ridge Mountains. The mobile belt thus corresponds to the Appalachian eugeosyncline and includes the quasi-cratonic margins. The western edge of the mobile belt parallels and lies to the west of what was originally the eastern edge of the North American continent during Cambro-Ordovician time as defined by Rodgers (Ref. 2.5-74).

2.5.2.2.3 Tectonic Differentiation of the Craton

The cratonic portion of the Appalachian Highlands is underlain by continental crust composed of 1000 million-year-old crystalline rocks which were deformed during the Grenvillian orogenic cycle. On the eastern edge of the craton, these rocks crop out at the surface as great anticlinoria. West of these elevated

anticlinoria lies an elongated, downwarped segment of the continental crust forming the asymmetric Appalachian basin. The floor of this basin is formed of Grenvillian rocks greatly depressed in the east (up to 40,000 feet below sea level) and gradually rising toward the west. The basin is filled with largely unmetamosphosed sedimentary rocks (both clastic and carbonate) ranging in age from Early Cambrian to Carboniferous. These rocks form a sedimentary wedge, thickening to the southeast, reflecting the asymmetry of the basin floor.

Blue Ridge - Highlands Tectonic Province

The eastern portion, termed here the Blue Ridge - Highlands, constitutes a tectonic province and is characterized by Grenvillian rocks deformed during the Paleozoic convergent stage.

Characteristically, the terrain is mountainous and exhibits exposure of some of the oldest rocks in the eastern U.S. (1000-1100 million years). Earthquakes no greater than Intensity VI are characteristic of this tectonic regime, and none have been related to specific structures.

Stable Interior Tectonic Province

The Stable Interior Tectonic Province of the western basin is characterized by the absence of intense deformation and the presence of shelf-delta type Paleozoic sediments. The rocks display gentle folding as opposed to the intensely folded and faulted rocks of the Fold and Thrust Belt immediately to the southeast. The largest significant earthquake to have occurred in this province (within the regional scope of this study) was the 1929 Attica, New York, event (initially cataloged as Intensity VII-VIII) approximately 168 miles from the station. This shock and an accompanying concentration of lesser events has been spatially related to the Clarendon-Linden Fault, an anomalous structure in the essentially untectonized rocks making up this portion of the Stable Interior. A small concentration of activity, apparently related to doming of the Adirondak massif, occurs 150 to 200 miles northeast of the station. With the exception of these moderately active areas, the province is virtually aseismic.

Fold and Thrust Belt Tectonic Province

The Fold and Thrust Belt tectonic province is characterized by tightly folded and thrust-faulted Paleozoic sedimentary strata deposited as flysch or molasse. The northwestern boundary of this province generally marks a transition between gently folded strata on the northwest (Stable Interior) and intensely folded and faulted strata on the southeast, thus marking the western limit of Paleozoic thrusting (Ref. 2.5-75).

The largest earthquake which has been recorded in the Fold and Thrust Belt tectonic provinces was the Giles County, Virginia, Intensity VIII shock of 1897, over 350 miles from the station, and in the southernly division of the Fold and Thrust Belt. Other earthquakes in this province are widely scattered with only two events as large as Intensity VI occurring within 200 miles of the station.

2.5.2.2.4 Tectonic Differentiation of the Mobile Belt

The mobile portion of the northern Appalachian orogen within the region of interest for this study includes the eastern cratonic margin, which is underlain by continental crust of predominantly Grenvillian age (Inner Piedmont and Conestoga Valley tectonic provinces).

The eastern cratonic margin is bounded on its western side by the Blue Ridge - Highlands tectonic province and on its eastern side by the easternmost extent of Grenvillian basement. This eastern boundary is interpreted principally from a line of gneiss domes of one billion year-old continental crust including the Pine Mountain belt, the Sauratown Mountains anticlinorium, the Baltimore Gneiss domes, and, possibly, the Chester dome of Vermont. This boundary corresponds to the eastern limit of the ancient continental margin of North America (Ref. 2.5-76 and 2.5-77). It also coincides with several significant geological and geophysical changes (Ref. 2.5-76). First, it parallels the main gravity high of the Appalachians (Ref. 2.5-78). Second, it is marked by contrasting seismic refraction profiles that reflect deep crustal contrast. Finally, it is a zone of faulting, contrasting structural style and contrasting metamorphic facies.

This cratonic margin is divided into two tectonic provinces north of Virginia, the Conestoga Valley province and the Inner Piedmont province. The boundary between these provinces corresponds to the Martic Line in Pennsylvania (Ref. 2.5-79) and the southward extension of Cameron's Line in western Connecticut.

Conestoga Valley Tectonic Province

The Conestoga Valley tectonic province is characterized by a miogeosynclinal assemblage overlapping an older clastic assemblage. Triassic Basins of the Newark Group are characteristic of the Conestoga Valley province (and, to a lesser extent, the Inner Piedmont and Coastal Plain) and are found in the area between Massachusetts and North Carolina. Triassic rocks have been encountered in borings at Bowling Green and Edgehill, Virginia and near Brandywine, Maryland.

These basins were formed during Triassic time as downfaulted and folded elongate graben structures. Non-marine arkosic sediments and intercalated lava flows filled these basins as they were down-faulted and tilted. At the close of the period, the processes of erosion continued to modify the topography of the eastern section to form the base for deposition of Coastal Plain sediments.

In the Triassic Basins and associated down-faulting, intrusions of Trio-Jurassic age are cut and displaced, indicating a post-Trio-Jurassic age for some of the faulting. Similar intrusions in the Inner Piedmont are not disrupted or offset in this manner. Earthquakes no larger than Intensity VI have been noted in the Conestoga Valley province, although some small events, up to Intensity VI, are reported and have been tentatively associated with Triassic basin border faults.

Inner Piedmont Tectonic Province

The Inner Piedmont Tectonic Province is characterized by a eugeosynclinal assemblage over an older clastic assemblage, which is characterized in this region by a northeast-southwest trending belt of Precambrian to early Paleozoic schists, gneisses, slate, metaconglomerates and some igneous intrusions. These rocks are interrelated in a complex manner by faulting and folding.

Earthquakes ascribed to the Inner Piedmont should include the boundary (Inner Piedmont-Coastal Plain) Intensity VII events at Wilmington, Delaware (1871) and Asbury Park, New Jersey (1927) as well as several Connecticut valley events of Intensity VII which occurred over 200 miles from the station, albeit, in the Inner Piedmont Province (Ref. 2.5-73). No larger events have been recorded in this province and none of the historical shocks can be satisfactorily related to specific structures. The Inner Piedmont is, in general, apparently the most seismically active portion of the area within 200 miles of the station. Concentrations of moderate events are apparent in the New York City area in and the Central Virginia seismic zone near Charlottesville as described by Bollinger (Ref. 2.5-80). Both of these zones are characterized by low to moderate seismic activity. Seismicity elsewhere in the province is relatively rare and apparently random.

Coastal Plain Tectonic Province

The Coastal Plain tectonic province is characterized by the development of a miogeosynclinal wedge during the advanced phases of the final crustal divergence. In the region south and east of the station, this province is characterized by a stratigraphic sequence of interbedded sands, gravels, clays and silty sands of both marine and continental origin. These materials were deposited on the downwarped basement complex from Early

Cretaceous to Quaternary time. The strata wedge out at the Fall Zone to form a wedge-shaped mass that thickens to the southeast. The average dip of these strata varies from 75 feet per mile within the Cretaceous sediments to approximately 10 feet per mile in the upper Tertiary formations.

Few geologic structures are known in the Coastal Plain Province. The Salisbury Embayment is a structural low in the basement rocks between Newport News, Virginia and Atlantic City, New Jersey. The Embayment is marked by a deep accumulation of Mesozoic and Cenozoic sediments, which approach a thickness of 3,500 to 7,500 feet at the Maryland coastline. The feature is fairly prominent in the basement rocks but loses form in the younger sedimentary sequences, suggesting that it is predominately a pre-Tertiary feature. The Coastal Plain underwent regional epeirogenic movements from Pliocene to Quaternary time, which lifted a portion of the continental terrace above sea level .

The significant seismic activity in the Coastal Plain includes the Intensity X event at Charleston, South Carolina, and, for the sake of conservatism, the Wilmington, Delaware event of Intensity VII.

2.5.2.3 Correlation of Earthquake Activity with Geologic Structures or Tectonic Provinces

Only a few of the historical earthquakes in the northeastern United States can be satisfactorily related to specific structures at this time. Therefore, a consideration of the significant events which could influence the seismic design for the Susquehanna SES will rely, for the most part, on an approach based on the tectonic settings discussed above. To augment the tectonic province approach, the concept of the seismic zones within the provinces as discussed by Bollinger (Ref. 2.5-80) and Hadley and Devine (Ref. 2.5-81) will be addressed.

Those events which constitute the largest earthquakes of record in the Eastern United States and which embrace all significant considerations for the Safe Shutdown Earthquake for the station, are listed below:

1) The large events (maximum Intensity IX) such as those in the St. Lawrence Valley and Ottawa-Bonnechere Graben area

2) The large events such as those (maximum MM Intensity VIII) which occurred in the Cape Ann Massachussetts area

3) The (originally catagorized as Intensity VII-VIII) Attica shock (1929) in western New York State

4) The Intensity (IX to X) Charleston, South Carolina earthquake in the Coastal Plain

5) The Intensity VIII Giles Co., Virginia earthquake of 1897

6) The Intensity VII events such as those shocks which have been recorded in and around New York City, Wilmington, Delaware, Asbury Park, New Jersey, and Lake George, New York, and

7) The Intensity VI events which occur only infrequently in the general region

St. Lawrence Valley

The St. Lawrence Valley and the Ottawa-Bonnechere Graben area are contained in the Ottawa Basin tectonic province (Ref. 2.5-73). Earthquakes as large as Intensity IX are reported in this region. The structural interpretation shows that this area is the extension of a transverse trough and mobile zone into the stable interior (Ref. 2.5-73).

Because of the obvious historical confinement of seismic activity to this region marked by an intraplate weakness, recurrence of such large shocks are expected to remain in the area and are thus not translatable to the station.

Boston-Cape Ann

The large (Maximum Intensity VIII) events in the Boston-Cape Ann area were formerly historically associated with the Boston-Ottawa trend of earthquake activity (Ref. 2.5-82) which included the Ottawa-Bonnechere Graben area. However, a recent re-evaluation (Ref. 2.5-73) has resulted in the identification of tectonic regimes which separate the former "Boston-Ottawa trend" into specific tectonic provinces. On the basis of this, the Cape Ann Intensity VIII event, being the largest event to have occurred in the Avalon Platform province (Ref. 2.5-73) would be restricted to a distance from the site of no less than 250 miles. Moreover, according to Ballard and Uchupi (Ref. 2.5-83), it is possible that the significant Boston-Cape Ann seismic activity is associated with the faulted northwestern boundary of the Avalon Platform.

For these reasons it is not deemed necessary to translate this activity (Maximum VIII) out of the Avalon Platform.

Western New York

The shock of 1929 near Attica, New York is anomalous with respect to the exceedingly sparce seismicity of this portion of the

Stable Interior. It does mark, however, a noted concentration of earthquakes which are spatially related to the well-recognized feature of the immediate area, the Clarendon-Linden Structure (Ref. 2.5-114). It is generally accepted that any recurrence of a similar event would be confined to the Attica area. Therefore, the postulation of a recurrence of this shock at the closest approach of Stable interior to the station is not warranted. A recurrence of the largest event at any location along the Clarendon-Linden Structure could result in only minimal ground motion at the station (less than Intensity IV).

Charleston, South Carolina

The largest events to occur in the eastern United States were the events of approximately Intensity X at Charleston, South Carolina in 1886.

The concentration of seismic activity (over 400 events) in the immediate vicinity of Charleston is unique to the Atlantic Coastal Plain; moreover, such a confined density of epicenters is unmatched anywhere in the central and eastern United States, with the possible exception of the New Madrid, Missouri region. On the strength of this areal distribution alone, it would be concluded that a specific tectonic anomaly is responsible for this localized activity. Independent lines of investigation have recently suggested a structural regime which may be responsible for the observed seismicity. On the basis of seismic reflection profiles parallel to the coast of South Carolina, Dillon (Ref. 2.5-84) has reported evidence of northwest-trending faults in the continental shelf along the South Carolina coast, and states that this would seem to be the only zone of active faulting in the United States south of Cape Hatteras and east of the Appalachians. Possible evidence of faulting is noted in the basement rocks offshore and in the Tertiary rocks of the continental margin. This possible faulting aligns with the northwest-trending seismic zone (Ref. 2.5-80) and has been postulated to be the extension of an active oceanic fracture zone into the continental block (Ref. 2.5-84, and 2.5-82).

More locally, a mild, breached fold in the shallow sediments several miles west-southwest of Charleston has been identified by Colquhoun and Comer (Ref. 2.5-85) as the Stono Arch. The axis of this arch trends west-northwest and has possible associated faulting. The trend of this structure is aligned with, and grossly parallel to, the seismic zone and the offshore structure discussed above, and represents the only known deformation in the immediate vicinity of Charleston. Thus, it may be a near-surface expression of the more regional (and deeper) anomaly suggested by offshore reflection surveys and magnetic anomaly trends (Ref. 2.5-86).

Transverse to the strike of these structural features are the northeast-trending axes of two structural highs which are identified along the coast, from Savannah, Georgia to just south of Charleston, as the Beaufort-Burton High and the Yamacraw Ridge (Ref. 2.5-84). According to Dillon et al. (2.5-84), the Beaufort-Burton High may be a shallow expression of the deeper lying Yamacraw Ridge. The intersection of these structures with the suggested northwest trends in the vicinity of Charleston may, at least, be an expression of deeper basement complexity in the area, and lends support to a definition of structure responsible for the well-defined cluster of seismic activity in the Charleston area. No other structural anomalies of significance are known in this area of the Coastal Plain. Therefore, the unique density of earthquake activity in the Charleston area is considered to be associated with localized structure, the character and extent of which are only grossly suggested at the present time. In this respect, an earthquake similar to the largest Charleston shock would be expected to recur in the same locale, and would not be subject to translation throughout the Atlantic Coastal Plain tectonic province.

Giles County, Virginia

The Giles County, Virginia earthquake of 1897 is the largest shock to have occurred in the southern Appalachian region. It is listed (Ref. 2.5-72) as Intensity VIII, and occurred in the Southern Appalachian Seismic Zone near its intersection with the Central Virginia Seismic Zone (Ref. 2.5-80), more than 350 miles from the station. This intersection is marked by a definite break in the continuity of the activity of the northeast-trending Southern Appalachian Seismic Zone and lies well to the south of an area of apparent differentiation of the system of tectonic stresses along the Appalachians called the Central Appalachian Salient in southern Pennsylvania.

This salient was probably initiated during early crustal divergence in late Precambrian time (Ref. 2.5-73 and 2.5-87) resulting in a profound difference between the northern and southern portions of the Appalachian orogen as evidenced by three stages of the orogen's development:

1. In late Precambrian the initial rifting stage developed with a bend, offsetting the northern and southern portions of the continental margin.

2. During the end of the convergent stage (middle to late Paleozoic), the Alleghanian orogeny was pronounced only in the south and translation was restricted to the northern Appalachians.

3. In the Jurassic during the final rifting stage, different stress regimes prevailed in the northern and southern portions (Ref. 2.5-88).

The Central Appalachian Salient occurs where the NNE to NE trends, common to the Appalachians, change to EW in the vicinity of 40°N latitude. The EW trend has been interpreted to be a major crustal structure by several authors (Ref. 2.5-89 and 2.5-90) based largely on circumstantial evidence of interpreted offsets of geophysical anomalies, isopach contours, and geologic map patterns. Drake and Woodward (Ref. 2.5-90) have suggested that 80-90 miles of dextral offset has occurred on this feature and that no evidence of post-Cretaceous movement has been found. Figure 2.5-8 shows the approximate location of this structure as defined by Woodward (Ref. 2.5-89).

Even though its surficial expression cannot be well defined, the Central Appalachian Salient clearly divides the northern and southern portions of the orogen. This is borne out by inspection of the historical seismicity shown on Figure 2.5-8, wherein consistent changes in the seismicity within the described provinces are noted from north to south. The virtually aseismic character of that portion of the Fold and Thrust province containing the station has been noted by the Nuclear Regulatory Commission (Ref. 2.5-91).

Thus, in consideration of the tectonic development, the inferred geological and geophysical evidence, and seismicity, the existence of a fundamental boundary between the northern and southern orogen is herein considered and illustrated as a zone on Figure 2.5-8.

This change of seismotectonic style is further corroborated by Hadley and Devine (Ref. 2.5-81, Sheet 3) who have shown a seismotectonic province generally recognizing the earthquake activity of Bollinger's (Ref. 2.5-80) Southern Appalachian Seismic Zone within the Fold and Thrust Belt. At its northern extent, their boundary stops at the southern Pennsylvania border about 125 miles from the station. They describe the zone as an area where epicentral distribution or relation to known structure indicates a limiting structural factor, and where at least one earthquake of Intensity VII or VIII (Giles County event, 1897) has been recorded.

Because of (1) the notable change in tectonic style in the Fold and Thrust Belt Province South of the Pennsylvania border, (2) the reduction in seismicity north of the Central Appalachian Seismic Zone, (3) the assignment of a different seisotectonic character to the Fold and Thrust Belt south of Pennsylvania, and (4) the historical record which shows a sparcity of earthquakes in Pennsylvania, we consider that a translation of an Intensity

VIII event (Giles County recurrence) closer than 100 miles to the Susquehanna SES is not warranted.

Intensity VII Events

Consideration must be given to the likelihood of Intensity VII events which are known to occur occassionally in this region of the northeast. Within 200 miles of the station, nine shocks of Intensity VII have been documented. Five of these are early reports (1568, 1574, 1584, 1592 and 1791 - See Table 2.5-2) of concentrated activity in the Connecticut Basin about 200 miles from the site. This area lies within the Inner Piedmont Tectonic Province. The other four occurred within the Piedmont province, or at its (eastern) boundary with the Coastal Plain near New York City and northern Delaware. The closest approach to the station of the tectonic province containing these events would be 50 miles. Such an event would attenuate to about Intensity V even on unconsolidated materials at the station, according to conservative central U.S. attenuation characteristics (Ref. 2.5-92), and would be less on competent rock.

The Intensity VII event at Lake George in northern New York, although over 200 miles from the station, is spatially associated with a general concentration of smaller earthquakes. Hadley and Devine (Ref. 2.5-81, Plate C) confine this Lake George event to (1) a tectonic province whose nearest approach to the Susquehanna SES is greater than 150 miles, and (2) a bounded area of seismic activity "in which known faults are associated with epicentral alignments or distribution in such a way as to indicate that movements on the known faults or closely related faults have been the source of recorded earthquakes." This seismic area boundary approaches no closer than 150 miles to the station.

It is seen, then, that Intensity VII events can be confined to approaches of tectonic provinces, seismic zones, and/or structure which are no closer than 50 miles to the Susquehanna SES.

Intensity VI Events

Within 200 miles of the station, a few scattered Intensity VI events are noted (Figure 2.5-8). The two closest events occur about 48-60 miles due south at the closest approach of their tectonic province and are, at least spatially, related to Triassic border faults (Ref. 2.5-81). Several others are concentrated in the immediate vicinity of the Clarendon-Linden structure in northwestern New York State. It should be noted that in the station province (Fold and Thrust Belt) no Intensity VI events occur north of the Central Appalachian Salient in southern Pennsylvania, a distance of over 100 miles from the station. In the Stable Interior, an Intensity VI event in northern New York, 180 miles north of the station, is not related to known structure, and could conservatively be translated to the

closest approach of its province to the station, a distance of about 40 miles.

2.5.2.4 Maximum Earthquake Potential

The previous section defined the maximum potential earthquake in terms of the closest postulated approach of maximum historical events to the Susquehanna SES. Consideration was given in each case to a conservative utilization of tectonic province models, recognized seismic zones, and/or any associated tectonic structure. The resulting candidates for the Safe Shutdown Earthquake (SSE) are:

Intensity (MM)	Closest Approach to Site	Maximum Site Intensity
X	450 miles	< V
IX	> 300 miles	V
VIII	100 miles	V-VI
VII	50 miles	V-VI
VI	40 miles	IV-V

In deriving the maximum Intensity to be felt at the station from the above candidates, the attentuation curves developed for the central United States (Ref. 2.5-92) were used. These curves are the most conservative available for the United States in that both western (California) and eastern (Canada, New York, Charleston, S.C.) data show a greater attentuation of Intensity with distance than does the central United States experience. It should be noted, also, that such attentuation relations are based on isoseismal maps which tend to record the maximum Intensity felt in a given locale, usually on poor soil conditions. It is likely, then, that the Intensities (damage potential) actually experienced on solid foundation material of the Susquehanna SES would be somewhat less than those levels specified in the foregoing table which are used in the Safe Shutdown Earthquake derivation in Subsection 2.5.2.6 below.

From inspection of the above candidates, a station intensity of less than VI is the maximum consistent with the tectonic model, seismic zones, and/or associated structure. This is entirely in keeping with the historical earthquake record which shows that the area of the station is virtually aseismic. Moreover, there are no known faults which appear capable of generating other than minor disturbances well below damaging levels of ground motion.

2.5.2.5 Seismic Wave Transmission Characteristics

The static and dynamic properties of the subsurface materials at the station are presented in Subsection 2.5.4.2. The analyses presented in this referenced section are based on characteristic ground motion and significant frequencies generated by the maximum potential earthquake described above and quantified below.

2.5.2.6 Safe Shutdown Earthquake (SSE)

As a result of the derivations discussed above, an SSE of less than Intensity VI is the maximum earthquake consistent with tectonic models and historical evidence presented for the site. However, an SSE generating a horizontal ground acceleration of 10 percent of gravity (g) has been selected in compliance with the minimum design requirement of the regulatory agencies.

To justify the conservative nature of this design level as an anchor for design response spectra, the acceleration/intensity correlations which have been developed for an Intensity of VI are discussed, although this Intensity is not expected to be felt at the station on the basis of the discussions above.

Recent correlations between Intensity and peak horizontal ground acceleration have made use of currently available data for the western United States (Ref. 2.5-113) and worldwide (Ref. 2.5-93). The results of these current studies do not differ greatly from prior parallel studies, but are generally more conservative. Therefore, these investigations can be used as a general guide for an expected value of acceleration (from an Intensity VI event) on which to anchor design response spectra. These studies show an expected acceleration level of about 6 to 7 percent of gravity as a result of a Intensity VI event. On the basis of these relationships, the design acceleration for the Susquehanna SES for structures founded on rock is conservatively selected as the required minimum of 10 percent g in accordance with 10CFR100, Appendix A. This level is used to anchor the design response spectra shown on Figure 2.5-27. For structures founded on soil, the NRC required that the SSE be increased 50 percent or to 0.15 g in order to accommodate any amplification of ground motion in the soil overlying the bedrock. It should be noted, however, that the maximum earthquake for the station is less than Intensity VI, which correlates with an acceleration of no more than 0.06 g. If this value were increased by 50% to accommodate amplification due to soils, the resulting SSE for structures founded on soil would not be more than 0.10 g. Thus, the selected value of 0.15 g provides a large margin of safety. The 0.15 g value is applied at foundation level.

The duration of strong motion from the SSE is not expected to exceed 5 seconds (Ref. 2.5-95 and 2.5-96) and in all probability, would be considerably less at frequencies critical to design. Duration of motion from a larger, more distant event such as the Charleston, South Carolina event (X) would be relatively longer than that from the design event, but the low accelerations which are characteristics of long period motion from distant large events will be adequately enveloped by response spectra anchored at the minimum level of 10 percent g.

2.5.2.7 Operating Basis Earthquake (OBE)

On the basis of the historical seismicity described above wherein a maximum Intensity felt at the station from historical earthquakes was no larger than IV, an Operating Basis Earthquake (OBE) which would, during the life of the facility, generate a ground acceleration at the station no higher than 5 percent g (1/2 SSE) has been selected. This level of acceleration will not be exceeded by the occurrence of even an Intensity V shock adjacent to the station or a recurrence of large, regional events at a distance. According to Trifunac and Brady (1975), a felt Intensity of V will result in acceleration levels below 4 percent g. Return periods for such ground motion at the station are of an extremely low order of probability, as evidenced by the fact that no local (Pennsylvania) shocks have been reported as felt at the station. Figure 2.5-28 is the design spectra anchored at the OBE level of 5 percent g. For structures founded on soil, an OBE of 0.08 g was used for design.

2.5.3 SURFACE FAULTING

Based on the data contained in Subsections 2.5.1 and 2.5.2 and the interpretations and conclusions therein, there is no capable fault (Appendix A, 10 CFR, Part 100) within at least five miles of the Susquehanna Steam Electric Station.

A detailed description of the lithologic, stratigraghic and structural conditions at the site and the surrounding region is contained in Subsection 2.5.1. All historical reported earthquakes within 50 miles of the site, and all earthquakes within 200 miles of the site with magnitudes (Richter) greater than 3.0 or MM intensities greater than III are detailed in Subsection 2.5.2.

The above referenced information clearly indicates that surface faulting is not of significance to the Susquehanna Steam Electric Station.

2.5.4 STABILITY OF SUBSURFACE MATERIALS AND FOUNDATIONS

2.5.4.1 Geologic Features

General. The upper bedrock at the site area includes the Middle Devonian Mahantango Formation. The upper part of the Mahantango is a dark gray siltstone, with bedding generally delineated by thin, consistent, light gray, fine-grained sandstone stringers. Beneath the upper member, the Mahantango is comprised of 120 to 150 ft of dark gray, hard calcareous siltstone, typically having bedding obscure to absent and displaying cleavage. This member which supports the power block structures is harder, more massive, and more resistant to erosion than the upper member.

The irregular bedrock surface underlying the site is the result of a combination of preglacial weathering and stream erosion, glacial scour, later erosion by glacial melt waters, and the varying resistance of the rock units to erosion. The bedrock is blanketed by till and glacial outwash which grades upward from a gravelly boulder zone to a surface layer of silty fine sands and sandy silt. The surface layer is believed to be reworked loess. The maximum thickness of overburden is around 40 ft in the southern half of the site, with bedrock occasionally cropping out at the surface. North of the east-west bedrock ridge situated just north of the reactors, the glacial deposits fill a valley eroded into bedrock to a depth exceeding 100 ft.

Structurally, the site is situated on the north limb of the Berwick Anticlinorium; its axis passes just south of the site. The anticlinorium trends east-northeast and plunges gently to the northeast. As with the regional picture, folding is the most characteristic feature of the site area. Minor faulting in the form of small bedding-plane slips and intraformational shear zones occur, but they are of no significance to the site. They apparently developed during the Paleozoic (more than 200 million years ago) during the Appalachian orogeny. The zones are typically healed with calcite and quartz (Additional description of site geologic conditions is presented in Subsection 2.5.1.2).

All Seismic Category I plant structures except the spray pond, the Engineered Safeguard Service Water (ESSW) pumphouse and pipeline are founded on bedrock. The ESSW pipeline trench is excavated partly in soil and partly in rock. Most of the other major plant structures, including the cooling towers, are also founded on bedrock.

Site geologic and foundation conditions are entirely suitable for the construction and operation of the Susquehanna SES.

2.5.4.1.1 Areas of Potential Subsidence, Uplift, or Collapse

The potential for significant uplift or subsidence at the site, due to man's activities or geologic conditions such as regional warping, is negligible.

The shallowest carbonate rock that may be present beneath the site is the Onondaga Formation, above which occurs more than 2,000 ft of Middle Devonian shales and siltstones (Figure 2.5-14). At that depth the Onondaga Formation, if present, would not be expected to have a significant potential for subsidence or collapse even if it contained solution cavities. No coal beds are present beneath the site; the nearest coal measures are about 3-1/2 miles north of the site near Shickshinny. Rocks in the site area have no known potential for oil or gas production. The nearest oil or gas field is located 25 miles northeast of the site. Precise leveling surveys and other data in the literature provide no indication that the Site is in an area experiencing any abnormal regional warping, uplift, or subsidence.

More detailed discussion of the potential for uplift or subsidence at the site is presented in Subsection 2.5.1.2.5.3.

2.5.4.1.2 Previous Loading History of the Foundation Materials

Bedrock at the site had been buried and deformed during the Appalachian orogeny (over 200 million years ago) with sufficient intensity to impose secondary cleavage in places and to mobilize calcite and to some extent quartz, resulting in a hard, indurated rock lacking the bedding-plane fissility normally associated with less well indurated silty shales and shaly siltstones. During Quaternary time, at least two ice lobes advanced over the site; the only direct effect this additional load might have on the bedrock at the site would be a tendency to scour loose or weathered rock from the rock-soil interface. Any pre-existing surficial deposits not removed by the glaciers would have been preconsolidated and thereby strengthened by ice loading.

Sufficial material at the site consists of glacial drift largely, if not wholly, deposited by the Olean advance of the Wisconsin ice sheet. These deposits, which are described in Subsection 2.5.4.1, would be expected to have varying consolidation or preloading characteristics depending on local depostional history. Soils in the spray pond excavation (including the ESSW pumphouse excavation) and in the pipeline excavations leading to the spray pond are of particular interest because they support Seismic Category I facilities in these areas. Here geologic mapping shows that these soils consist of well stratified outwash

sands and gravels, together with poorly stratified to unstratified kame-like gravels (refer to geologic maps, Figures 2.5-15 and 2.5-18). They evidently were deposited during or subsequent to stagnation of the final ice advance in the site area, since they are not overlain by a till blanket, nor do the strata show structural evidence of having been overriden by ice. Geologic evidence, therefore, indicates that the stratified surficial materials exposed in the excavations for the soil-supported Seismic Category I facilities are likely to be normally consolidated and not preloaded by ice during or subsequent to deposition.

2.5.4.1.3 Structures and Zones of Weathering, Disturbance, or Weakness in Foundation Materials

Foundation materials consist of two basic types; namely, glacial till and outwash in the spray pond area, and siltstone or indurated slaty shale in the remainder of the principal plant foundations.

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A geologic map of the foundation rock for the principal plant structures is presented on Figure 2.5-18. It shows joints, shears, attitudes of bedding and other features. All foundations shown were excavated so that structures are founded on firm bedrock, the Devonian Mahantango Formation. Geologic sections through the foundations are shown on Figure 2.5-19. A geologic map for the spray pond is presented on Figure 2.5-15.

The southwestern tip of the spray pond is cut into bedrock while the remainder is excavated in glacial materials. The thickness of the glacial deposits beneath the bottom of the spray pond ranges from zero at the rock contact to 93 ft at the eastern end of the pond. The foundation for the pumphouse structure, located at the southeastern corner of the pond, is underlain by 35 to 60 ft of glacial material. The water circulation pipelines, between the pumphouse and the plant, intersect bedrock at an elevation of 668 ft, approximately 260 ft southeast of the pumphouse.

The vicinity of the spray pond is situated over a glacial or preglacial, east-west trending bedrock valley as outlined by contours on top of bedrock (Figure 2.5-17). Total relief of the bedrock surface is about 130 ft. The valley is filled with dense, permeable gravelly and sandy glacial outwash and till deposits that attain a maximum thickness of about 110 ft in the spray pond area. They were deposited during the Olean substage (early Wisconsinan) of the Wisconsin glaciation which occurred at least 50,000 years ago, and there is a possibility that at least some of the bedrock erosion and overlying glacial deposits are the result of an earlier Illinoian glaciation (refer to Subsection 2.5.1.2). In general, the deposits consist of a sequence of sand, gravel, and boulders overlain by sand and gravel, overlain in turn by silty sand. The entire sequence is highly variable in grain size distribution and sorting, and contains discontinuous pockets of similar materials. As a rule, grain size decreases and sorting increases toward the top of the sequence. The glacial materials in the deposit are noncalcareous; most of the rock particles consist of indurated sandstones. The origin and composition of the deposit are such that it is not susceptible to significant weathering or alteration.

In the power block and cooling tower foundations, the principal structural feature is a minor anticline, the axis of which trends about N85°E and was exposed in the radwaste and Unit 1 cooling tower foundations (Figure 2.5-18). South of this feature, bedding generally dips gently south with minor undulations; to the north, beds dip more steeply north. Bedding, which generally strikes N70 to 85°E, is obscure in the foundations; the foundation rock is quite massive and is not characterized by weak zones developed along bedding or cleavage planes. Where

observed, the bedding planes as a rule are smooth and uncontorted with only minor undulations. In the turbine and reactor building foundations, the bedding dips 5 to 10 degrees to the south, while north of the circulating water pumphouse the dips are in places up to 5 to 8 degrees north to northeast, reflecting this minor undulatory variability of bedding. Small-scale folds a few feet in dimension occur but are not prevalent in the site area; one such small anticlinal fold was recognized at the north edge of the circulating water pumphouse excavation. Cleavage is variably developed, strikes generally parallel to the strike of bedding, and dips steeply south.

Jointing in the rock excavated for foundations is fairly well developed. Figure 2.5-18 shows the principal joints encountered at foundation grade, which is at sufficient depth below the top of the rock to be in essentially unweathered material. Here joints are tight and either uncoated, or coated with calcite or a mixture of quartz and calcite. Few joints at foundation level contained significant iron staining; some iron-stained joints are mapped in the radwaste foundation area. Toward the surface these joints generally become more heavily iron-stained with greater degree of weathering, and calcite coatings tend to be leached out, resulting in open joints, in joints partly coated with quartz, or in clay-filled joints in the zone of weathering. The major joint set strikes east-northeast (N60-85°E), and dips vertically (within 15° of vertical). Other steeply dipping to vertical joint sets strike northwest to north-northwest, and north-south. Less steeply dipping joints generally have an east-northeast strike; one group dips gently northward at 10-18°, and another, in the southern part of the excavation, dips 50-60°SE. Some of the joint surfaces, particularly the low-angle joints, are slickensided. In addition to these principal joints, high-angle, discontinuous, white calcite and quartz-calcite veinlets are typically exposed locally throughout principal plant foundations.

A few minor shear planes, originally recognized in the cores obtained during the early phase of site exploration, were exposed during foundation excavation and are mapped on Figure 2.5-18. One shear plane, traced from the northeast corner of the Unit 1 reactor foundation northward to the Unit 1 cooling tower, was found to be oriented parallel to bedding and is denoted "bedding plane shear A" on Figure 2.5-18. The surfaces of the bedding plane shear are healed with 1/4 to 3/4 in. thick laminae of calcite, siltstone and some quartz. The calcite laminae are approximately 1/16 in. thick, alternating with thinner siltstone laminae. The entire exposed area of this bedding plane contains prominent slickensides trending N30° to 40°W, with a 6° to 7°SE plunge. Updip and closer to the top of the rock, the bedding plane contains a 1/2 to 1 in. wide, iron-stained zone, and it also shows extensive leaching of the minerals filling the shear. In places the adjacent rock is weathered to a granular sandy

soil. The calcite which fills the bedding plane shows no sign of crushing. It should be emphasized that the weathering and staining on the bedding plane shear occurs only near the top of the rock where surface water and groundwater could penetrate along the plane; at foundation grade which is well below the weathered zone, the unweathered laminae have the properties of firm rock. In places the bedding plane shear is apparently not a prominent feature in the unweathered rock. For example, it was identified only as a slickensided surface with associated jointing in boring 105 and as horizontal jointing planes in boring 351 (shown in profile on Figure 2.5-19).

Foundation mapping reveals that the bedding plane shear is warped in conformance to the folding of bedding. The shear can be traced northward to the excavation for the Unit 1 cooling tower (Figure 2.5-18). Measured attitudes of bedding show that the axis of the minor anticline described above occurs near the foundations for pedestals 6 and 7 of the cooling tower. South of pedestal 7, the bedding plane shear dips gently south; north of pedestal 6, the bedding plane shear dips gently north. Additional subsurface data from bore holes farther south and north strongly suggest that the configuration of the bedding plane closely follows the undulations in bedding (Figure 2.5-19). No evidence was found at the site to indicate that the shear plane substantially deviates from bedding planes.

During excavation, this bedding plane shear was traced updip to its intersection with the top of rock at a steep, glacially eroded contact. The eroded rock surface was continuous across the trace of the bedding plane, without displacement or offset (Figures 2.5-20a through 2.5-20g). Since the erosion of the rock surface would necessarily have occurred prior to the deposition of the overlying glacial deposits, which have been established as being more than 50,000 years old (refer to Subsection 2.5.1.2.2.1), this relationship shows that any displacement along bedding plane shear A occurred more than 50,000 years ago. In reality, the most probable age of the shearing is pre-Triassic or over 200 million years ago. This is indicated by regional relationships plus the fact that the shear plane is folded (A detailed presentation and analysis of the relationship between site and regional structure is presented at the end of Subsection 2.5.1.2.3.2).

A second bedding plane shear (Shear B), a few feet below and parallel to the first bedding plane shear, was exposed near the northwest corner of the Unit 1 turbine building foundation. It is similar in appearance but apparently more restricted in areal extent than the first. Two vertical, calcite-filled joints cut across this second bedding plane (Figures 2.5-20a through 2.5-20g). The calcite in these vertical joints is continuous across the bedding plane with no offset, showing that the joints were

formed and the calcite was deposited in the joints, subsequent to development of the slickensides on the bedding plane.

The conclusion stated in the PSAR (p. 2.7-2) regarding the significance of these shear planes is still appropriate and deserves restatement: "Minor bedding plane slips at depth have been observed in the site area, both north and south of the interior ridge. Those slips have not experienced movements in more than 200 million years. A minor slip of this nature could be exposed in any large excavation anywhere in the area; however, it would not affect the structural design of the facilities".

All deformational features observed in rock at the site are geologically old and are not significant to plant structures. In unweathered rock, minor shears that do occur are tightly healed with calcite and quartz mineralization, and joints are likewise tight and unweathered. All foundations for plant structures designed to rest on sound rock were excavated to, or into, unweathered bedrock. No structurally weak zones were encountered in these foundations (Refer to Subsection 2.5.1.2.5.5).

Further description of depth of weathering and geologic structures at the site and in the foundations is presented in Subsections 2.5.1.2.3.2 and 2.5.1.2.5.6.

2.5.4.1.4 Unrelieved Residual Stresses in Bedrock

No indications were found during excavation and construction at the site of the presence of any significant stress in bedrock (refer to Subsection 2.5.1.2.5.8 for additional discussion).

2.5.4.1.5 Potential for Unstable or Hazardous Rock or Soil Conditions

Foundation rock at the site is a hard, indurated, unweathered siltstone, a member of the Middle Devonian Mahantango Formation. Similar materials underlie the site to a depth of at least 1,000 ft. This rock contains no unstable minerals and provides highly stable foundation conditions.

Soils at the site are glacial in origin, deposited mostly by flowing glacial meltwater, much under torrential conditions. The soil is noncalcareous. Most of the rock fragments consist of indurated sandstones. The origin and mineralogy of these soils is such that they present no hazardous conditions (refer to Subsection 2.5.1.2.5.7).

2.5.4.2 Properties of Subsurface Materials

A few of the safety-related principal plant structures are founded on soil. These structures consist of the Engineered Safeguard Service Water (ESSW) pumphouse, the spray pond, and portions of the Seismic Category I pipeline linking the reactor building to the spray pond. Most other plant structures are founded on rock. The location of these structures is shown on Figure 2.5-24; soil and rock foundations are identified on Figure 2.5-17A.

The static and dynamic engineering properties of the site bedrock and overburden soils were determined by field investigation and laboratory testing. The results of laboratory testing of the materials sampled from the project site are covered in two reports (Ref. 2.5-97 and 2.5-98).

A detailed study of the soil properties at the site of the spray pond and ESSW pumphouse is given in Subsection 2.5.5.

2.5.4.2.1 Properties of Foundation Rock

The Category I reactor buildings and diesel generator building, as well as the non-Category I turbine and radwaste buildings (see Figure 2.5-24) are founded on unweathered siltstone bedrock. The siltstone, a member of the Mahantango Formation of Devonian age, is hard and indurated, and in the foundations area is lithologically homogeneous with bedding generally not well defined, and lacking the bedding plane fissility usually associated with less well indurated shaly siltstones and silty shales. In places the rock exhibits cleavage, further evidence of its indurated nature.

In the area of the principal plant structures, bedrock bedding where observed generally dips gently (less than 10°) south; locally, such as north of the circulating water pumphouse, beds dip slightly north. At the north end of the radwaste building and the north side of the Unit 1 cooling tower, bedding dips more steeply north. The cleavage is steeply inclined to the south. Minor slickensided bedding plane shears and joint planes occur in the foundations as described in Subsections 2.5.4.1 and 2.5.1.2.3. All such shears beneath the principal plant foundations are fully healed with unweathered calcite and quartz mineralization and do not adversely affect the strength and competence of the foundation rock. Further evidence of the healed nature of these shears is furnished by the RQD values and core recovery rates in borings that penetrated bedding plane shear A (refer to Figure 2.5-18 and discussion in Subsection 2.5.4.1) at elevations below the bottom of the foundation of the principal plant structures, such as in borings 302, 309, and 314. In all cases RQD values are above 35 percent through the shear plane; in most cases, RQD values exceed 80 or 90 percent and core

recovery was close to 100 percent (Further information on foundation geologic conditions is presented in Subsection 2.5.4.1).

Typical values of unconfined compressive strength of unweathered siltstone underlying the principal plant foundations range from 3,650 to 16,000 psi (see Table 2.5-3). The modulus of deformation determined from these laboratory tests on core samples range from 3.1×10^6 to 9.4×10^6 psi. These values indicate strong, competent rock.

P-wave measurements were made by Dames and Moore in the laboratory on individual core specimens. The cores were from borings 303, 314, and 315 which are located, respectively, near the Unit 1 turbine building condensate pump pit at the center of the Unit 1 reactor, and at the center of the Unit 2 reactor. The average seismic P-wave velocity determined for 10 samples at or below foundation grade beneath power block structures is 13,236 fps. For three samples from boring 303 in the Unit 1 turbine building, the average V value is 14,272 fps, or approximately 14,000 fps. These determinations are listed in Tables 2.5-4 and 2.5-5.

Rock quality designation (RQD) measurements made by Dames and Moore on rock cores from below the foundation elevations in the reactor, turbine, radwaste, diesel generator, and circulating water pumphouse foundations exceed 80 percent (refer to boring logs, Ref 2.5-97).

In the reactor area, cross-hole and down-hole measurements of in situ seismic velocities show high values. The measurements were made by Weston Geophysical Engineers, Inc., June 8 - August 6, 1971 using boreholes 105, 303, 307, 314, 315, and 316 (refer to Figure 2.5-29). Values obtained from the cross-hole array for the elevation interval 550-640 ft MSL are 16,000 fps for the P-wave velocity and 7500 fps for the S-wave velocity in the reactor area (design elevation of bottom of reactor foundations, 639 ft MSL). The results of the down-hole measurements yield values that are slightly lower, by a factor of about 15 percent; that is, a V value of about 14,000 fps and V of about 6,200 fps. These in situ results are in good agreement with the laboratory determinations. Additional cross-hole and up-hole in situ seismic velocity measurements were made in the spray pond area (Ref 2.5-99). Results of the cross-hole explorations at the site are further discussed in Subsections 2.5.4.2.2 and 2.5.4.4.

Plate load tests were carried out on sound rock near the center of the Units 1 and 2 reactor building excavation in the vicinity of boring 105 (refer to Figure 2.5-18). Plates 24, 13.5, and 8 in. in diameter were subjected to successively increasing total loadings of 7, 22, and 60 tons per square foot (tsf), respectively. A total deflection of .062 in. occurred when the 24 in. plate was loaded to a maximum of 7 tsf. An additional deflection of 0.036 in. was recorded on subsequent loading to 22 tsf, and another 0.036 in. of deflection on application of the

60 tsf maximum load, producing a total settlement of 0.134 in. for the three-stage loading to 60 tsf. Recovery of the rock by elastic rebound upon release of these loads was substantial: 68, 75, and 80 percent repeatable elastic recovery of the total deflections were recorded after release of the 7, 22, and 60 tsf loadings, respectively. Additional deflections due to cyclic loading were small. Application of 14 cycles of load at 7, 15, and 30 tsf resulted in additional settlements of only 0.012, 0.003, and 0.002 in., respectively, over the corresponding single loadings. These results are consistent with the high modulus values and seismic velocities of the foundation rock, and indicate structurally strong, competent material for foundations in unweathered rock.

It is concluded from the engineering properties of the unweathered bedrock of the Mahantango Formation that the rock provides adequate support for the major plant structures under both static and dynamic conditions. Settlement of structures under static loading is insignificant. It consists of pseudo-elastic compression of the underlying rocks and occurs essentially upon load application. Moreover, the bedrock will undergo no loss of strength and will experience negligible additional settlement under earthquake loading.

A summary of the properties of the foundation rock is compiled in Table 2.5-5.

2.5.4.2.2 Properties of Foundation Soils

The results of detailed explanation of the soils in the spray pond area are given in Subsection 2.5.5. Only information on the properties of the pumphouse foundation soils is given in this subsection.

The natural soils at the pumphouse site are normally consolidated and consist predominantly of sand, gravel, cobbles, and boulders. The soils are poorly stratified, starting as sand or sandy gravel at the surface and grading to mostly cobbles and boulders near bedrock. The depth of the soil deposit below foundation grade ranges from about 35 ft at the south end of the pumphouse to about 60 ft at the north end. A subsurface cross-section through the pumphouse site is shown on Figure 2.5-30, cross-section D-D. The soils below the foundation level are predominantly sandy gravels with large amounts of cobbles and boulders. The properties of these sandy and gravelly soils are as follows:

a) Grain Size Distribution

Grain size distribution tests were made on most of the split spoon samples for classification purposes. Sieve

and hydrometer analyses were performed according to ASTM Procedure D-422. The range of grain size curves is shown on Figure 2.5-31. The mean grain size (D_{50}) of the gravelly soils, which are the predominant material below the pumphouse, was found to be in the range of 4.5 to 25.0 mm. Wherever the sand is present below the pumphouse, the D_{50} size is in the range of 0.14 to 3.0 mm.

b) Relative Density

Relative density data were derived from standard penetration test results using the Gibbs and Holtz procedure (Ref 2.5-100). This procedure is valid for normally consolidated sands.

Values of relative density obtained in this way are summarized on Figure 2.5-32. A direct comparison of relative density from 'N' values given in Figure 2.5-32 and from undisturbed samples and/or in situ density tests cannot be made because no relative density tests were made. The soil deposits are glacial in nature. The deposits are quite variable in particle size and sorting and contain discontinuous sand pockets and gravel pockets. Grain size in general increases with depth. At the foundation level of the pumphouse, the maximum sizes of the particles are in the range of 3 to 12 inches. Undisturbed tube samples could not be obtained in the gravelly soils. The gravel also will influence the results of in situ density tests so that they may not represent the in situ condition as a whole. The Standard Penetration resistance versus elevation is given on Figure 2.5-33. The 'N' values will be influenced by gravel. Because of this the higher blowcounts were not considered representative of site conditions. A value of N = 40 was selected for design. Of the 49 standard penetration tests made beneath the foundation level at the ESSW Pumphouse, 43 exceeded 40 blows per foot. Of the 6 values that were less than 40 blows per foot only one was less than 30 blows per foot.

c) Static and Dynamic Shear Strength

Undisturbed sampling of gravelly soils was not possible. Therefore, shear strength testing was conducted only on the sands. The shear strength of the gravelly soils was then conservatively assumed to be equal to that of the sands.

The details of the testing procedures and selection of design strengths are given in Subsection 2.5.5. The effective angle of internal friction was selected from the test data to be 35° (Figure 2.5-34). The cyclic shear stress ratios at the two effective consolidation pressures 1.0 ksf and 6.0 ksf were determined to be 0.320 and 0.260, respectively, for 5 loading cycles (Figure 2.5-35, Subsection 2.5.5). A linear relationship was assumed in computing cyclic shear

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stress ratios at other effective consolidation pressures.

d) Shear Wave Velocity and Shear Moduli

Cross-hole shear wave velocity measurements were performed by Weston Geophysical Engineers, Inc. (Ref 2.5-99). Compressional and shear wave velocities were measured in situ to depths of about 100 ft. The average shear wave velocities obtained from the measurements are given on Figure 2.5-36.

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Shear moduli were computed from the values of shear wave velocity:

$$G = \frac{\gamma}{g} y_S^2$$

Where:

G	=	shear modulus, psf
γ	=	unit weight, pcf
g	=	gravitational acceleration, ft/sec²
V_S	=	shear wave velocity, fps

A discussion on how the shear modulus is influenced by the confining pressure, the strain amplitude, and the relative density is given in Subsection 2.5.5.2.

2.5.4.3 Exploration

The location of all field explorations is shown on the plot plan, Figure 2.5-22.

A total of approximately 250 exploratory borings was made in soil and rock at the site. Borings were logged in detail; boring logs are contained in Refs. 2.5-97, 2.5-98 and 2.5-99 and Appendix 2.5C. The soils were classified in accordance with the Unified Soil Classification System. Rock logs include RQD (rock quality designation) values. Coring in rock was performed using NX double-tubed coring equipment.

Drilling was conducted in late 1970 (100 and 200 series borings) to establish general geologic relationships over the site area and to determine general soil and rock conditions at the site. A more intensive program (300 series borings) was conducted in the Spring of 1971 to define foundation conditions in the principal plant structures area. Four 45-degree angle holes were drilled in the reactor area. Additional exploration drilling was necessary to locate the site for the Susquehanna River intake and discharge structures (700-800 series borings), to define soil and rock conditions at the spray pond and ESSW pumphouse site (1100 series and some 400 series borings), and to investigate foundation conditions for the cooling towers (borings B1 to B10) and the railroad spur and bridge over State Highway 11 (borings 417 to 455 and 929 to 940). Because of the safety-related

(Category I) function of the spray pond and ESSW pumphouse, the exploration program for these facilities was comprehensive and included split spoon and undisturbed samples, laboratory testing, hydrologic surveys, permeability tests, and seismic cross-hole and up-hole surveys. After completion of geologic borings, static water levels were measured in some of the borings drilled on the site. Perforated plastic pipes were installed in a number of the borings to allow collection of future water level data. These borings are denoted on the plot plan, Figure 2.5-22.

Forty-seven test pits were excavated by backhoe at selected locations to observe soil and rock conditions. Two north-south trenches totalling over 700 ft in length were excavated to obtain information on physical properties, structure, and variability of the near-surface materials at the site. Logs of the test pits and trenches are compiled in Appendix 2.5C.

A geologic map of the Category I and other principal plant foundations is presented on Figure 2.5-18. A geologic map of the excavation for the spray pond is shown on Figure 2.5-15. Geologic profiles are identified on Figures 2.5-18, 2.5-22, 2.5-30 and shown on Figures 2.5-19, 2.5-21, 2.5-30, 2.5-40 and 2.5-56.

Photographs depicting significant features in the principal plant foundation excavations are shown on Figures 2.5-20a through 2.5-20q.

2.5.4.4 Geophysical Surveys

Seismic refraction profiles and cross-hole, up-hole and down-hole measurements were conducted at the site during the Fall of 1970, Summer of 1971, and Summer of 1974. The seismic refraction lines totalled over 40,000 lf of coverage. They are identified on Figure 2.5-29. The refraction profiles collected in Appendix 2.5C.

As interpreted from refraction measurements, overburden at the site consists of a surficial layer of unconsolidated, unsaturated material up to 15 ft thick, constituting at least in part the soil horizon, underlain by more consolidated, partly or fully saturated till and compact outwash, which extend to the bedrock surface. Compressional (P-wave) velocity of the surficial material is typically about 1500 fps. Velocities in the lower till and outwash material generally range between 3,000 and 4,500 fps, although in some places velocities attain 6,000 fps (north-south baseline at boring 107).

The refraction survey obtained a persistent P-wave value of 12,000 to 14,000 fps for unweathered bedrock, which in many places is coincident with the top of rock. Frequently, however,

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lower velocities were recorded in a zone 0 to 20 ft thick near the top of rock. These lower compressional velocities are in the range of 6,000 to 9,000 fps, and are indicative of the zone of surficial weathering near the top of rock. At the site, material having a P-wave velocity of 4,000 to 6,000 fps may represent either dense soil or more thoroughly weathered or fractured bedrock; construction experience at the site indicates that here such material is generally correlative with dense soil.

Seismic cross-hole velocity measurements were performed in the reactor and spray pond areas, the principal sites of the Category I structures. Two arrays were employed in the spray pond area; namely, a north-south array across the location of the ESSW pumphouse, and an east-west array over the approximate location of lowest top of rock elevations in the spray pond. Both latter arrays provided data from which were calculated values for the dynamic moduli of the soil materials. In addition, down-hole measurements were made in the reactor area and up-hole measurements in the spray pond area. Figure 2.5-29 shows the borings that were used for the cross-hole arrays.

In the spray pond area, the seismic characteristics of the subsurface materials as measured in each array are similar. The material overlying bedrock has a P-wave velocity ranging from 4,200 to 4,800 fps and an S-wave velocity ranging from 1,600 to 1,900 fps. It is overlain by lower velocity material at about elevation 658 at the ESSW pumphouse location and at about elevation 643 farther west in the spray pond. At the ESSW pumphouse, this upper material has P-wave and S-wave velocity ranges of 2,300 to 2,400 fps and 1,300 to 1,350 fps, respectively, while farther west beneath the pond the materials between approximate elevations 643 and 673 have P-wave and S-wave velocity ranges of 3,000 to 3,300 fps and 1,450 to 1,500 fps, respectively. Table 2.5-6 summarizes the results of the seismic velocity measurements in the spray pond area and lists dynamic moduli computed from these data.

In the reactor area, cross-hole measurements were made on material above and below foundation grade. Above foundation grade the bedrock was weathered to a depth of about 10 ft below original top of rock. P-wave velocity of this weathered material was 7,600 fps and S-wave velocity, 3,600 fps. A P-wave velocity of 14,800 fps for the interval 640 to 660 ft MSL indicates the top of unweathered rock is at about 660 ft. At and below foundation grade in the reactor area high seismic velocities were recorded (V_p = 16,000 fps, V_s = 7,500 fps) indicating the presence of strong, competent foundation rock. Table 2.5-7 lists the results of the in situ cross-hole velocity measurements made in the reactor area; Table 2.5-5 lists the moduli values.

Further discussion of the properties of the underlying soils and bedrock are given in Subsections 2.5.4.2 and 2.5.5.

2.5.4.5 Excavations and Backfill

2.5.4.5.1 Extent of Seismic Category I Excavations, Fills, and Slopes

Figure 2.5-37 shows the location and limits of excavations, fills, and backfills associated with Seismic Category I structures at the site. Typical foundation sections for seismic Category I structures are shown.

2.5.4.5.2 Excavation Methods and Dewatering

2.5.4.5.2.1 Excavations in Rock

All Seismic Category I rock foundations were carried to or well below unweathered bedrock. Rock foundations for the turbine and radwaste buildings, although they are not Seismic Category I structures, were prepared according to the same general procedures and criteria used in preparing the Seismic Category I rock foundations.

Excavation of rock proceeded by initial ripping of any weathered surficial rock material followed where necessary by line blasting and presplitting in holes drilled to provide slopes of 1 horizontal to 4 vertical. Essentially vertical slopes in unweathered rock proved stable throughout the duration of construction and no special protective measures were required. Weathered rock was cut on slopes of 1 horizontal to 2 vertical. In a few places, wire mesh was used for protection of higher weathered rock slopes that were exposed for extended periods.

The surface of the excavated foundation rock was scaled to remove loose debris and jetted with water or air to remove loose fragments and to prepare the surface for concrete. Before placement of structural concrete or concrete backfill to design elevation, all Seismic Category I foundations were inspected by an engineering geologist to verify the suitability of the rock and its proper surface preparation to receive concrete. All foundation rock bearing a Seismic Category I structure was geologically mapped (see Figure 2.5-18).

Foundations for each of the cooling towers (nonseismic-Category I structures) consist of 40 individual pedestals supporting the columns and extended to bedrock. Excavation proceeded by cutting a ring trench and preparing for each pedestal a suitable surface in unweathered or partly weathered bedrock by ripping or blasting as necessary, followed by scaling and jetting.

During construction of principal plant structures founded on rock, excavations extended below the water table and some dewatering was required. Due to the low permeability of the rock, groundwater inflow was small. Dewatering was accomplished by surface drains and sumps.

2.5.4.5.2.2 Excavations in Soil

The excavation for the spray pond and ESSW Pumphouse was predominantly in soils. Excavation proceeded initially by using large earth moving equipment, then finished by using more refined procedures. On completion of excavation, the surface layer of the natural soil formation was recompacted as follows:

a) For soils having not more than 12 percent passing the No. 200 sieve size, 80 percent relative density as determined by ASTM D2049

b) For all other soils, 95 percent of maximum dry density as determined by ASTM D1557

Test Results are included in Appendix 2.5C. The location of test specimens with respect to the spray pond is shown on Figure 2.5-59. A statistical analysis of the test results was made and is summarized on Figure 2.5-60. The required compaction was met or exceeded. 17

A protective concrete mat was immediately placed over the compacted soil under the ESSW Pumphouse and a minimum of 5 in. thick reinforced concrete liner placed over the entire spray pond area.

All temporary slopes in soil were formed at a maximum slope of 1 1/2 horizontal to 1 vertical. The temporary slopes in the vicinity of the ESSW Pumphouse were protected with a 3 in. layer of concrete to maintain the natural soil formation intact. All permanent slopes in soil were formed at a slope of 3 horizontal to 1 vertical.

The excavation for the Seismic Category 1 pipelines in soil was carried out similarly. All slopes were cut at a maximum of 1 1/2 horizontal to 1 vertical. The minimum clearances were 1 ft beneath the pipe and 2 ft to the sides.

2.5.4.5.3 Backfill and Compaction

Generally, the excavated area, for a minimum distance of 10 ft surrounding the major structures, was backfilled with a non- 17

7 corrosive lean mix concrete known as sand-cement-flyash backfill. A minimal amount of backfilling has taken place using granular
17 backfill, with the exception of the spray pond and vicinity addressed later in this section.

The Seismic Category I pipelines were generally backfilled with the sand-cement-flyash; otherwise granular material was used.

Buried Seismic Category I electrical ductbanks are composed of reinforced concrete encasements around plastic or metal ducting; the concrete encasement being cast directly against the excavated grade. Granular or sand-cement-flyash backfill was used the same as for buried pipes.

The properties of these respective backfills were as follows:

a) Sand-Cement-Flyash

Weight	-	110 lb/cu ft minimum
Slump	-	3 in. minimum
	-	6 in. maximum
Strength	-	40 psi minimum at 28 days

b) Granular

Granular backfill was well-graded, sound, dense, and durable material. It consisted of sand, gravel or crushed rock and did not contain any topsoil, humus, brush, roots, peat, sod, cinders, shale, rubbish or other perishable materials, or portions of clay, waste concrete, trash, or frozen material. No more than five percent by weight passed the No. 200 sieve. The maximum size of the material was 4 in. in confined areas where hand tamping was required and 6 in. in other areas.

The placement specification of these respective backfills was as follows:

a) Sand-Cement-Flyash

Sand-cement-flyash backfill was either mixed at the batch plant or obtained from an offsite source, conveyed to the point of placement by truck, and placed in lifts not exceeding 30 in. in height. The maximum rate of pour did not exceed 4 ft/hr. It was vibrated in place with approved equipment. It was protected from freezing temperatures for a minimum of 3 days.

b) Granular

Granular backfill was placed in maximum 8 in. loose horizontal layers, moisture conditioned, and compacted

to at least 80 percent relative density as determined by ASTM D2049.

Backfill material within 2 ft of structures and in areas where large construction equipment could not be used or where there was a danger of damage to structures was compacted to the specified density by hand operated equipment.

Some areas beneath the spray pond concrete liner were filled. The material and placement specification for this type of fill (arbitrarily designated Fill Type A) was as follows: 2

Fill Type A, Material

The maximum size of this material was 4 inches and no more than 5 percent by dry weight passed the No. 200 sieve.

Fill Type A, Placement

Fill Type A was placed in maximum 6 inch uncompacted layers, moisture conditioned to obtain the required compaction, and compacted to at least 80 percent relative density as determined by ASTM D2049.

The area to the south and south-east of the spray pond was filled in a controlled manner. The material and placement specification for this type of fill (arbitrarily designated Fill Type 'B') was as follows: 17

Fill Type B, Material

The maximum size of this material was 12 inches and no more than 35 percent by dry weight passed the No. 200 sieve.

Fill Type B, Placement

Fill Type B was placed in a 15 inch maximum uncompacted layer thickness, moisture conditioned to obtain the required compaction, and compacted to satisfy both of the following requirements:

a) At least 80% relative density as determined by ASTM D2049 for material having not more than 12% passing the No. 200 sieve or 90% of maximum dry density as determined by ASTM D1557 for all other material.

b) Irrespective of the compacting effort required to satisfy part a) above, the fill was compacted in one of the following manners as a minimum effort:

i) Using a crawler tractor having a weight at least equal to that of a D8 Caterpillar tractor with bulldozer blade. Each track overlapped the preceding track by not less than four inches. When the tractor has made one entire coverage of an area in this manner, it was considered to have made one pass. Each fill lift was compacted with four passes.

ii) Using a vibratory roller of minimum weight 20,000 pounds having a roller width of approximately 78 inches and a diameter of approximately 60 inches. The roller had a vibrator frequency range of between 1100 and 1600 vibrations per minute and had a minimum vibratory dynamic force of 40,000 pounds. The roller speed did not exceed 3 mph and each track overlapped the preceding one by at least 4 inches. When the roller had made one entire coverage of an area in this manner, it was considered to have made one pass. Each fill lift was compacted with four complete passes.

iii) Using a hand controlled vibratory compactor in locations inaccessible by tractor or vibratory compactors was on the basis of the demonstrated ability of the compactor to compact the material to the same density as the contiguous backfill.

Test results are included in Appendix 2.5.C. The location of test specimens with respect to the spray pond is shown on Figure 2.5-59. A statistical analysis of the test results was made and is summarized on Figure 2.5-60. The required compaction was met or exceeded.

To compute the lateral pressures acting on subterranean walls, all backfill was conservatively assumed to be granular. The static and dynamic engineering properties of this granular backfill was assumed as follows:

Bulk unit weight, γ_b	=	135 pcf
Saturated unit weight, γ_S	=	140 pcf
Coefficient active earth pressure, K_A	=	0.30
Coefficient earth pressure "at-rest", K_o	=	0.70

The computation of static and dynamic lateral soil pressures acting on subterranean walls is addressed in Subsection 2.5.4.10.2.

2.5.4.5.4 Bedding Material for Seismic Category I Pipes and Electrical Duct Banks

The bedding material was sand-cement-flyash as defined in Section 2.5.4.5.3.

The excavation was made to original ground or in sand-cement-flyash backfill to required bedding subgrade. The bedding subgrade was inspected and verified to be sound and dense meeting visual requirements for backfill adequate for support of bedding material, thus meeting specification intent. The subgrade was also inspected for unsuitable material such as water, frozen, organic or deleterious material. Such material, when found, was removed.

The sand-cement-flyash bedding material was either mixed at the batch plant or obtained from an approved offsite source. The sand-cement-flyash was then placed in lifts not exceeding 30 inches in height nor 4 feet per hour. For pipes the pour was brought to the pipe spring line and was allowed to set for duct banks the bedding was not placed until the duct bank concrete reached the required strength. Sand-cement-flyash was then poured to the top of the duct bank and allowed to set.

Installation of the bedding material is not part of the quality control inspection proceedure.

Analysis of the relevant field test for bedding material is included in the summary given in Figure 2.5-61

2.5.4.6 Groundwater Conditions

Special measures for control of groundwater levels beneath Seismic Category I plant structures founded on rock are not required. However, control of groundwater levels and seepage is needed at the spray pond; discussion of design criteria for stability of the spray pond is presented in Subsection 2.5.5.

Periodic water level readings were obtained in the vicinity of the principal plant (power block) structures between December 1970 and August 1972. Groundwater fluctuations ranged from 1.5 ft in drill holes 209, 311, to 6.2 ft in drill hole 213.

The maximum groundwater level measured in the plant structures area during this preconstruction period ranged from approximately 690 ft at the west edge of the site of the turbine building, to about 655 ft at the east edge of the site of the reactor buildings (refer to Figure 2.5-55). These levels were obviously influenced by the topographic high of 749 ft just west of the site of the power block structures. However, subsequent excavation and grading in these areas preclude water levels from rising to this height in the future.

During construction, the area just west of the power block structures was graded to elevation 710 ft or less. Excavations for the foundations of the principal plant structures extended below the water table and some minor dewatering was required. Due to the low permeability of the rock, groundwater inflow was small and was confined to seepage from fractures. Dewatering was accomplished by pumping from low areas and sumps. Where seeps were noted issuing from fractures in the rock, holes were drilled

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into the fractures and pipes caulked in the holes to control water while the mudmat was placed. In the foundation for the reactor building (elevation 639 ft) and in the turbine condensate pump pit (at elevation 635 ft), hydrostatic pressure caused lifting of small areas of the 3 inch thick concrete mudmat that had been placed over the impervious membrane. Approximately 20 relief wells drilled through the mudmat released the pressure and allowed the mat to settle back to its original position. The weight of the structural concrete slab subsequently placed on this mudmat was more than sufficient to resist any uplift pressures.

The highest seeps noted in the foundation rock during construction were at elevation 642 ft in the radwaste building excavation and at about the same elevation in the pipe trench in the southern part of the Unit 2 turbine building. Some seeps were also noted in the foundation rock for the reactor buildings at elevation 639 ft and in sumps below this. To the west of the turbine building in the circulating water pumphouse excavation, water was noted to enter the excavation to an elevation of approximately 660 ft. Hydrostatic lifting (described above) of the impervious membrane did not occur at foundation elevations above 640 ft.

Additional information with regard to groundwater monitoring and water table fluctuations in the principal plant structures area is provided in Subsection 2.4.13 and Tables 2.4-31 and 2.4-32.

At the spray pond, water level information taken between July 29, 1974 and August 4, 1975, and from January through March 1977, indicate a minimum water level fluctuation of 4.0 ft recorded at observation wells 1111 and 1113, and a maximum fluctuation of 7.0 ft in 1115. Additional discussion of groundwater fluctuations in the spray pond area can be found in Subsection 2.5.5. Because groundwater levels at the pond will be higher than the maximum projected flood elevation (refer to Figure 2.5-38 and Subsection 2.4.3, respectively), flooding conditions will not significantly affect the groundwater levels.

Local wells within two miles of the plant site were inventoried and the information is given in Table 2.4-22.

Groundwater flows away from the principal plant structures area to the north, east, and south. However, the predominant direction of flow is to the east and southeast at gradients of 0.05 and 0.06, respectively. The flow rate in bedrock is estimated to be less than 1 ft per day as discussed in Subsection 2.4.13. Groundwater contours at the site are shown on Figure 2.5-38.

Permeability of the intact bedrock at the site is less than 1 ft/year. The average permeability of the glacial materials at

the spray pond is 2,000 ft/year; however, this value has been considerably exceeded in some tests. For a complete description of permeability at the spray pond and plant structures areas, consult Subsections 2.5.5 and 2.4.13, respectively. Measured permeability values may be found in Table 2.4-33 and 2.4-34.

2.5.4.7 Response of Soil and Rock to Dynamic Loading

2.5.4.7.1 Response of Rock to Dynamic Loading

Rock at the site would be unaffected by dynamic loading from earthquakes. During historical time, no Pennsylvania earthquakes have been felt at the site. Approximately 14 earthquakes originating outside Pennsylvania could have been felt at the site, but with a probable maximum intensity of only IV on the Modified Mercalli Scale. Ground motion at this intensity would have had no effect on the site.

The compressional and shear wave velocities of sound, unweathered foundation rock in the reactor area (V_p = 14,000 to 16,000 fps; V_S = 6,200 to 7,500 fps) indicate that the rock possesses a high rigidity and provides effective resistance against dynamic loads for all structures founded upon it (refer to Table 2.5-5). Such rock will not be subject to any loss of strength under earthquake loadings.

2.5.4.7.2 Response of Soil to Dynamic Loading

The analysis of earthquake-induced soil strain and settlement of the spray pond and ESSW pumphouse are given in Subsection 2.5.5. If the sands at the site behave like dry sand during an earthquake, the settlement will be less than 0.05 in. If the sand deposits are saturated and excess pore pressures develop, they will reconsolidate following the earthquake and settlements up to 1.2 in. at the east end of the pond and up to 1.0 in. at the ESSW pumphouse may be expected.

The bearing capacity of the pumphouse mat footing was evaluated by the following equation (Ref. 2.5-115):

$$q_d' = 1/2\ B\ \gamma\ N_\gamma + D_f\ (N_q - 1)$$

Where:

q'_d	=	ultimate bearing capacity
B	=	width of the footing
γ	=	unit weight of the soil
D_f	=	depth of surcharge
N_γ, N_q	=	bearing capacity factors

This equation was derived for the static condition; however, a conservative evaluation of the bearing capacity for the dynamic condition can be made by assuming that, during dynamic loading, the footing has an effective width equal to 1/3 of the actual footing (Ref. 2.5-115). Substituting all values given in Subsection 2.5.4.10.2 into the equation but using B=21.3 ft instead of 64 ft, the ultimate bearing capacity was calculated to be 52 kips/sq ft. The corresponding factor of safety against bearing failure is 17.

2.5.4.7.3 Soil Structure Interaction

Soil structure interaction has been addressed in Subsection 3.7.2.4. The analysis and design of buried pipelines has been addressed in Subsection 3.7.3.12.

2.5.4.8 Liquefaction Potential

For the soil supported spray pond, ESSW pumphouse and Seismic Category I pipelines, the liquefaction potential was evaluated. The soil underneath these structures is predominantly sand, gravel, cobbles, and boulders.

The liquefaction potential of the soils beneath the spray pond and the ESSW pumphouse is discussed in detail in Subsection 2.5.5. The minimum factor of safety against liquefication for these structures was found to be 1.26, which is larger than the minimum acceptable factor of safety of 1.20.

The soil supported Seismic Category I pipelines do not offer a worse situation regarding liquefaction potential in comparison with spray pond, since the pipelines are underlain by the same glacerial deposits as the spray pond area and the depth to the maximum predicted water level is greater.

2.5.4.9 Earthquake Design Bases

The design bases for the SSE and OBE are addressed in Subsections 2.5.2.6 and 2.5.2.7.

2.5.4.10 Static Stability

2.5.4.10.1 Static Stability of Safety-Related Structures Supported on Rock

The reactor buildings, control structure, and the diesel generator building, all of which are Seismic Category I structures, are founded on sound, unweathered siltstone bedrock. The Seismic Category I pipelines linking the reactor buildings with the spray pond are trenched partly in soil and partly in bedrock.

The strength of the unweathered bedrock amply accommodates the loads of the plant providing highly stable foundation conditions. As measured in the Seismic Category I reactor area, compressional velocities are in the range of 14,000 to 16,000 fps; shear wave velocity ranges between 6,200 and 7,500 fps. Static deformational moduli as measured on rock cores vary between 3.1 to 9.4×10^6 psi (refer to Table 2.5-3). Measurements of unconfined compressive strength of unweathered foundation rock from the vicinity of the principal plant structures were between 3,650 and 16,000 psi (Table 2.5-3). Static properties of the foundation rock are summarized in Table 2.5-5. Loads induced by the plant structures are less than the allowable bearing pressure of the rock and far below the ultimate bearing capacity. The structural loads will produce no significant total or differential settlement of the foundations.

Safety-related structures founded on rock were designed for a hydrostatic groundwater loading caused by a maximum groundwater level of 665 ft. This is higher than the expected maximum water level, as discussed in Subsection 2.4.13.

2.5.4.10.2 Static Stability of Safety-Related Structures Supported on Soil

The mat footing of the ESSW pumphouse is 112 ft long, 64 ft wide, and 3 ft thick. The total dead and live loads are 20,000 kips and 2,100 kips, respectively. The corresponding unit pressures are 2.80 ksf and 0.30 ksf, respectively. The bottom of the mat is at elevation 657 ft.

The ultimate bearing capacity of the mat can be estimated by the following equation (Ref. 2.5-115):

$$q'_d = 1/2\ B\ \gamma\ N_\gamma + D_f\ (N_q - 1)$$

Where:

q'_d = ultimate bearing capacity

B = width of the mat = 64 ft

γ = unit weight of the soil = 30 pcf

D_f = depth of surcharge, conservatively assumed to be zero

N_γ, N_q = bearing capacity factors

= 38, and 33, respectively (Ref. 2.5-115) corresponding to $\phi = 35^\circ$ (Subsection 2.5.4.2.2)

The ultimate bearing capacity of the mat foundation was found to be 158 kips/sq ft. The factor of safety was computed to be 51, which indicates no danger in overstressing the supporting granular soil. Therefore, the allowable bearing pressure and settlement of the mat footing were evaluated by the method of limiting settlements suggested by Peck, Hanson, and Thornburn (Ref. 2.5-116). The allowable bearing pressure for a maximum settlement not to exceed 2 in. was computed by the formula:

$$q_a = 0.22\ C_a\ C_n\ N_w$$

Where:

q_a = allowable bearing pressures, tsf

N = number of blows per foot in the standard penetration test

C_n, C_w = correction factors for "N", for the effects of overburden pressure and location of groundwater surface

A conservative N value of 40 was selected to represent the soils below the mat foundation (Elevation 657 ft, Figure 2.5-38). The Standard Penetration Tests below the foundation level were made at an average overburden pressure of about 6,000 psf (Figure 2.5-39); the corresponding correction factor C_n was obtained from Figure 19.6 of Ref. 2.5-115 to be 0.63. Assuming that the groundwater surface is at 7 ft below the mat and no surcharge, the correction factor C_w was computed to be 0.55 by equation 19.4 of Ref. 2.5-115.

The allowable bearing pressure was computed to be 6.0 kips/sq ft based on the values of N, C_h, and C_w given above. At this bearing pressure, the settlement of the mat foundation should be less than 2 in. and the differential settlement should be less than 3/4 in. Therefore, by proportion, for a design total pressure of 3.1 kips/sq ft, the corresponding maximum and differential settlements would be less than 1 in. and 1/2 in., respectively. Settlement in sand and gravel deposits occurs almost simultaneously with the application of load. Since more than 80 percent of the total load is dead load, then less than 0.2 in. of settlement is expected after the completion of the construction.

The structural stability of the ESSW pumphouse is discussed in Subection 3.8.4 and 3.8.5.

The sustained load from the spray pond is less than the weight of overburden removed; therefore, there is an adequate factor of safety against overstressing the underlying soil. Soils rebound during excavation in granular soils of the type at the spray pond is insignificant.

The maximum predicted elevation of the water table is below the base of the spray pond and ESSW pumphouse; therefore, dewatering was not necessary and hydrostatic water loadings were not considered in the design of these structures. A full discussion of the water table in this vicinity is in Subsection 2.5.5.

The lateral earth pressure acting on subterranean walls of Seismic Category I structures was computed assuming granular backfill having the properties stated in Subsection 2.5.4.5.3. The coefficient of earth pressure "at-rest" was used. Additionally, the walls were designed for surcharge loadings and dynamic soil pressures as appropriate. The typical pressure diagrams and combinations are shown on Figure 2.5-39.

Water levels in the spray pond area are discussed in Subsection 2.5.5.1.2. Contours on the groundwater table in the spray pond area are shown on Figure 2.5-38. Profiles of measured and projected profiles of the groundwater table beneath the spray pond are shown on Figure 2.5-40.

2.5.4.11 Design Criteria

2.5.4.11.1 Design Criteria of Safety-Related Structures on Rock

As summarized in Subsection 2.5.4.9, the plant structures founded on rock are designed for a maximum acceleration of 0.10g from an occurrence of the SSE event. From consideration of its engineering properties, it is evident that the foundation rock will not be measurably affected by seismic loadings, and negligible additional foundation settlement will accompany these maximum potential dynamic loads. The maximum contemplated total static and dynamic loads of 40 tsf are only a fraction of the bearing capacity of the rock, thus ensuring an ample margin of safety.

2.5.4.11.2 Design Criteria of Safety-Related Structures on Soil

As summarized in Subsection 2.5.4.9, the spray pond slopes are designed for a maximum acceleration of 0.15g from an occurrence of the SSE event at the site. The spray pond riser pipe columns, the Seismic Category 1 buried pipes, and the ESSW pumphouse are designed for a maximum acceleration of 0.15g from an occurrence of the SSE event at the site.

The allowable bearing pressure under both static and dynamic conditions satisfies these conditions:

a) Sustained dead load plus live load with a minimum factor of safety of 3

b) Sustained dead load plus maximum live load with a minimum factor of safety of 2

c) Sustained dead load plus live load keeping settlement within tolerable limits.

At the spray pond, a liner has been designed to restrict the seepage rate from the pond in order to limit buildup of a groundwater mound in the glacial materials underlying the pond. The pond has been designed for a maximum groundwater elevation of 665 ft. Detailed description of design criteria for control of groundwater levels and seepage at the spray pond and the stability of the pond are in Subsection 2.5.5.

2.5.4.12 Techniques to Improve Subsurface Conditions

2.5.4.12.1 Foundations in Rock

No special treatment was required to improve foundation conditions beneath the Seismic Category I structures bearing on rock. During construction, high loads were carried by the gantry crane rails, one of which was adjacent to the top of the temporary vertical slope on the east side of the reactor building excavation. As a precautionary measure to ensure stability of this slope during construction, tensioned rock bolts were installed in the slope. One large pothole was encountered in the Unit 1 turbine building area, necessitating overexcavation of some 23 ft below design base elevation. The resulting hole, which was in fresh, unweathered rock, was backfilled with 574 cubic yards of concrete (f_c^1 = 2,000 psi) to foundation grade.

2.5.4.12.2 Foundations in Soil

No improvement of the natural soil formation at this site was required.

2.5.4.13 Subsurface Instrumentation

2.5.4.13.1 Instrumentation for Rock Foundations

Since settlements are negligible for the safety-related facilities founded on rock (refer to Subsections 2.5.4.7 and 2.5.4.10), no instrumentation to monitor such settlements is necessary.

2.5.4.13.2 Instrumentation for Soil Foundations

The foundation design for the ESSW pumphouse was based on measured soil parameters obtained by field and laboratory testing. The actual settlement should not exceed tolerable limits for the structure and its piping connections. A systematic monitoring program was therefore instituted to study the settlement performance of the structure. The following instrumentation program was carried out:

a) Permanent Bench Marks: Two permanent bench marks were installed as reference points for measurements.

b) Settlement Pins: A total of six settlement pins were cast into the structural mat and 5 settlement pins were installed in the pumphouse floor at Elev. 685'-6". Details are shown on Figure 2.5-41 and Figure 2.5-62. A survey reading was taken on each pin at approximately monthly intervals. The total settlement and differential settlement of the mat foundation was therefore deduced.

Survey readings will be taken on the five pins located at ESSW pumphouse floor elevation 685'-6". These readings will be recorded once a year until 1983. This will give recordings of at least 4 years from the pumphouse completion. In addition to the annual reading a survey of these pins shall be conducted after any of the following events:

1.) Earthquake

2.) 100 year storm

3.) Major leakage or break in a water pipe in the pumphouse fill area.

Results are shown on Table 2.5-8.

2.5.4.14 Construction Notes

During construction of the spray pond liner, cracking was observed in several areas, the most extensive being the area along the southwest edge of the spray pond. The remainder of the cracking was distributed between areas just north and south of the spillway and two small areas located along the north and south central portions. The cracks along the southwest and spillway area were approximately 50 feet in length while the cracks along the central area averaged 7 to 10 feet in length. The cracks in all areas ranged from 1/2" to 1 1/2" in depth. The cracks located above elevation 676'-6" and the cracks wider than 1/16" below elevation 676'-6' were "V" groved to a depth of 1/2" and sealed with Horn Flex L sealant A manufactured by W. R. Grace Co. Cracks below elevation 676'-6" and having widths smaller than 1/16" were left as is.

The hairline cracking which is predominant in the southwest section of the pond is coincident with the concrete liner being placed directly on bedrock. Since the liner in contact with the bedrock is more restrained during the initial concrete curing and shrinkage period it has been determined that these shrinkage forces were the major cause for cracks in this area. In addition two slabs in this area were displaced by hyprostatic uplift

forces causing some additional cracking. This uplift occurred during the construction phase when the pond was empty of water. The hydrostatic uplift pressure was relieved by means of 2 inch diameter core drills thru the liner. These relief holes were then filled with grout just prior to filling the pond with water.

A slab located south of the spray pond spillway was displaced by means of frost heave and resulted in cracking. This action also took place during the construction phase when the pond was empty of water. The displaced section was removed and repaired in accordance with section 7.14 of specification C36. The cracks were repaired as described above. 20

Uplift due to hydrostatic pressure up to design elevation and frost heave are of no design concern when the pond is filled with water as required during plant operation.

In areas where the liner was placed on soil very little hairline cracking has occurred. As a result there has been no indication of cracks being caused by soil settlement.

2.5.5 STABILITY OF SLOPES

Natural slopes at the site are depicted in the site topographic map, Figure 2.4-1. Final plant grades are shown on Figure 2.5-24.

Few rock slopes are present at the site that need to be considered with respect to possible adverse effects on the safety-related operation of the plant. Within the area impounded by the spray pond, bedrock forms a portion of the southwest slope, cut on a gradient of 3 horizontal to 1 vertical. North of the spray pond, a natural slope formed on Trimmers Rock sandstone rises at a maximum gradient of 2 horizontal to 1 vertical to a height of approximately 380 ft above the bottom of the pond (refer to Figure 2.5-56). As discussed in Subsection 2.5.5.2.3.1, such rock slopes would present no significant hazard to safety related plant structures. 2

The soil slopes to be considered are those forming and surrounding the spray pond.

2.5.5.1 Slope Characteristics

The slopes analyzed include the cut slopes of the spray pond and the slopes of the railroad embankment adjacent to the spray pond. The failure of either slope could affect the normal operation of

the spray pond. The stability of these slopes is also dependent upon the stability of the spray pond itself. Therefore, the safety analyses of the slopes and the stability of the spray pond are investigated and discussed together in this section.

The cut slopes of the spray pond consist of two portions separated by a 20 ft service road; both were made at 3 horizontal to 1 vertical (Figures 2.5-42 and 2.5-43). The lower portion is a 17.5 ft slope between the service road (Elevation 685.5) and the pond bottom (Elevation 668). The upper portion extended from the service road to daylight, the height of the slopes varies from

0 ft at the east end to about 40 ft at the west end of the pond. Except for a few cut slopes that are made in bedrock, the majority of the slopes are made of granular material.

The slopes of the railroad embankment adjacent to the spray pond were made of shot-rock. The slopes are at 3 horizontal to 1 vertical with a maximum height of 30 ft.

2.5.5.1.1 Geologic Conditions

The vicinity of the spray pond is situated over a glacial, or preglacial, east-west trending bedrock valley as outlined by contours on top of bedrock (Figure 2.5-17). These contours indicate that the bedrock surface of the valley was eroded about 100 ft below the average elevation of bedrock to the south and considerably more than that below bedrock elevations to the north. Total relief of the bedrock surface is about 130 ft. The valley is filled with dense gravelly and sandy glacial outwash and till deposits which attain a maximum thickness of about 110 ft in the spray pond area. They were deposited during the Olean substage (early Wisconsinan) of the Wisconsin glaciation, which occurred approximately 50,000 years ago, and there is a possibility that some of the bedrock erosion and overlying glacial deposits are the result of an earlier Illinoian glaciation known to have occurred here (refer to Subsection 2.5.1.2). In general, the deposits are normally consolidated and consist of a sequence of sand, gravel, and boulders overlain by sand and gravel, overlain in turn by silty sand. The entire sequence is highly variable in grain size distribution and sorting, and contains discontinuous pockets of similar materials. As a rule, grain size decreases and sorting increases toward the top of the sequence. Topsoil of variable thickness, consisting of brown sandy silt and organic matter, overlies the glacial drift.

Bedrock beneath the spray pond is correlated with the uppermost strata of the Middle Devonian Mahantango Formation. Strata of the overlying Trimmers Rock Formation crop out along the ridge north of the spray pond; the contact between these two formations is buried by glacial material, but has been inferred from drill hole data to occur immediately north of the spray pond along the buried south-facing bedrock slope (refer to Subsection 2.5.1.2.2.2 and Figures 2.5-40 and 2.5-56). The strata, which consist of dark gray, noncalcareous siltstone with fine sandstone stringers in the upper Mahantango grading to more sandy material in the Trimmers Rock Formation, strike N75°E and dip 15° to 40° north.

The southwestern tip of the spray pond is cut into bedrock while the remainder is excavated in glacial materials. The thickness

of the glacial deposits beneath the bottom of the spray pond range from zero at the rock contact to 93 ft at the eastern end of the pond. The ESSW pumphouse structure located at the southeastern corner of the pond is underlain by 40 to 80 ft of

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glacial material. The water circulation pipelines between the pumphouse and the plant overlie glacial material having a maximum depth of 65 ft. They intersect bedrock at an elevation of 668 ft, approximately 260 ft southeast of the pumphouse (refer to Figure 2.5-17A).

North of the spray pond, the Trimmers Rock Formation forms a steep ridge rising approximately 380 ft above the spray pond. The south-facing slope of this ridge is essentially a rock slope underlain by resistant sandstone thinly mantled with soil and rock fragments. The sandstone is massive to flaggy and exposures exhibit well developed joint systems. The lower portions of the Trimmers Rock are less sandy and occur beneath the surface from the base of this high ridge southward to the northern part of the
3 | spray pond area (Figure 2.5-56).

Geologic conditions elsewhere at the site are reviewed in Subsection 2.5.4.1.

2.5.5.1.2 Groundwater Conditions

The groundwater table elevations and contours shown on Figure 2.5-38 are based on water level measurements made June 30, 1971 in the vicinity of the major plant structures, and on measurements made August 6, 1974 in the spray pond area. Water level measurements in the plant structures area were discontinued before the observation wells in the spray pond area were installed, and the wells were destroyed during construction of the plant. The water level data show that the groundwater table is in bedrock beneath the major plant structures, whereas beneath most of the spray pond it is in the glacial drift. Modification (lowering) of the water table by excavation in the major plant structures area is described in Subsection 2.4.13.5. However, some movement of groundwater from the plant structures area toward the spray pond to the north can still be expected, even though the major direction of movement is toward the Susquehanna River to the east. The direction of groundwater movement from the spray pond is also easterly toward the Susquehanna River. The undisturbed groundwater table elevation beneath the southwest end of the spray pond is about 670 ft where it is in bedrock. At the east end of the pond, it is in soil at an elevation of
3 | 615 ft.

The observation wells installed in the spray pond area have not been monitored long enough to allow a close determination of a

maximum high water level. Monitoring of the observation points was discontinued in August 1975 and was resumed in January 1977. The recorded measurements suggest that, in some cases, up to 11 ft of fluctuation has occurred. However, the measurements taken during October 1974 are considered to be incorrect; therefore, they are not included in the evaluation. Eliminating those measurements, the maximum fluctuation is 7 ft (Table 2.5-9). Intermittent measurements of water levels at observation wells in the area of the principal plant structures were taken by Dames & Moore over a period of 11 months (1970 through 1971). These data indicate fluctuation of less than 10 ft. Using these limited data, it is estimated that the maximum rise of groundwater levels beneath the spray pond will not be greater than 10 ft above those on August 6, 1974.

2.5.5.1.3 Field Sampling and Testing

The field exploration for the spray pond was carried out from June 27, 1974 through August 15, 1974. The drilling subcontractor was American Drilling and Boring Company of Providence, Rhode Island. The boring locations are shown on Figure 2.5-44.

At the time of the investigation, the spray pond area had been used as a spoil area for excavation from the plant site. As much as 33 ft of soil and rock was dumped above natural ground. The majority of this was in the east half of the spray pond area. At the west end of the spray pond, a railroad fill consisting in large part of shot rock skirted the spray pond. The railroad fill was 30 ft deep at Boring 1120. The area between Borings 1110 and 1107 was the only area without any spoil.

Underlying the spoil material is glacial drift which in turn overlies siltstone bedrock. The depth of glacial material varies from 0 ft at Borings 1118 and 1121 to 108 ft at Boring 1104. The bedrock surface generally slopes to the east. At the southwest end of the site, bedrock is exposed at ground surface. The natural soils consist predominantly of sand, gravel, cobbles, and boulders. The soils are poorly stratified, starting as sand or sandy gravel at the surface and grading to mostly cobbles and boulders near bedrock. However, cobbles and boulders were encountered at various depths in most of the borings. Some of the sands and gravels were silty. Generalized sections through the pond area are given on Figure 2.5-30.

Twenty-five test borings were drilled. Ten holes were completed for the geophysical survey, ten for permeability and five as groundwater observation wells. Also shown on Figure 2.5-44 are borings in the 300 and 400 series made in 1971 and 1972 (Ref. 2.5-97 and 2.5-98). Information provided by these early borings

was used for preparing the generalized sections given on Figure 2.5-30. The details of the drilling and sampling program are included in Subsection 2.5.5.3 along with logs of borings.

Permeability tests, using either packers or driven casing to isolate zones to be tested, were conducted in nine holes in the spray pond site. The method of analysis used is described in US Bureau of Reclamation Earth Manual, Designation E-18. One hole (1124) was constructed for permeability testing using the field permeameter method, as described in the US Bureau of Reclamation Earth Manual, Designation E-19. Locations of these test holes are shown on Figure 2.5-44, and results of the tests are listed in Table 2.5-10.

The tests were conducted primarily to determine permeability characteristics of the glacial drift and the contact zone between the glacial drift and the bedrock (siltstone of the Mahantango formation). Permeability testing of the Mahantango Formation was performed during investigation of the railroad bridge (Table 2.5-11). The siltstone beneath the spray pond is similar to that tested at the railroad bridge, and these data are taken as representative of the intact bedrock beneath the spray pond.

One of the test sections in the spray pond was isolated in the weathered and fractured siltstone (Boring 1117) immediately below the contact with the glacial drift. The calculated average permeability of that test (Table 2.5-10) is markedly higher than any of the tests performed in the intact bedrock, as would be expected. The exploratory holes in the spray pond area penetrated no more than 10 ft of the more permeable weathered bedrock. Three of the tests (Borings 1112, 1113, and 1114) measured permeability of the contact zone (including from 5 to 10 ft of the weathered bedrock with overlying glacial drift in the test section), and the balance of tests in the spray pond measured permeability of different materials in the glacial drift.

The boring logs indicate that the glacial drift is primarily outwash deposits consisting of permeable sands and gravels, with some discontinuous lenses of less permeable silty sands. The materials tend to be coarser and, presumably, more permeable toward the base of the deposits filling the small valley. The tests summarized in Table 2.5-10 indicate that the permeability of these materials varies considerably. Permeability of the predominant sand and gravel deposits is greater than 2,000 ft/yr (Borings 1111 and 1115). The silty sand lenses are much lower in permeability (Boring 1122 through 1125).

These data indicate that the average permeability of the glacial drift is considerably higher than that of the intact bedrock. The range of permeability in the glacial drift is greater, with permeability of some silty sands as low as some of the bedrock.

The maximum measured permeability of intact bedrock is 277 ft/yr, and the median value of the 41 tested intervals (Table 2.5-11) is 81 ft/yr. Assigning an average permeability of 200 ft/yr to the bedrock appears conservative. For purposes of seepage analysis, it can be assumed that bedrock is impermeable and groundwater movement occurs in the glacial drift.

The high permeability of the glacial outwash deposits is indicated by the two tests in which the capacity of the measuring equipment was exceeded. Also, during drilling of eight of the exploratory holes, there was considerable difficulty because of loss of drilling fluid (see Table 2.5-12). Commonly, it was necessary to drive casing to seal off highly permeable zones. The coarse nature of these lost-circulation zones precluded attempts to perform meaningful permeability tests. Further, the permeable nature of the glacial drift is demonstrated by the performance of the two plant site water wells for construction use (Figure 2.5-38). Each of these wells has a capacity of 150 gpm, and at least one is operating continuously. These wells draw from a maximum of 60 ft of saturated glacial drift. From the relationship of specific capacity of a water well to the thickness of the aquifer, the permeability of the aquifer can be estimated (Ref. 2.5-101). This method indicated 4,000 ft/yr as the apparent minimum average permeability at these wells.

An average permeability of 2,000 ft/yr for the glacial drift was used in the seepage analysis. Considering the evidence that highly permeable materials are present, the results of the permeability tests, and the yield from the wells, assumption of an average permeability of 2,000 ft/yr is conservative in relating seepage losses to groundwater levels and safety against liquefaction.

In the seepage analyses, the possible differences of vertical and horizontal permeabilities must be considered. The vertical permeability of glacial outwash deposits can be as small as one-fifth the horizontal permeability. Because groundwater in the saturated zone moves in a predominantly horizontal direction, the effective permeability is the horizontal permeability. In analyzing seepage through the unsaturated zone, however, movement of groundwater may be predominantly vertical; thus, the possibly lower vertical permeabilities were considered. Beneath the spray pond lenses of materials with low permeability are thin and discontinuous and therefore do not appear to cause a significantly lower permeability in the vertical direction. This is confirmed by the fact that no perched water has been detected in the area.

2.5.5.1.4 Laboratory Testing

2.5.5.1.4.1 General

In general, the granular deposits underlying the spray pond consist of silty sand at shallow depth, underlain by sandy gravel with boulders and cobbles. The test program was conducted only on the sands because of difficulties in collecting undisturbed gravel samples. The undisturbed samples were obtained in sand zones which had lower standard penetration blow counts than in the coarser material. The relative locations of soil samples for which the tests were made are shown on the generalized cross sections E and F, on Figure 2.5-45.

The laboratory test results are summarized in Table 2.5-13. For detailed information on test procedures and results, see Ref. 2.5-102.

2.5.5.1.4.2 Grain Size Distribution

Grain size determinations were made on most of the split spoon samples and on Shelby tube samples for classification purposes and to determine the D_{50} size that can be used as an index for evaluating the potential susceptibility of granular soils to liquefaction.

Sieve and hydrometer analyses were performed according to ASTM Procedure D 422-63, 1972. The range of grain size curves for the granular deposits is shown on Figure 2.5-31. The mean grain sizes (D_{50}) of the samples of sand and gravel were found to be in the range of 0.14 to 3.0 mm and 4.5 to 25.0 mm, respectively.

2.5.5.1.4.3 Unit Weight

Unit weights were obtained for all undisturbed Shelby tube samples on which strength tests were performed. The undisturbed samples were obtained by cutting the Shelby tubes into approximately 7 in. lengths by a tube cutter. The length and weight of each sample section was determined while in the tube for unit weight computations. The unit weight is required to determine the relative density of the site soils.

2.5.5.1.4.4 Maximum-Minimum Densities

Maximum dry density values were obtained using two procedures; namely, impact compaction and vibratory compaction. Both tests were performed on samples obtained by mixing bulk samples from Test Pit No. 1.

The impact compaction tests were performed using ASTM Procedure D 1557-70, method D, modified so that each of the five layers was compacted with 20 blows of a 10 lb hammer dropping 18 in., i.e., a total compaction energy equal to 20,000 ft.lb/ft^3 of soil. The vibratory compaction test was performed according to ASTM Procedure D 2049-69 using a 0.1 cu ft mold and the wet method.

The maximum dry density obtained from these two tests were 106.1 pcf and 108.2 pcf, respectively.

The minimum dry density was also performed on bulk samples obtained from Test Pit No. 1 according to ASTM Procedure D 2049-69. The minimum dry density obtained was 91.5 pcf.

The relative density of the in situ soils was determined using the maximum and minimum densities.

2.5.5.1.4.5 Relative Density

Relative density data were obtained from two sources: densities of the undisturbed Shelby tube samples were correlated to the maximum-minimum dry densities, and correlations were made with standard penetration test results obtained during the drilling operations using the Gibbs and Holtz procedure (Ref 2.5-100).

The relative densities based on maximum-minimum dry densities were determined using the relationship:

$$D_d = \frac{\gamma_{max} (\gamma - \gamma_{min})}{\gamma(\gamma_{max} - \gamma_{min})} \times 100$$

as given in ASTM Procedure D 2049-69, where

D_d = relative density, percentage

γ_{max} = maximum dry density, pcf

γ_{mim} = minimum dry density, pcf

γ = dry density of undisturbed samples, pcf

There was insufficient data to directly determine the relative density of each of the samples of in situ soils. Therefore, the relative density of undisturbed samples was not used in the analyses. The design engineering properties of the site soils were based on tests on undisturbed samples. During drilling operation, a Standard Penetration Test was performed every 3 ft in each of the drill holes. From these data and the values of effective overburden pressure at the location of each Standard Penetration Test, the relative densities were determined from the correlation between standard penetration resistance, effective overburden pressure, and relative density of granular soils given by Gibbs and Holtz (Ref. 2.5-100). The Gibbs and Holtz procedure is valid for normally consolidated sands. Values of relative density obtained in this way are summarized on Figure 2.5-46.

2.5.5.1.4.6 Static Triaxial Shear Test

Eight static consolidated-drained triaxial tests were performed on undisturbed samples. The purpose of the test was to obtain the strength data required to evaluate the static stability of the cut slopes.

The tests were carried out in triaxial cells and the test specimens were saturated by the back pressure method. The saturation was checked by determining the value of Skempton's B coefficient (Ref. 2.5-103). Specimens were considered to be saturated when the B coefficient was 0.95 or higher. The specimens were obtained by cutting the 3 in. Shelby tubes into 7 in. lengths. They were then extruded and trimmed. The specimens were consolidated isotropically under effective consolidation pressures ranging from 0.50 to 6.0 ksf. These confining pressures represent the range of effective overburden pressures at the site. The results of these tests are presented on Figure 2.5-34 which also shows the selected design parameters.

2.5.5.1.4.7 Cyclic Triaxial Shear Tests

Twenty-five cyclic loading triaxial shear tests (CR) were performed to determine the cyclic shear strength of the soils. Sixteen tests were performed on undisturbed samples. Nine tests were performed on remolded samples. Undisturbed specimens were prepared in the same manner as for static triaxial tests. After completion of the tests on selected undisturbed specimens, they were oven dried, broken down, and compacted by vibration to the same dry density as the original undisturbed specimen.

Test specimens were saturated as in the case of the static triaxial tests and consolidated under an isotropic pressure equal to either 1.0 ksp or 6.0 ksf. During the cyclic shear tests, a

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symmetrical cyclic deviator stress was applied at a constant frequency ranging between 1 cycle per 2 to 3 seconds while measuring axial deformation, axial load, and pore pressure continuously by means of electric transducers and a chart recorder. The results of all CR tests including the number of cycles to reach a total strain of 5 percent are given in Table 2.5-14. The undisturbed specimens were generally found to be more resistant to cyclic loading than the corresponding remolded specimens prepared at the same dry density. The loss of shear resistance may be due to changes in the original soil structure and destruction of slight cementation which exists in the soil in the undisturbed state. The test results on undisturbed samples are shown on Figure 2.5-35.

Also shown on Figure 2.5-35 are the results of four cyclic triaxial tests reported by Dames & Moore (Ref. 2.5-98). In general, the Dames & Moore samples yielded higher cyclic strength. The reason for the difference may be due to the difference in the method of sampling. The undisturbed samples tested by GEI (Ref. 2.5-102) were sampled with a thin-walled Shelby tube sampler which was pushed by hydraulic pressure in accordance with ASTM D1587-67. However, the undisturbed samples tested by Dames & Moore were obtained with the "Dames & Moore" sampler. The area ratio of the "Dames & Moore" sampler is large compared to the thin-walled Shelby tube sampler, and the greater area ratio may result in greater disturbance to the sample. Since the amount of disturbance could not be evaluated and since the GEI samples yielded lower cyclic strength, the Dames & Moore results were not used in the liquefaction analysis for conservatism.

2.5.5.2 Design Criteria and Analyses

2.5.5.2.1 Design Criteria for Spray Pond

The design criteria adopted for the analysis of the spray pond and the slopes surrounding the spray pond include criteria for ground surface acceleration, liquefaction, and slope stability.

2.5.5.2.1.1 Ground Surface Acceleration

The horizontal ground accelerations used for design of the spray pond are 0.15g for the Safe Shutdown Earthquake (SSE) and 0.08g for the Operating Basis Earthquake (OBE).

2.5.5.2.1.2 Liquefaction

For the most adverse water level conditions at the spray pond site, the factor of safety provided against liquefaction should not be less than 1.2 for the SSE condition.

2.5.5.2.1.3 Slope Stability

The slopes in the area of the spray pond must be designed to provide a minimum factor of safety of 1.5 for the static condition and 1.1 when subjected to an SSE event.

2.5.5.2.2 Design Analyses for Spray Pond

The design analyses, including the seepage analysis, liquefaction potential of the spray pond, stability of slopes, and the earthquake induced settlement, are given in the following four Subsections.

2.5.5.2.2.1 Spray Pond Seepage Analysis

The total inventory that determines the spray pond capacity includes sufficient water to compensate for losses that could occur over the 30 day shutdown period. Additionally, seepage losses must be controlled during normal operation so that the groundwater table is not artificially elevated to a level that would aggravate the safety margin against liquefaction. Seepage analyses were made to determine what design parameters are required for the spray pond to meet these restrictions. It was first determined that seepage from an unlined pond does not meet these restrictions, and that a lining of the pond is required. The second case determines the design parameters for a lining that will sufficiently control the quantity of seepage to satisfy liquefaction requirements.

To maintain the groundwater level below the levels necessary to ensure an adequate factor of safety against liquefaction, an unsaturated zone must be maintained beneath the spray pond. A liner must be designed that will sufficiently restrict seepage and prevent groundwater levels from rising above the design levels. Seepage from the pond will increase the total groundwater underflow beneath the pond and develop a groundwater mound. That is: Total Underflow = Pond Seepage and Base Flow (the present underflow).

The groundwater flow path from the pond is eastward along the trough in bedrock which is filled with glacial deposits. The downstream discharge point of the groundwater mound is assumed to

be at the surface at elevation 600 near a present spring. The quantity of underflow, Q, beneath the pond may be calculated using Darcy's law:

$$Q = KIA$$

Where:

Q = quantity of underflow (ft^3/yr)

K = permeability (ft/yr)

I = average hydraulic gradient (ratio)

A = cross sectional area of flow path (sq ft)

The controlling permeability for this case is the average permeability of the glacial drift, 2,000 ft/yr. The average hydraulic gradient may be taken as the difference in elevation between the elevation of the assumed discharge point, 600 ft, and the elevation of the water table beneath the center of the pond over the distance between the two points, 1,850 ft. Gradients were determined for several assumed elevations beneath the pond. The average cross-sectional area of saturated glacial draft along the flow path was determined for each assumed elevation of the groundwater mound beneath the pond.

An average base underflow of 4.3×10^5 ft^3/yr was calculated for the undisturbed groundwater conditions, represented by water levels shown on Figure 2.5-38. The amount of total underflow was then calculated for several assumed groundwater elevations beneath the center of the pond. The seepage which is producing the groundwater mound can be determined by subtracting the base flow of 4.3×10^5 ft^3/yr from the total underflow. Then, by using a form of Darcy's Law (Ref. 2.5-105):

$$q = K \frac{p + d}{d}; \text{ and } Q = qA$$

Where:

q = quantity of seepage through one square foot of liner assumed to be saturated

K = effective liner permeability (ft/yr)

p = head of water in pond (ft)

d = liner thickness (ft)

Q = effective seepage losses through the liner (ft^3/yr)

A = area of the spray pond (sq ft)

The seepage loss as related to maximum groundwater level beneath the pond can be calculated. Then, the liner thickness and permeability that would restrict the amount of seepage sufficiently to maintain the selected groundwater elevation can be calculated. This provides a relationship between liner parameters and the elevation of the groundwater mound. The groundwater elevations beneath the pond that would be maintained by specific liner parameters are listed in Table 2.5-15 and shown on Figures 2.5-38 and 2.5-40. The relationship between seepage losses and the groundwater elevation beneath the pond is shown on Figure 2.5-47. To maintain the groundwater level below the maximum allowable level determined by the liquefaction analysis, 665¼ft, a reinforced concrete liner has been constructed. The concrete liner has expansion, contraction and construction joints at appropriate spacing to control cracking. Both expansion and construction joints have impermeable rubber waterstops in corporated.

The relationship between the thickness of the liner, permeability of the liner and seepage loss is shown on Figure 2.5-57. The permeability of reinforced concrete is conservatively 1×10^{-2} feet per year. The minimum thickness of liner provided over the entire pond is 5 inches. Therefore, during normal operation, the seepage loss from the spray pond is estimated to be 5.9×10^4 gallons per 30 days. More than 5 times this amount is required to raise the groundwater level to the design value of 665 feet which was used for the liquefaction analysis.

Accident conditions and their possible effect on the integrity of the liner and on seepage losses have also been examined.

Should a tornado-generated missile having a frontal area of 20 square feet puncture the liner, the additional leakage would be 1.6×10^4 gallons per 30 days. This volume of water would not be sufficient to elevate the water table to an unacceptable level.

In the event of an SSE, the soils analysis covering slope stability, discussed in Subsection 2.5.5.2.2.3.2, shows that the cut slopes will remain stable. No credit is taken in the analysis for the presence of the concrete liner. Subsection 2.5.5.2.2.4 discusses the settlement which might result from an earthquake induced motion. The relative settlement across the pond would be very small, less than 1 inch in 500 ft. It is therefore anticipated that the liner will not undergo any significant displacement as a result of an SSE. Some additional cracking could occur. However, since a very conservative approach has been taken in providing a liner with a permeability well below that required to establish liquefactor potential, the additional cracking can be tolerated.

Seepage from the spray pond will be monitored by periodic measurements of water levels in observation wells near the spray pond; refer to Subsection 2.4.13.4 for discussion of monitoring programs and location of observation wells. Additionally, actual seepage losses from the pond will be measured by periodically

recording water levels in the pond and by measuring amount of precipitation in a rain gauge and rate of evaporation in an evaporating pan located near the spray pond.

2.5.5.2.2.2 Liquefaction Potential

2.5.5.2.2.2.1 Method of Analysis

The evaluation of the liquefaction potential of the soils at the site was made by comparing the shear strength of the soils under cyclic loading conditions to the dynamic shear stress induced in the soils by the vibratory motion associated with the SSE. The ratio of shear strength to induced shear stress is termed the factor of safety against liquefaction. Since both the shear strength of the soils and the induced shear stresses are dependent on depth below ground surface, determinations of the factor of safety against liquefaction were made at various depths.

Soil profiles and the corresponding groundwater levels representative of the site conditions were chosen for the study. The SSE was applied at the ground surface and deconvolved downward to bedrock using the SHAKE 3 Computer Program (Ref. 2.5-106).

The soil profiles used in the analyses were conservatively assumed to consist only of sand even though they included gravel, boulders, and cobbles in places as discussed in Subsection 2.5.5.1. Based on limited information available (Ref. 2.5-107), the resistance to liquefaction of gravel, boulders, and cobbles is equal or better than that of sand. For instance Kishida (Ref. 2.5-112) has indicated that soils with D less than 2 mm and with uniformity coefficients less than 10 are most susceptible to liquefaction (Ref. 2.5-98). The saturated unit

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weight of the sand was taken to be 130 pcf and the buoyant unit weight to be 67.5 pcf. The spray pond was simulated as a material with a low shear modulus value of 1.0 ksf. Because water does not transmit shear waves, the simulation was necessary so that the computer program SHAKE 3 could be used to compute the shear stresses induced by the earthquake. Use of this small modulus has an insignificant influence on the induced shear stresses.

2.5.5.2.2.2.2 Soil Profiles and Positions of Groundwater Table

As disclosed by the field investigation, the thickness of overburden varies at the site of the spray pond. The bedrock contours are shown on Figure 2.5-17. At the southwest end, bedrock was exposed at the ground surface and over 90 ft of granular deposits were encountered at the northeast end. Therefore, to evaluate the liquefaction potential, three soil profiles were chosen to represent three thicknesses of overburden. The depths from the bottom of the pond (Elevation 668 ft) to the bedrock for the three soil profiles were 93 ft (Profile 1 - east end of spray pond), 57 ft (Profile 2 - central section, and pumphouse), and 20 ft (Profile 3 - west end of spray pond). The predicted maximum groundwater levels that will occur beneath a lined pond as discussed in Subsection 2.5.5.2.2.1 were used at each profile.

To evaluate the liquefaction potential at other locations in the spray pond, the same soil profiles were used and the groundwater table was varied in accordance with the predicted maximum water table elevations given on Figure 2.5-40.

Figure 2.5-48 shows the soil profiles and the maximum groundwater levels used at Profiles 1, 2, and 3 in the analyses.

2.5.5.2.2.2.3 Shear Moduli

Cross-hole shear wave velocity measurements were performed during August and September 1974 (Ref. 2.5-99). Compressional and shear wave velocities were measured in situ to depths of about 100 ft by the cross-hole procedure. The average shear wave velocities obtained from the measurement are presented on Figure 2.5-36. As shown on the figure, the shear wave velocity increases linearly with depth. The average shear wave velocities used for soil and rock in the liquefaction analyses are also shown.

The shear moduli of sand were computed from the values of shear wave velocity as follows:

$$G = \frac{\gamma}{g} V_S^2$$

Where:

G = shear modulus, psf

= unit weight, pcf

g = gravitational acceleration, ft/sec²

V_S = shear wave velocity, fps

The shear modulus is influenced by the confining pressure, the strain amplitude, and the relative density and, in general, these can be related by the equation:

$$G = 1000\ K_S\ (\bar{\sigma}_m)\ 1/2 \qquad \text{(Ref. 2.5-108)}$$

Where:

G = shear modulus, psf

k_S = a variable parameter, dependent on relative density, shear wave velocity, and strain amplitude

$\bar{\sigma}_m$ = mean principal effective stress, psf

In the liquefaction analysis, the shear modulus values at the corresponding effective confining pressures obtained from the above equation, were used as initial values at very small strains. The strain-compatible shear moduli were then determined from the curve of shear modulus-shear strain relationship as given by Seed and Idriss (Ref. 2.5-108).

2.5.5.2.2.2.4 Cyclic Shear Strength

The results of cyclic triaxial shear tests are given in Table 2.5-14 and on Figure 2.5-35. The results are given in terms of the cyclic shear stress ratio $(\sigma_1-\sigma_3)\ cy/2\bar{\sigma}_c$ and the number of loading cycles required for the test specimen to reach a total axial strain of 5 percent, where:

$(\sigma_1-\sigma_3)cy$ = cyclic deviator stress

σ_c = effective consolidation pressure

The selected design cyclic shear strength is given on Figure 2.5-35. Based on results of the site seismicity study and on the SSE

having a magnitude less than 6, the cyclic shear strength at 5 equivalent uniform load cycles was considered appropriate and this was used in evaluating the liquefaction potential at the pond (Ref. 2.5-96).

The cyclic tests were performed at two effective consolidation pressures, 1.0 and 6.0 ksf. These pressures were selected to envelope the actual field conditions. However, the 1.0 ksf was selected as a lower limit for the testing pressure. Testing of sand samples at very low pressure may not be relative. From the test results, the cyclic shear stress ratios at these two effective consolidation pressures were determined to be 0.320 and 0.260, respectively, for 5 loading cycles. A linear relationship was assumed in computing cyclic shear stress ratios at other effective consolidation pressures. The cyclic triaxial testing conditions differ from field conditions and to account for these differences, and also to permit the use of effective vertical pressures instead of effective consolidation pressures, a correction factor, C_y, must be applied to the test results before using them in liquefaction analyses. The correction factor is a function of relative density and values have been published by Seed and Idriss (Ref. 2.5-108). A value of 0.57 was used in the analyses. This corresponds to an average field relative density of 50 percent for normally consolidated sands at the site. Using the above data, the following relationship was established between field cyclic shear strength, τ, and the effective vertical pressure, $\bar{\sigma}$:

$$\text{Cyclic Shear Strength, } \tau = 0.57\,\bar{\sigma}\,(0.332-0.012\bar{\sigma}_N)$$

or

$$\tau = \bar{\sigma}\,(0.189 - 0.0068\,\bar{\sigma})$$

(τ and $\bar{\sigma}$ in ksf)

The above expression permits the calculation of the cyclic shear strength at any depth down to bedrock.

2.5.5.2.2.2.5 Determination of Dynamic Shear Stresses

The vibratory motion of the SSE was applied at the ground surface and deconvolved downward to the bedrock; thus, inducing shear stresses into the soil. The synthetic time history of ground surface acceleration during the SSE was used with a maximum acceleration of 0.15g as discussed in Subsection 2.5.4.9.

The maximum shear stresses developed at various depths within the soil during the SSE were calculated using the computer program SHAKE 3 developed by Schnabel, Lysmer, and Seed (Ref. 2.5-106).

In addition to the SSE time history and maximum ground acceleration, the computer program uses the following parameters:

a) Unit weights of the subsurface strata and depth to the groundwater table

b) Damping ratios of the subsurface strata and the variation of these damping ratios with shear strain

c) Shear moduli of the subsurface strata and the variation of these moduli with shear strain

The output of the SHAKE 3 computer program provides values of the peak shear stresses induced in the various strata during an SSE event. However, for the liquefaction potential analysis, the equivalent uniform average shear stress is required. The average shear stress during the SSE has been taken to be equal to 0.65 times the peak shear stress (Ref. 2.5-107).

The results of the average shear stress determinations for the various cases analyzed are compared with the cyclic shear strength.

2.5.5.2.2.2.6 Design Earthquake

The synthetic acceleration time history as discussed in Subsection 3.7b.1.2 was used in the evaluation of liquefaction potential.

Based on the site seismicity studies discussed in Subsection 2.5.5.1.2, the ground acceleration of 0.15g was adopted for the SSE for structures founded on soil.

2.5.5.2.2.2.7 Results of Liquefaction Analyses

2.5.5.2.2.2.7.1 Liquefaction Potential Under the Design SSE

The average shear stresses induced by the SSE of 0.15g and the corresponding shear strengths and factors of safety, for three different profiles and various groundwater levels, are given in Table 2.5-16. The factors of safety are also shown on Figure 2.5-49.

Based on these values, it was possible to obtain and to interpolate the factor of safety at any particular location in the pond for the predicted maximum groundwater elevation as shown on Figure 2.5-38. On Figure 2.5-50 the factors of safety at

seven selected locations are shown along with the information on the elevation of maximum predicted water table and bedrock. The minimum factor of safety was found to be 1.26, which is larger than the minimum acceptable factor of safety of 1.20, as given in Subsection 2.5.5.2.1.

As indicated by the results shown on Figure 2.5-50, the factor of safety decreases as the groundwater table rises, and at the same water level the factor of safety decreases as the depth to bedrock increases.

2.5.5.2.2.2.7.2 Variations of Shear Moduli and Damping Ratios for Evaluation of Liquefaction Potential

As mentioned in Subsection 2.5.5.2.2.2.5, the "standard" relationship between the effective strain and the dynamic properties (shear moduli and damping ratios) given by Seed and Idriss (Ref. 2.5-108) was used in the liquefaction analysis for estimating induced cyclic stresses. To evaluate the effects of possible variations of these relationships, liquefaction studies were made initially in which the values of shear moduli and damping ratios were varied by ±30 percent. The most critical soil profile (Profile 2) with the maximum predicted water table at 665 ft was used in the study.

The study included the following cases:

a) Varying shear moduli ±30 percent, damping ratios remain unchanged

b) Varying damping ratios by ±30 percent, shear moduli remain unchanged

c) Change both shear moduli and damping ratios by ±30 percent

The average induced cyclic shear stresses, shear strengths, and factors of safety, along with the results using the standard relationship, are summarized in Table 2.5-17. The maximum change of shear stress is found to be about 3 percent. This reduces the minimum factor of safety to 1.23, but it is still larger than the acceptable value of 1.2 (Subsection 2.5.5.2.1). The results of this study indicate that the effects of variations of moduli and damping are small and do not change the conclusion that there is an adequate factor of safety against liquefaction.

A subsequent review of the results in Table 2.5-17 were made to determine the effect of varying the damping ratio by ±50% and the shear modules by ±50%. The review was made by projecting the

results. A plot showing the effect on the factor of safety is given on Figure 2.5-50A.

2.5.5.2.2.2.7.3 Results of Liquefaction Analyses Using Real Earthquake Records

All the results of the liquefaction study, presented in the previous sections, were based on the design SSE of 0.15g at the ground surface. Some real earthquake records obtained at sites with comparable geologic conditions and in the same range of magnitude were available and were used to check the liquefaction potential. Three rock records: Golden Gate (M = 5.3, 1957), Helena (M = 6.0, 1935), and Parkfield (temblor Station, M = 5.6, 1966) were used for this purpose.

Liquefaction studies were made using these records applied at rock outcropping for obtaining cycle shear stresses in the soil.

The resulting factors of safety along with the factor of safety for the design SSE (Bechtel synthetic) are summarized on Table 2.5-18 and Figure 2.5-51. The minimum factors of safety obtained from the real records were larger than the ones obtained from the synthetic earthquake. The stresses induced by both the design SSE and the real earthquakes are also shown on Figure 2.5-52.

2.5.5.2.2.3 Slope Stability Analyses

2.5.5.2.2.3.1 Stability of Rock Slopes

The southwestern tip of the spray pond is cut into bedrock. However, since the cut slope is 3 horizontal to 1 vertical, the slope will obviously be stable, considering the engineering properties of the bedrock as discussed in Subsection 2.5.4. A detailed analysis of the stability of such a slope in rock is therefore not required.

North of the spray pond, the Trimmers Rock Formation forms a steep ridge rising approximately 380 ft above the bottom of the spray pond. The south-facing slope of this ridge is essentially a rock slope underlain by flaggy, resistant sandstone thinly mantled with soil and rock fragments. The closest approach of this slope to the spray pond is along the northern perimeter of the pond; the toe of slope at elevation 710-720 ft is at least 150 ft from the top of the north slope of the pond at elevation 700-727 ft (Refer to Figure 2.5-24 for final site grades in this area). The maximum slope along the ridge is about 2 horizontal to 1 vertical, and an overall slope of 3-1/2 horizontal to 1 vertical, a relatively flat slope for rock. Bedding in the rock dips approximately 30° to the north into the slope; thus, it is favorably oriented for slope stability. Data of McGlade (Ref. 2.5-56, p. 108) indicate that natural slopes eroded on Trimmers Rock strata are "steep and stable". In consideration of the competency of the rock forming the slope and the favorable orientation of rock structure, together with the fact that such gentle rock slopes are normally stable in this region, it is concluded that there is an ample margin of safety against failure of the slope north of the spray pond.

2.5.5.2.2.3.2 Stability of Slopes in Soil

Stability analyses were performed for the spray pond cut slopes and the fill slopes of the railroad embankment that are immediately adjacent to the pond. Both cut and fill slopes are constructed at 3 horizontal to 1 vertical. Except a small portion of cut slopes that are made in bedrock, the majority of cut slopes and all the fill slopes are made of granular materials. The granular materials range from sand and sandy gravel for the cut slope to the shot rocks for the embankment slopes. The shot rocks were obtained from the main plant excavation.

To evaluate the stability of the slopes, the effective angle of internal friction of the sand deposits was found to be 35° from

the test data shown on Figure 2.5-34. For the sandy gravel and the shot rock, the effective angle of internal friction was conservatively assumed to be the same as that of the sand.

The pond is lined and the maximum predicted groundwater level is below the bottom of the slopes. Therefore, the infinite slope analysis and the yield acceleration analysis by Seed and Goodman (Ref. 2.5-109) are considered appropriate for evaluating the stability of the slopes.

For static conditions, the infinite slope analysis was used to determine the factor of safety of soil slopes:

$$FS = \frac{\tan\phi}{\tan_i}$$

Where:

ϕ = friction angle of sand

i = inclination of slope

Therefore, for $\phi = 35°$ and $i = \tan^{-1}(1/3)$, the factor of safety under static condition is found to be 2.10.

For the dynamic condition the yield acceleration analysis was used. The yield acceleration is defined as the acceleration at which sliding will begin to occur. The yield acceleration coefficient (k_y) is defined as:

$$k_y q = \tan(\phi - i)q$$

where ϕ and i were defined in the previous paragraph. For $\phi = 35°$ and $i = \tan^{-1}(1/3)$, k_y is found to be 0.297. Compared to the SSE of 0.15q, the factor of safety for the dynamic condition would be:

$$FS = \frac{0.297}{0.150} = 1.98$$

Consequently, the railroad embankment slopes and the cut slopes will be stable under both the static and dynamic conditions.

2.5.5.2.2.4 Earthquake-Induced Soil Strain and Settlement

Two methods were used for estimating the earthquake induced settlement. Seed and Silver have proposed a method for determining settlement of sands that are sufficiently free draining in the field such that excess pore pressure cannot develop during an earthquake (Ref. 2.5-110). Lee and Albaisa

have proposed a method that accounts for the reconsolidation settlement that results from dissipation of excess pore pressure following the earthquake (Ref. 2.5-111).

Following the procedure suggested by Seed and Silver (Ref. 2.5-110), the distributions of average induced shear strain as a function of depth for the soil profiles shown on Figure 2.5-48 were plotted and are shown on Figure 2.5-53.

It was conservatively assumed that the relationship between vertical strain and cyclic shear strain for sand at 60 percent relative density, as shown on Figure 8b of Ref. 2.5-110, was applicable for the sand deposits at the site. The corresponding values of vertical strain were then interpolated based on the values of shear strain as shown on Figure 2.5-53. The settlement of each layer was obtained by multiplying the layer thickness and the vertical strain. The total settlement was then obtained by summing up the settlements of all layers. By this approach, the vertical settlements at three Profiles 1, 2, and 3 shown on Figure 2.5-48 were found to be 0.05, 0.03, and 0.01 in., respectively. The results of these computations are summarized in the first half of Table 2.5-19.

To estimate the vertical settlement resulting from dissipation of excess pore pressure following the earthquake, the procedure proposed by Lee and Albaisa (Ref. 2.5-111) was followed. The results of cyclic triaxial shear tests carried out by GEI (Ref. 2.5-99) and an analysis using the SHAKE 3 computer program were used in addition to the experimental data shown on Figures 6 and 7 of Ref. 2.5-111.

The stress ratio causing liquefaction is related to field conditions by the equation (Ref. 2.5-104):

$$\frac{\sigma dc}{2\sigma a} = \frac{1}{C_r} \left(\frac{\tau}{\sigma'_o} \right)$$

in which

$\frac{\sigma dc}{2\sigma a}$ = stress ratio

and C_r = correction factor

τ = shear stress induced

σ'_o = effective overburden pressure

The stress ratios developed at various depths can be computed when the shear stress induced during an earthquake and the effective overburden pressure are known. The stress ratios developed at various depths were computed based on the values of

induced shear stress given in Table 2.5-16, the computed effective overburden pressure, and a correction factor $C_\gamma = 0.57$. Entering these stress ratios on Figure 2.5-35, the corresponding number of cycles (N_1) to reach a total axial strain of 5 percent are obtained. A nondimensional cycle ratio (N/N_1) is computed for each depth with N equal to 5.

Figure 2.5-54 was prepared to show the relationship between the peak excess pore pressure and cycle ratio. The correlation was based on the data obtained from the cyclic triaxial shear tests on undisturbed samples (Ref. 2.5-99). This figure is similar to Figure 9 of Ref. 2.5-111. The peak excess pore pressure ratio ($\Delta \mu/\sigma 3c$) that will occur during an earthquake can be estimated from Figure 2.5-54 using the cycle ratio (N/N_1) obtained previously.

The volumetric strains were estimated from Figures 6 and 7 of Ref. 2.5-110 using the peak excess pore pressure data in Figure 2.5-54. Figure 6 of Ref. 2.5-110 was used first to obtain the volumetric strains for sand at 50 percent relative density. These strains were then corrected to correspond to strains in sand at 60 percent relative density by multiplying by a factor of 0.8 obtained from the curve shown on Figure 7 of Ref. 2.5-110.

Lee and Albaisa assumed in their paper (Ref. 2.5-110) that vertical strain is equal to the measured volumetric strain in triaxial tests. However, when lateral movement is restricted as is the case of the soil deposit at the Susquehanna site, the vertical strain is one-half of such volumetric strain. Therefore, the volumetric strains obtained by the procedure of Lee and Albaisa were divided by two to obtain the appropriate vertical strains for the site conditions. The vertical settlement of each layer was then determined by multiplying the thickness of each layer by the vertical strain in the layer. The total settlement was obtained by summing up the settlements of each layer. The results of these computations are summarized in the second part of Table 2.5-19.

The values of total vertical settlement are also summarized below:

	Resulting from Compaction of Dry Soils (Inches)	Resulting from Reconsolidation of Saturated Soils (Inches)
PROFILE 1 (East End of Pond)	0.05	0.1 - 1.2
PROFILE 2 (Central Section, Pumphouse, etc.)	0.03	0.1 - 1.0
PROFILE 3 (West End of Pond)	0.01	0.01 - 0.2

Based on the results given above, it is apparent that if the sands at the site behave like dry sand during an earthquake, then the settlement will be less than 0.05 in. However, if the sand deposits are saturated and excess pore pressures develop, they will reconsolidate following the earthquake and settlements up to 1.2 in. at the east end of the pond and up to 1.0 in. at the ESSW pumphouse may be expected.

The settlements given above were based on soil profiles consisting of sand deposits (Figure 2.5-48); the imposed dead and live loads on the mat footing of the pumphouse were not considered. The imposed weight will increase the confining pressure of the soil, resulting in a higher reconsolidation volumetric strain. However, according to Lee and Albaisa (Ref. 2.5-111), the influence of confining pressure is not strong and is only signficant for developed excess pore pressure ratios greater than about 0.6. As shown on Table 2.5-19, the only soil that may develop a pore pressure ratio greater than 0.6 is at a depth below 43 ft near the bedrock of Profile 2. The soil at that depth, however, has a higher relative density than the 60 percent used for estimating the settlement. Since the volumetric strain decreases rapidly as the relative density increases (Ref. 2.5-111), the net combined effects of a larger pore pressure ratio developed and a higher relative density would result in a smaller volumetric stain. Therefore, the results given above are still valid for the additional imposed weight at the surface.

2.5.5.3 Logs of Borings

Logs of 25 test borings and two test pits are presented in Appendix 2.5C.

Standard Penetration Tests were made in 16 of these holes at 3 ft intervals. Undisturbed samples were taken in five of the split spoon holes where soil conditions permitted. Ten holes were completed for the geophysical survey, 10 for permeability and five as groundwater observation wells. The locations of borings and test pits are shown on Figure 2.5-44. Holes 1118, 1119, and 1121 were planned but not drilled.

Based on the results of the Standard Penetration Test borings, six borings (1106A, 1107A, 1110A, 1112A, 1113A, and 1115A) were drilled immediately adjacent to six of the Standard Penetration Test borings for undisturbed sampling of strata in which, based on classification of the split spoon samples, it was believed undisturbed samples could be obtained. Two test pits were dug with a Case backhoe to a depth of 12 ft to obtain bulk samples.

Due to the large amounts of oversize material encountered, drilling operations were slow and difficult. Frequent mud losses hampered drilling operations in spite of using additives in the drilling fluid. The additives included walnut shells, sawdust, cotton waste and Quick-gel. In some holes, it was necessary to case the hole in order to continue drilling.

Soil sampling consisted of both split spoon (Standard Penetration Test) and undisturbed sampling. The split spoon sampling was carried out in accordance with ASTM D1586-67. The undisturbed sampling was conducted in accordance with ASTM D1587-67. The undisturbed sampling was carried out using both Shelby tube and pitcher barrel sampler methods. In both cases, the sample tube were 3 in. outside diameter, 3 ft long and the tubes were of 16 gage steel.

Undisturbed samples were difficult to obtain due to the large amount of gravel, cobbles, and boulders in the glacial drift. The majority of undisturbed samples were obtained from Borings 1106, 1106A, 1107A, 1113A, and 1122. Where possible, every attempt was made to obtain samples below the proposed bottom elevation of the spray pond (Elevation 668 ft).

All undisturbed samples were handled in the same manner. The top end of the tube was cleaned out; a piece of plastic followed by damp paper towels was inserted and a plug was then formed with microcrystalline wax. The bottom end of the tube was trimmed and an expandable packer was installed. The packers were perforated with a 1/16 in. diameter hole for drainage of free water from the sample. Both ends of the tube were capped and dipped in wax. The samples were stored vertically with the expandable packer on the bottom in the subcontractor's equipment trailer in special boxes supplied for this purpose. The temperatures were well above freezing during the time they were stored so no provisions for heating were necessary.

Soil samples selected for the laboratory testing are indicated on the boring logs. The following symbols were used on the boring logs to indicate the type of laboratory test conducted.

CR - Cyclic Consolidated - Undrained Triaxial Test

S - Consolidated - Drained Triaxial Test

Gs - Specific Gravity Determination

Grain Size - Grain Size Determination

2.5.5.4 Compacted Backfill

Compacted fill is placed at the southeast corner of the spray pond to satisfy freeboard requirements. This fill has been placed with a maximum slope of 3 horizontal to 1 vertical. The material, placement, and testing specifications were as follows:

a) Material

Well graded, sound, dense, durable material. It does not contain any topsoil, roots, brush, logs, trash or waste material, ice, or snow. The maximum size of the material is 12 in. and no more than 35 percent by weight passed the No. 200 sieve.

b) Placement

The material was placed in uniform horizontal layers so that when compacted it was free from lenses, pockets, and layers of material differing substantially in grading from surrounding material. It was not placed on frozen ground. Placement for which moisture conditioning was required was suspended whenever the ambient temperature reached 34°F and falling.

The compaction requirements were specified as follows:

Fill shall be placed in a 15 in. maximum uncompacted layer thickness, moisture conditioned to obtain the required compaction, and compacted to satisfy both of the following requirements:

a) At least 80 percent relative density as determined by ASTM D2049 for material having not more than 12 percent passing the No. 200 sieve or 90 percent of maximum dry density as determined by ASTM D1157 for all other material.

b) Irrespective of the compacting effort to satisfy part a) above the fill shall be compacted in one of the following manners as a minimum effort:

i) Using a crawler tractor having a weight at least equal to that of a D8 Caterpillar tractor with bulldozer blade. Each track shall overlap the preceding track by not less than 4 in. When the tractor has made one entire coverage of an area in this manner, it will be considered to have made one pass. Each fill lift shall be compacted with four passes.

ii) Using a vibratory roller of minimum weight 20,000 lb having a roller width of approximately 78 in. and a diameter of approximately 60 in. The roller shall have a vibrator frequency range of between 1,100 and 1,600 vibrations per minute and have a minimum vibratory dynamic force of 40,000 lb. The roller speed shall not exceed 3 mph and each track shall overlap the preceding one by at least 4 in. When the roller has made one entire coverage of an area in this manner, it shall be considered to have made one pass. Each fill lift shall be compacted with four complete passes.

iii) Using a hand controlled vibratory compactor in locations inaccessible by tractor or vibratory roller. Approval to use hand controlled vibratory compactors will be on the basis of the demonstrated ability of the compactor to compact the material to the same density as the contiquous backfill.

c) Testing

The testing requirements were specified as follows:

The in situ density of the fill shall be determined in accordance with ASTM D1556 and performed at a frequency of at least one test per lift and every 10,000 sq ft on plan.

Tests in accordance with ASTM D2049 or ASTM D1557, as appropriate, shall be carried out on the same material extracted for the ASTM D1556 test. The frequency of this testing shall be once in every 10 ASTM D1556 tests.

Gradation tests in accordance with ASTM D422 shall be carried out at least twice in each 8 hours during placing operations.

The railroad embankment to the north of the spray pond was constructed out of rock, obtained from the main plant area excavation.

The material and placement specifications were as follows:

a) Material

Fill shall consist of rock derived from Class B and C excavation having a maximum size of 20 in. Class B and C excavations are defined as follows:

1) Class B Excavation

Rock that cannot be excavated except by systematic ripping.

Ripping shall not be judged necessary when the material can be cut by a bulldozer in the following manner: A fifty-three and one-half (53 1/2) inch high bulldozer blade with standard rock or corner bits mounted on a catepillar D-8 or equal tractor having 270 net flywheel horsepower moved through forty (40) feet of travel shall fill even with the top with a minimum angle of repose of forth-five (45) degrees, or a volume of ten (10) cubic feet per one linear foot of width of the blade.

2) Class C Excavation

Rock that cannot be excavated except by systematic drilling and blasting.

Blasting shall not be judged necessary if the rock can be ripped by a tractor rated at not less than 385 net flywheel horsepower, equipped with a single shank beam, parallelogram type (72" for deep arrangement), and weighing not less than 40 tons fully equipped; i.e., with dozer blade, ripper and other accessories. Drawbar pull will not be less than the following ratios:

1st gear - 95,000 lbs at 1 mph

2nd gear - 48,000 lbs at 2 mph

3rd gear - 30,000 lbs at 3 mph.

Fill shall be placed in lifts not exceeding 24 in. in uncompacted thickness and in such a manner so as to produce a well graded matrix.

b) Placement

Fill shall be compacted in one of the following manners:

1) Using a crawler tractor having a weight at least equal to that of a D8 Caterpillar tractor with bulldozer blade. Each track shall overlap the preceding track by not less than 4 in. When the tractor has made an entire coverage of an area in this manner, it will be considered to have made one pass. Each fill lift shall be compacted with four passes.

2) Using a vibratory roller of minimum weight 20,000 lb having a roller width of approximately 78 in. and a diameter of approximately 60 in. The roller shall have a vibrator frequency range of between 1,100 and 1,600 vibrations per minute and have a minimum vibratory dynamic force of 40,000 lb. The roller speed shall not exceed 3 mph and each track shall overlap the preceding one by at least 4 in. When the roller has made one entire coverage of an area in this manner, it shall be considered to have made one pass. Each fill lift shall be compacted with four complete passes.

3) Using a hand controlled vibratory compactor in locations inaccessible by tractor or vibratory roller. Approval to use hand controlled vibratory compactors will be on the basis of the demonstrated ability of the compactor to compact the material to the same density as the contiguous backfill.

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3.4 WATER LEVEL (FLOOD) DESIGN

As discussed in Section 2.4, all Seismic Category I structures are secure against flooding due to probable maximum flood (PMF) of the Susquehanna River or probable maximum precipitation (PMP) on the area surrounding the plant. Therefore, special flood protection measures are unnecessary. The Seismic Category I structures have, however, been designed for hydrostatic loads resulting from groundwater, as discussed in Section 3.8. The groundwater table is at elevation 665 MSL in the main plant area.

A postulated break in the cooling tower basins or of the water delivery pipes to the basin could result in a build-up of water against the walls of either or both of the ESSW pumphouse and the turbine building. In the event of such water build-up breaching the turbine buidling wall, water that would not be intercepted by the floor drains or grilles and thus would flow through the turbine building to the reactor building would be prevented from endangering equipment in the latter by means of watertight doors. Flood water building up against the ESSW pumphouse would also be prevented from entering the building by means of watertight doors. Impact forces and water pressure due to flood water will not endanger the integrity of the ESSW pumphouse.

All safety-related systems are located in the Reactor Building, Diesel Generator Building, Control Structure and the Engineered Safeguard Service Water (ESSW) Pumphouse.

Sufficient physical separation between these buildings is provided to prevent internal spreading of any floods from one building to another.

Redundant Engineered Safety Features, pumps and drives, heat exchangers and associated pipes, valves and instrumentation in the reactor building subject to potential flooding, are housed in separate watertight rooms. Seismic Category I level detectors trip alarms in the main control room when the water level in any room exceeds the set point. Isolation of the floor drainage lines from these rooms is provided by outside manual valves.

All other rooms in the reactor building and control structure containing safety related equipment which are subject to potential flooding by process fluid leakage or fire protection water are provided with at least one open floor drain.

Floods in excess of the approximately 80 gpm floor drain capacity increase the water level in the affected area and are released through the door-to-floor clearance of these rooms.

Refer to Subsection 9.3.3 for a detailed description of the reactor building and control structure drainage system.

The four diesel generator sets are housed in individual water tight compartments within the diesel generator building. Floor drain line branches from each of these compartments are equipped with check valves to prevent backflooding from the common sump.

The ESSW pumphouse is divided into two redundant compartments. Flooding from internal leakage would, therefore, only affect one of the redundant pump sets. The control and electrical panels are mounted on minimum 4 inch high concrete pads or structural supports. Operating floor openings allow drainage of any leakage to the ESSW pump suction space below or to a reserve sump space that could be emptied with a portable pump.

3.5 MISSILE PROTECTION

Where possible, the Seismic Category I and safety related structures, equipment, and systems are protected from missiles generated by internal rotating or pressurized equipment through basic station component arrangement so that, if equipment failure occurs, the missile does not cause the failure of these structures, equipment, or systems. Where it is impossible to provide protection through plant layout, suitable physical barriers will be provided to isolate the credible missiles or to shield the critical system or component. Also, redundant Seismic Category I components are suitably protected so that a single missile cannot simultaneously damage a critical system component and its backup system. Table 3.2-1 provides a tabulation of safety related structures, systems, and components, along with their applicable seismic category and quality group classification. 16

Section 3.12 - Separation Criteria for Safety Related Mechanical and Electrical Equipment provides a detailed discussion of protection from missles, such as equipment separation and redundancy, to preclude damage to the systems necessary to achieve and maintain a safe plant shutdown. 10

3.5.1 MISSILE SELECTION AND DESCRIPTION

3.5.1.1 Internally Generated Missiles (Outside Primary Containment)

There are two general sources of postulated missiles outside the primary containment:

a) Rotating component failure missiles

b) Pressurized component failure missiles

3.5.1.1.1 Rotating Component Failure Missiles

The systems located outside the primary containment have been examined to identify and classify potential missiles. The basic approach is to ensure design adequacy against generation of missiles, rather than to allow missile formation and then containing their effects.

Catastrophic failure of rotating equipment, such as pumps, turbines, fans, and compressors leading to the generation of missiles, is not considered credible. Massive and rapid failure

of these components is incredible because of the conservative design, material characteristics, inspections, quality control during fabrication and erection, and prudent operation as applied to the particular component.

It has been concluded that large, massive rotating components, such as the various ECCS pumps and motors, fans, and compressors outside the primary containment, do not have sufficient energy to move the masses of their rotating parts through the housings in which they are contained.

Similarly, it is concluded that the HPCI and RCIC turbines cannot generate missiles. Overspeed tripping devices ensure that the HPCI and RCIC turbines will not reach runaway speed where component failure could take place.

However, even with this conservative design, the RCIC and HPCI turbines are located in separate compartments so that any turbine missile will affect only one division of equipment.

This is also true for other large rotating safety related equipment, such as pumps, fans, and compressors. Redundant equipment is normally located in different areas of the plant or separated by walls, so that a single missile from a rotating mass will not damage both redundant systems.

3.5.1.1.2 Pressurized Component Failure Missiles

The following potential internal missile donors from pressurized equipment were investigated:

10 a) High Energy Piping

Pressurized components in systems where service temperature exceeds 200°F or service pressure exceeds 275 psig were evaluated as to their potential for becoming missiles. Pipe whip restraints were provided at possible breakpoints of these high energy lines, which may impact on safety related equipment or structures (see Section 3.6).

Additional attention has been given to ensure that safety relief valves and valve headers are not credible missiles. All SRV headers are restrained in accordance with the pipe whip criteria described in Section 3.6 to ensure that in the event of a circumferential type break of the header, no
10 missile would result.

The safety relief valves are attached to welded, Schedule 150 sweepolet fittings on the headers. The design of this attachment includes all dynamic loads that may be associated with the SRV discharge. This attachment is not a postulated

break location in accordance with the criteria stated in Section 3.6.2. Verification of this will be available upon completion of the stress report. The SRV header is designed and built to the conservative requirements of the ASME Section III, Class 1, Code and as such is subject to the ASME Section XI Inservice Inspection requirements. This inspection plus the RCPB leak detection capability would provide early indication of any possible failure in this area. 10 16 16

Therefore, it is concluded that the likelihood of missiles from high energy piping, which may impact on safety related equipment, is remote.

b) Valve Bonnets 10

Valves of ANSI 900 psig rating and above, constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code, are pressure seal bonnet type valves. For pressure seal bonnet valves, valve bonnets are prevented from becoming missiles by the retaining ring, which would have to fail in shear, and by the yoke, which would capture the bonnet or reduce bonnet energy. 10

The bonnet bolts preload the pressure seal gasket so the valve will be sealed when it is not under pressure. When pressurized, the valve is sealed by process fluid pressure and the bonnet bolts are under no load. All ASME III Class I, 900 # bonnet-seal type valves were analyzed per ASME B & PV Code, Section III. Standard calculation pressure used in these analyses was given by Figure NB-3545.1-2 for weld-end valves. Using the typical pressure seal valve shown in Figures 3.5-9 and 3.5-10 as an example, the total thrust load on the retaining ring and valve body was calculated. The results are listed in Table 3.5-7. The results show both the retaining ring and valve body meet the NB-3227 requirement while using a calculation pressure which is much higher than the normal operating pressure of the valve. 10 16

The majority of valves inside containment have massive valve operators which are supported by the yoke. For these valves, the valve operators act as an additional limitation to the yoke becoming a missile. 10

For a yoke clamp to fail, one would have to assume that the retaining ring fails completely and instantaneously so that the bonnet could strike the yoke. The yoke is normally under no load and complete failure of the yoke clamp is not considered credible. 16

Because of the highly conservative design of the retaining ring of these valves, bonnet ejection is highly improbable and hence bonnets are not considered credible missiles.

Most valves of ANSI rating 600 psig and below are valves with bolted bonnets. Valve bonnets are prevented from becoming missiles by limiting stresses in the bonnet-to-body bolting material by requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III, and by designing flanges in accordance with applicable code requirements. Even if bolt failure were to occur, the likelihood of all bolts experiencing simultaneous complete severance failure is remote. The widespread use of valves with bolted bonnets and the low historical incidence of complete severance failure of bonnets confirm that bolted valve bonnets need not be considered as credible missiles.

10 | c) Valve Stems

Valve stems are not considered potential missiles if at least one feature in addition to the stem threads is included in their design to prevent ejection. Valves with backseats are prevented from becoming missiles by this feature. In addition, air or motor operated valve stems will be effectively restrained by the valve operators.

10 | d) Temperature detectors

Temperature or other detectors installed on piping or in wells are evaluated as potential missiles if a single circumferential weld would cause their ejection. This is highly improbable, since a complete and sudden failure of a circumferential weld is needed for a detector to become a missile. In addition, because of the spatial separation of redundant safety related equipment, a small missile such as a detector, assuming the circumferential weld fails completely, is not likely to hit redundant safety related equipment.

10 | e) Nuts and Bolts

Nuts, bolts, nut and bolt combinations, and nut and stud combinations have little stored energy and thus are of no concern as potential missiles.

f) Blind Flanges

Bolted blind flanges are not considered credible missiles because of the extremely unlikely occurrence of all bolts experiencing simultaneous complete severence failure as discussed in (b) above.

10

g) Safety Relief Valve and Main Steam Isolation Valve Accumulators.

Pressurized ASME III vessels such as SRV and MSIV accumulators are not considered credible missiles. These accumulators are operated at a maximum pressure and

temperature of 150 psig and 150°F. These vessels have low stresses and operate in the "moderate energy" range and therefore any failures would be a slot type and not of concern for missile generation. 10

3.5.1.2 Internally Generated Missiles (Inside Containment)

There are three general sources of postulated missiles inside the primary containment: 16

a) Rotating component failure missiles

b) Pressurized component failure missiles.

c) Gravitationally generated missiles. 10

3.5.1.2.1 Rotating Component Failure Missiles

The most significant pieces of rotating equipment in the primary containment are the recirculation pumps and motors. GE Licensing Topical Report NEDO-10677, submitted to the NRC, contained a discussion of the potential overspeed of a recirculation pump due to LOCA blowdown flow past the pump impeller and the possible results of such overspeed. That report also presents a decoupler concept to protect the pump motor under such conditions.

In a letter to the NRC dated November 6, 1975, GE wrote that an analytical study has shown that a decoupling device is not needed, and that the NEDO-10677 report should be rescinded.

The following results were outlined in the GE letter to the NRC:

a) If a break were to occur in the pump discharge pipe, either a guillotine or longitudinal break, the maximum calculated resultant pump speed would be 110 percent of rated. In this analysis, the flow choking at the volume diffuser inlet area in the pump casing determines the differential feed and volumetric flow rate used to predict pump speed during blowdown. Longitudinal breaks up to one pipe cross-sectional area were considered. 10

b) For a longitudinal break in the pump suction pipe, the maximum calculated pump speed in the reverse direction would be 140 percent of rated. This speed does not result in mechanical motor damage. Longitudinal breaks up to one pipe cross-sectional area were considered.

c) For a guillotine suction pipe break the maximum calculated pump speed in the reverse direction is 710

percent of rated, which is a destructive overspeed of the motor. However, the initial torque for this event is 40 times the rated motor torque and this is sufficient to decouple the motor from the pump by mechanical failure of the pump to motor shaft. Mechanical failure is calculated to occur at 5 to 10 times the rated motor torque with or without a decoupler device in the drive train. Thus an inherent self decoupling would exist for this case.

On November 19, 1976, the NRC wrote GE a letter stating that applicants must file a formal application for amendment of their construction permit or operating license before they would be released from their commitment to install the decoupler.

The letter also stated that "any such application to delete the decoupler from a boiling water reactor design must include a thorough safety evaluation setting forth the reasons why a recirculation pump decoupler is no longer necessary".

GE has completed such a safety analysis report on a generic basis, in a letter from E.A. Hughes (GE) to R.C. DeYoung (NRC), January 18, 1977, "GE Recirculation Pump Potential Overspeed".

It is concluded in the above letter that destructive pump overspeed can result in certain types of missiles. A careful examination of shaft and coupling failures shows that the fragments will not result in damage to the containment or to vital equipment.

(1) Low Energy Missiles (Kinetic energy less than 1,000 ft-lbs):

Low energy level missiles may be created at motor speeds of 300% of rated, through failure of the end structure of the rotor. The structure consists of the retaining ring, the end ring, and the fans. Missiles potentially generated in this manner will strike the overhanging ends of the stator coils, the stator coil bracing, support structures, and two walls of one-half inch thick steel plate. Due to the ability of these structures to absorb energy, it is concluded that missiles would not escape this structure. It is at this point frictional forces would tend to bring the overspeed sequence to a stop.

(2) Medium Energy Missiles (Kinetic energy less than 20,000 ft-lbs):

In the postulated event that the body of the rotor were to burst, medium energy missiles could be created. The likelihood that these missiles would escape the motor is considered less than the likelihood of escape for the low-energy missiles described above, due to the additional amount of material constraining missile escape, such as the stator coil, field coils, and stator frame directly adjacent to the rotor.

(3) The Motor as a Potential Missile:

Since bolting is capable of carrying greater torque loads than the pump shaft, pump bolt failure is precluded. Since pump shaft failure decouples the rotor for the overspeed driving blowdown force, only those cases with peak torques less than that required to fail the pump shaft (five times rated) will have the capability to drive the motor to overspeed. When missile generation probabilities are considered along with a discussion of the actual load bearing capabilities of the system, it is evident that these considerations support the conclusion that it is unrealistic that the motor would become a missile.

It is concluded that the other rotating components inside the containment such as fans and chillers do not have sufficient energy to move the masses of their rotating parts through the housings in which they are contained.

In addition, redundant safety related components are located in different areas of the containment, so that a rotating component failure missile will not damage both redundant components.

3.5.1.2.2 Pressurized Component Failure Missiles 1

A discussion of the potential for missile generation from the failure of pressurized components, e.g. valve stems, valve bonnets, and temperature element assemblies, is presented in Subsection 3.5.1.1.2. That discussion is also applicable to pressurized components inside containment. 1

3.5.1.2.3 Gravitationally Generated Missiles

Components necessary for the operation and safety of the reactor are designed to remain in place and functioning during all design basis conditions. Equipment which is not necessary for operation, startup testing, or safety is removed from the containment or seismically supported and secured in place prior to operation to ensure that it will not become a missile during plant operation or during a safe shutdown earthquake. Therefore, during reactor operation and following a LOCA all equipment inside containment is secured. During maintenance when such equipment is returned to the containment or made operational administrative and procedural methods will be used to ensure that significant damage is not caused to safety equipment even when the reactor is in the shutdown condition. 10 16

3.5.1.3 Turbine Missiles

An analysis was performed to evaluate the probability of damage from postulated turbine missiles to safety related components at Susquehanna SES. The probability of unacceptable damage to safety related components due to turbine missiles has been calculated as 2.61E-10 per unit per year for the two turbine trains. Based on this low probability, the turbine missile hazard is not considered as a design basis event for Susquehanna SES. In the following, the data used in this analysis along with salient features of the analysis are described.

3.5.1.3.1 Turbine Placement and Orientation

The safety related structures are those in which a single strike by a postulated turbine missile could result in a loss of the capability to function in a manner necessary to meet the requirements of 10CFR100.

At Susquehanna SES, these are the reactor buildings, diesel generator building, the control structure, and the ESSW pumphouse.

The locations of these buildings with reference to the turbine are shown on Figure 3.5-5. The figure also shows the ±25 degree missile ejection zone with respect to the low-pressure turbine wheels for each turbine unit within reach of the plant structure.

3.5.1.3.2 Missile Identification and Characteristics

The turbine generators at Susquehanna SES are manufactured by GE. Each unit consists of a tandem-compound six-flow, nonreheat, 1800 rpm turbine, directly connected to a synchronous generator. The turbine has 38 in. last stage buckets.

GE has performed a study (Ref 3.5-1) to determine the characteristics of the missiles which can be expected as a result of a turbine burst. The methodology is discussed in their memo reports on hypothetical turbine missiles (Ref. 3.5-1, 3.5-2 and 3.5-3). Each of the seven stages in a typical LP turbine were analyzed. Significant similarities were found in the dimensions, shapes, weights, and initial energies of missiles from adjacent stages. These similarities justify grouping the stages to simplify the probability calculations. Three stage groups are considered. Stage Group I includes the first three stages. Stage Group II consists of stages four through six. The last stage is included in Stage Group III. It is postulated that four types of missiles can be ejected by wheels in each stage group. These missile types are illustrated by Figure 3.5-1. The characteristics of the missiles postulated for each stage group are given in Table 3.5-1.

3.5.1.3.3 Probability Analysis

The probability of turbine missile damage is expressed as:

$$P4 = P1\ P2\ P3 \qquad \text{(Eq. 3.5-1)}$$

where:

P4 = probability of turbine missile damage, per year

P1 = probability of a turbine failure resulting in the ejection of a missile, per year

P2 = probability that a missile will strike a barrier that houses a critical plant component, given that a missile has been ejected from the turbine, and

P3 = probability that a missile will spall a barrier, thus damaging a critical plant component, given that a missile has been ejected from the turbine and has struck the barrier.

P1 , P2 , and P3 are evaluated using a methodology that considers turbine characteristics, turbine failure mechanisms, plant layout, and barrier types.

The analysis considered 18 missile ejection points representing the two turbine trains. It was assumed that Stage Group I and Stage Group II missiles can be ejected from the centers of the six hoods so an ejection point was placed at each of those locations. Each of these ejection points includes six Stage Group I wheels and six Stage Group II wheels. Stage Group III missiles can originate at each of the two end wheels in each hood. The remaining 12 ejection points were located accordingly.

The procedure for calculating the total P2 and P2xP3 for each target is discussed below. The probabilities for each target were obtained from:

$$P_r = \sum_{i=1}^{6} (W_i^I P_i^I + W_i^{II} P_i^{II}) + \sum_{i=7}^{18} W_i^{III} P_i^{III} \qquad \text{(Eq. 3.5-2)}$$

where the superscripts refer to the Stage Groups and the subscript i refers to a particular ejection point.

In P2xP3 calculations P_i^I is defined as:

$$P_i^I = \text{Max}\ \{(P2 \times P3)_i^{Ia}, (P2 \times P3)_i^{Ib}, (P2 \times P3)_i^{Ic}, (P2 \times P3)_i^{Id}\} \qquad \text{(Eq. 3.5-3)}$$

$$= (P2 \times P3)_i^I$$

where the superscripts a, b, c, and d refer to missile type. In P2 calculations P_i^I is defined as:

$$P_i^I = \text{Max}\ \{(P2)_i^{Ia}, (P2)_i^{Ib}, (P2)_i^{Ic}, (P2)_i^{Id}\} \text{ if } (P2 \times P3)_i^I = 0$$

$$P_i^I = (P2)_i^{Ia} \text{ if } (P2 \times P3)_i^I = (P2 \times P3)_i^{Ia}$$

$$P_i^I = (P2)_i^{Ib} \text{ if } (P2 \times P3)_i^I = (P2 \times P3)_i^{Ib}$$

$$P_i^I = (P2)_i^{Ic} \text{ if } (P2 \times P3)_i^I = (P2 \times P3)_i^{Ic}$$

$$P_i^I = (P2)_i^{Id} \text{ if } (P2 \times P3)_i^I = (P2 \times P3)_i^{Id}$$

P_i^{II} and P_i^{III} are similarly defined.

W_i^I, W_i^{II}, and W_i^{III} are weighting factors associated with each ejection point. They are the probabilities that a particular wheel included in ejection point i fails, given that the turbine has failed. GE states that the wheels fail with equal probability (See References 3.5-1 and 3.5-3). Since there is a total of 42 wheels per turbine, it follows that:

$$W_i^I = W_i^{II} = 6/42 \text{ and}$$

$$W_i^{III} = 1/42$$

The contributions from each turbine are computed in a similar fashion.

If;

$$PT = PT1 + PT2, \text{ then} \qquad \text{(Eq. 3.5-4)}$$

$$PT1 = \sum_{i-1}^{3} (W_i^I P_i^I + W_i^{II} P_i^{II}) + \sum_{i-7}^{12} W_i^{III} P_i^{III} \quad \text{and}$$

$$PT2 = \sum_{i=4}^{6} (W_i^I P_i^I + W_i^{II} P_i^{II}) + \sum_{i-13}^{18} W_i^{III} P_i^{III}$$

Note that if $P_i^I = P_i^{II} = 1$ (i=1,2,3) and $P_i^{III} = 1$ (i=7,--,12) then:

$$PT1 = \sum_{i=1}^{3} (\frac{6}{42} + \frac{6}{42}) + \sum_{i=7}^{12} (\frac{1}{42}) = 3 \times \frac{12}{42} + 6 \times \frac{1}{42} = 1$$

The P4 for the target is obtained by multiplying its P2xP3 by P1. The P2, P2xP3, and P4 for each of the units on the site is computed by summing the probabilities for the targets in the unit. Average values of P3 for the individual targets and units are calculated by dividing P2xP3 by P2.

The analysis assumes that each missile damages at most one target and ignores ricochets. The second assumption is justified by the geometry of the targets which were conservatively defined to be entire buildings. The first assumption avoids double counting the effects of the missile under consideration. The calculation of P2 and P2xP3 is discussed below in detail as is the value of P1 used in the analysis.

3.5.1.3.4 Missile Generation Probability (P1)

In general, two specific overspeed conditions are postulated for which the missile generation probability values (P1) and the missile characteristics are evaluated:

a) Design overspeed (low speed burst): This is 120 percent of rated speed of the turbine and is based on the precept that, should the turbine speed governing system be incapacitated so that the turbine is tripped by the overspeed trip mechanism, the attained speed will not exceed 120 percent of rated speed. Disc failure would occur at this speed as a result of undetected material deficiencies leading to brittle fracture.

b) Destructive overspeed (high speed burst): This is 180 percent of rated speed and is the lowest calculated speed at which any low-pressure rotor disc will burst based on the average tangential stress being equal to the maximum ultimate tensile strength of the disc material, assuming no flaws or cracks in the disc.

At either overspeed condition, it is postulated that the rupture of one disc will do sufficient damage to the unit so that further overspeeding and additional missile generation will not occur.

GE has established that the probability of missile generation at the design overspeed conditions is statistically insignificant. The probability of disc failure leading to the ejection of a missile at the destructive overspeed is calculated by GE as 1.5E-7 in the estimated 30-year life of the turbine. This corresponds to a yearly probability of occurrence of destructive overspeed turbine missiles of 5.0E-9. For further details of this analysis for estimating the missile generation probabilities, reference is made to GE's memo reports. See Ref 3.5-3. Turbine overspeed protection is also discussed in subsection 10.2.2.6.

3.5.1.3.5 Calculation of Strike Probability (P2)

Figure 3.5-2 illustrates a Cartesian coordinate system used to specify the direction of missile ejection from the turbine. The x axis corresponds to the turbine shaft and the y axis is normal to the shaft in the horizontal plane.

The direction of missile ejection is specified by two angles: ϕ subtends the y axis and the projection of the ejection vector on the y-z plane, ψ is the angle from the y-z plane to the ejection vector. Two additional angles are derived from ϕ and ψ : ϕ' is the vertical ejection angle measured from the x-y plane to the ejection vector; ψ' is the horizontal angle subtended by the y axis and the projection of the ejection vector on the x-y plane. The angles are related by the following formulae:

$$\phi' = \sin^{-1} (\sin \phi \cos \psi) \qquad \text{(Eq. 3.5-5)}$$

$$\psi' = \tan^{-1} \frac{\tan \psi}{\cos \phi} \qquad \text{(Eq. 3.5-6)}$$

If the effect of air resistance is discounted, the missile will follow a parabolic trajectory which lies within the vertical plane defined by the formula: $x/y = \tan \psi'$.

An equation for the trajectory may be derived from elementary physics:

$$z = r \tan \phi' - \frac{g r^2}{2 (V \cos \phi')^2} \qquad \text{(Eq. 3.5-7)}$$

In the above equation, r is the horizontal distance from the point of missile ejection; V is the ejection velocity; ϕ' is the vertical ejection angle defined in Equation 3.5-5; and g is the gravitational constant.

From Equations 3.5-5, 3.5-6, and 3.5-7, it can be seen that a missile trajectory is determined by the two angular variables ϕ and ψ, which determine ϕ', and by the ejection velocity V. The principle of the calculation of the strike probability P2 is to determine, out of the range of possible values, the range of ϕ, ψ, and V which define missile trajectories intersecting the target. Functions must be defined, $P(\phi)$, $P(\psi)$, and $P(V)$, which determine the probability of missile ejection over the range of each of the three variables. P2 is then the product of the probability distributions integrated over the range corresponding to missile strike trajectories:

$$P2 = \int_{\phi 1}^{\phi 2} \int_{\psi 1(\phi)}^{\psi 2(\phi)} \int_{V_0(\phi,\psi)}^{V2(\phi,\psi)} P(\phi)\, P(\psi)\, P(v)\, d\phi\, d\psi\, dv \qquad \text{(Eq. 3.5-8)}$$

23

The ejection probability distribution for the angle ϕ is assumed to be uniform over the 360° arc about the turbine axis:

$$P(\phi)\, d\phi = \frac{d\phi}{2\pi} \qquad \text{(Eq. 3.5-9)}$$

The probability distribution for the angle ψ is assumed to be uniform within some range ψmin to ψmax specified by the turbine manufacturer:

$$P(\psi)\, d\psi = \frac{d\psi}{\psi max - \psi min} \; ; \quad \psi min \leq \psi \leq \psi max \qquad \text{(Eq. 3.5-10)}$$

$$P(\psi)\, d\psi = 0; \quad \psi < \psi min \quad \text{or} \quad \psi > \psi max$$

The ejection probability distribution P(V) for the specified range of possible ejection velocities is uniform with V:

$$P(V)\ dV = \frac{dV}{V\ max - V\ min}\ ;\ V\ min \leq V \leq V\ max \qquad \text{(Eq. 3.5-11)}$$

$$P(V)\ dV = 0\ ;\ \ V < V\ min \text{ or } V > V\ max$$

This analysis conservatively neglects the protection against low trajectory missiles by the moisture separator and its radiation shields. This effect would be treated by generating a non-uniform velocity distribution based on equations 3.5-19, 3.5-20, 3.5-21, 3.5-22, 3.5-23 and 3.5-24 and the full range of projected missile rim areas described by equations 3.5-28, 3.5-29 or 3.5-30. The maximum missile velocity would correspond to the residual velocity calculated based on Vmax and the minimum
23 projected rim area. The minimum velocity would correspond to the residual velocity based on Vmin and the maximum projected rim area.

Since equation 3.5-8 is not readily integrable over ϕ and ψ, a numerical integration is necessary to complete the evaluation of P2.

An appropriate range, $\phi 1$ to $\phi 2$, is selected for which a target strike is possible, typically, $0° < \phi < 90°$. The range is divided into I discrete ejection angles, ϕ_i, each representing an angular increment of width $\Delta\phi$.

For each value of $\phi 1$, the angular interval $\psi 1(\phi 1)$ to $\psi 2(\phi 1)$ corresponding to target strike trajectories must be determined. The projection of $\psi 1(\phi 1)$ and $\psi 2(\psi 1)$ on the x-y plane defines the angles $\psi 1'$ and $\psi 2'$ which subtend the target at the missile origin as illustrated on Figure 3.5-3. $\psi 1'$ and $\psi 2'$ may be calculated from simple geometry; $\psi 1(\phi 1)$ and $\psi 2(\phi 1)$ are obtained by inverting Formula (Eq. 3.5-6).

$$\psi_1(\phi_i) = \tan^{-1}(\tan\psi_1' \cos\phi_i) \qquad \text{(Eq. 3.5-12)}$$

$$\psi_2(\phi_i) = \tan^{-1}(\tan\psi_2' \cos\phi_i)$$

If $\psi 1 \geq \psi$ max. or $\psi 2 \leq \psi$min. the target cannot be struck.

This angular range in ψ is divided into J discrete ejection angles ψij, each representing an angular increment of width $\Delta\psi i$ The numerical intergration thus treats J values of ψ for each value of ϕ or a total of I x J missile ejection directions each representing a solid angle of area $\Delta\phi\Delta\psi i$.

For each direction of missile ejection specified by ϕi and ψij Eq. 3.5-7 for the parabolic trajectory is inverted to solve for the range of ejection velocities which correspond to trajectories intersecting the target.

The vertical plane containing the trajectory intersects the target at a minimum distance r1 and a maximum distance r2 from the turbine. Equation 3.5-7 is inverted to solve for V2, the maximum ejection velocity corresponding to a target strike, by inserting r2 for r, and Zr, the target roof elevation relative to the turbine, for Z. If V2>Vmax., V2 is set equal to Vmax. The minimum velocity corresponding to target strike, Vo, can be determined by inserting in Equation 3.5-7 r1 for r and the ground elevation with respect to the turbine for Z. If Vo > Vmin, Vo is set equal to Vmin. If Vo ≥ Vmax no strikes are possible. Thus a missile ejected at velocity V2 will intersect the far edge of the target roof. A missile ejected at velocity Vo will intersect the lower edge of the target wall nearest the turbine. Missiles ejected at velocities greater than V2 will overshoot the target; those ejected at velocities less than Vo will fall short. 23

To distinguish between strikes on the target roof and strikes on the wall, an additional ejection velocity is determined. V1 corresponds to a trajectory intersecting the upper edge of the target wall. It is determined by inserting r1 for r and Zr for Z in Equation 3.5-7. If V1 ≥ Vmax, V1 is set equal to Vmax. The ejection velocity range Vo to V1 then corresponds to wall strikes, the range V1 to V2 to roof strikes. The trajectories corresponding to these three ejection velocities are illustrated on Figure 3.5-4.

For the ejection angles $\phi 1$ and ψij, an increment in the strike probability, P2ij, is calculated for the formula:

$$P2ij = \frac{\Delta\phi}{2\pi}\left(\frac{\Delta\psi i}{\psi \max - \psi \min}\right)\left(\frac{V2\ (\phi i,\ \psi ij) - V0\ (\phi i,\ \psi ij)}{V \max - V \min}\right) \qquad \text{(Eq. 3.5-13)}$$

The increments are summed to obtain the strike probability:

(Eq. 3.5-14)

$$P2 = \sum_{i=1}^{I} \sum_{j=1}^{J} P2ij$$

3.5.1.3.6 Calculation of the Damage Probability (P3)

A missile striking a concrete wall with sufficient impact to cause spalling, which is the ejection of concrete fragments from the inner wall face, may constitute a hazard, even if the wall is not penetrated. Thus, a missile impact that causes spalling from the walls of a target structure is considered the threshold of target damage.

This analysis used the formulation presented below to predict the minimum velocity required to initiate spalling on a barrier.

The data from low velocity missile impact test (Ref. Eq. 3.5-12, 3.5-13) have enabled development of empirical relationships defining the concrete element thickness for threshold of spalling by low velocity solid steel missiles.

23 For solid steel missiles:

(Eq. 3.5-15)

$$Tss = 15.5 \frac{W^{0.4} \; Vs^{0.5}}{\sqrt{f'c} \; D^{0.2}}$$

where

Tss = thickness for threshold of spalling for solid steel missiles (in.)

W = missile weight (lbs.)

D = missile diameter (in.)

f'c = concrete strength (psi)

Vs = missile striking velocity (fps)

Equation (3.5-15) defines the threshold of spalling for low velocity missiles. The Corps of Engineers' equation (Ref. 3.5-14)

is selected to define the threshold of spalling for high velocity nondeformable missiles. This equation is rewritten in the following form:

$$Tcs = \frac{384}{\sqrt{f'c}} \frac{W}{D^{1.785}} \left[\frac{Vs}{1000}\right]^{1.5} + 2.80D$$

where Tcs = thickness for threshold of concrete spalling (Eq. 3.5-16)

For high velocity missiles, this equation is considered more reliable than other available empirical relationships since it is based on the most extensive accumulation of experimental data from tests involving high velocity nondeformable missiles with a large variation of missile size and weight. Also, a comparison of equations (3.5-15 and 3.5-16) reveals a convergence in predicted thicknesses at intermediate velocities.

The thickness versus velocity curves defined by these equations for a particular missile may or may not intersect, depending upon missile weight and diameter. However, the difference in predicted thickness is small at a velocity Vt where the two curves become parallel. For solid steel missiles, the value of Vt would be:

23

(Eq. 3.5-17)

$$Vt = 425 \frac{D^{1.585}}{W^{0.6}}$$

where Vt = transition velocity (fps)

The velocity Vt defines the transition velocity between equations 3.5-15 and 3.5-16. For striking velocity less than Vt, threshold of spalling is defined by equation 3.5-15.

At striking velocities greater than Vt, the value of Ts would be between Tcs and Tss and converge toward Tcs as Vs approaches 2Vt.

The value of T in this velocity range would therefore be closely represented by:

$$T_S = Tcs - (Tcst - Tsst) \cdot \left(2 - \frac{Vs}{Vt}\right); 1 < \frac{Vs}{Vt} < 2 \qquad \text{(Eq. 3.5-19)}$$

where

Vt = transition velocity from equation 3.5-17

Tcst = Tcs at velocity Vt from equation 3.5-16

Tsst = Tss at velocity Vt from equation 3.5-15

For striking velocities greater than 2Vt, the threshold of spalling is determined from equation 3.5-16.

Although the presently available low velocity test data do not allow independent development of perforation formulae in the low velocity range, an improved estimate of the thickness for threshold of perforation can be obtained by correlation of test
23 data with low velocity and high velocity equations for perforation and spalling.

It is noted that the thickness for perforation for low velocity missiles should converge toward that predicted by the Corps of Engineers equation (3.5-19) for perforation as the missile velocity increases and approaches the lower limits of applicability of the Corps of Engineers' equation. This equation can be rewritten in the following form:

$$Tcp = \frac{350}{\sqrt{f'_c}} \frac{W}{D^{1.785}} \left[\frac{Vs}{1000}\right]^{1.5} + 1.94D \qquad \text{(Eq. 3.5-19)}$$

where

Tcp = thickness for threshold of perforation by Corps of Engineers' equation (in.)

The relationship between thickness to spall and thickness to perforate for lower velocity missiles can be obtained from

equations (3.5-16) and (3.5-19) at the transition velocity Vt (Eq. 3.5-17).

The estimated thickness for threshold of perforation Tp for low velocity solid steel missiles would therefore be:

$$Tp = \frac{(Tss)\ (Tcpt)}{Tcst} \qquad Vs \leq Vt$$

(Eq. 3.5-20)

where

Tp = thickness for threshold of perforation

Tss = thickness for threshold of spalling from equation 3.5-15

Tcpt = thickness obtained from equation 3.5-19 with Vs equal to Vt

Tcst = thickness obtained from equation 3.5-16 with Vs euqal to Vt

Vs = striking velocity

Vt = transition velocity for solid steel missiles from equation 3.5-17

As the striking velocity increases above Vt, the thickness for threshold of perforation would be between that given by equation (3.5-20) and Tcp obtained from equation (3.5-19), and would converge toward Tcp as Vs approaches 2 Vt. Therefore, Tp would be closely approximated by: 23

(Eq. 3.5-21)

$$Tp = Tcp - \left[Tcpt - \frac{Tsst\ (Tcpt)}{Tcst}\right] \left[2 - \frac{Vs}{Vt}\right] \quad 1 \leq \frac{Vs}{Vt} \leq 2$$

where

Tsst = the value of Tss from equation (3.5-15) at Vs equal to transition velocity Vt (equation 3.5-17)

When Vs exceeds 2 Vt, Tp is obtained from equation (3.5-19).

Accounting for multiple element concrete barriers involves determination of the residual velocity, Vr, after perforating an element of the barrier and applying this velocity as the striking velocity, Vs, for impact calculations on the next barrier in the series.

Test data indicate that the velocity of spall particles, when the striking velocity is equal to or greater than that for threshold of perforation Vp, see Appendix E, is:

$$Vr = \frac{W\ Vs}{W + \gamma T\ (D + \frac{T}{2})^2} \qquad \text{(Eq. 3.5-22)}$$

where

Vr = residual or spall velocity (fps)

W = weight of missile (pounds)

Vs = striking velocity (fps)

γ = density of concrete (lb./ft.^{3})

D = diameter of missile (ft.)

T = thickness of concrete element (ft.)

Also, when Vs exceeds Vp by about 20%, the residual velocity of the missile can be closely approximated by equation (3.5-22). When Vs is equal to Vp, the spall particle velocity is represented by equation (3.5-22) but the residual velocity of the missile is essentially zero. When the striking velocity is between Vp and that for threshold of spalling, spall velocities can be estimated by a linear interpolation between a value of V equal to 10 fps when Vs is that for threshold of spalling to Vr determined
23 from equation (3.5-22) with Vs equal to Vp.

An estimate of the steel element thickness for threshold of perforation for nondeformable missiles is provided by equation (3.5-23) (which is a more convenient form of the Ballistic Research Laboratory (BRL) equation (Ref. 3.5-15) for perforation of steel plates with material constant taken as unity):

$$Tp = \frac{(Ek)^{2/3}}{672\ D} \qquad \text{(Eq. 3.5-23)}$$

$$Ek = \frac{Mm\ Vs^2}{2}$$

where:

Tp = steel plate thickness for threshold of perforation (in.)

Ek = missile kinetic energy (ft./lbs.)

Mm = mass of the missile (lb.-sec.^{2}/ft.)

Vs = missile velocity (fps)

D = missile diameter (ft.)

It may be necessary to account for several steel elements. Analysis of missile barriers composed of several elements involves determining the residual velocity ,Vr, after perforation of one element and using this value for the striking velocity ,Vs, on the next element. The following formula is used to determine the residual velocity Vr:

$$Vr = (Vs^2 - Vp^2)^{1/2} \quad \text{for } (Vp < Vs) \qquad \text{(Eq. 3.5-24)}$$

$$Vr = 0 \quad \text{for } (Vp \geq Vs)$$

where:

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Vr = residual velocity of missile after perforation of an element of thickness t (fps)

Vs = striking velocity of the missile normal to target surface (fps)

Vp = velocity required to just perforate an element (fps)

In order to apply the equations developed above, it is necessary to use an equivalent diameter. This diameter D, is calculated on the bases of the projected rim area of the turbine misile from:

$$D = \frac{\sqrt{4A}}{\pi} \qquad \text{(Eq. 3.5-25)}$$

where A is the projected area of the missile.

The projected rim area of a missile depends on the missile orientation about its axis.

For fragment group A and fragment group B let:

$$A1 = R3\ T2 + R2\ (T1 - T2) \qquad \text{(Eq. 3.5-26)}$$
$$A2 = (R3 - R2)\ T2 + (R2 - R1)\ T1 \qquad \text{(Eq. 3.5-27)}$$

Where R1, R2, R3, T1 and T2 are illustrated by figure 3.5-1. Values of these parameters for each fragment group are given in Table 3.5-1.

The projected area of fragment group A missiles is then given by:

$$A(\theta) = \begin{cases} A1 - (A1-A2)\cos(\pi/6 + \theta) & 0 \leq \theta \leq \pi/6 \\ A1\,(1 + \sin(\theta-\pi/6)) & \pi/6 < \theta \leq \pi/3 \\ \sqrt{3}A1 \sin\theta & \pi/3 < \theta \leq \pi/2 = \theta s \end{cases} \quad \text{(Eq. 3.5-28)}$$

where θ describes the orientation of the missile. $\theta = 0$ corresponds to the orientation with minimum projected rim area and θs corresponds to the orientation with maximum projected area.

Similarly, for fragment group B missile:

(Eq. 3.5-29)

$$A(\theta) = \begin{cases} A1 - (A1 - A2)\cos(\pi/6 + \theta) & 0 \leq \theta \leq \pi/6 \\ A1 \sin\theta + A2 \cos(\pi/6 + \theta) & \pi/6 < \theta \leq \pi/3 \\ A1 \sin\theta & \pi/3 < \theta \leq \pi/2 = \theta s \end{cases}$$

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For fragment group C and D missile:

(Eq. 3.5-30)

$$A(\theta) = A3\,(\cos\theta + \sin\theta) \qquad 0 \leq \theta \leq \pi/4 = \theta s$$

where A3 = XT2 and X and T2 are illustrated by Figure 3.5-1. Values of X andT2 are given in Table 3.5-1 for each stage group.

Equations 3.5-15, 3.5-16 and 3.5-18 are used to generate ranges of damage initiation velocities Vw0 to Vw1 and Vr0 to Vr1, corresponding to minimum ($\theta = 0$) and maximum ($\theta = \theta s$) projected areas of the missile, for the walls and roof of the target, respectively. If there are barriers adjacent to the target wall or roof, equations 3.5-19, 3.5-20, 3.5-21 and 3.5-22 or 3.5-23 and 3.5-24 are used to increment these velocities so that they correspond to the minimum normal impact velocities which will result in perforation of the barrier system with sufficient residual velocity to spall the target wall or roof.

For normal impact velocities between Vw0 and Vw1, there will be some projected areas for which wall damage is not possible. The

minimum normal impact velocity is calculated for two additional values of θ. The results of these calculations are used to generate a third order polynomial which gives the maximum value of θ for which wall damage is possible for normal impact velocity V.

(Eq. 3.5-31)

$$\theta w\ (V) = AwV^3 - BwV^2 + CwV - Dw \quad (Vwo \leq V \leq Vw1)$$

The probability that wall damage will occur is simply,

$$\theta w\ (V)/\ \theta s \qquad \text{(Eq. 3.5-32)}$$

where it is assumed that the distribution of missile orientation is uniform. A similar expression is developed for the roof.

(Eq. 3.5-33)

$$\theta r\ (V) = ArV^3 - BrV^2 + CrV - Dr \quad (Vro \leq V \leq Vr1)$$

For strikes on the target wall, the calculation of the normal component of impact velocity is simplified since, for parabolic trajectories, the horizontal velocity component is constant, $V\cos\phi'$.

Defining αw as the angle between the wall normal and the vertical plane containing the trajectory, the normal impact velocity component is given by: 23

(Eq. 3.5-34)

$$Vnw = V\cos\ \phi'\ \cos\alpha w$$

The range of the normal components of the missile velocities which result in wall strikes is:

(Eq. 3.5-35)

$$Vno = Vo\cos\ \phi'\ \cos\alpha w \leq Vnw \leq V1\cos\ \phi'\ \cos\ \alpha w = Vn1$$

For each value of ϕi and ψij, an increment in the damage probability, P3ijw, can now be calculated.

Let $\phi' ij = \sin^{-1}(\sin\ \phi i\ \cos\ \psi ij)$ and $Cij = \cos\ \phi ij\ \cos\alpha w$

If $Vn1 \leq Vwo$, $P3inw = 0$

If $Vno \leq Vwo < Vn1 \leq Vw1$

$$P3ijw = \int_{Vwo}^{Vn1} \frac{\theta w\ (Vn)}{\theta s} \frac{d\ Vn}{(V\ max - V\ min)\ Cij} \qquad \text{(Eq. 3.5-36)}$$

(Eq. 3.5-37)

If $V_{no} \le V_{wo} < V_{w1} < V_{n1}$,

$$P3ijw = \int_{V_{w0}}^{V_{w1}} \frac{\theta_w (V_n)}{\theta_s} \frac{dV_n}{(V_{max} - V_{min})\ C_{ij}} + \frac{V_1 - \frac{V_{w1}}{C_{ij}}}{(V_{max} - V_{min})}$$

(Eq. 3.5-38)

If $V_{wo} < V_{no} < V_{n1} \le V_{w1}$

$$P3ijw = \int_{V_{n1}}^{V_{no}} \frac{\theta_w (V_n)}{\theta_s} \frac{d\ V_n}{(V_{max} - V_{min})\ C_{ij}}$$

(Eq. 3.5-39)

If $V_{wo} \le V_{no} < V_{w1} \le V_{n1}$

$$P3ijw = \int_{V_{no}}^{V_{w1}} \frac{\theta_w (V_n)}{\theta_s} \frac{d\ V_n}{(V_{max} - V_{min})\ C_{ij}} + \frac{V_1 - \frac{V_{w1}}{C_{ij}}}{(V_{max} - V_{min})}$$

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(Eq. 3.5-40)

If $V_{no} \ge V_{wo}$, $P3ijw = \frac{V_1 - V_o}{(V_{max} - V_{min})}$

To simplify the calculations for the roof, it was assumed that the roof is at the same elevation as the turbine. This is conservative for slabs which are above the turbine. Under this assumption, the range of vertical velocities with which the missile strikes the roof is:

(Eq. 3.5-41)

$$V_{n2} = V_1 \sin \phi'_{ij} \le V_{nr} \le V_2 \sin \phi'_{ij} = V_{n3}$$

The computation of the increments P3ijr due to roof strikes is similar to the calculation of the increments P3ijw. To obtain

P3ijr, replace Vno with Vn2, Vn1 with Vn3, θw with θr, Vwo with Vro, Vw1 with Vr1, and Cij with sin $\phi'ij$ in the expressions for P3ijw. The probability that the target is damaged, given a turbine failure, is

$$P2 \times P3 = \sum_{i=1}^{I} \sum_{j=1}^{J} \frac{\Delta\phi}{2} \frac{\Delta\psi}{\psi\,max - \psi min} (P3ijw + P3ijr) \qquad \text{(Eq. 3.5-42)}$$

An average P3 for the target is calculated by setting:

(Eq. 3.5-43) 23

$$P3 = \frac{P2 \times P3}{P2}$$

3.5.1.3.7 Probability of Turbine Missile Damage (P4)

Table 3.5-8 summarizes the pertinent data for the targets considered. Table 3.5-9 locates the ejection points used to model the turbine. No barriers for addition to the wall and slabs of the target structures were accounted for.

The results of this probability analysis are shown in Tables 3.5-2 and 3.5-3. Table 3.5-2 presents the values of P2, P3, P2 x P3, and P4 computed for Unit 1 targets due to each of the two turbine trains. The targets, denoted by letter symbols, are marked on Figure 3.5-5. Table 3.5-3 presents the total values of P2, P3, P2 x P3, and P4 for each of the Unit 1 targets and the totals for the entire unit. Because of the symmetrical arrangement of the SSES, the results for Unit 2 are identical. Table 3.5-3 shows that the total probability of turbine missile damage for the two turbine trains is 2.61E-10 per unit per year.

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3.5.1.4 Missiles Generated by Natural Phenomena

Only tornado generated missiles are considered. Table 3.5-4 lists the missiles considered in the design. The structures designed for tornado generated missiles are listed in Table 3.3-2.

3.5.1.5 Site Proximity Missiles

No offsite design basis hazards were identified in Section 2.2.3. Fragments or missiles which might reach the plant walls, even though not associated with a design basis event, would include rifle bullets fired by vandals, fragments from a truck explosion, and gas bottles propelled by pressurized gases.

The velocity decay of fragments due to air friction is given approximately by:

$$V/V_o = \exp \ .004 \ R/W_f^{1/3}$$

where, V, V_o are the present and initial velocities

R is the range in feet

W_f is the fragment weight in ounces

(Ref. 3.5-6).

For a range of 3,000 feet (out beyond the fenceline or to U.S. Highway 11), the ratio V/V_o is equal to 6×10^{-6} for a one ounce fragment, 1.3×10^{-3} for a six ounce fragment.

Maximum theoretical fragment velocities from a TNS detonation are 10,000 to 12,000 fps (feet per second), with a 6,000-8,000 fps representing typical maximum velocities achieved in explosions. Striking velocity of 10-15 fps might be reached at plant range. No damage to the concrete walls of the plant would result from such a fragment.

High velocity rifles can achieve muzzle velocities over 4,000 fps; ordinary rifle muzzle velocity is about 2,500 fps.

The velocity decay law given above does not apply because of the aerodynamic properties of the spinning bullet.

The velocity decay law given abov does not apply because of the aerodynamic properties of teh spinning bullet. However, decay to at least 1/3 muzzle velocity would occur over a 3,000 foot range. The penetration of a two ounce projectile from a high-velocity weapon at point blank range would be about five inches into

concrete of 4,000 psi strength concrete. Rifle bullets could penetrate the metal siding of the roof cap, but would not have a trajectory for energetically entering the spent fuel pool if fired from the fenceline.

Gas bottles, which weigh 100 lbs. or more may achieve velocities of up to 900 fps if they are allowed to fall over, breaking off the valve fitting to create a rocket. The initial surge could create a missile somewhat more severe than tornado missile "D" (285 lbs., 6" diameter by 15' long steel pipe at 230 fps, NRC Standard Review Plan, Sect. 3.5.1.4). It could spall out five inches of concrete at point blank range.

None of these hypothetical situations would cause energetic spalling or penetration of the critical concrete walls, which are designed at 36-inch thickness of 4,000 psi or better reinforced concrete.

Railcars have also been considered as a hypothetical missile hazard to the plant. A rail spur int the reactor building will service the plant from the rail trackage along the west river bank. Use of this former mainline trackage is discontinued, except for plant service. Two factors are expected to eliminate the hazard potential from rumanary or uncontrolled railcars on the spur: 1) the topography is upslope from the river bank; 2) two derailers are in use, and a third will be installed when the plant is in operation.

3.5.1.6 Aircraft Hazards

3.5.1.6.1 Airport Operations

The aircraft operations identified in Subsection 2.2.2.5 which could constitute a potential hazard at the site were take-off/landing movements at the Berwick Airport and commercial flights in the Federal airways.

The Berwick Airport is four miles west-southwest of the site. It has a single grass strip of 2,300 feet, basically east-west, with its axis making an angle of 16° with the plant. It usage is limited to general aviation light aircraft. See table 3.5-5 for a breakdown of aircraft movements at Berwick Airport by type of aircraft. A 3,600 lb. single engine aircraft was chosen for anlyzing aircraft impacts. This represents a conservative choice since 96% of all aircraft movements at Berwick Airport involve aircraft of less than 2,500 lbs.

A survey of the airport traffic is being conducted, including that from a flight school in operation at the airport. Annual movements are estimated at about 13,000; 4,000 in private usage

and 9,000 by the three aircraft of the flight school. Traffic is quite variable, peaking at over 50 daily on weekends, lighter during the week. Operations of twin engine aircraft are estimated at 28 annually, from current observations.

All traffic patterns are used at the airport, but traffic tends to prefer the river as a guide, turning inland from the river and into the approach about two miles from the east end of the runway. Thus traffic in the southeast quadrant of the airport is greater than the northeast quadrant, in which the site lies. The ridgeline running north of Berwick and the site contributes to this traffic preference.

Wind patterns for the airport are 47 percent from the west, 42 percent form the east, and 11 percent null (crosswind or calm). The 13,000 movements estimated for the annual traffic can be allocated 45-55 percent, east to west. Thus eastbound takeoffs (toward the site) are estimated at 2925 annually, and westbound landings would be estimated at 3575 annually. The complement of this traffic, 3575 westbound takeoffs and 2925 eastbound landings, passes over Berwick, away from the site. Eastbound takeoffs of twin engine aircraft are estimated at six annually, with eight westbound landings.

Five Federal Vortac airways near the site were identified in Subsection 2.2.2.5, Figure 2.2-2. Of these, V-106, passing 3.5 miles from the site, has traffic of eights flights daily, or 3,000 annual movements. V-232, nine miles distant, has a traffic potential of about 18,000 movements annually, but actual traffic is about 9,000 movements (Newark-Cleveland, Cleveland-New York and Chicago-New York flights).

There is no scheduled commercial traffic on V-499 or V-164 and only two flights per day on V-188/226. These latter flights are negligible in comparison with V-232 because of the greater distance.

3.5.1.6.2 Aircraft Crash Probability

Airport Operations

A model of the probability of an airplane crash beyond the end of the runway was developed by Eisenhut (1973) and has been adopted as a criterion of the NRC Standard Review Plan for Section 3.5.1.6 (Ref. 3.5-7). This model specifies that the crash probability in an annular sector between four and five miles from the end of a runway and 30° on each side of the runway centerline is 1.2×10^{-8} per square mile for either landing or takeoff by an aircraft in U.S. general aviation.

Commercial Aircraft

A model for the probability that an aircraft crash will result at a distance x normal to its flight path was developed by Solomon (Ref. 3.5-8 and 3.5-9) using a negative exponential distribution. Solomon's model requires estimation of the deviation, which was determined by a 20° angle from the flight path. A deviation is estimated here by referencing Solomon's angle at 14,000 feet, and adding one mile for deviation in the airway. This gives the probability:

$$f(x) = \frac{1}{3.94} \exp(-x/1.97)$$

where x is in miles. The probability that an enroute crash will occur is about 0.45×10^{-9} per flightpath mile, based upon U.S. commercial aviation performance in the period 1970-75 (Ref. 3.5-10). Thus, the crash probability at the site per passage in each of the Federal airways can be estimated by the product of these two probabilities.

V-499	2.5×10^{-11} per square mile (3 miles)
V-106	1.9×10^{-11} per square mile (3.5 miles)
V-164	0.38×10^{-11}
V-232	0.12×10^{-11} per square mile (9 miles)
V-188/226	0.02×10^{-11} per square mile (13 miles)

3.5.1.6.3 Critical Target Area for the Plant

The aircraft crash probability estimates the chance that a given aircraft maneuver (passage, takeoff, or landing) will result in striking the ground within a specific unit area, without regard to the obstruction of surrounding objects. If the assumed flight ray of the aircraft in crashing should pass through an obstructing object, then the probability of crash into that obstructing object will be the same as for the associated unit ground area.

The types of target areas which will apply to the Susquehanna plant, depending on the type of crash event considered, include:

(i) the roof or plan area (augmented by the wingspan of the striking aircraft), which may be projected to an equal ground area regardless of the angle of flight path slope;

(ii) the wall shadow area, Bh cot A, which is the ground area projected by a wall of vertical dimension h, width B

normal to the flight path (and augmented by the wingspan of the striking aircraft), with an aircraft glidepath making an angle A with the ground. Since the Berwick Airport lies at an angle of 16° from the north-south face of the plant, the width normal to the assumed glidepath would be B cost 16° for the east-west walls and B sin 16° for the north-south facing walls; and

(iii) the skid area, associated with a ground strike ahead of a wall, followed by sliding into the wall. The controlling parameters for the target area of this mode are the width of the wall normal to the skid path (augmented by the wingspan of the striking aircraft), and the distance the wreckage will slide.

The structures identified as critical to nuclear safety are the reactor building, the diesel generator building, the control structure adjacent to the reactor building, and the ESSW pumphouse. However, the design of the reinforced portion of these structures is adequate to preclude penetraton or internal spalling of concrete fragments resulting from the design impact of any of the single-engine craft observed at the Berwick Airport. Specifically, a Beechcraft A36 impacting horizontally at 100 mph was used as the model to determine the vulnerability of the critical structures. The modeled impact consists of a 100 mph crash normal to the concrete wall. This glide angle is in general use of aircraft target cross-section analyses (Ref. 3.5-11). The exception to the involnerability of the critical structures to single-engine aircraft impacts is the opening in the reactor building roof for the spent fuel pools. A plane could penetrate the decking material and cause entry of energetic wreckage fragments into the pool. Although the presence of fragments or debris in the pool would not be considered a problem, their energetic entry into the pool has a potential of causing the rupture of the cladding of some of the fuel elements stored there.

For aircraft with an assumed glide angle in crashing of 15°, the zone of impact into the roof cap which could generate pieces of plane wreckage falling into the pool, or propel part of the decking into the pool, is the decking wall ahead of the pool, and the roof from the wall to the far side of the pool. Although an impact into the roof beyond the pool opening could bring portions of the deck sheeting down onto or into the pool, that situation is not expected to result in rupture of fuel elements. Not all impacts ahead of the pool opening would be expected to result in fuel element rupture, since the roof support syste is capable of absorbing considerable impact energy; however, all such impacts are counted as critical. The support system for the metal decking, or roof cap, is designed to isolate collapse of an impacted section without transmitting the collapse throughout the entire cap. The portion of the cap structure over the spent fuel pool openings is limited to four bays supported at the five

locations 26.5, 27.5, 29, 30.5 and 31. (See figures 3.5-6 and 3.5-7). These four bays comprise a width of 108 feet in the plant cross section.

For eastbound aircraft, the roof span from the wall to the far side of th pool involves a decing span of 101 feet, from stations M to S. The roof cap wall exposed to eastbound movements rises about 47 feet above the control structure roof. Aircraft impacting the control structure roof may also skid across it and impact the roof cap wall. A skid distance of 49 feet across the control structure roof must be included in the plant target area. However, the target section of the plant is partially shielded from approaches from the Berwick Airport by the cooling tower of Unit 2. The minimum diameter of ths tower is 301 feet. After augmenting this minimum radius by 16 feet, half of the wingspan for a typical single-engine plane, the shadow of a 16° ray from the airport is found to fall across the center of the control room structure (See figure 3.5-8). Thus, at least half of the wall exposure, and about 60% of the roof area involved, is shielded from eastbound aircraft movements.

For westbound aircraft, the impact exposure involves the roof decking wall opposite the pool opening, and the decking between stations Ua and Q, a span of about 118 feet. There is no skid area for westbound movements. The resultant plant target areas for ssingle-engine craft are tabulated in Table 3.5-6. The width dimensions in the cross-sections are augmented by the 32 foot wingspan assumed as representative of single-engine craft, except for situations involving shadowing. The plant target areas are approximately 15.8×10^{-4} and 5.7×10^{-4} square miles for westbound and eastbound craft, respectively.

For the few twin engine aircraft exposures, it has been established that an impact of the heaviest twin engine aircraft which can use the Berwick Airport could induce spalling in some of the critical structures (specifically, a Cessna 400). Although none of the twin engine aircraft movements observed at the Berwick Airport are as heavy as the Cessna 400, the plant critical structures have been used conservatively as the target area. The results are tabulated in Table 3.5-6. A wingspan of 50 feet and a glide slope of 15° in crashing is used in the calculations. A shadowing of the reactor building by 50% and the control structure by 40%, is used for eastbound movements. The skidding length is based upon a roughness factor of 2.5 in the formula:

$$\text{Skid (mi)} = 2.52 \times 10^{-6} \ (V^2)$$

where V is in miles per hour (Ref 5).

The target area for commercial jet aircraft flying along the trace of the Vortac airways can be computed using the east face of the reactor and diesel generator buildings as an average

exposure. This is not the greatest target area for the plant, but is selected as an average of the exposures around the compass. On the west side, there is shielding from the cooling towers, and there are several ground obstacles to skidding, which comprises the largest portion of target area. As a crash event, a 727-type wingspread (110 feet) and a power-out glide impacting at 290 mph are assumed. The resultant plant target area of about. 04 mi^2 is tabulated in Table 3.5-6.

3.5.1.6.4 Striking Probabilities

The probability that an aircraft might strike the Susquehanna SES, resulting in a potential nuclear safety hazard, is the product of:

the annual traffic (number of aircraft) (3.5.1.6.1)

The crash probability (events per mi^2) (3.5.1.6.2)

The applicable target area (mi^2) (3.5.1.6.3)

The computations for the three types of aircraft considered are:

Single-engine

Eastbound takeoffs: 2925 movements x $1.2 \times 10^{-8}/mi^2 \times 5.7 \times 10^{-4}\ mi^2$ = 2.00×10^{-8} per year.

Westbound landings: 3575 movements x $1.2 \times 10^{-8}/mi^2 \times 15.8 \times 10^{-4}\ mi^2$ = 6.78×10^{-8} per year.

Twin Engine

Eastbound takeoffs: 6 movements x $1.2 \times 10^{-8}/mi^2 \times .0064\ mi^2$ = $.05 \times 10^{-8}$ per year

Westbound landings: 8 movements x $1.2 \times 10^{-8}/mi^2 \times .0064\ mi^2$ = $.17 \times 10^{-8}$ per year.

Commercial Aircraft

In V-232 18,000 movements x $0.12 \times 10^{-11}/mi^2 \times .04\ mi^2$ = $.09 \times 10^{-8}$ per year.

In V-106 3,000 movements x $1.9 \times 10^{-11}/mi^2 \times .04\ mi^2$ = $.23 \times 10^{-8}$ per year.

The sum of these event probabilities at the Susquehanna SES site is about 9.3×10^{-8}.

3.5.2 SYSTEMS TO BE PROTECTED

3.5.2.1 Missile Protection Design Philosophy

Systems that are reviewed for missile protection are listed in Subsection 3.12.2.

For internally generated missiles, protection is provided through basic station component arrangement so that, if equipment failure occurs, the missile does not cause the failure of a Seismic Category I structure or any safety related system. Where it is impossible to provide protection through station layout, suitable physical barriers are provided whose function is either to isolate the missile or to shield the critical system or component. In addition, redundant Seismic Category I components are suitably protected so that a single missile cannot simultaneously damage a critical component and its backup system.

3.5.2.2 Structures Designed to Withstand Missile Effects

Seismic Category I structures are designed to withstand postulated external or internal missiles which may impact them. Table 3.3-2 is a list of the structures designed to withstand external tornado generated missiles, and the safety related equipment which they protect. The missiles are listed in Table 3.5-4.

3.5.3 BARRIER DESIGN PROCEDURES

The structures and barriers are designed in accordance with the procedures detailed in Reference 3.5-5. The procedures include:

a) Prediction of local damage (penetration, perforation, and spalling) in the impact area including estimation of the depth of penetration

b) Estimation of barrier thickness required to prevent perforation

c) Prediction of the overall structural response of the barrier and portions thereof to missile impact.

The use of a ductility ratio higher than 10 but less than the allowables given in Reference 3.5.5 will be governed by the following conditions:

(1) Reinforced concrete barriers

The allowable displacement of reinforced concrete flexure members can be based on an upper limit for plastic hinge rotation r_θ follows:

$$r_\theta = 0.0065 \frac{d}{c} \leq 0.07$$

where

d = distance from compression face to centroid of tensile steel reinforcement (inch)

c = distance from compression face to the neutral axis at ultimate strength (inch)

This condition is given in section C.3.5 of Appendix C and commentary to Appendix C of ACI 349-76.

(2) Steel barriers

To insure the ability of a steel beam to sustain fully plastic behavior and thus to possess the assumed ductility at plastic hinge formation, it is necessary that the elements of the beam section meet minimum thickness requirements sufficient to prevent local buckling failure.

The conditions to preclude local buckling as given in AISC Manual are satisfied.

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3.5.4 REFERENCES

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TABLE 3.5-2

PROBABILITIES FOR UNIT 1 TARGETS DUE TO EACH TURBINE[1]

	UNIT 1 TURBINE				UNIT 2 TURBINE			
Target	P2	P3	P2XP3	P4	P2	P3	P2xP3	P4
A Reactor Building[2]	5.95E-2	.752	4.47E-2	2.24E-10	4.47E-4	.993	4.43E-4	2.22E-12
B Steam Tunnel	3.25E-3	.892	2.90E-3	1.45E-11	1.59E-5	.776	1.23E-5	5.16E-14
C Control Structure	8.31E-4	.882	7.33E-4	3.66E-12	8.31E-4	.882	7.33E-4	3.66E-12
D Diesel Generator Building	3.75E-3	.664	2.49E-3	1.25E-11	8.16E-5	1.00	8.16E-5	4.08E-13
E ESSW Pumphouse	1.04E-5	.956	9.97E-6	4.98E-14	5.71E-6	.967	5.52E-6	2.76E-14

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(1) Unit 2 is symmetrical to Unit 1 so the probabilities are identical.

(2) Includes the spent fuel pool.

TABLE 3.5-3

TOTAL PROBABILITIES FOR UNIT 1[1]

	Target	P2	P3[3]	P2xP3	P4
A	Reactor Building[2]	5.99E-2	.754	4.51E-2	2.26E-10
B	Steam Tunnel	3.27E-3	.891	2.91E-3	1.46E-11
C	Control Structure	1.66E-3	.882	1.47E-3	7.33E-12
D	Diesel Generator Building	3.84E-3	.671	2.57E-3	1.29E-11
E	ESSW Pumphouse	1.61E-5	.960	1.55E-5	7.75E-14
	Total	6.87E-2	.759	5.21E-2	2.61E-10

(1) Unit 2 is symmetrical to Unit 1 so the probabilities are identical.
(2) Includes the spent fuel pool.
(3) P3 = P2xP3/P2

TABLE 3.5-8

TURBINE TARGET PARAMETERS

Target	X1	X2	Y1	Y2	Coordinates (Ft.) Y1	Coordinates (Ft.) Y2	Wall Thickness (In.)	Slab Thickness (In.)
Unit 1 RB North Wall	163.0	163.0	116.5	253.5	729.0	818.1	36.0	-
Unit 1 RB West Wall North of Tunnel Vent Structure	104.0	163.0	116.5	116.5	729.0	818.1	76.0	-
Unit 1 Steam Tunnel Vent Structure North Wall	104.0	104.0	100.3	116.5	729.0	771.0	84.0	-
Unit 1 Steam Tunnel Vent Structure West Wall	66.0	104.0	100.3	100.33	729.0	771.0	54.0	-
Unit 1 Steam Tunnel Vent Structure Above El. 778	66.0	104.0	100.3	116.5	771.0	818.1	36.0	27.4
Control Structure	-66.0	66.00	56.0	100.1	729.0	825.63	36.0	74.5
Unit 1 Fuel Pool	9.5	50.0	144.0	201.0	818.1	819.1	-	0
Unit 1 RB Roof	0.0	163.0	116.5	253.5	818.1	818.1	-	27.0
Unit 2 RB West Wall South of Tunnel Vent Structure	-163.0	-104.0	116.5	116.5	729.0	818.1	36.0	-
Unit 2 Vent Structure West Wall Below El. 771	-104.0	- 66.0	100.33	100.33	729.0	771.0	54.0	-
Unit 2 " " " " Above El. "	-104.0	- 66.0	100.3	100.33	771.0	818.1	34.0	-
Unit 2 " " " " Roof Slab	-104.0	- 66.0	100.33	116.5	818.1	818.1	-	27.0
Unit 2 Spent Fuel Pool	- 30.0	- 9.5	144.0	201.0	818.1	818.1	-	0
Unit 2 RB Roof Slab	-163.0	0.0	116.5	253.5	818.1	818.1	-	27.0
Diesel Generator Building Roof	161.2	285.2	193.5	227.0	737.0	737.0	-	18.0
Diesel Generator Building Roof	211.2	237.2	227.0	261.0	737.0	737.0	-	18.0
Diesel Generator Building Roof	163.21	211.2	277.0	275.5	723.0	723.0	-	18.0
Diesel Generator Building Roof	211.2	237.2	261.0	273.5	723.0	763.0	-	18.0
Diesel Generator Building Roof	237.21	285.2	227.0	273.5	713.0	722.0	-	18.0
ESSW Pump House (South End)	945.0	982.5	320.0	408.5	685.5	716.0	24.0	24.0
ESSW Pump House (North End)	982.5	1000.8	320.0	408.5	615.5	685.5	-	4.0

(1) Relative to an origin air El. 0'0' and the intersection of the plant and with the turbine axis. The positive X-axis runs north of the origin. The positive Y-axis runs east of the origin.

TABLE 3.5-9

TURBINE MISSILE EJECTION POINT PARAMETERS

Missile Ejection Point Coordinates (ft)			Missile Deflection Angle Range
X	Y	t	
195.8	0.0	733.5	0 to 25
187.8	"	"	-5 to 5
179.8	"	"	-25 to 0
161.0	"	"	0 to 25
153.0	"	"	- 5 to 5
145.0	"	"	-25 to 0
126.2	"	"	0 to 25
118.2	"	"	-5 to 05
110.2	"	"	-25 to 0
-110.2	"	"	0 to 25
-118.2	"	"	-5 to 5
-126.2	"	"	-25 to 0
-145.0	"	"	0 to 25
-153.0	"	"	-5 to 5
-161.0	"	"	-25 to 0
-179.8	"	"	0 to 25
-187.8	"	"	-5 to 5
-195.8	"	"	-25 to 0

(1) See Table 3.5-8

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 CONCRETE CONTAINMENT

The Susquehanna primary containments Units 1 and 2 are boiling water reactor, Mark II (over/under) types.

3.8.1.1 Description of the Containment

3.8.1.1.1 General

The primary containment is an enclosure for the reactor vessel, the reactor coolant recirculation loops, and other branch connections of the reactor coolant system. Essential elements of the primary containment are the drywell, the pressure suppression chamber that stores a large volume of water, the drywell floor that separates the drywell and the suppression chamber, the connecting vent pipe system between the drywell and the suppression chamber, isolation valves, the vacuum relief system, and the containment cooling systems and other service equipment.

The primary containment (as shown in Figures 3.8-1 through 3.8-8) is in the form of a truncated cone over a cylindrical section, with the drywell in the upper conical section and the suppression chamber in the lower cylindrical section. These two sections comprise a structurally integrated reinforced concrete pressure vessel, lined with welded steel plate and provided with a steel domed head for closure at the top of the drywell. Connection of the drywell head to the top of the drywell wall is shown on Figure 3.8-9. The drywell floor is a reinforced concrete slab structurally connected to the containment wall as shown on Figure 3.8-10.

The primary containment is structurally separated from the surrounding reactor building except at the base foundation slabs where a cold joint between the two adjoining foundation slabs is provided.

3.8.1.1.1.1 Dimensions

The dimensions of the primary containment are as follows:

a) Inside Diameter

1) Suppression chamber - 88 ft 0 in.

2) Base of drywell - 86 ft 3 in.

3) Top of drywell - 36 ft 4 1/2 in.

b) Height

1) Suppression chamber - 52 ft 3 in.

2) Drywell - 87 ft 9 in.

c) Thickness

1) Base foundation slab - 7 ft 9 in.

2) Containment wall - 6 ft 0 in.

3.8.1.1.2 Base Foundation Slab

The containment base foundation slab is a 7 ft 9 in. thick reinforced concrete mat. The top of the base foundation slab is lined with a carbon steel liner plate.

3.8.1.1.2.1 Reinforcement

The base foundation slab is reinforced with #18, Grade 60 rebar at top and bottom faces. The average rebar spacing is 18 in. Shear reinforcement consists of #8 and #9 vertical and inclined ties. Mechanical ("Cadweld") splices are used for splicing all main reinforcing bars. Figure 3.8-11 shows plan and section views of reinforcement.

3.8.1.1.2.2 Liner Plate and Anchorages

The steel liner plate is 1/4 in. thick and is anchored to the concrete slab by structural steel beams embedded in the concrete and welded to the plate. See Figure 3.8-12 for details of the liner plate and anchorages. All liner plate weld seams less than 1/2 inch thick are provided with a leak chase system.

3.8.1.1.2.3 Pedestal and Suppression Chamber Column Base Liner Anchorages

Figures 3.8-13 and 3.8-14 show the base foundation slab liner anchorages for the reactor pedestal and the suppression chamber columns respectively. For the pedestal anchorage, B-series "Cadweld" sleeves are welded to the top and bottom surfaces of the thickened base liner to permit anchorage of the pedestal vertical rebar into the base foundation slab. Metal studs are welded to the top and bottom surfaces of the thickened base liner in order to transfer radial and tangential shear forces from the pedestal to the base foundation slab. For the suppression chamber column anchorage, pipe caps are welded to the thickened base liner, where the column anchor bolts penetrate the base liner, to ensure the leak-tight integrity of the base liner.

3.8.1.1.3 Containment Wall

The containment wall is a 6 ft 0 in. thick reinforced concrete wall. The inside surface of the containment wall is lined with a carbon steel liner plate.

3.8.1.1.3.1 Reinforcement

The containment wall is reinforced with #18, Grade 60 rebar at inner and outer faces. The inner rebar curtain consists of two meridional layers and one hoop layer. The outer rebar curtain consists of one meridional layer, two hoop layers and two helical layers. Shear reinforcement consists of #6 horizontal and inclined ties. Mechanical ("Cadweld") splices are used for splicing all main reinforcing bars. Figures 3.8-15 and 3.8-16 show section and developed elevation views of suppression chamber and drywell wall reinforcement respectively.

3.8.1.1.3.2 Liner Plate and Anchorages

The steel liner plate is 1/4 in. thick and is anchored to the concrete wall by structural tee vertical stiffeners spaced horizontally every 2 ft. Horizontal plate stiffeners and horizontal structural channels spaced vertically every 5 ft provide additional stiffening. See Figures 3.8-17 and 3.8-18 for details of the liner plate and anchorages.

Around the containment liner plate penetrations, the liner is reinforced in accordance with ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition. See Subsection 3.8.1.1.3.3 for a further description of penetrations. Loads from internal containment attachments such as beam seats and pipe restraints are transferred directly into the containment concrete wall. This is accomplished by thickening the liner plate and attaching to it structural weldments to transfer to the concrete any type of load without relying on the liner plate or its anchorages. Where internal containment attachment loads are large, the structural weldments penetrate the liner plate rather than being welded to opposite sides of the liner plate. This was done to eliminate the possibility of lamellar tearing. See Subsection 3.8.1.1.3.4 for a further description of internal containment attachments.

3.8.1.1.3.3 Penetrations

General

Services and communications between the inside and outside of the containment are performed through penetrations. Basic penetration types include the drywell head, access hatches (equipment hatches, personnel lock, suppression chamber access hatches, CRD removal hatch), pipe penetrations, and electrical penetrations. Penetrations consist of a pipe with a plate flange welded to it. The plate flange is embedded in the concrete wall and provides an anchorage for the penetration to resist normal operating and accident pipe reaction loads. The pipe is also welded to the containment liner plate to provide a leak-tight penetration.

Meridional and hoop reinforcement are bent around typical penetrations as shown on Figures 3.8-19 and 3.8-20. Additional local reinforcement in the hoop and diagonal directions is added at all large penetrations as shown on Figures 3.8-19 and 3.8-20. Local thickening of the containment wall at penetrations is generally not required. See Subsection 3.8.2.1 for a further description of penetrations.

Pipe Penetrations

Details of typical pipe penetrations are shown on Figure 3.8-21. There are two basic types of pipe penetrations. For piping systems containing high temperature steam or water, a sleeved penetration is furnished, thereby providing an air gap between the containment concrete wall and the hot pipe. This air gap is large enough to maintain the concrete temperature in the area of the penetration below 200°F. A flued head outside the containment connects the process pipe to the pipe sleeve. For piping systems containing low temperature water, an unsleeved penetration is furnished. For this type of penetration, the process pipe is welded directly to the pipe penetration.

Electrical Penetrations

Figure 3.8-22 shows a typical electrical penetration assembly used to extend electrical conductors through the containment. The assembly is sized to be inserted in the 12 in., Schedule 80 penetration nozzles that are furnished as part of the containment. The penetrations are hermetically sealed and provide for leak testing at design pressure.

Equipment Hatches and Personnel Lock

Two 12 ft 2 in. I.D. equipment hatches are furnished in the drywell wall. One of these equipment hatches includes an 8 ft 7 in. I.D. personnel lock. Figure 3.8-23 shows details of reinforcement around the equipment hatches. Additional meridional, hoop, helical, and shear reinforcement is provided to account for local stress concentrations at the opening. The shell is thickened at the equipment hatches to accommodate the additional rebars.

Drywell Head Assembly

The drywell head lower flange assembly is anchored to the top of the drywell wall by one-third (108) of the total number of meridional reinforcing bars in the inner curtain as shown on Figure 3.8-9.

Suppression Chamber Access Hatches

Two 6 ft 0 in. I.D. access hatches are furnished in the suppression chamber wall. Figure 3.8-24 shows a detail of reinforcement around the suppression chamber access hatches. Additional local reinforcement in the meridional, hoop, and diagonal directions is added as shown on Figure 3.8-24.

3.8.1.1.3.4 Internal Containment Attachments

Drywell Floor Embedments

The drywell floor is attached to the containment wall by a structural weldment at the junction of the two structural components shown on Figure 3.8-10. Radial force and bending moment carried by the drywell floor main reinforcement is transferred to the containment wall by cadwelding the drywell floor rebar to the top and bottom flanges of the structural weldment. The top and bottom flanges of the structural weldment penetrate the thickened containment liner plate and are embedded deeply into the containment concrete wall. Flexural shear in the drywell floor is transferred to the containment wall through the web of the structural weldment, which is welded to opposite sides of the containment liner plate.

Beam Seat Embedments

Beam seats are provided to support the drywell platforms. A typical beam seat embedment is shown on Figure 3.8-25.

Pipe Restraint Embedments

Pipe restraints are provided to prevent pipe whip for all high energy piping systems. Typical pipe restraint embedments are shown on Figure 3.8-26.

Seismic Truss Embedments

The seismic truss provides lateral support for the reactor vessel. A typical seismic truss embedment in the drywell wall is shown on Figure 3.8-27.

Snubber Embedments

Snubbers dampen the vibratory motion of piping systems due to seismic or any other dynamic loading. A typical snubber embedment in the drywell wall is shown on Figure 3.8-28.

3.8.1.1.3.5 External Containment Attachments

There are no major external structural attachments. A 2 in. wide separation gap is provided between the containment and the surrounding reactor building to prevent interaction of the two structures. The only place where the containment is in contact with the reactor building is at the base foundation slabs where a cold joint between the two adjoining foundation slabs is provided.

3.8.1.1.3.6 Steel Components Not Backed by Structural Concrete

A description of steel portions of the containment that are not backed by concrete, such as the drywell head, equipment hatches, personnel lock, suppression chamber access hatches, CRD removal hatch, and piping and electrical penetrations, is given in Subsection 3.8.2.

3.8.1.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design and construction of the containment are listed in Table 3.8-1 and given a reference number.

The reference numbers for the concrete containment are 10A, 12A, 1C, 2C, 3C, 6C and 2K.

The reference numbers for the liner plate and anchorages are 4C, 1H, 1J and 1K.

3.8.1.3 Loads and Loading Combinations

3.8.1.3.1 General

Table 3.8-2 lists the loading combinations used for the design and analysis of the containment. The loading combinations are in compliance with those given in Reference 12A of Table 3.8-1. The loading combinations shown in Table 3.8-2 do not include the hydrodynamic loads.

The containment has also been analyzed and designed for hydrodynamic loads from main steam safety/relief valve discharge and LOCA. For a definition of these loads and loading combinations including hydrodynamic loads, refer to GEs "Mark II Containment Dynamic Forcing Functions Information Report" (NEDO-21061), and the "Susquehanna Plant Design Assessment Report".

3.8.1.3.2 Description of Loads

Normal Loads: Those loads encountered during normal plant operation and shutdown, including dead loads, live loads, thermal loads due to operating temperature, and other permanent loads

contributing stress such as hydrostatic loads. Dead and live loads are described in Subsection 3.8.1.3.2.1 and 3.8.1.3.2.2 respectively.

Severe Environmental Loads: Those loads sustained during severe environmental conditions, including those induced by the operating basis earthquake (OBE) and the design basis wind. Loads due to OBE are discussed in Section 3.7 and Subsection 3.8.1.3.2.6. Wind loads are discussed in Section 3.3.

Extreme Environmental Loads: Those loads sustained during extreme environmental conditions, including those induced by the safe shutdown earthquake (SSE) and the design basis tornado. Loads due to SSE are discussed in Section 3.7 and Subsection 3.8.1.3.2.6. Tornado loads are discussed in Section 3.3.

Abnormal Loads: Those loads sustained during abnormal plant conditions. Such abnormal plant conditions include the postulated rupture of high-energy piping. Loads induced by such an accident include elevated temperatures and pressures within or across compartments, and jet impingement and impact forces associated with such ruptures. Loads due to postulated rupture of piping are discussed in Section 3.6.

3.8.1.3.2.1 Dead Load

Dead load includes the weight of the structure plus any other permanent loads contributing stress, such as hydrostatic loads.

3.8.1.3.2.2 Live Load

Live load includes those loads expected to be present when the plant is operating, such as movable equipment, piping, cables, and lateral earth pressure.

3.8.1.3.2.3 Design Basis Accident Pressure Load

The design basis accident (DBA) is defined as a loss of coolant accident (LOCA) that produces the largest containment pressure. Transients resulting from the design basis accident are presented in Subsection 6.2.1 and serve as the basis for the containment internal design pressure of 53 psig.

3.8.1.3.2.4 Thermal Loads

The temperature gradients through the containment wall are shown on Figure 3.8-29 for the operating and the postulated design accident conditions. The design accident temperature gradient shown on Figure 3.8-29 occurs five minutes after LOCA. This transient temperature gradient is used for the design of the containment since it produces the largest stresses in the structure.

Thermal effects anticipated at the time of the structural acceptance test are insignificant because changes in temperature inside and outside the containment during the Unit 1 structural acceptance test were small. Therefore, thermal effects at the time of the structural acceptance test are insignificant.

3.8.1.3.2.5 Wind and Tornado Loads

Wind and tornado loads are not considered because the containment is surrounded by the reactor building.

3.8.1.3.2.6 Seismic Loads

a) Loads from the Operating Basis Earthquake result from ground surface horizontal acceleration of 0.05 g, and vertical ground surface acceleration of 0.033 g, acting simultaneously.

b) Loads from the Safe Shutdown Earthquake result from ground surface horizontal acceleration of 0.10 g, and vertical ground surface acceleration of 0.067 g, acting simultaneously.

3.8.1.3.2.7 External Pressure Load

The containment shell is designed to withstand an external pressure of 5 psi differential.

3.8.1.3.2.8 Missile and Pipe Rupture Loads

The containment wall is designed to withstand the missile and pipe rupture loads due to a postulated rupture of a 26 in. diameter main steam pipe, which produces the largest loads on the

containment wall. These loads include the effects of jet impingement, pipe whip, and pipe reaction. An equivalent static load of 1000 kips is considered. This load includes an appropriate dynamic load factor to account for the dynamic nature of the load. See Section 3.6 for a further discussion of postulated pipe rupture loads.

3.8.1.4 Design and Analysis Procedures

3.8.1.4.1 General

This subsection describes the procedures used for the design and analysis of the containment. The description does not include the effects of hydrodynamic loads from main steam safety/relief valve discharge and LOCA. For a description of the design and analysis procedures that consider the effects of hydrodynamic loads refer to GE's "Mark II Containment Dynamic Forcing Functions Report" (NEDO-21061) and the "Susquehanna Plant Design Assessment Report".

The analysis procedure consists of two parts. First, the uncracked forces, moments, and shears for both axisymmetric and non-axisymmetric loads are determined. Axisymmetric loads are dead load, live load, design accident pressure load, vertical seismic load, and operating and design accident thermal loads. Non-axisymmetric loads are horizontal seismic load and localized missile and pipe rupture load. The second part consists of taking into account the expected cracking of the concrete and determining the concrete and reinforcing steel stresses and strains. The liner plate is not considered to resist any load.

The 3D/SAP computer program (Appendix 3.8A) is used to determine the uncracked forces, moments, and shears due to axisymmetric loads. The operating and design accident temperature gradients are computed using ME 620 computer program (Appendix 3.8A). For transient loads such as design accident pressure and thermal loads, the most critical combination of these loads is considered.

The forces, moments, and shears in the uncracked structure due to seismic loads are determined per Bechtel Topical Report BC-TOP-4-A (Ref. 2K of Table 3.8-1.) The effect of variations in the values of structural and foundation parameters on the modal frequencies is considered. See Section 3.7 for a description of the containment seismic analysis. The 3D/SAP program is used to analyze the containment for non-axisymmetric loads due to missile and postulated pipe rupture.

The CECAP computer program (Appendix 3.8A) is used to determine the extent of concrete cracking and the concrete and rebar stresses and strains. The input data for the CECAP program consists of the uncracked forces, moments, and shears calculated by the 3D/SAP and seismic analysis programs. The CECAP program models a single element of unit height, unit width, and depth equal to the thickness of the wall or slab. The program assumes isotropic, linear elastic material properties and uses an iterative technique to obtain stresses considering their redistribution due to cracking. The program determines the redistribution of thermal stresses due to the relieving effect of concrete cracking.

3.8.1.4.2 Containment Wall

Figure 3.8-30 shows the 3D/SAP finite element model used to analyze the containment wall for axisymmetric loads. A 10 degree wedge of the containment is modeled using solid finite elements having linear elastic, isotropic material properties. The model includes the containment wall, base foundation slab, drywell floor, reactor pedestal and the foundation material. Boundary conditions are imposed on the analytical model by specifying nodal point forces or displacements. Referring to Figure 3.8-30, the nodal points lying along Boundary A are allowed to move within the X-Z plane, and Boundary B within the X-Y plane. Points along Boundary C are prevented from moving in the radial direction and points along Boundary D are prevented from moving in the hoop direction. Nodal forces, moments, and shears are applied to Boundaries E and F to account for reaction loads from the drywell head and reactor vessel and reactor shield wall respectively.

Figure 3.8-31 shows the 3D/SAP finite element model used to analyze the drywell wall for non-axisymmetric missile and pipe rupture loads. A 180 degree half model of the drywell wall consisting of linear elastic, isotropic, solid finite elements is used. Referring to Figure 3.8-31, the nodal points lying along Boundary A are allowed to move within the X-Z plane. Points along Boundary B are prevented from moving in the vertical and radial directions. Nodal forces, moments, and shears are applied to Boundary C to account for reaction loads from the drywell head.

Tangential shears caused by seismic loads are totally resisted by helical reinforcing bars and concrete. No tangential shear is taken by the concrete. The tangential shear is considered as diagonal tension and compression components. The helical reinforcing bars resist diagonal tension and the concrete resists diagonal compression. In calculating the reinforcing steel requirement, the helical reinforcement is designed to resist stresses due to

design accident pressure and thermal loads as well as tangential shears caused by seismic loads.

3.8.1.4.3 Base Foundation Slab

Figure 3.8-32 shows the 3D/SAP finite element model used to analyze the base foundation slab. A 180 degree half model of the base foundation slab consisting of linear elastic, isotropic, solid finite elements is used. The model includes the base foundation slab, a portion of the containment wall and the foundation material. Referring to Figure 3.8-32, the nodal points lying along Boundary A are allowed to move within the X-Z plane, and Boundary B within the X-Y plane. Points along Boundary C are prevented from moving in the radial direction. Axisymmetric forces, moments, and shears calculated using the 3D/SAP containment model and seismically-induced, tangential shears are applied to Boundary D. The height of the model is chosen so that the overturning moment caused by the tangential shear is the same as the overturning moment determined by the seismic analysis. In order to be able to consider uplifting of the base foundation slab from its foundation, a thin layer of foundation material is provided immediately beneath the foundation slab. If the computer output indicates tension in any of these thin foundation elements, the modulus of elasticity of these elements is reduced to almost zero. Then a second computer run is made and any additional uplift is identified. Further iterations and modifications of foundation material properties are made until the complete extent of uplift is determined. Uplift does not result in overstressing the containment foundation.

3.8.1.4.4 Analysis of Areas Around Equipment Hatches

Figure 3.8-33 shows the 3D/SAP finite element model used to analyze the areas of the containment wall around the equipment hatches. A 60 degree wedge of the containment wall is modeled using solid finite elements having linear elastic, isotropic material properties. To reduce the size of the analytical model, Boundary A follows the vertical plane of symmetry of the equipment hatch. The points delineating the outermost boundaries of the model are located at a sufficient distance from the opening so that the behavior of the model along the boundaries is compatible with that of the undisturbed shell. Referring to Figure 3.8-33, the nodal points lying along Boundary A are allowed to move within the X-Z plane, and Boundary B within the X-Y plane. Points along Boundary C are prevented from moving in the hoop direction. Axisymmetric forces, moments, and shears calculated using the 3D/SAP containment model are applied to

Boundary D. Seismic loads calculated by the seismic analysis are applied locally to the elements. Seismically induced, tangential shears around the equipment hatches are resisted by helical reinforcing bars and concrete in compression.

3.8.1.4.5 Liner Plate and Anchorages

The design and analysis of the liner plate and anchorages is per Bechtel Topical Report BC-TOP-1 (Ref. 1K of Table 3.8-1).

3.8.1.5 Structural Acceptance Criteria

3.8.1.5.1 Reinforced Concrete

3.8.1.5.1.1 Working Stress Method

The preoperational testing condition listed in Table 3.8-2 is designed according to the stress limitations of ACI 318, Section 8.10 except that the maximum permissible tensile stress for reinforcement shall be 0.5 Fy. This criterion conforms to Reference 12A of Table 3.8-1.

3.8.1.5.1.2 Strength Method

The factored load combinations listed in Table 3.8-2 are designed according to the strength method of ACI 318. The following allowable stresses are used:

a) Concrete

   1) Compression - 0.85 f'c

   2) Tension - not permitted

   3) Radial shear - ACI 318-71 (Chapter 11)

   4) Tangential shear - not permitted

b) Reinforcing Steel

   1) Tension - 0.90 Fy

   2) Compression - 0.90 Fy

The allowables are defined as:

f'c = Specified compressive strength of concrete

Fy = Specified yield strength of reinforcing steel

3.8.1.5.2 Liner Plate and Anchorages

The allowable strain in the liner plate due to design basis accident thermal load is 0.5 percent. This value is based on ASME Code, Section III (Ref. 1J of Table 3.8-1), Figure I-9.1 which permits an allowable strain of approximately 2 percent for
1 10 cycles. Since the graph in Figure I-9.1 does not extend below 10 cycles, 10 cycles are conservatively used for the DBA instead of one cycle.

The allowable forces on the liner plate anchorages are in accordance with Bechtel Topical Report BC-TOP-1 (Ref. 1K of Table 3.8-1).

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

3.8.1.6.1 Concrete Containment

The concrete and reinforcing steel materials for the containment
11 are discussed in Appendix 3.8B. Concrete design compressive strengths are given in Table 3.8-11.

3.8.1.6.2 Liner Plate, Anchorages, and Attachments

3.8.1.6.2.1 Materials

Liner plate materials conform to the requirements of the standard specifications listed below:

Item	Specification
Liner plate (less than 1/2 in. thick)	ASTM A 285, Grade A

Liner plate (1/2 in. thick or thicker)	ASME SA-516, Grade 60 or 70 conforming to the requirements of ASME Boiler and Pressure Vessel Code (ASME B&PV Code), 1971 Edition with Addenda through Summer 1972, Section III, Article NE-2000
Anchorages and attachments other than pipe restraints	ASTM A36
Pipe restraint attachments	ASTM A441

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3.8.1.6.2.2 Welding

Liner plate and structural steel welding conform to the applicable portions of Part UW of Section VIII of the ASME B&PV Code. Specifically, Paragraph UW-26 through UW-38 inclusive apply in their entirety. The welding of liner plate butt welds and attachments that penetrate the liner plate is performed by either the shielded metal arc or the automatic submerged arc process. The minimum number of individual weld layers for welds that must maintain leak-tightness is two. Welders and weld procedures are qualified in accordance with either Section IX of the ASME Code or AWS D1.1.

3.8.1.6.2.3 Materials Testing

Liner plate material 3/4 in. thick or over is impact tested at 0°F or below as required by the ASME Code. Liner plate or attachment material subjected to transverse tensile stress is vacuum degassed and ultrasonically tested in accordance with ASME Code, Section III, NB-2530 and conforms to the requirements of Article NE-2000 of Section III.

3.8.1.6.2.4 Nondestructive Examination of Liner Plate Seam Welds

Nondestructive examination of liner plate welds is performed in accordance with Regulatory Guide 1.19, Revision 1 except that for leak chase testing, the leak chase pressure is 115 percent of design pressure instead of 100 percent of design pressure, and the pressure is held for 15 minutes instead of two hours. This exception is considered justifiable since any significant leakage (i.e. any pressure decay in excess of the rated accuracy of the pressure gage) will be determined within 15 minutes.

Spot radiographic examination is performed for all radiographable liner plate seam welds. Radiography is performed in accordance with Section V, Article 3 of the ASME Code. Personnel performing radiographic examinations are qualified in accordance with the Society for Non-Destructive Testing's Recommended Practice No. SNT-TC-1A, Supplement A, plus any additional requirements of the ASME Code, Section V. Acceptance standards are in accordance with Paragraph UW-51, of Section VIII, Division 1 of the ASME Code. The first 10 ft of weld for each welder and welding position is 100 percent radiographed. Thereafter, one 12 in. long radiograph is taken for each welder and weld position in each additional 50 ft increment of weld. A minimum of 2 percent of all liner seam welds are examined by radiography. For

nonradiographable welds, the length of weld needed to meet the 2 percent requirement is accounted for by additional radiographs of that length for the accessible welds.

Where nonradiographable weld joints are used, the entire length of weld is magnetic particle examined. All magnetic particle examinations conform to the ASME Code, Section V. Personnel performing magnetic particle examinations are qualified in accordance with SNT-TC-1A plus any additional requirements of the ASME Code, Section V. Acceptance standards are in accordance with the ASME Code, Section VIII, Division 1, Appendix VI.

The vacuum box soap bubble test is performed on all accessible liner plate weld seams. A 5 psi minimum pressure differential is maintained for a minimum time of 20 seconds. The leak detecting solution is continuously observed for bubbles that indicate leaks. If a leak is detected, the defective weld is repaired and reinspected by vacuum box testing.

Welds that are inaccessible for vacuum box testing are 100 percent liquid penetrant tested. Liquid penetrant examinations conform to the ASME Code, Section V. Personnel performing liquid penetrant examinations are qualified in accordance with SNT-TC-1A plus any additional requirements of the ASME Code, Section V. Acceptance standards conform to the ASME Code, Section VIII, Division 1, Appendix VIII.

A leak chase system is provided on liner plate seam welds less than 1/2 in. thick on the base foundation slab liner plate and on that portion of the suppression chamber wall liner plate that is below the suppression pool water level. This system will allow periodic leak testing of welds that are submerged in the suppression pool. It also provides a secondary leak-tight barrier at the liner plate weld seams. Following installation of the leak chase system, the leak chase system is pressurized to 63 psig. The pressure is monitored by valving off the air supply and measuring any pressure decay with a pressure gage. Any pressure decay in excess of the rated accuracy of the pressure gage within 15 minutes is cause for rejection of that portion of the liner plate seam welds and the leak chase system. Any leaks are repaired, and following repair, the affected portion of the leak chase system is retested.

3.8.1.6.2.5 Quality Control

Quality control requirements are discussed in Appendix D and amendments to the PSAR for the construction phase.

3.8.1.6.2.6 Erection Tolerances

The specified erection tolerances for the liner plate are as follows:

a) The slope of any 10 ft section of cylindrical liner plate, referred to true vertical, does not exceed 1:180. The deviation from theoretical slope of any 10 ft section of conical liner plate, measured within a vertical plane, does not exceed 1:120.

b) The cylindrical shell is plumb within 1/400 of the height. The vertical axis of the conical shell, as established at the top and bottom of the conical section, is plumb within 1/400 of the height.

c) The radial dimension to any point on the liner plate does not vary from the design radius by more than ± 1 in., and at any given elevation the maximum diameter minus the minimum diameter shall not exceed 4 in., except that there is a radial tolerance of ± 2 in. for local out-of-roundness. Radial measurements are taken at 24 locations spaced equally around the containment at any elevation. Local out-of-roundness tolerance is used for not more than two measurements at any given elevation and is not used at adjacent measurements.

d) Plates joined by butt welding are matched accurately and retained in position during the welding operation. Misalignment in completed joints shall not exceed the requirements of Paragraph UW-33 of Section VIII, Division 1 of the ASME Code.

e) The levelness of anchorages placed in the base foundation slab is within ±1/4 in. of the theoretical elevation over the entire area, plus a local tolerance of ±1/8 in. in any 30 ft length.

Actual deviations from the above were handled in accordance with the procedures covered in Subsection 3.8.1.6.2.5.

3.8.1.7 Testing and In-service Surveillance Requirements

3.8.1.7.1 Preoperational Testing

3.8.1.7.1.1 Structural Acceptance Test

This subsection briefly describes the Unit 1 containment structural acceptance test. For a more detailed description, refer to the "SSES, Unit 1 Containment Structure, Structural Integrity Test Report".

The Unit 1 containment structural acceptance test was performed after completion of the containment structure but prior to installation of piping and equipment. The reactor vessel was installed at the time of the test and the suppression chamber was filled with water to the normal level. The Unit 2 containment structural acceptance test will be performed after completion of the containment including all piping and equipment. The Unit 1 test was a prototype test and, therefore, internal concrete strains were measured. The Unit 2 test will be a non-prototype test and, therefore, internal concrete strains will not be measured.

The Unit 1 test was done and the Unit 2 test will be done in accordance with Regulatory Guide 1.18, Revision 1, except for the following:

a) A continuous increase in containment pressure, rather than incremental pressure increases, was used. This is considered justifiable since data observations at each pressure level were made rapidly. Rapidly is defined as requiring a time interval for the data point sample sufficiently short so that the change in pressure during the observation would cause a change in structural response of less than five percent of the total anticipated change. Also, the maximum rate of pressurization was limited to 3 psi/hr to ensure that the structure would respond to the pressure load without any time lag.

b) The distribution of measuring points for monitoring radial deflections was selected so that the as-built condition could be considered in the assessment of the general shell response. In general, the locations of measuring points for radial deflections was in agreement with Regulatory Guide 1.18, Figure B, except point 1. Point 1 was provided at a distance of two times the wall thickness (12 ft) above the base mat. This variation was made to properly predict the containment behavior

near the base mat to wall connection. If point 1 was provided at a height of three times the wall thickness (18 ft), it would be located close to point 2 (suppression chamber wall midheight is 26 ft) and would not yield any additional behavior pattern of the containment.

c) Some of the strain gage instrumentation was farther from the equipment hatch than 0.5 times the wall thickness (3 ft) as required by Regulatory Guide 1.18, Paragraph C.5. This was required in order to clear reinforcement and is considered justifiable since the intent of the Regulatory Guide, ie, to demonstrate the structural integrity of the containment, was met.

d) Tangential deflections of the containment wall adjacent to the equipment hatch were not measured because the predicted values of tangential deflection were small and it would have been difficult to obtain fixed reference points for measurement of local tangential deflections.

e) Triaxial concrete strain measurements were not used to evaluate the concrete strain distribution because the measured strain values could not be properly interpreted. The difficulty in interpreting the data was due to the large size of the strain gages relative to the wall thickness. The concrete stain was evaluated using linear strain measurements in the meridional and hoop directions.

f) Humidity inside the containment was not measured during the test since it does not affect the response of the structure.

The containment was pneumatically pressurized to 1.15 times the design accident pressure as shown on Figure 3.8-34. The drywell floor was tested to 1.15 times the design downward differential pressure.

Structural measurements were taken at peak pressure and peak differential pressure as well as at intermediate stages. Measured structural data include the following:

1) Radial and vertical deflections of the containment

2) Internal concrete strains

3) External concrete surface cracks.

The above data were measured for the containment and for the largest opening which are the two equipment hatches. Since the

areas of the containment wall around the equipment hatches are of identical design, only one of the hatches was instrumented. See Figures 3.8-35 and 3.8-36 for the locations of deflection measuring devices for the containment and the equipment hatch respectively. See Figures 3.8-37 and 3.8-38 for the location of strain gage instrumentation for the containment and the equipment hatch respectively. Strain gages were located within the walls and slabs at the rebar layers in the direction of the main reinforcement. An inspection of external concrete surface cracks was performed at six locations. Each crack inspection area was at least 40 sq ft. Figure 3.8-39 shows the locations of the crack mapping areas.

Deflections and strains were calculated prior to the test. A 15 percent margin was added to the calculated values of deflection and strain to arrive at the predicted values. The FINEL computer program (Appendix 3.8A) was used to calculate the deflections and strains for the containment. The program performs a finite element, static analysis of axisymmetric structures with axisymmetric loading. Special material properties that can be considered include bilinearity in compression and bilinearity or cracking in tension. Figure 3.8-40 shows a vertical section through the model. Points along Boundary A are prevented from moving in the vertical direction and points along Boundary B are prevented from moving in the radial direction. Concrete, reinforcing steel, and liner plate materials are included in the model. The SUPERB computer program (Appendix 3.8A) was used to calculate the predicted deflections and strains for the equipment hatch. Figure 3.8-41 shows the analytical model of the equipment hatch. Shell elements are used to represent the containment wall around the equipment hatch and the drywell floor. Points along Boundary A are allowed to move within the X-Z plane, and Boundary B within the X-Y plane. Points along Boundary C are prevented from moving in the hoop direction, and points along Boundary D are prevented from moving in the radial direction. Nodal forces, moments, and shears are applied to Boundary E to account for the reaction loads from the upper portion of the drywell wall.

Deflections and strains measured during the test were less than or equal to the predicted values at all critical locations. Thus, the design of the containment provides an adequate safety margin against internal pressure. Figure 3.8-42 shows a comparison between measured and predicted deflections for the containment at peak pressure. Figure 3.8-43 shows a comparison between measured and predicted deflections for the equipment hatch at peak pressure. The maximum strain occurs at midheight of the suppression chamber wall. Figures 3.8-44 through 3.8-48 compare measured and predicted strains at this location. Very little concrete cracking was observed. Figure 3.8-49 shows the cracks mapped at midheight of the drywell wall where the greatest amount of concrete surface cracks were observed.

3.8.1.7.1.2 Leak Rate Testing

Preoperational leak rate testing is discussed in Subsection 6.2.6.

3.8.1.7.2 In-service Leak Rate Testing

In-service leak rate testing is discussed in Subsection 6.2.6.

3.8.2 ASME CLASS MC STEEL COMPONENTS OF THE CONTAINMENT

This subsection pertains to the ASME Class MC steel components of the concrete containment that form a portion of the containment pressure boundary and are not backed by structural concrete. These components include the drywell head assembly, the equipment hatches and personnel lock, the suppression chamber access hatches, the CRD removal hatch, and piping and electrical penetrations.

3.8.2.1 Description of the ASME Class MC Components

3.8.2.1.1 Drywell Head Assembly

The drywell head provides a removable closure at the top of the containment for reactor access during the refueling operation. The drywell head assembly consists of a 2:1 hemi-ellipsoidal head and a cylindrical lower flange. The lower flange is supported on the top of the drywell wall as shown on Figure 3.8-9. The head is made of 1-1/2 in. thick plate and is secured with 80 2-3/4 in. diameter bolts at the 4 in. thick mating flange. Double rubber gaskets are provided at the head-to-lower flange connection to permit local leakage testing of the gaskets. The inside diameter (ID) of the drywell head at the mating flange is 37 ft 7-1/2 in. A 24 in. diameter double-gasketed manhole is provided in the drywell head.

Figure 3.8-50 shows details of the drywell head assembly.

3.8.2.1.2 Equipment Hatches and Personnel Lock

Two 12 ft 2 in. ID equipment hatches are furnished in the drywell wall to permit the transfer of equipment and components into and out of the drywell. One hatch is furnished with a double-gasketed flange and a bolted dished door. The other hatch is furnished with a double-gasketed flange and a bolted personnel lock. The personnel lock is an 8 ft 7 in. ID cylindrical pressure vessel with inner and outer flat bulkheads. Interlocked, double-gasketed doors are furnished in each bulkhead. A quick-acting, equalizing valve vents the personnel lock to the drywell to equalize the pressure in the two systems when the doors are opened and then closed. The two doors in the personnel lock are mechanically interlocked to prevent them from being opened simultaneously and to ensure that one door is closed before the opposite door can be opened. The personnel lock has an ASME Code N-stamp. See Figures 3.8-51 and 3.8-52 for details of the equipment hatch and the equipment hatch with personnel lock respectively.

3.8.2.1.3 Suppression Chamber Access Hatches

Two 6 ft 0 in. ID access hatches are furnished in the suppression chamber wall to permit personnel access and the transfer of equipment and components into and out of the suppression chamber. Each hatch is furnished with a double-gasketed flange and a bolted flat cover. See Figure 3.8-53 for details of the suppression chamber access hatches.

3.8.2.1.4 CRD Removal Hatch

One 3 ft 0 in. ID CRD removal hatch is furnished in the drywell wall to permit transfer of the control rod drive assemblies into and out of the drywell. The hatch is furnished with a double-gasketed flange and a bolted flat cover. See Figure 3.8-54 for details of the CRD removal hatch.

3.8.2.1.5 Penetrations

The entire length of any penetration sleeve is considered an MC component and, as such, is designed in accordance with Subsection NE of the ASME B&PV Code, Section III. See Subsection 3.8.1.1.3.3 for a description of the containment penetrations. Figures 3.8-21 and 3.8-22 show details of typical pipe and electrical penetrations respectively.

3.8.2.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design and construction of the containment are listed in Table 3.8-1 and given a reference number.

The reference numbers for the ASME Class MC components are 7C, 1H, 1J, and 1K.

3.8.2.3 Loads and Loading Combinations

3.8.2.3.1 General

Table 3.8-3 lists the loading combinations used for the design and analysis of the ASME Class MC components. The loading combinations comply with Regulatory Guide 1.57. The loading combinations shown in Table 3.8-3 do not include the hydrodynamic loads.

The ASME Class MC components have also been analyzed for hydrodynamic loads from main steam safety/relief valve discharge and LOCA. For a definition of loads and loading combinations including hydrodynamic loads, refer to GE's "Mark II Containment Dynamic Forcing Functions Information Report" (NEDO-21061), and the "Susquehanna Plant Design Assessment Report".

3.8.2.3.2 Description of Loads

3.8.2.3.2.1 Dead and Live Load

For a description of dead and live load, see Subsections 3.8.1.3.2.1 and 3.8.1.3.2.2 respectively.

3.8.2.3.2.2 Design Basis Accident Pressure Load

The MC components are designed for a containment design basis accident internal pressure of 53 psig. The personnel lock is also designed for a design basis accident internal pressure of 53 psig.

3.8.2.3.2.3 External Pressure Load

The MC components are designed for a containment external pressure of 5 psi differential.

3.8.2.3.2.4 Thermal Loads

The operating and postulated design accident temperatures for the MC components are as follows:

	Temperature (°F)	
Condition	Drywell	Suppression Chamber
Operating	135	90
Design Accident	340	220

Thermal cycles used in design are as follows:

a) Startup and shutdown - 500 cycles, 105°F range

b) Design Basis Accident - 1 cycle, 220°F range.

3.8.2.3.2.5 Seismic Loads

The MC components are designed for acceleration values, which are calculated using methods described in Bechtel Topical Report BC-TOP-4-A (Ref. 2K of Table 3.8-1).

The following acceleration values are used for the design of the drywell head assembly:

a) Operating Basis Earthquake - 1.0g horizontal, ±0.4g vertical

b) Safe Shutdown Earthquake - 1.5g horizontal, ±0.6g vertical

The following acceleration values are used for the design of all other class MC components:

a) Operating Basis Earthquake - 0.4g horizontal, ±0.3g vertical

b) Safe Shutdown Earthquake - 0.6g horizontal, ±0.4g vertical

3.8.2.3.2.6 Missile and Pipe Rupture Loads

The drywell head assembly is designed for a local pipe rupture load of 48,000 lb uniformly distributed over a circular area of 0.56 sq ft at any location on the drywell head. This load is due to the postulated rupture of the 6 in. diameter reactor vessel head spray pipe, which produces the largest load on the drywell head.

The equipment hatches are designed for a pipe rupture load of 1,200,000 lb uniformly distributed over a circular area of 12 ft diameter.

The CRD removal hatch is designed for a pipe rupture load of 160,000 lb uniformly distributed over a circular area of 3 ft diameter.

The loads on the equipment hatches and the CRD removal hatch are due to the rupture of a 28 in. diameter recirculation loop outlet pipe, which produces the largest load on the components.

The above values of static load include an appropriate dynamic load factor to account for the dynamic nature of the load. See Section 3.6 for a further discussion of pipe rupture loads.

3.8.2.4 Design and Analysis Procedures

3.8.2.4.1 Drywell Head Assembly

The analysis of the drywell head assembly is done using the thin shell computer program E0781 (Appendix 3.8A). This program calculates the stresses and displacements in thin-walled, elastic shells of revolution when subjected to static edge, surface, and/or temperature loads with an arbitrary distribution over the surface of the shell.

The drywell head assembly is divided into two analytical models. Figure 3.8-55 shows the drywell head model and the lower flange model. Displacement compatibility of the two models at the mating flange surface is maintained in the analysis. Boundary conditions are imposed on the analytical models by specifying boundary forces or displacements. Referring to Figure 3.8-55, the translation and rotation of the top of the drywell wall are imposed as boundary conditions to Boundary A. Boundary forces

applied to Boundary B are calculated in accordance with thin shell theory.

3.8.2.4.2 Access Hatches

Access hatches, including the equipment hatches, personnel lock, suppression chamber access hatches and CRD removal hatch, are designed as pressure retaining components. The portions of the sleeves not backed by concrete are designed and analyzed according to the provisions of Section III, Subsection NE of the ASME B&PV Code.

At the junction of the hatch cover to the flange on the sleeve, where local bending and secondary stresses occur, the computer program E0119 (Appendix 3.8A) is used for analysis. This program is also used for the analysis of the flat head covers.

3.8.2.4.3 Pipe and Electrical Penetrations

For nuclear Class I flued head penetrations, the stress calculations are performed according to the requirements of Article NB-3200 of the ASME B&PV Code, Section III for design, normal and upset, emergency, and faulted conditions. Nuclear Class II flued head penetrations are designed for the most severe condition which is the faulted condition. The stress calculations are performed using acceptable simplified equations or finite element computer program.

For Class IE electrical cable penetrations, the procedures used in design and analysis are in compliance with Subsection NE of the ASME Code, Section III, Division 1. The stress calculations were performed using acceptable simplified equations shown in Appendix A-5000 of the ASME Code, Section III.

3.8.2.5 Structural Acceptance Criteria

Table 3.8-3 lists the allowable stress criteria used for the design and analysis of the ASME Class MC components. The criteria comply with Regulatory Guide 1.57 except that the Code addendum (Summer 1973) applicable to the Regulatory Guide is subsequent to the Code addendum used for the design of the MC components (Summer 1972).

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

3.8.2.6.1 Materials

3.8.2.6.1.1 General

All carbon steel materials conform to the requirements of Article NE-2000, Materials, Section III of the ASME B&PV Code, 1971 Edition, with addenda through Summer 1972. Stainless steel materials for the CRD supply and return pipe penetrations conform to the requirements of Subsection NC of Section III of the ASME B&PV Code, 1971 Edition, with addenda through Summer 1972.

3.8.2.6.1.2 Drywell Head Assembly

Item	Specification
Drywell head and lower flange	SA-516, Grade 70, normalized
Bolts	SA-320, Grade L43
Nuts	SA-194, Grade 7

3.8.2.6.1.3 Access Hatches

Item	Specification
Sleeve and cover	SA-516, Grade 60 or 70, normalized
Bolts	SA-193, Grade B7
Nuts	SA-194, Grade 7

3.8.2.6.1.4 Penetrations

Item	Specification
Carbon steel sleeves	SA-333, Grade 1 or 6 or SA-516, Grade 60 or 70 normalized
Carbon steel caps for spare penetrations	SA-234, Grade WPB
Stainless steel sleeves for CRD supply and return penetrations	SA-312, Grade TP 304
Stainless steel fittings for CRD supply and return penetrations	SA-182, Grade F 304

3.8.2.6.2 Welding

Welding conforms to the requirements of Subsection NE, Section III, ASME B&PV Code, except all welding of the CRD supply and return penetrations conforms to the requirements of Subsection NC of Section III of the ASME B&PV Code. All pressure boundary welds are full penetration welds of double welded, bevel type. Welders and weld procedures are qualified in accordance with either Section IX of the ASME Code or AWS D1.1.

Penetrations, access hatches, and the drywell head flange are postweld heat treated in accordance with Article NE-4000 of Section III of the ASME Code. Penetrations are preassemblied into the liner plate sections and postweld heat treated as complete subassemblies.

3.8.2.6.3 Materials Testing

Impact testing as required by the ASME Code is performed at 0°F or below.

3.8.2.6.4 Nondestructive Examination of Welds

All welds between penetrations and liner plate, access hatches and liner plate, and pressure retaining welds not backed by concrete are examined in accordance with Article NE-5000 of Section III of the ASME Code. Nondestructive examination complies with Regulatory Guide 1.19.

3.8.2.6.5 Quality Control

Quality control requirements for the construction phase are discussed in Appendix D and amendments to the PSAR.

3.8.2.6.6 Erection Tolerances

The specified erection tolerances for ASME Class MC steel components of the containment are as follows:

a) Suppression chamber penetrations are within 1 in. of their design elevations and circumferential locations.

b) Drywell penetrations are within 1 in. of their design circumferential locations. Critical penetrations, such as main steam, feedwater, core spray, etc, are within 1 in. of their design elevations. All other drywell penetrations vary from within 1 in. of design elevations for penetrations near the base of the drywell wall to within 2 in. of design elevations for penetrations near the top of the drywell wall.

c) Alignments of penetrations are within 1 degree of the design alignments.

d) The average elevation of the mating flange between the drywell head and the lower flange is within 3 in. of the design elevation. The mating flange is within 1/2 in. of level.

Actual deviations from the above were handled in accordance with procedures covered in Subsection 3.8.2.6.5.

3.8.2.7 Testing and In-service Inspection Requirements

3.8.2.7.1 Preoperational Testing

3.8.2.7.1.1 Structural Acceptance Test

The drywell head assembly, equipment hatches, suppression chamber access hatches, CRD removal hatch, and pipe and electrical penetrations are pneumatically tested to 1.15 times the design accident pressure during the containment structural acceptance test. See Subsection 3.8.1.7.1.1 for a description of the structural acceptance tests.

The personnel lock is pneumaticallly tested to 1.15 times the design accident pressure, following shop fabrication and following field erection, to verify its structural integrity.

The CRD supply and return pipe penetrations are hydrotested to 1.5 times the design pressure of 1510 psig following shop fabrication in accordance with the ASME Code, Section III, Subsection NC.

3.8.2.7.1.2 Leak Rate Testing

Leaktightness of the containment Class MC components that are pressure retaining is verified during the integrated leak rate test. See Subsection 6.2.6 for a description of the containment integrated leak rate test.

The personnel lock is leak rate tested to 100 percent of the design accident pressure following shop fabrication and following field erection. The maximum allowable leak rate is 0.2 percent of the weight of the contained air in 24 hr when measured at ambient temperature and test pressure.

3.8.2.7.2 In-service Leak Rate Testing

In-service leak rate testing is discussed in Subsection 6.2.6.

3.8.3 CONTAINMENT INTERNAL STRUCTURES

3.8.3.1 Description of the Internal Structures

The internal structures of the containment perform the following major functions:

a) Support and shield the reactor vessel

b) Support piping and equipment

c) Form the pressure suppression boundary.

The containment internal structures are constructed of reinforced concrete and structural steel. The containment internal structures include the following:

a) Drywell floor

b) Reactor pedestal

c) Reactor shield wall

d) Suppression chamber columns

e) Drywell platforms

f) Seismic truss

g) Reactor steam supply system supports

Figures 3.8-1 through 3.8-8 show an overview of the containment including the internal structures.

3.8.3.1.1 Drywell Floor

The drywell floor serves as a barrier between the drywell and suppression chamber. It is a reinforced concrete circular slab with an outside diameter of 88 ft 0 in. and a thickness of 3 ft 6 in. See Figure 3.8-56 for details of the drywell floor reinforcement.

The drywell floor is supported by the reactor pedestal, the containment wall, and 12 steel columns. The connection of the drywell floor to the containment wall is shown on Figure 3.8-10. The drywell floor is penetrated by 87 24-in. diameter vent pipes. Additional reinforcement is furnished at vent pipe penetrations. See Subsection 6.2.1 for a description of the vent pipes.

A 1/4 in. thick carbon steel liner plate is provided on top of the drywell floor and anchored to it. The liner plate prevents bypass of the vent pipes during LOCA. Refer to Subsection 6.2.1 for a description of the bypass leakage requirements. Figure 3.8-57 shows the drywell floor liner plate and anchorage system.

3.8.3.1.2 Reactor Pedestal

The reactor pedestal is a 82 ft high, upright cylindrical reinforced concrete shell that rests on the containment base foundation slab and supports the drywell floor, reactor vessel, and reactor shield wall as well as drywell platforms, pipe restraints, and recirculation pumps. The connection of the reactor pedestal to the base foundation slab is shown on Figure 3.8-13. The reactor pedestal below the drywell floor has a 19 ft 7 in. inside diameter and a 5 ft 1 in. wall thickness. The reactor pedestal above the drywell floor has a 20 ft 3 in. inside diameter and a 4 ft 5 in. wall thickness. The thickness at the top of the pedestal is increased to 5 ft 4 in., where it supports the reactor vessel and the reactor shield wall. See Figures 3.8-58 and 3.8-59 for details of reinforcement. Openings are provided in the reactor pedestal to permit flow of air and suppression pool water into and out of the pedestal cavity. Additional reinforcement is furnished at openings. A 1/4 in. thick carbon steel form plate is provided on the inside and outside surfaces of the reactor pedestal below the drywell floor. This plate acts as a concrete form during construction and preserves the water quality of the suppression pool by preventing the leaching of chemicals from the reactor pedestal concrete into the suppression pool.

3.8.3.1.3 Reactor Shield Wall

The reactor shield wall is a 49 ft high upright cylindrical shell which rests on the top of the reactor pedestal and provides primary radiation shielding as well as supports for pipe restraints and drywell platforms. The reactor shield wall is constructed of inner and outer carbon steel plates and unreinforced concrete between the two plates. See Figure 3.8-60 for details of the reactor shield wall. The reactor shield wall has a 25 ft 7 in. inside diameter and a 1 ft 9 in. wall thickness. The outer steel plate is 1-1/2 in. thick and is designed to withstand any local pipe restraint and drywell platform attachment loads. The inner steel plate is 1/2 in. thick and is designed to act with the outer plate to withstand local and nonlocalized loads. The inner and outer plates are connected with steel bars spaced on 2 ft 6 in. centers. The annular space between the inner and outer plates is filled with

unreinforced concrete. The concrete is used for radiation shielding only and is not relied upon as a structural element. Normal density concrete is used in the top and bottom portions of the reactor shield wall. High density concrete is used at the midheight of the reactor shield wall opposite the reactor core for additional radiation shielding. The reactor shield wall is connected to the top of the reactor pedestal by 48 2-in. diameter, high strength anchor bolts as shown on Figure 3.8-61. The seismic truss and seismic stabilizer, which provide lateral support to the reactor vessel, are attached to the top of the reactor shield wall. Penetrations with hinged doors or removable plugs are provided in the reactor shield wall to facilitate piping connections to the reactor vessel and to provide access for in-service inspection. The wall thicknesses of penetration sleeves are large enough to prevent local stress concentrations in the inner and outer plates.

3.8.3.1.4 Suppression Chamber Columns

Twelve hollow steel pipe columns are furnished to support the drywell floor. Each column is 52 ft 6 in. long, 42 in. outside diameter, with a 1-1/4 in. wall thickness as shown on Figure 3.8-62. The columns are connected to the base foundation slab at the bottom and to the drywell floor at the top with embedded anchor bolts. Figure 3.8-14 shows the connection to the base foundation slab.

3.8.3.1.5 Drywell Platforms

Platforms are furnished at five elevations in the drywell to provide access and support to electrical and mechanical components. The platforms consist of structural steel framing with steel grating. Builtup box shapes are used for beams that must resist biaxial bending. Beams that span between the pedestal or shield and the containment wall are provided with sliding connections at one end. Thus, no thermal axial loads are developed in the beams and no radial loads are imposed on the pedestal, shield, or containment wall. See Figures 3.8-63 through 3.8-67 for details of the drywell platforms.

3.8.3.1.6 Seismic Truss and Seismic Stabilizer

The seismic truss and the seismic stabilizer provide lateral support for the reactor vessel during earthquake and pipe rupture loading. The seismic truss spans between the containment wall and the reactor shield wall, and the seismic stabilizer spans

between the reactor shield wall and the reactor vessel. For a description of the seismic stabilizer, see Section 3.9 . The seismic truss is shaped like an eight-pointed star and is fabricated of steel plates. See Figure 3.8-68 for details of the seismic truss. Figure 3.8-27 shows the connection of the seismic truss to the containment wall. This connection is designed to allow vertical and radial movement of the seismic truss relative to the containment wall but to prevent tangential movement.

3.8.3.1.7 Reactor Steam Supply System Supports

The steam supply system piping and pumps are supported by hangers, which in turn are supported by the reactor pedestal, reactor shield, and drywell platforms. A description of these supports is given in Section 3.9. In addition, the reactor vessel itself is supported on the reactor pedestal by 120, 3-1/4 in. diameter, high strength anchor bolts as shown on Figure 3.8-61. The reactor vessel is supported laterally by the seismic truss and seismic stabilizer as discussed in Subsection 3.8.3.1.6.

3.8.3.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design and construction of the containment internal structures are listed in Table 3.8-1 and given a reference number.

The reference numbers for the drywell floor are 10A, 12A, 1C, 2C, 3C, 6C, and 2K.

The reference numbers for the drywell floor liner plate and anchorages are 4C, 1H, 1J, and 1K.

The reference numbers for the reactor pedestal are 7A, 10A, 12A, 1C, 2C, 3C, 6C, and 2K.

The reference numbers for the reactor shield wall are 1B, 6C, 1H, and 2K.

3 The reference numbers for the suppression chamber columns are 1H, 2H, 3H, 1J, and 2K.

3 The reference numbers for the drywell platforms and seismic truss are 1B, 1H, 2H, 3H and 2K.

3.8.3.3 Loads and Loading Combinations

3.8.3.3.1 General

Tables 3.8-2, 3.8-2a and 3.8-4 through 3.8-7 list the loading combinations used for the design and analysis of the containment internal structures. The loading combinations shown in these tables do not include hydrodynamic loads.

The internal structures have also been analyzed for hydrodynamic loads from main steam safety/relief valve discharge and LOCA. For a definition of loads and loading combinations including hydrodynamic loads, refer to GE's "Mark II Containment Dynamic Forcing Functions Information Report" (NEDO-21061) and the "Susquehanna Plant Design Assessment Report".

3.8.3.3.2 Drywell Floor and Reactor Pedestal

Table 3.8-2 lists the loading combinations used for the design of the drywell floor. The loading combinations are in compliance with those given in Reference 12A of Table 3.8-1.

Table 3.8-2a lists the loading combinations used for the design of the reactor pedestal. The loading combinations are in compliance with those given in SRP Section 3.8.3.II.3.

3.8.3.3.2.1 Description of Loads

Dead Load, Live Load, and Seismic Loads

For a description of dead load, live load, and seismic loads, see Subsections 3.8.1.3.2.1, 3.8.1.3.2.2 and 3.8.1.3.2.6 respectively.

Design Basis Accident Pressure Load

The drywell floor and the reactor pedestal are designed for the following pressures:

a) Maximum pressure: 53 psig in the drywell and the suppression chamber

b) Maximum differential pressure: 28 psig (53 psig in the drywell and 25 psig in the suppression chamber).

Thermal Loads

The temperature gradients through the drywell floor and the reactor pedestal are shown on Figure 3.8-69 for the operating and the postulated design accident condition. The design accident temperature gradients shown on Figure 3.8-69 occur five minutes after LOCA. These transient temperature gradients are used for the design of the drywell floor and the reactor pedestal because they produce the largest stresses in the structure.

Thermal effects anticipated at the time of the structural acceptance test are insignificant since changes in temperature inside and outside the containment during the test will be small.

Missile and Pipe Rupture Loads

The drywell floor and the reactor pedestal are designed to withstand the missile and pipe rupture loads due to a postulated rupture of a 28 in. diameter recirculation loop pipe, which produces the largest loads on the structures. These loads include the effects of jet impingement, pipe whip, and pipe reaction. An equivalent static load of 1030 kips is considered. This load includes an appropriate dynamic load factor to account for the dynamic nature of the load. See Section 3.6 for a further discussion of postulated pipe rupture loads.

3.8.3.3.3 Reactor Shield Wall

The reactor shield wall is designed using the elastic working stress design methods of AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings", dated 1969, Part 1. Table 3.8-4 lists the load combination used for the design of the reactor shield wall. Since this loading condition combines the design basis accident loads with the maximum seismic loads, it is the most severe loading condition and other, less severe load combinations are not considered.

3.8.3.3.3.1 Description of Loads

Dead Load, Live Load, and Seismic Loads

For a description of dead load, live load, and seismic loads, see Subsections 3.8.1.3.2.1, 3.8.1.3.2.2 and 3.8.1.3.2.6 respectively.

Design Basis Accident Pressure Load

The reactor shield wall is designed for internal pressure due to a postulated pipe rupture at the connection of the pipe to the reactor vessel nozzle safe end. The following two pressure conditions are considered:

a) Maximum unbalanced pressure: pressure condition shortly after pipe break, which produces the largest lateral load on the reactor shield wall, as shown in Figure 6A-3b.

b) Maximum uniform pressure: 70 psig internal pressure.

Thermal Loads

The temperature gradients through the reactor shield wall are shown on Figure 3.8-70 for the operating and the postulated design accident conditions. The design accident temperature gradient shown on Figure 3.8-70 occurs five minutes after LOCA. This transient temperature gradient is used for the design of the reactor shield wall since it produces the largest stresses in the structure.

Missile and Pipe Rupture Loads

The reactor shield wall is designed to withstand the missile and pipe rupture loads due to a postulated rupture of any high energy pipe that penetrates the reactor shield wall and connects to the reactor vessel, such as recirculation and feedwater pipes. These loads include the effects of jet impingement, pipe whip, and pipe reaction. Equivalent static loads are considered, which include an appropriate dynamic load factor to account for the dynamic nature of the load. See Section 3.6 for a further discussion of postulated pipe rupture loads.

3.8.3.3.4 Suppression Chamber Columns

The suppression chamber columns are designed using the plastic design methods of AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings", dated 1969, Part 2. Table 3.8-5 lists the load combinations used for the design of the suppression chamber columns. The columns are designed to resist the reaction loads from the drywell floor for the LOCA conditions. Subsection 3.8.3.3.2 includes a description of the loads for the drywell floor. The abnormal loading conditions govern the design since they include the design basis accident pressure load, which is the critical load for columns.

3.8.3.3.5 Drywell Platforms

The drywell platforms are designed using working stress design methods except for the pipe restraints supported on the platforms. The pipe restraints are designed to undergo local inelastic deformations due to postulated pipe rupture loads. However, there is no loss of function of the pipe restraints since they will restrain any postulated pipe whip. The built-up box beams that support the pipe restraints are designed to withstand all postulated pipe rupture loads. Design accident pressure and operating and design accident thermal loads do not affect the design of the drywell platforms. For the design of box beams, seismic loads due to dead weight of the beams may be neglected since these loads are insignificant relative to the pipe rupture loads. For the design of the framing beams, seismic loads due to dead weight of the beams are small and may be neglected since these beams are laterally braced by other framing beams and by the grating. The uniform design live load for the grating and framing beams is 200 psf. The live load for the framing beams also includes the gravity load, thermal reaction load, and seismic SSE reaction load of all piping and equipment supported on the beams. Table 3.8-6 lists the load combinations used to design the drywell platforms. Pressure, thermal and seismic loads are not considered since they are not critical.

3.8.3.3.6 Seismic Truss

The seismic truss is designed using working stress design methods. It is designed primarily for lateral seismic loads. However, it is also designed for jet impingement loads due to the postulated rupture of a 26 in. diameter main steam pipe. Design accident pressure and operating and design accident thermal loads do not affect the design of the seismic truss. Table 3.8-7 lists the load combination used to design the seismic truss. Pressure and thermal loads are not considered since they are not critical.

3.8.3.4 Design and Analysis Procedures

This section describes the procedures used for the design and analysis of the containment internal structures. The description does not include the effects of hydrodynamic loads from main steam safety/relief valve discharge and LOCA. For a description of the design and analysis procedures that consider the effects of hydrodynamic loads, refer to GE's "Mark II Containment Dynamic Forcing Functions Report" (NEDO-21061) and the "Susquehanna Plant Design Assessment Report".

3.8.3.4.1 Drywell Floor

The design and analysis procedures used for the drywell floor are similar to those used for the containment wall. Used for the analysis are 3D/SAP, CECAP, ME620, and seismic analysis computer programs (Appendix 3.8A). See Subsection 3.8.1.4.1 for a detailed description of the analysis procedures.

Figure 3.8-71 shows the 3D/SAP finite element model used to analyze the drywell floor for all loads other than seismic loads. A 15 degree wedge of the drywell floor is modeled using solid finite elements having linear elastic, isotropic material properties. One vertical boundary plane goes through a suppression chamber column and the other is halfway between two columns. The model includes the drywell floor, suppression chamber wall, reactor pedestal below the drywell floor, and a suppression chamber column. Boundary conditions are imposed on the analytical model by specifying nodal point forces or displacements. Referring to Figure 3.8-71, the nodal points lying along Boundary A are allowed to move within the X-Z plane, and Boundary B within the X-Y plane. Points along Boundary C are prevented from moving in the hoop direction. Points along Boundary D are prevented from moving in the radial direction to account for the restraining effect of the inner portion of the drywell floor. Nodal forces, moments, and shears are applied to Boundaries E and F to account for reaction loads from the drywell wall and reactor pedestal above the drywell floor respectively.

Analytical techniques as described in Bechtel Topical Report BC-TOP-4-A (Ref. 2K of Table 3.8-1) are used to analyze the drywell floor for seismic loads.

3.8.3.4.2 Drywell Floor Liner Plate and Anchorages

The design and analysis of the drywell floor liner plate and anchorages is in accordance with Bechtel Topical Report BC-TOP-1 (Ref. 1K of Table 3.8-1).

3.8.3.4.3 Reactor Pedestal

The reactor pedestal is designed for axisymmetric loads using the FINEL computer program (Appendix 3.8A). The program performs a finite element, static analysis of axisymmetric structures with axisymmetric loading. Both concrete and reinforcing steel materials are included in the model. Special material properties include bilinearity in compression and bilinearity or cracking in tension. The operating and design accident temperature gradients

are computed using ME620 computer program (Appendix 3.8A). For transient loads such as design accident pressure and thermal loads, the most critical combination of these loads is considered. Figure 3.8-40 shows a vertical section through the FINEL model of the containment used to analyze the reactor pedestal below the drywell floor. Points along Boundary A are prevented from moving in the vertical direction and points along Boundary B are prevented from moving in the radial direction.

Figure 3.8-72 shows the FINEL model used to analyze the reactor pedestal above the drywell floor. The model includes the reactor pedestal above the drywell floor and portions of the reactor vessel and the reactor shield wall. Local thermal effects at the top of the reactor pedestal due to heat input from the reactor vessel are determined by using the ME620 computer program (Appendix 3.8A). Referring to Figure 3.8-72, nodal points along Boundary A are prevented from moving in the vertical and radial directions. Nodal forces, moments, and shears are applied to Boundaries B and C to account for reaction loads from the reactor vessel and the reactor shield wall respectively.

Non-axisymmetric loads on the reactor pedestal include seismic loads and reactor vessel and reactor shield reaction loads. Seismic forces, moments, and shears are calculated as described in Section 3.7. Vertical forces, horizontal shears, and overturning moments at the base of the reactor shield wall are determined as described in Subsection 3.8.3.4.4. These loads are applied to the top of the reactor pedestal. Concrete and reinforcing steel stresses in the reactor pedestal due to the above loads are calculated using the design methods of ACI 307. ACI 307 includes equations for determining the neutral axis of reinforced concrete cylindrical shells subjected to axial force and overturning moment. The position of the neutral axis satisfies the equilibrium of internal stresses and external forces and moments.

Concrete and reinforcing steel stresses due to axisymmetric and non-axisymmetric loads are combined to determine the total stress. Additional meridional, hoop, and shear reinforcement is provided at the top of the pedestal as shown in Figure 3.8-59 to resist local loads on the pedestal from the reactor vessel and the reactor shield. The seismically-induced tangential shears on the reactor pedestal are considerably less than the seismically-induced tangential shears on the containment wall. Therefore, helical reinforcement is not provided in the reactor pedestal in order to resist tangential shears. Meridional and hoop reinforcement is designed to carry the entire tangential shear by shear friction using the design methods of ACI 318-71.

3.8.3.4.4 Reactor Shield Wall

The reactor shield wall is analyzed in two stages. First, the effect of openings on the behavior of the reactor shield is investigated. This is done to determine whether the shield may be analyzed as an axisymmetric cylindrical shell without openings or whether the openings cause local stress concentrations. Loads considered for this analysis are design accident pressure and postulated pipe rupture loads. The EASE computer program (Appendix 3.8A) is used for this analysis. Figure 3.8-73 shows the finite element model. A full 360 degree section of the reactor shield wall is modeled using plate elements having linear elastic, isotropic material properties. One 64 in. diameter recirculation outlet penetration and two adjacent 48 in. diameter recirculation inlet penetrations are included in the model. Smaller finite elements are used in the area of the openings to obtain an accurate stress gradient. Referring to Figure 3.8-73, points along Boundary A are prevented from moving in the vertical and radial directions. Boundary B is a free edge. The results of this analysis confirm that there are no significant local stress concentrations in the shield around the openings. This is due to the stiffening of the shell that is provided by the thick-walled penetration sleeves. Therefore, the use of an axisymmetric analytical model without openings is justified.

The second stage analyzes the reactor shield wall as an axisymmetric shell. For axisymmetric loads, which include dead load and design accident thermal load, the FINEL computer program is used. The most critical temperature gradient as determined by the ME620 computer program (Appendix 3.8A) is considered. The FINEL program performs a finite element, static analysis of axisymmetric structures with axisymmetric loading. For non-axisymmetric loads, which include design accident pressure load, seismic load, and pipe rupture load, the ASHSD computer program (Appendix 3.8A) is used. The ASHSD program performs an elastic, finite element, static, or dynamic analysis of axisymmetric structures with non-axisymmetric loading. The distribution of non-axisymmetric load around the shell is approximated by a Fourier series expansion. Figure 3.8-74 shows a vertical section through the model used for FINEL and ASHSD programs. Points along Boundary A are prevented from moving in the vertical and radial directions. For non-axisymmetric loads, Boundary B at the connection of the seismic truss to the containment wall is prevented from moving in the radial direction. Total stresses in the reactor shield wall are determined by summing the axisymmetric and non-axisymmetric stresses.

3.8.3.4.5 Suppression Chamber Columns

Axial force, shear, and moment in the columns due to axisymmetric loads, such as dead load and design accident pressure and thermal loads, are determined using the FINEL computer program (Appendix 3.8A). Figure 3.8-40 shows the FINEL model of the containment used to analyze the suppression chamber columns. A description of the program and the boundary conditions is given in Subsection 3.8.3.4.3. Since the FINEL program can consider only axisymmetric structures, the 12 columns are modeled as an equivalent cylinder having the cross-sectional area and axial stiffness of the columns. Axial force in the columns is calculated from the axial stress determined by the FINEL program. Shear and moment in the columns are calculated from relative displacements of the drywell floor and the base foundation slab determined by the FINEL program.

Axial force, shear, and moment in the columns due to seismic loads are determined using several methods. Axial force in the columns due to horizontal seismic load is determined using the ASHSD program (Appendix 3.8A). Figure 3.8-75 shows the model. Axisymmetric shell and solid finite elements having linear elastic, isotropic material properties are used. Nodal points lying along Boundary A are prevented from moving in the vertical direction and points along Boundary B are prevented from moving in the radial direction. The load applied to the ASHSD model is the seismic horizontal shear and overturning moment for the containment calculated as described in Section 3.7.

Shear and moment in the columns due to horizontal seismic load are determined using the analytical procedures described in Bechtel Topical Report BC-TOP-4-A (Ref. 2K of Table 3.8-1). The lumped mass model of the containment including columns and vent pipes is shown in Figure 3.8-76. Since the vent pipes are laterally braced to the columns, shear and moment are produced in the columns due to seismic motion of the vent pipes.

Axial force in the columns due to vertical seismic load is determined by applying the vertical forces calculated from the containment seismic analysis to the drywell floor at its connections to the containment wall and the reactor pedestal. The vertical force transmitted to the columns through the drywell floor is calculated considering the relative vertical stiffnesses of the containment wall, reactor pedestal, and columns.

The postulated rupture of a 28 in. diameter recirculation loop pipe produces a vertical jet impingement load on the top of the drywell floor and, therefore, produces loads in the columns. Axial force, shear, and moment in the columns due to jet force is calculated by the CE 668 computer program (Appendix 3.8A). The program performs a static, linear elastic analysis of flat slabs

of arbitrary dimensions subjected to arbitrary loading. Figure 3.8-77 shows the 180 degree model of the drywell floor. A vertical jet force is applied along the axis of symmetry and the reaction is calculated in the column adjacent to the applied load. Edges of the drywell floor along Boundaries A and B are considered to be fixed supports. Nodal points at the columns are fixed in the plane of the model.

The total axial force, shear, and moment in the columns for all load combinations are determined by summing the results of the separate analyses. Stability of the columns for the most critical load combination is checked using the plastic design methods of AISC, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings", dated 1969, Part 2 (Ref. 1H of Table 3.8-1).

3.8.3.4.6 Drywell Platforms

The drywell platforms are designed using conventional elastic design methods which conform to the AISC Specification, 1969, Part 1 (Ref. 1H of Table 3.8-1).

3.8.3.4.7 Seismic Truss

Seismic forces in the seismic truss are calculated using the methods described in Bechtel Topical Report BC-TOP-4-A (Ref. 2K of Table 3.8-1). Axial force, shear force, and moment in the seismic truss due to postulated pipe rupture loads are calculated using moment distribution. Figure 3.8-78 shows the rigid frame model including boundary conditions.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Reinforced Concrete

The allowable stresses for the reinforced concrete portions of the containment internal structures are the same as the allowable stresses for the reinforced concrete portions of the containment. See Subsection 3.8.1.5.1 for a description.

3.8.3.5.2 Drywell Floor Liner Plate and Anchorages

1 The structural acceptance criteria for the drywell floor liner plate and anchorages are the same as the structural acceptance criteria for the containment liner plate and anchorages see subsection 3.8.1.5.2 for a description.

3.8.3.5.3 Structural Steel

Structural steel portions of the containment internal structures include the reactor shield wall, suppression chamber columns, drywell platforms, and seismic truss.

For normal loading conditions, the allowable stresses are in accordance with the AISC Specification (Ref. 1H of Table 3.8-1).

For extreme environmental and abnormal loading conditions, the allowable stresses are as follows:

a) Bending - 0.90 Fy

b) Axial tension or compression - 0.85 Fy except, where allowable stress is governed by requirements of stability (local or lateral buckling), allowable stress shall not exceed 1.5 Fs.

c) Shear - 0.50 Fy

The allowables are defined as:

Fs = Allowable stress according to the AISC Specification, Part 1 (Ref. 1H of Table 3.8-1)

Fy = Specified yield strength of structural steel

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

3.8.3.6.1 Concrete Containment Internal Structures

The concrete and reinforcing steel materials for the containment internal structures are discussed in Appendix 3.8B. Concrete
11 design compressive strengths are given in Table 3.8-11.

3.8.3.6.2 Drywell Floor Liner Plate, Anchorages, Attachments

3.8.3.6.2.1 Materials

Liner plate materials conform to the requirements of the standard specifications listed below:

Item	Specification
Liner plate (less than 1/2 in. thick)	ASTM A 285, Grade A
Liner plate (1/2 in. thick or thicker)	ASME SA-516, Grade 60 or 70 conforming to the requirements of ASME Boiler and Pressure Vessel Code (ASME B&PV Code), 1971 Edition with Addenda through Summer 1972, Section III, Article NE-2000, Materials
Anchorages and attachments	ASTM A 36

3.8.3.6.2.2 Welding

Welding requirements for the drywell floor liner plate and anchorages are the same as the welding requirements for the containment liner plate and anchorages. See Subsection 3.8.1.6.2.2 for a description of the welding requirements.

3.8.3.6.2.3 Nondestructive Examination of Liner Plate Seam Welds

Nondestructive testing of liner plate welds is performed in accordance with Regulatory Guide 1.19, Revision 1.

Liner plate seam welds are 100 percent magnetic particle examined. Liner plate seam welds are also 100 percent vacuum box soap bubble tested. Welds that are inaccessible for vacuum box testing are 100 percent liquid penetrant tested. Examination procedures, personnel qualification, and acceptance standards are in accordance with Subsection 3.8.1.6.2.4.

3.8.3.6.2.4 Erection Tolerances

The specified levelness of anchorages placed in the drywell floor is within ±1-1/4 in. of the theoretical elevation over the entire area, plus a local tolerance of ±1/8 in. in any 30 ft length. Actual deviations from the above were handled in accordance with procedures covered in Subsection 3.8.3.6.6.

3.8.3.6.3 Reactor Shield Wall and Seismic Truss

3.8.3.6.3.1 Materials

Item	Specification
Inner and outer plates, seismic truss, pipe restraints, etc.	ASTM A 588, Grade A or B
Internal stiffeners	ASTM A 36
Seismic Truss Male Stabilizer Block	ASME SA 181, Grade II

3.8.3.6.3.2 Welding and Nondestructive Examination of Welds

Welding and nondestructive examination is performed in accordance with AWS D1.1.

3.8.3.6.3.3 Materials Testing

The 1-1/2 in. thick outer plate and other plates subjected to transverse tensile stress are vacuum degassed and ultrasonically tested in accordance with supplementary requirements S-1 and S-8.1 respectively of ASTM A 20-72a.

3.8.3.6.3.4 Erection Tolerances

The specified erection tolerances for the reactor shield are as follows:

a) The radial dimension from the as-built centerline of the reactor vessel to any point on the reactor shield is within 3/8 in. of the theoretical radius.

b) The top of the reactor shield is set within 1/4 in. of its theoretical elevation.

c) The azimuths of the shield penetrations are within 1/2 in. of the theoretical azimuths.

d) Seismic truss members do not deviate from axial straightness by more than 1/1000 of axial length.

Actual deviations from the above were handled in accordance with procedures covered in Subsection 3.8.3.6.6.

3.8.3.6.4 Suppression Chamber Columns

3.8.3.6.4.1 Materials

The column shafts, base plates, and top plates are fabricated of ASME SA-516, Grade 70 material.

3.8.3.6.4.2 Welding

Weld procedures and qualifications conform to the provisions of Section IX and Section VIII, Division 1 of the ASME Boiler and Pressure Vessel Code, 1971 Edition with addenda through Summer 1972. All welders are qualified in accordance with Section IX of the ASME Code.

3.8.3.6.4.3 Nondestructive Examination of Welds

Nondestructive examinations conform to Section V of the ASME B&PV Code, 1971 Edition with addenda through Summer 1972. All personnel performing nondestructive examination are qualified in accordance with the American Society for Nondestructive Testing's Recommended Practice No. SNT-TC-1A and its applicable

supplements. Acceptance standards conform to Section VIII, Division 1 of the ASME Code.

3.8.3.6.4.4 Fabrication and Erection Tolerances

1 The specified fabrication and erection tolerances for suppression chamber columns are as follows:

a) The outside diameter, based on circumferential measurements, does not deviate from the theoretical outside diameter by more than 0.5 percent.

b) Out-of-roundness, defined by the difference between the maximum and minimum diameters related to the theoretical diameter, is in accordance with ASME Code, Section VIII, Division 1, Paragraph UG-80.

c) The finished length does not differ from the theoretical length by more than 1/4 in.

d) The finished column shaft does not deviate from straightness by more than 1/8 in. in 1 ft, with a maximum for the full length of 1/1000 of the total length.

4 e) Erection tolerances are in accordance with the AISC Specification (Ref. 1H and 2H of Table 3.8-1).

Actual deviations from the above were handled in accordance with procedures covered in Subsection 3.8.3.6.6.

3.8.3.6.5 Drywell Platforms

3.8.3.6.5.1 Materials

Item	Specification
Box beams	ASTM A 441
Rolled shapes	ASTM A 36
Connection bolts	ASTM A 325

3.8.3.6.5.2 Welding and Nondestructive Examination of Welds

Welding and nondestructive examination is performed in accordance with AWS D1.1.

3.8.3.6.5.3 Erection Tolerances

Erection tolerances for the drywell platforms are in accordance with AISC Specification (Ref. 2H of Table 3.8-1)

3.8.3.6.6 Quality Control

Quality control requirements for construction are discussed in Appendix D and amendments to the PSAR.

3.8.3.7 Testing and In-service Inspection Requirements

3.8.3.7.1 Preoperational Testing

3.8.3.7.1.1 Structural Acceptance Test

The drywell floor is tested to 1.15 times the design downward differential pressure. See Subsection 3.8.1.7.1.1 for a description of the structural acceptance tests.

Deflections and strains of the drywell floor measured during the Unit 1 test were less than the predicted values. Thus, the design of the drywell floor provides an adequate safety margin against internal pressure. Figure 3.8-79 shows a comparison between measured and predicted deflections for the drywell floor at peak differential pressure.

3.8.3.7.1.2 Leak Rate Testing

Preoperational leak rate testing is discussed in Subsection 6.2.6.

3.8.3.7.2 In-service Leak Rate Testing

In-service leak rate testing is discussed in Subsection 6.2.6.

3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES

This section gives information on all Seismic Category I structures except the primary containment and its internals. It also describes safety related non-Seismic Category I structures. The structures included in this section are as follows:

Seismic Category I Structures

Reactor Building

Control Building

Diesel Generator Building

Engineered Safeguards Service Water Pumphouse

Spray Pond

Non-Seismic Category I, Safety Related Structures

Turbine Building

Radwaste Building

The general arrangement of these structures is shown on Figures 3.8-80 through 3.8-103.

3.8.4.1 Description of the Structures

Reactor Building

Refer to Figures 3.8-80 through 3.8-89.

The reactor building encloses the primary containment, and provides secondary containment when the primary containment is in service during power operation. It also serves as containment during reactor refueling and maintenance operations, when the primary containment is open. It houses the auxiliary systems of the nuclear steam supply system, new fuel storage vaults, the refueling facility, and equipment essential to the safe shutdown of the reactor.

The reactor building, up to and including the operating floor, is of reinforced concrete on a mat foundation. The bearing walls are of reinforced concrete and are designed as shear walls to resist lateral loads. The floors are of reinforced concrete supported by a steel beam and column framing system and are designed as diaphragms to resist lateral load. The framing runs in both east-west and north-south directions, with the exterior ends of the beams supported by either the bearing walls or steel columns. The steel columns are supported by base plates on the mat foundation. The reinforced concrete walls and floors meet structural as well as radiation shielding requirements. Where structurally permissible, concrete block masonry walls are used at certain locations to provide better access for erection and installation of equipment. The block walls also meet the radiation shielding requirements.

The reactor building superstructure above the operating floor is a steel structure. The structural steel framing supports the roof, metal siding, and overhead cranes. The framing consists of a series of rigid frames connected by roof and wall bracing systems. The roof consists of built-up roofing on metal deck.

The refueling facility is located above the containment structure. It consists of spent fuel pool, fuel shipping cask storage pool, steam dryer and separator storage pool, reactor cavity, skimmer surge tank vault, and load center room. The facility is supported by two reinforced concrete girders running north-south, spanning over the containment. The girders are supported at the ends by concrete walls and at intermediate points by steel box columns. A gap is provided between the bottom of the girders and the top of the containment to ensure that loads from the refueling facility are not transferred to the containment. The walls and slabs of the spent fuel pool, the fuel shipping cask storage pool, the reactor cavity, and the steam dryer and separator storage pool are lined on the inside with a stainless steel liner plate. The facility meets the radiation shielding requirements.

The reactor building is separated from the primary containment by a gap, except at the foundation level, where a cold joint is provided between the two mats. A gap is also provided at the interface of the reactor building with the diesel generator and turbine buildings.

Control Building

Refer to Figures 3.8-80 through 3.8-88.

The control building houses the control room, the cable spreading rooms, computer and relay room, the battery room, H&V equipment room, off-gas treatment room, and the visitors' gallery for the control room.

The control building is structurally integrated with the reactor building. It is a reinforced concrete structure on a mat foundation. The bearing walls are of reinforced concrete and are designed as shear walls to resist lateral loads. The floors and roof are of reinforced concrete supported by steel beams, and are designed as diaphragms to resist lateral loads. The beams span in the east-west direction and are supported by the bearing walls at the ends. The reinforced concrete walls and floors meet structural as well as radiation shielding requirements. Where structurally permissible, concrete block masonry walls are used at certain locations to provide better access for erection and installation of equipment. The block walls also meet the radiation shielding requirements.

The control building is separated from the turbine building by a gap, except at the foundation level, where a cold joint is provided between the two mats.

Diesel Generator Building

Refer to Figures 3.8-92 and 3.8-93.

The diesel generator building houses the diesel generators essential for safe shutdown of the plant.

The diesel generators are separated from each other by concrete walls. A concrete overhang on the east side of the building serves as an air intake plenum. A concrete plenum for diesel exhaust is located on the roof.

It is a reinforced concrete structure on a mat foundation. The bearing walls are of reinforced concrete and are designed as shear walls to resist lateral loads. The floors and roof are of reinforced concrete supported by steel beams, and are designed as diaphragms to resist lateral loads. The south side of the building interfaces with the reactor building; there, a reinforced concrete wall is provided from foundation up to the design high water table level and then a steel frame is provided up to the roof. Where structurally permissible, concrete block masonry walls are used at certain locations to provide better access for erection and installation of equipment.

The diesel generators are supported by reinforced concrete pedestals. The pedestals are separated from the operating floor by a gap to allow for their independent vibration.

Engineered Safeguards Service Water (ESSW) Pumphouse

Refer to Figure 3.8-94.

The ESSW Pumphouse contains the emergency service water (ESW) and residual heat removal (RHR) pumps and the weir and discharge conduit for the spray pond.

It is a two-story reinforced concrete structure on a mat foundation. The bearing walls are of reinforced concrete and are designed as shear walls to resist lateral loads. The operating floor and roof are of reinforced concrete supported by steel beams and are designed as diaphragms to resist lateral loads. A mezzanine floor composed of grating over steel beams is provided to support the heating and ventilating equipment.

Spray Pond

Refer to Figures 3.8-95 through 3.8-98.

The spray pond is a reservoir, free form in shape, which holds approximately 25 million gal of water during normal operation. The water surface area is approximately eight acres and has a depth of approximately 10 ft 6 in. It is designed so that normal operating water is retained in excavation alone, ie, not by constructed embankments. Embankments are provided to ensure a minimum freeboard of 3 ft and to direct flood water away from safety related facilities in a controlled manner.

The ESSW pumphouse is located at the southeast corner of the spray pond. A reinforced concrete liner covers the entire spray pond and is integrated with the outer walls of the ESSWP.

The water level in the pond is controlled by a weir housed in the ESSW pumphouse. During normal operation, excess water is discharged into the Susquehanna river via a conduit from the ESSW pumphouse.

An emergency spillway is provided at the east end of the pond. The only anticipated use of this spillway will be either during a malfunction of the discharge conduit leading out of the ESSW pumphouse or during certain postulated flood conditions. This is discussed in Subsection 2.4.8.

The ESW and RHR pipes enter the south side of the pond and traverse to the spray bank areas buried in 18 in. of concrete, provided as missile protection. Concrete columns support the riser pipes in the spray bank areas.

Turbine Building

Refer to Figures 3.8-80 through 3.8-84, 3.8-88, 3.8-90, and 3.8-91.

The turbine building is divided into two units with an expansion joint separating the two units. It houses two in-line turbine generator units and auxiliary equipment including condensers, condensate pumps, moisture separators, air ejectors, feedwater heaters, reactor feed pumps, motor-generator sets for reactor recirculating pumps, recombiners, interconnecting piping and valves, and switchgears.

Two 220-ton overhead cranes are provided above the operating floor for service of both turbine generator units. Two reinforced concrete tunnels, one for each unit, are provided for the off-gas pipelines at the foundation level between the recombiners and the radwaste building. Reinforced concrete tunnels are also provided for the main steam lines below the operating floor from the reactor building to the condenser areas of the turbine generators.

The turbine building rests on a reinforced concrete mat foundation. The superstructure is framed with structural steel and reinforced concrete. Rigid steel frames support the two 220-ton cranes. They also resist all transverse (east-west) lateral loads. Steel bracings resist longitudinal (north-south) lateral loads above the operating floor. Below this level, reinforced concrete shear walls transfer all lateral loads to the foundations.

A seismic separation gap, also serving as an expansion joint, is provided near the center of the building between the two units. Seismic separation gaps are also provided at the interface of turbine building with the reactor, control, and radwaste buildings.

The floors of the turbine building are of reinforced concrete on structural steel beams. They are designed as diaphragms for lateral load transfer to the shear walls. The roof is built-up roofing on metal decking.

Exterior walls are precast reinforced concrete panels except for the upper 30 ft, which are metal siding.

Interior walls required for radiation shielding or fire protection are constructed of reinforced concrete block. These walls are not used as elements of the load resistant system.

The turbine generator units are supported on freestanding reinforced concrete pedestals. The mat foundations for the pedestals are founded on rock at the same level as the base mat

for the turbine building. Separation joints are provided between the pedestals and the turbine building floors and walls to prevent transfer of vibration to the building. The operating floor of the building is supported on vibration damping pads at the top edge of the pedestal.

Radwaste Building

Refer to Figures 3.8-99 through 3.8-103.

The radwaste building houses systems for receiving, processing, and temporarily storing the radioactive waste products generated during the operation of the plant.

It is a reinforced concrete structure on a mat foundation. The bearing walls are of reinforced concrete and are designed as shear walls to resist lateral loads. The floors and roof are of reinforced concrete supported by a beam and column framing system and are designed as diaphragms to resist lateral loads. The columns are supported by base plates on the mat foundation. The reinforced concrete walls and floor meet structural as well as radiation shielding requirements. Where structurally permissible, concrete block masonry walls are used at certain locations to provide better access for erection and installation of equipment. The block walls also meet the radiation shielding requirements.

The radwaste building is separated from the turbine building by a gap.

3.8.4.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design, fabrication, and construction of the structures listed in Subsection 3.8.4 are shown in Table 3.8-1 and include reference numbers 10A, 1B, 1H, 2H, 3H, 1J, 2K, 3K, and 1L. |1 |4 |3

3.8.4.3 Loads and Load Combinations

The following loads and load combinations are considered in the design of Seismic Category I structures (other than the containment).

3.8.4.3.1 Description of Loads

For a general description of loads, see Subsection 3.8.1.3.2.

3.8.4.3.2 Load Combinations

Table 3.8-8 describes the load combinations applicable to the reactor building. Table 3.8-9 contains the load combinations applicable to Seismic Category I structures other than the reactor building. Table 3.8-10 describes the load combinations used in the design of the turbine and the radwaste buildings.

3.8.4.4 Design and Analysis Procedures

The structures described in Subsection 3.8.4.1 are designed to maintain elastic behavior under various loads and their combinations. The loads and the load combinations are fully described in Subsection 3.8.4.3. All reinforced concrete components of the structure are designed by the strength method per ACI 318 (Ref 10A of Table 3.8-1). All structural steel components are designed by the working stress method per AISC specification (Ref 1H of Table 3.8-1).

Determination of wind and tornado loads is described in Section 3.3.

Seismic design of structures is described in Section 3.7. The buildings are analyzed dynamically.

Design of structure for missile protection is covered in Subsection 3.5.3.

Computer programs STRESS and ICES STRUDL-II (Ref 1 and 2 respectively of Subsection 3.8.4.8) are used to analyze structural steel framing.

The refueling facility of the reactor building is designed based on finite element analysis by use of computer program MRI/STARDYNE 3 (Ref 3 of Subsection 3.8.4.8).

The spray pond is basically a soil structure. Its design is discussed in Subsection 2.5.5.

Concrete masonry blockwalls in all Seismic Category I structures have been analyzed dynamically as described in Section 3.7b.3.1.5. They are designed for out-of-plane and in-plane inertia forces generated by the mass of the blockwall and attachment loads, combined with other loads as described in Tables 3.8-8 and 3.8-9. Walls in the turbine and radwaste buildings have been designed for seismic loads per UBC (Ref. 1L of Table 3.8-1).

3.8.4.5 Structural Acceptance Criteria

Reinforced Concrete

The reinforced concrete structural components are designed by the strength method per ACI 318 (Ref 10A of Table 3.8-1) for loads and load combinations described in Subsection 3.8.4.3. The margins of safety are contained in the capacity reduction factors (ø) specified in the code.

Structural Steel

The structural steel components are designed by the working stress method per AISC specification (Ref 1H of Table 3.8-1) for loads and load combinations described in Section 3.8.4.3. The allowable stresses for different load combinations are indicated therein.

Concrete Block Masonry Walls

All masonry blockwalls are reinforced walls and do not act as shear walls. Masonry blockwalls are designed by the working stress method per UBC (Ref. 1L of Table 3.8-1). The allowable loads per UBC Tables 24-B or 24-H (special inspection) are modified as described in Tables 3.8-8, 3.8-9 and 3.8-12, except as noted below.

For double wythe walls designed as composite sections and having concrete or grout infill thickness of 8 inches or more, the allowable shear or tension between masonry block and infill is $1.1\sqrt{f'_m}$ i.e. 43 p.s.i. However, the actual design stress does not exceed 15 p.s.i. For other double wythe walls, allowable shear/tension stress is assumed to be zero at the interface.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

3.8.4.6.1 Concrete and Reinforcing Steel

The concrete and reinforcing steel materials are discussed in Appendix 3.8B. Concrete design compressive strengths are given in Table 3.8-11. Materials for concrete block masonry walls are discussed in Appendix 3.8C.

3.8.4.6.2 Structural Steel

3.8.4.6.2.1 Materials

The various structural steel components conform to the following specifications:

Item	Specification
Beams, girder, and plates	ASTM A36 and ASTM A588
Box columns including base plates and cap plates	ASTM A588
Structural tubing	ASTM A500 and ASTM A501
High strength bolts	ASTM A325 and ASTM A490
Studs	AWS D1.1

3.8.4.6.2.2 Welding and Nondestructive Testing

Welding and nondestructive testing is performed in accordance with either AWS D1.1 (Ref. 1B of Table 3.8-1) or Section IX of the ASME Code (Ref. 1J of Table 3.8-1).

3.8.4.6.2.3 Fabrication and Erection

The fabrication and erection of structural steel conforms to the AISC specification (Ref. 1H, 2H and 3H of Table 3.8-1).

3.8.4.6.2.4 Quality Control

Quality control of structural steel for the construction phase is discussed in Appendix D of the PSAR and amendments to the PSAR.

3.8.4.6.3 Special Construction Techniques

Techniques involved in the construction of Seismic Category I structures are standard construction procedures.

3.8.4.7 Testing and In-service Inspection Requirements

Testing and in-service inspection are not required for Seismic Category I structures (other than the containment).

3.8.4.8 Computer Programs Used in the Design and Analysis of Other Seismic Category I Structures

1) STRESS, Department of Civil Engineering, Massachusetts Institute of Technology

2) ICES STRUDL-II, Department of Civil Engineering, Massachusetts Institute of Technology

3) MRI/STARDYNE (Version 3), Control Data Corporation.

For other computer programs refer to Subsection 2.5.5 and Section 3.7

3.8.5 FOUNDATIONS

This subsection describes foundations for all Seismic Category I structures except the spray pond. The spray pond is basically a soil structure and its design is discussed in Subsection 2.5.5. Descriptions of foundations for safety related non-Seismic Category I structures, such as the turbine building and the radwaste building, are also included in this section.

3.8.5.1 Description of the Foundations

Typical details of the foundations for various structures are shown on Figure 3.8-104.

Reinforced concrete mat foundations have been provided for all structures. The mats rest on sound rock except the ESSW pumphouse mat is supported by natural soil.

All bearing walls of the structures are rigidly connected to the foundation mat. Where steel columns are provided, they are attached to the mat by base plates and anchor bolts. The bearing walls and the steel columns carry all the vertical loads from the structure to the mat. Horizontal shears due to wind, tornado, and seismic loads are transferred to the shear walls by the roof and floor diaphragms. The shear walls transfer the horizontal shears to the foundation mat and from there to the foundation medium through friction. Also, as shown on Figure 3.8-104, the sides of the base mats of all the structures except the ESSW pumphouse are keyed to the foundation rock all around by poured concrete, which helps in transferring the horizontal shears to the foundation rock. The edges of the ESSW pumphouse base mat are poured directly against the excavated slopes of the natural soil formation.

A mudmat (unreinforced concrete layer) is provided between the base of the foundation mat and the foundation medium. Except for the ESSW pumphouse, a waterproofing membrane is provided in the mudmat and on the outside face of peripheral subterranean walls. Perforated pipes are provided around the periphery of the buildings to collect groundwater seepage and drain it to the sumps. Waterproofing membrane under the ESSW pumphouse foundation mat is not considered necessary as the predicted groundwater table at the pumphouse site is well below the foundation mat (refer to Subsection 2.5.5).

Peripheral subterranean walls are designed to resist lateral pressures due to backfill, groundwater, and surcharge loads, in addition to dead loads, live loads, and seismic loads.

Containment: The containment foundation is described in Subsection 3.8.1.

Reactor Building and Control Building: The foundation mats of the reactor and control buildings are poured monolithically.

The reactor building foundation mat is approximately 4 ft 9 in. thick and is reinforced typically with #11 bars at 12 in. centers at top and bottom in both the north-south and east-west directions. The mat surrounds the containment mat, with a cold joint separating the two.

The control building foundation mat is about 2 ft 6 in. thick and is reinforced typically with #8 bars at 12 in. centers at top and bottom in the north-south direction and #11 bars at 12 in. centers at top and #8 bars at 12 in. centers at bottom in the east-west direction. A cold joint is provided between the control and the turbine building mats.

Diesel Generator Building: The foundation mat of the diesel generator building is approximately 2 ft 6 in. thick and is reinforced typically with #9 bars at 12 in. centers at top and bottom in both the north-south and east-west directions. A cold joint is provided between the diesel generator pedestal mat and the diesel generator building

ESSW Pumphouse: The foundation mat of the ESSW pumphouse is about 3 ft thick and is reinforced typically with #9 bars at 12 in. centers at top and bottom in both the north-south and east-west directions.

Turbine Building: The turbine building mat is approximately 2 ft 6 in. thick and is reinforced typically with #6 bars at 12 in. centers at top and bottom in both the north-south and east-west directions. A cold joint is provided between the turbine pedestal mat and the turbine building mat.

Radwaste Building: The radwaste building mat is about 3 ft thick and is reinforced typically with #9 bars at 12 in. centers at top and bottom in both the north-south and east-west directions.

3.8.5.2 Applicable Codes, Standards, and Specifications

The codes, standards, and specifications used in the design, fabrication, and construction of foundations of structures are listed in Table 3.8-1.

3.8.5.3 Loads and Load Combinations

The loads and load combinations used in the design of the containment foundation are described in Subsection 3.8.1.3. The loads and load combinations used in the design of foundations of other Seismic Category I structures are discussed in Subsection 3.8.4.3. In addition, the following load combinations are considered to determine the factors of safety against sliding and overturning due to winds, tornadoes, and seismic loads, and against flotation due to groundwater pressure:

a) D+H+W

b) D+H+W'

c) D+H+E

d) D+H+E'

e) D+F

where:

D, W, W', E, and E' are as described in Subsections 3.8.1.3 and 3.8.4.3 and H and F are as follows:

H = Lateral earth pressure

F = Buoyant force due to groundwater pressure.

3.8.5.4 Design and Analysis Procedures

The foundations are generally designed to maintain elastic behavior under different loads and their combinations. The loads and the load combinations are described in Subsection 3.8.5.3. The design and analysis of the reinforced concrete mat foundations have been carried out in accordance with ACI 318 (Ref 10A of Table 3.8-1).

The bearing walls and the steel columns carry all the vertical loads from the structure to the foundation mat. The lateral loads are transferred to the shear walls by the roof and floor diaphragms, which then transmit them to the foundation mat. Determination of overturning moment due to seismic loads is discussed in Subsection 3.7.2.14.

Except for ESSW pumphouse, settlement of the foundations of Seismic Category I structures is considered negligible as the foundations are supported by sound rock. The settlement of the ESSW pumphouse mat is considered in the design and is discussed in Subsection 2.5.4.

As explained in Subsection 3.8.5.1 and shown in Figure 3.8-104, the sides of the foundation mats (except for the ESSW pumphouse) are keyed to the rock by poured concrete, which resists sliding of the mats. Stability against sliding for the ESSW pumphouse is maintained by the friction on the underside of the basemat and passive resistance of the soil against the edge of the mat.

Detailed description of the foundation rock and soil is contained in Subsections 2.5.4 and 2.5.5. For design purposes, the

allowable bearing pressures of rock and soil are 40 and 2.5 tons/sq ft respectively. The calculated bearing pressures for loads and load combinations described in Subsection 3.8.5.3 do not exceed these allowable values.

The design and analysis of the containment foundation mat are discussed in detail in Subsection 3.8.1.4.

3.8.5.5 Structural Acceptance Criteria

The foundations of all Seismic Category I structures are designed to meet the same structural acceptance criteria as the structures themselves. These criteria are discussed in Subsections 3.8.1.5 and 3.8.4.5. In addition, for the additional load combinations delineated in Subsection 3.8.5.3, the minimum allowable factors of safety against overturning, sliding, and flotation are as follows:

Load Combination	Minimum Factors of Safety		
	Overturning	Sliding	Floatation
a) D+H+W	1.5	1.5	-
b) D+H+W'	1.1	1.1	-

c)	D+H+E	1.5	1.5	-	-
d)	D+H+E'	1.1	1.1		-
e)	D+F	-	-		1.1

The calculated factors of safety exceed the above minimum factor of safety.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The foundations of Seismic Category I structures are constructed of reinforced concrete. The concrete and reinforcing steel materials are discussed in Appendix 3.8B. Concrete design compressive strengths are given in Table 3.8-11. Techniques involved in the construction of these foundations are standard construction procedures.

3.8.5.7 Testing and In-service Inspection Requirements

The containment foundation is load tested during the structural acceptance test as described in Subsection 3.8.1.7. An in-service surveillance program to monitor the settlement of the ESSW pumphouse foundation has been instituted. Detailed discussion of the program is contained in Subsection 2.5.4. Testing and in-service inspection is not necessary for foundations of all other Seismic Category I structures.

TABLE 3.8-1 (pg 1 of 5)

LIST OF APPLICABLE CODES, STANDARDS, RECOMMENDATIONS, AND SPECIFICATIONS

Reference Number	Designation	Title	Edition
(A) American Concrete Institute			
1A	ACI 211.1	Recommended Practice for Selecting Proportions for Normal and Heavyweight Concrete	1970
2A	ACI 214	Recommended Practice for Evaluation of Compression Test Results of Field Concrete	1965
3A	ACI 301	Specifications for Structural Concrete for Buildings	1972
4A	ACI 304	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete	1973
5A	ACI 305	Recommended Practice for Hot Weather Concreting	1972
6A	ACI 306	Recommended Practice for Cold Weather Concreting	1966 (1972)
7A	ACI 307	Specification for the Design and Construction of Reinforced Concrete Chimneys	1969
8A	ACI 308	Recommended Practice for Curing Concrete	1971
9A	ACI 309	Recommended Practice for Consolidation of Concrete	1972
10A	ACI 318	Building Code Requirements for Reinforced Concrete	1971
11A	ACI 347	Recommended Practice for Concrete Formwork	1968
12A	ACI 349	Criteria for Reinforced Concrete Nuclear Power Containment Structures (included in ACI Manual of Standard Practice, Part 2, 1973)	-
13A	ACI SP2	Manual of Concrete Inspection	1975
(B) American Welding Society			
1B	AWS D1.1	Structural Welding Code	1972 (Generally all work) 1975 (Some work after June 1975)
2B	AWS D12.1	Recommended Practice for Welding Reinforcing Steel and Connections in Reinforced Concrete Construction	1961
(C) US Nuclear Regulatory Commission			
1C	RG 1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of	Revision 1

TABLE 3.8-1 (Continued) (pg 2 of 5)

Reference Number	Designation	Title	Edition
		Category I Concrete Structures	Jan. 1973
2C	RG 1.15	Testing of Reinforcing Bars for Category I Concrete Structures	Revision 1 Dec. 1972
3C	RG 1.18	Structural Acceptance Test for Concrete Primary Reactor Containments	Revision 1 Dec. 1972
4C	RG 1.19	Nondestructive Examination of Primary Containment Liner Welds	Revision 1 Aug. 1972
5C	RG 1.54	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Power Plants	June 1973
6C	RG 1.55	Concrete Placement in Category I Structures	June 1973
7C	RG 1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	June 1973
8C	RG 1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel	Aug. 1973
9C	RG 1.69	Concrete Radiation Shields for Nuclear Power Plants	Dec. 1973
10C	RG 1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	Apr. 1975
(D) American Society for Testing and Materials			
1D	ASTM A519	Seamless Carbon and Alloy Steel Mechanical Tubing	1971, 1974, 1975
2D	ASTM A615	Deformed and Plain Billet Steel Bars for Concrete Reinforcement	1972, 1974, 1975
3D	ASTM C29	Unit Weight of Aggregate	1971
4D	ASTM C31	Making and Curing Concrete Test Specimens in the Field	1969
5D	ASTM C33	Concrete Aggregates	1971, 1974
6D	ASTM C39	Compressive Strength of Cylindrical Concrete Specimens	1972
7D	ASTM C40	Organic Impurities in Sands for Concrete	1966, 1973

TABLE 3.8-1 (Continued) (page 3 of 5) | 1

Reference Number	Designation	Title	Edition
8D	ASTM C87	Effect of Organic Impurities in Fine Aggregate on Strength of Mortar	1969
9D	ASTM C88	Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate	1971, 1973
10D	ASTM C94	Ready-Mixed Concrete	1973, 1974
11D	ASTM C109	Compressive Strength of Hydraulic Cement Mortars	1973, 1975
12D	ASTM C117	Materials Finer than No. 200 Sieve in Mineral Aggregates by Washing	1969
13D	ASTM C123	Lightweight Pieces in Aggregate	1969
14D	ASTM C127	Specific Gravity and Absorption of Coarse Aggregate	1968, 1973
15D	ASTM C128	Specific Gravity and Absorption of Fine Aggregate	1968, 1973
16D	ASTM C131	Resistance to Abrasion of Small Size Coarse Aggregate by Use of the Los Angeles Machine	1969
17D	ASTM C136	Sieve or Screen Analysis of Fine and Coarse Aggregates	1971
18D	ASTM C138	Unit Weight, Yield, and Air Content of Concrete	1973, 1974, 1975
19D	ASTM C142	Clay Lumps and Friable Particles in Aggregates	1971
20D	ASTM C143	Slump of Portland Cement Concrete	1971, 1974
21D	ASTM C150	Portland Cement	1973, 1974, 1976, 1978, 1980 | 18
22D	ASTM C215	Fundamental Transverse, Longitudinal, and Torsional Frequencies of Concrete Specimens	1960
23D	ASTM C231	Air Content of Freshly Mixed Concrete by the Pressure Method	1973, 1974, 1975
24D	ASTM C235	Scratch Hardness of Coarse Aggregate Particles	1968
25D	ASTM C260	Air Entraining Admixtures for Concrete	1973, 1974
26D	ASTM C289	Potential Reactivity of Aggregates	1971
27D	ASTM C295	Petrographic Examination of Aggregates for Concrete	1965
28D	ASTM C311	Sampling and Testing Fly Ash for Use as an Admixture in Portland Cement Concrete	1968

TABLE 3.8-1 (Continued)

Reference Number	Designation	Title	Edition
29D	ASTM C330	Lightweight Aggregates for Structural Concrete	1969, 1975
30D	ASTM C469	Static Modulus of Elasticity and Poisson's Ratio of Concrete in Compression	1965
31D	ASTM C494	Chemical Admixtures for Concrete	1971
32D	ASTM C566	Total Moisture Content of Aggregate by Drying	1967
33D	ASTM C618	Fly Ash and Raw or Calcined Natural Pozzolans for Use in Portland Cement Concrete	1973
34D	ASTM C637	Aggregates for Radiation Shielding Concrete	1973
(E)	American Association of State Highway and Transportation Officials		
1E	AASHTO T26	Quality of Water to be Used in Concrete	1970
2E	AASHTO T150	Percentage of Particles of Less Than 1.95 Specific Gravity in Coarse Aggregate	1949
3E	AASHTO T161	Resistance of Concrete Specimens to Rapid Freezing and Thawing in Water	1970
(F)	US Army Corps of Engineers		
1F	CRD C36	Test for Thermal Diffusivity of Concrete	1973
2F	CRD C39	Test for Coefficient of Linear Thermal Expansion of Concrete	1955
3F	CRD C119	Test for Flat and Elongated Particles in Coarse Aggregate	1953
(G)	American National Standards Institute		
1G	ANSI N45.2.5	Supplementary QA Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants.	1972
2G	ANSI N101.6	Concrete Radiation Shields	1972
(H)	American Institute of Steel Construction		
1H	AISC	Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings and Supplement Nos. 1, 2 and 3	1969

TABLE 3.8-1 (Continued)

Reference Number	Designation	Title	Edition	
2H	AISC	Code of Standard Practice for Steel Buildings and Bridges	1970 (Some work before) 1972 (Generally all work) 1976 (Some work after Sept. 1976)	
3H	AISC	Specification for Structural Joints Using ASTM A325 or A490 Bolts	1966, 1972 and 1976	3
4H	AISC	Specification for the design, fabrication and erection of Structural Steel for buildings	1978 (Some work after July 1977)	18
(J) American Society of Mechanical Engineers				
1J	ASME	ASME Boiler and Pressure Vessel Code, Sections II, III, V, VIII, and IX	1971 with Addenda through Summer 1972	
(K) Bechtel Power Corporation, San Francisco, California, Topical Reports				
1K	BC-TOP-1	Containment Building liner Plate Design Report	Revision 1 Dec. 1972	
2K	BC-TOP-4-A	Seismic Analyses of Structures and Equipment for Nuclear Power Plants	Revision 3 Nov. 1974	
3K	BC-TOP-9A	Design of Structures for Missile Impact	Revision 2 Sept. 1974	
(L) International Conference of Building Officials				
1L	UBC	Uniform Building Code	1973, 1976	

TABLE 3.8-2

LOAD COMBINATIONS FOR PRIMARY CONTAINMENT AND DRYWELL FLOOR 12

Notations:

S = Required capacity of the section based on the working stress design method and the allowable stresses in ACI 318-71, Section 8.10 except that the maximum allowable tensile stress for reinforcement shall be 0.5½Fy, where Fy is specified yield strength of reinforcing steel.

U = Required capacity of the section based on the strength design method described in ACI 318-71.

D = Dead load

L = Live load

T_t = Thermal effects anticipated at time of structural acceptance test.

T_o = Thermal effects during normal operating conditions including temperature gradients and equipment and pipe reactions.

T_a = Added thermal effects (over and above operating thermal effects) which occur during a design accident.

P = Design basis accident pressure load

R = Local force or pressure on structure due to postulated pipe rupture including the effects of steam/water jet impingement, pipe whip, pipe reaction, steam pressurization, and water flooding. 19

E = Load due to Operating Basis Earthquake.

E' = Load due to Safe Shutdown Earthquake.

B = Hydrostatic loading due to post-LOCA flooding of the primary containment to the reactor core.

P' = Pressure of atmosphere in the primary containment with the containment flooded to the reactor core.

P_v = External Pressure Load 12

The primary containment and drywell floor are designed for the following load combinations:

Condition	
Preoperational Testing	$S = 1.0D+1.0L+1.0T_t+1.15P$

TABLE 3.8-11

CONCRETE DESIGN COMPRESSIVE STRENGTHS

Structure	Concrete Design Compressive Strength, f'_c (psi)
Turbine generator pedestal	3000
All other Seismic Category I and safety-related, non-Seismic Category I structures and their associated foundation mats including: a) Containment (including its internal structures) b) Reactor Building c) Control Building d) Diesel Generator Building e) ESSW Pumphouse f) Spray Pond g) Turbine Building h) Radwaste Building	4000

APPENDIX 3.8B

CONCRETE, CONCRETE MATERIALS, QUALITY CONTROL, AND SPECIAL CONSTRUCTION TECHNIQUES

Materials, workmanship, and quality control are based on the codes, standards, recommendations and specifications listed in Table 3.8-1. These documents are modified as required to suit the particular conditions associated with nuclear power plant design and construction while maintaining structural adequacy. Extent of application and principal exceptions are indicated herein, and as follows:

ACI 301-72

a) Provisions of ACI 301-72, Chapter 12, Curing and Protection, shall be modified as follows:

i) Paragraph 12.2.1 shall be revised to read as follows:

"For concrete surfaces not in contact with forms, one of the following procedures shall be applied immediately after completion of placement and finishing except that the curing process may be interrupted as necessary not to exceed 8 hours providing requirements for weather protection are maintained. Such curing process may not be interrupted more than twice with a minimum of 8 hours elapsing between interruptions. If the curing is interrupted for up to 8 hours, the curing time shall be extended to provide a total of 7 days curing.

ii) Paragraph 12.2.3 shall be revised to read as follows:

"Curing in accordance with Section 12.2.1 and 12.2.2 shall be continued for at least 7 days in the case of all concrete except high-early-strength concrete for which the period shall be at least 3 days. Alternatively, if tests are made of cylinders kept adjacent to the structure and cured by the same methods, moisture retention measures may be terminated prior to 7 days when test results indicate that the average compressive strength has reached 70 percent of the specified strength, f'c. Required period of initial curing need not be greater than the lesser of the two periods. If one of the curing procedures of Section 12.2.1.1 through 12.2.1.4 is used initially, it may be

replaced by one of the other procedures of Section 12.2.1 any time after the concrete is one day old provided the concrete is not permitted to become surface dry during the transition. Curing during periods of cold weather shall be in accordance with Section 12.3.1."

iii) Paragraph 12.3.1 shall be deleted and replaced with the following:

"Initial curing and protection measures for the concrete during periods of cold weather shall be in accordance with the recommendations of ACI 306-66 (1972)."

b) Provisions of ACI 301-72, Chapter 14, Massive Concrete, shall be modified as follows:

i) Paragraph 14.4.1 shall be deleted and replaced with the following:

"The slump of the concrete as placed shall be 3" or less except that a tolerance of up to 2" above this indicated maximum shall be allowed for batches provided the average for all batches or the most recent 10 batches tested, whichever is fewer, does not exceed 3". Concrete of lower than usual slump may be used provided it is properly placed and consolidated."

ii) Paragraph 14.4.3. Delete the first sentence of the paragraph and substitute the following:

"Concrete shall be placed in layers approximately 24" thick."

iii) Paragraph 14.5.1 shall be deleted and replaced with the following:

"The minimum curing period shall be in accordance with Section 12.2.3."

iv) Paragraph 14.5.4. The requirement for controlled cooling at the conclusion of the specified heating shall be accomplished by leaving the cold weather protection in place at least 24 hours after heating is discontinued. In extremely cold weather, the field engineer shall require that additional measures be taken to prevent rapid cooling of the concrete by this method.

ACI 318-71

a) Provision of ACI 318-71, Chapter 5, "Mixing and Placing Concrete" shall be modified as follows:

i) Paragraph 5.5 shall be revised by the addition of the following new paragraph 5.5.3:

5.5.3 The curing requirements as described in Sections 5.5.1 and 5.5.2 above may be interrupted as necessary not to exceed 8 hours providing requirements for weather protection are maintained. Such curing process may not be interrupted more than twice with a minimum of 8 hours elapsing between interruptions. If the curing is interrupted for up to 8 hours, the curing time shall be extended to provide a total of 7 days curing.

b) Provisions of ACI 318-71, Chapter 6, Formwork, Embedded Pipes, and Construction Joints, shall be modified as follows:

i) Paragraphs 6.3.2.4, 6.3.2.5, 6.3.2.6 and 6.3.2.7 shall be deleted and replaced with the following:

6.3.2.4 "All piping and fittings shall be tested in accordance with the requirements of the code governing that piping system (e.g., ASME Boiler and Pressure Vessel Code, ANSI B 31.1, state or local plumbing codes, etc.) or in accordance with applicable design or technical specifications, or design drawings.

Whenever the piping system is not governed by such applicable codes, code cases or design documents, then such systems shall be tested for leaks prior to concreting. The testing pressure above atmospheric pressure shall be 50 percent in excess of the pressure to which the piping and fittings may be subjected, but the minimum testing pressure shall be not less than 150 psig. The pressure test shall be held for 4 hours with no drop in pressure except that which may be caused by air temperature."

6.3.2.5 "Drain pipes and other piping systems not governed by applicable codes and designed for pressures of not more than 1 psig need not be tested as required above."

6.3.2.6 "Piping systems which are not governed by applicable codes, code cases or design documents and which carry liquid, gas or vapor which is explosive or injurious to health, shall be retested in accordance with Section 6.3.2.4 subsequent to the hardening of the concrete."

6.3.2.7 "Piping systems may be energized with water not exceeding 50 psi nor 90°F if approved by the responsible Field Engineer".

Other piping systems, including systems governed by piping system codes or design documents exceeding 50 psi or 90°F or energized with other than water, may be energized 7 days after the concrete placement provided that the temperature does not exceed 150°F nor the pressure exceed 200 psig. Piping systems may be energized prior to and during the placement of concrete provided that: (a) the above temperature and pressure restrictions are applied, (b) the energized system is not shut down within 24 hours of concrete placement, and (c) if the pressure in the energized system drops, the lower pressure shall become the limiting pressure until the seven-day post-placement time limit has elapsed. Piping systems which have been energized within 24 hours of concrete placement may be reenergized at any time more than 24 hours after concrete placement up to the limiting pressure.

3.8B.1 CONCRETE AND CONCRETE MATERIALS - QUALIFICATIONS

3.8B.1.1 Concrete Material Qualifications

Cement

Cement is Type II, portland cement conforming to ASTM C150. Certified copies of material test reports showing chemical composition of the cement and verification that the cement being furnished complies with requirements are furnished by the manufacturer for each batch or lot.

Normal Weight Aggregate

Fine and coarse aggregates conform to ASTM C33. Aggregate source acceptability is based on the following test requirements:

Method of Test	Designation
Unit Weight of Aggregate	ASTM C29
Organic Impurities in Sands	ASTM C40
Effect of Organic Impurities in Fine Aggregate on Strength of Mortar	ASTM C87
Soundness of Aggregates	ASTM C88
Materials Finer Than No. 200 Sieve	ASTM C117
Lightweight Pieces in Aggregate	ASTM C123
Specific Gravity & Absorption of Coarse Aggregate	ASTM C127
Specific Gravity & Absorption of Fine Aggregate	ASTM C128
L. A. Abrasion	ASTM C131
Sieve or Screen Analysis of Fine & Coarse Aggregates	ASTM C136
Clay Lumps & Friable Particles	ASTM C142
Scratch Hardness of Coarse Aggregates	ASTM C235

Potential Reactivity of Aggregate	ASTM C289
Petrographic Examination	ASTM C295
Lightweight Aggregates	ASTM C330
Percentage of Particles of Less Than 1.95 Specific Gravity in Coarse Aggregate	AASHTO T150
Resistance of Concrete Specimens to Rapid Freezing and Thawing in Water	AASHTO T161
Flat and Elongated Particles	CRD C119

Coarse aggregate loss from the L.A. Abrasion Test (ASTM C131) using Grading A is limited to 40 percent by weight at 500 revolutions.

Coarse aggregate grading is for size numbers 4, 8, and 67 as defined in ASTM C33 and the quantity of flat and elongated particles is limited to 15 percent in any nominal size group.

When fine and coarse aggregates are tested per ASTM C117 to meet the requirements of ASTM C33, and when the results of any of the aggregate sizes exceeds the stated limits for fines, the aggregate is accepted, provided the total amount of aggregate fines in a given mix is not greater than the total amount permitted for each aggregate size at ASTM C33 limits.

High Density Aggregates

The requirements for high density aggregates are the same as for normal density aggregates except as noted below.

Fine and coarse aggregate conforms to ASTM C637 except that grading is as follows:

Sieve Size U.S. Std. Sq. Mesh	Percentage Passing Fine Aggregate (Sand)	Percentage Passing Coarse Aggregate 1-1/2 in.
2 in.		100
1-1/2 in.		95-100
3/4 in.		35-70
3/8 in.	100	10-30
No. 4	75-95	2-15
No. 8	55-85	0-10
No. 16	30-60	
No. 30	15-45	
No. 50	10-30	
No. 100	0-15	

The fineness modulus of the fine aggregate is not less than 3.2 nor more than 4.2. Both fine and coarse aggregate have a minimum bulk specific gravity of 4.0.

These aggregates are not tested per AASHTO T161 unless the structure is exposed to a design freeze-thaw environment and are also not tested per ASTM C330.

Certified test reports are prepared by an independent testing laboratory for each material shipment attesting to aggregate conformance to cleanliness requirements when tested per ASTM C117 and specific gravity requirements when tested per ASTM C127 and C128.

Pozzolan

Pozzolan, when used, conforms to ASTM C618 for Class F except that the maximum loss on ignition is 6 percent. Prior to shipment a minimum of one sample is taken and tested in accordance with ASTM C311 to demonstrate conformance with the above. Such documentation accompanies material shipment.

Mixing Water and Ice

Water and ice used in mixing concrete is free of injurious amounts of oil, acid, alkali, organic matter, or other deleterious substances as determined by AASHTO T26. Such water and ice does not contain impurities that would cause either a change in the setting time of portland cement of more than 25 percent or a reduction in compressive strength of mortar of more than 5 percent compared with results obtained with distilled water. The water and ice do not contain more than 250 ppm of chlorides as Cl, or more than 1000 ppm of sulphates as SO4. The pH range is between 4.5 and 8.5.

Admixtures

Air entraining admixtures, when used, conform to ASTM C260. Water reducing and retarding admixtures, when used, conform to ASTM C494 for types A and D. Types A and D are used in accordance with the manufacturer's recommendations. Certificates of conformance stating conformance to the applicable ASTM specification are furnished with each shipment. Use of calcium chloride is not permitted.

3.8B.1.2 Concrete Mix Design

Concrete Properties

Concrete properties required for each type of mix design are verified by testing for the applicable properties indicated below:

Property	Test Designation
Compressive Strength	ASTM C39
Unit Weight	ASTM C138
Slump	ASTM C143
Air Content	ASTM C231

The following additional properties of selected mix designs have been determined to ascertain material compatibility with design assumptions:

Static Modulus of Elasticity	ASTM C469
Static Poisson's Ratio	ASTM C469
Dynamic Modulus of Elasticity	ASTM C215
Dynamic Poisson's Ratio	ASTM C215
Thermal Diffusivity	CRD C36
Thermal Coefficient of Expansion	CRD C39

Concrete Mix Proportions

Proportions of ingredients are determined and tests conducted in accordance with ACI 211.1, except as noted below, for combinations of materials established by trial mixes. These proportioning methods provide required concrete strength,

durability, and unit weight while maintaining adequate workability and proper consistency to permit required consolidation without excessive segregation or bleeding.

The design strength (f'c) of mixes that contain pozzolan is measured at 90 days; for those that do not contain pozzolan, f'c is measured at 28 days. Three cylinders are tested for each mix design and age as follows:

Pozzolan Mix	Nonpozzolan Mix
3 days	3 days
7 days	7 days
28 days	28 days
90 days	

Concrete mixes for limited uses such as in radiation-sensitive facilities and high density concrete do not contain pozzolan. All other concrete mixes are based on use of approximately 15 to 20 percent pozzolan by weight as cement replacement. Further concrete mixes except limited application use, such as high density concrete, are based on 3 to 6 percent air entrainment for both 3/4 and 1-1/2 in. nominal maximum size coarse aggregate. These measures provide a concrete possessing both good freeze-thaw and sulphate resistance.

In lieu of establishing limits on water-cement ratio, the concrete is proportioned and mixed so as to be placed at specified slumps. The average slump at the point of placement is less than the "Working Limit", which is the maximum slump for estimating the quantity of mixing water to be used in the concrete. An "Inadvertency Margin" is the allowable deviation from the "Working Limit" for such occasional batches as may inadvertently exceed the "Working Limit". Jobsite tests have indicated that concrete with slumps at the "Inadvertency Margin" will produce acceptable quality concrete.

3.8B.1.3 Grout

Construction Grout

Construction grout for use at horizontal construction joints and similar applications is proportioned from the same materials as for concrete. Grout strength is determined in accordance with ASTM C109.

Starter Mix

Starter mixes are used in applications such as at the bottom of foundation slabs and in lieu of construction grout and are proportioned from the same materials as for concrete. These mixes are generally proportioned for a "Working Limit" slump 2 in. greater than the associated concrete mix. Trial mixes are prepared and tested for strength as described for general concrete mixes.

Nonshrink Grout

Nonshrink grout is prepared from proprietary materials such as Embeco LL-636 by Master Builders Company or Five Star Grout by US Grout Corporation. Such grouts are proportioned in accordance with the manufacturer's recommendations and are tested for expansion, compressive strength, and flow characteristics with maximum water content recommended by the manufacturer prior to use.

3.8B.2 CONCRETE AND CONCRETE MATERIALS - BATCHING, PLACING, CURING, AND PROTECTION

3.8B.2.1 Storage

Storage of aggregates, cement, pozzolan, and admixtures is in accordance with the recommendations of ACI 304.

3.8B.2.2 Batching, Mixing, and Delivering

Concrete for principal structures is provided as central mixed concrete from a batch plant located on the jobsite. Some limited amounts of concrete are obtained from an offsite batch plant. All such batch plant facilities are certified by the National Ready Mix Concrete Association (NRMCA) and measuring devices are calibrated at required intervals and more frequently when deemed appropriate.

Measuring of materials, batching, mixing, and delivering normal weight concrete conform to ASTM C94, Alternate No. 1 except as otherwise noted.

Regulatory Guide 1.69 has basically adopted ANSI N101.6. This ANSI standard is intrepreted to be applicable only to high density concrete serving as radiation shields and is therefore not used on this project. As the concrete has a dual function of providing shielding and structural adequacy, the standard

practices as described herein are adopted for normal weight concrete. When higher density concrete is required for shielding purposes, the practices adopted are in general agreement with those outlined in the ACI Journal of August 1975 report by ACI Committee 304: "High Density Concrete Measuring, Mixing, Transporting, and Placing".

The delivery of materials from the batching equipment is within the following limits of accuracy:

Material	Over and Under Percent Weight: Less than or equal to 30 percent of scale capacity	Over and Under Percent Weight: Greater than 30 percent of scale capacity
Cement	Minus 0 Plus 4	1
Pozzolan	Minus 0 Plus 4	1
Water	1	1
Ice	1	1
Aggregate equal to or smaller than 1-1/2	3 (See note below)	2
Admixture when batched separately	3	3

Note: Or plus or minus 0.3 percent of scale capacity, whichever is less.

3.8B.2.3 Placing

Placing of normal weight concrete is in accordance with the recommendations of ACI 304. Placing of high density concrete is as described above.

3.8B.2.4 Consolidation

Consolidation of concrete is in accordance with the recommendations of ACI 309.

3.8B.2.5 Curing

Curing of concrete is in accordance with the recommendations of ACI 308.

3.8B.2.6 Hot and Cold Weather Concreting

Measures taken to mitigate the effects of hot and cold weather during each step of the concreting operation are in accordance with ACI 305 and 306 respectively.

3.8B.3 CONCRETE AND CONCRETE MATERIALS-CONSTRUCTION TESTING

An independent concrete and concrete materials testing laboratory has been established at the project site to monitor the quality of such work and materials and to promptly report any deviations from specified conditions. Such testing personnel are qualified to meet the requirements of NRC Regulatory Guide 1.58. Procedures and tests for accomplishing such work are reviewed and accepted by Bechtel prior to use. Qualifications and procedures in use by Bechtel quality control personnel and extent of conformance to Regulatory Guide 1.94 are described in Section 3.13.

Production testing for concrete and concrete materials is as shown in Table 3.8B-1.

Materials that do not meet test requirements are not used in the construction.

If the measured concrete temperature, slump, unit weight, or air content falls outside the limits specified, a check is made. In the event of a second failure, the load of concrete represented is not used in the construction.

Concrete cylinder test results are reviewed for compliance with Chapter 17 of ACI 301 and are evaluated in accordance with ACI 214.

Materials or portions thereof that do not meet the above criteria but may inadvertently be used are handled as described in Appendix D and amendments to the PSAR.

3.8B.4 CONCRETE REINFORCEMENT MATERIALS - QUALIFICATIONS

Reinforcing steel for concrete structures conforms to ASTM A615, Grade 60, including Section S1 for bar sizes 14 and 18. Certified copies of material test reports indicating chemical composition, physical properties and dimensional compliance are furnished by the manufacturer for each heat.

When permitted by the design drawings, reinforcing steel is furnished by the supplier to special chemistry requirements to enhance reinforcing weld characteristics. The chemistry of such bars meets the following chemical analysis requirements expressed in maximum percentage by weight:

C	-	0.50%	P	-	0.05%
Mn	-	1.30%	S	-	0.05%

Weld splicing of reinforcing is not performed in the primary containment structures.

Each bundle of reinforcing steel is tagged to ensure unique heat traceability during production, while in transit and into storage. During storage and installation reinforcing steel is collectively traceable to the group of certified material test reports received.

Prior to installation at the jobsite all reinforcing steel is subjected to a testing program meeting the requirements of NRC Regulatory Guide 1.15. Any reinforcing steel which does not meet these requirements is not used in the construction.

Sleeves for reinforcing steel mechanical splices conform to ASTM A519 for Grades 1018 and 1026. Certified copies of material test reports indicating chemcial composition and physical properties are furnished by the manufacturer for each sleeve lot.

3.8B.5 CONCRETE REINFORCEMENT MATERIALS - FABRICATION

3.8B.5.1 Bending Reinforcement

Hooks and bends are fabricated in accordance with ACI 318 Chapter 7.1. Bars partially embedded in concrete are bent subject to the following conditions.

Bending Partially Embedded Reinforcement

The minimum distance from existing concrete surface to the beginning of bend and the minimum inside diameter of bend is:

Bar Size	Min. Dist. from Surface to Beginning of Bend	Min. Inside Bend Diameter
No. 3 through No. 8	3 Bar Diameters	6 Bar Diameters
No. 9, No. 10, No. 11	4 Bar Diameters	8 Bar Diameters
No. 14, No. 18	5 Bar Diameters	10 Bar Diameters

Bars No. 3 to No. 5 inclusive may be bent cold once. Heating is required for subsequent straightening or bending.

Bars No. 6 and larger may be bent and straightened, provided that heating is used.

When heat is used, it is applied as uniformly as possible over a length of bar equal to 10 bar diameters, and is centered at the middle of the arc of the completed bend. The maximum bar temperature is between 1100 and 1200°F, and maintained at that level until bending (or straightening) is complete.

Temperature-measuring crayons or a contact pyrometer is used to determine the temperature. Heat is applied in such a way as to avoid damage to the concrete. Care is taken to prevent rapid quenching of heated bars.

Straightened bars are visually inspected to determine whether they are cracked, reduced in cross-section, or otherwise damaged. Any damaged portions are removed and replaced.

3.8B.5.2 Splicing Reinforcement

Lap Splices

In general, lapped splices are used for No. 11 and smaller bars. Such lap splices are in accordance with Sections 7.5, 7.6, and 7.7 of ACI 318.

Mechanical Splices

In general, mechanical (Cadweld) splices are used for all No. 14 and No. 18 splices, for splices across liner plates and in lieu of standard hooks when a plate anchorage is required or desirable. To obtain an effective level of quality control for this splicing process, a qualification, inspection, testing, and acceptance program in accordance with NRC Regulatory Guide 1.10 has been used. Welding of splice sleeves to liners, or other plates and shapes is in accordance with AWS D1.1.

Welded Splices

Whenever both lap and mechanical splices have been determined to be impractical, welded splices are used on a case-by-case approval basis. Such welding is performed by qualified welders using a procedure conforming to the basic recommendations of AWS D12.1.

3.8B.5.3 Placing Reinforcement

Reinforcement is securely tied with wire and held in position by spacers, chairs, and other supports to maintain placement accuracy within the tolerances established for reinforcement protection and the design requirements.

3.8B.5.4 Spacing Reinforcement

Spacing of reinforcement is in accordance with Sections 3.3.2, 7.4.1, and 7.4.5 of ACI 318.

3.8B.5.5 Surface Condition

Reinforcement surface condition at the time of concrete placement is in accordance with Section 7.2 of ACI 318.

3.8B.6 CONCRETE REINFORCEMENT MATERIALS - CONSTRUCTION TESTING

Inspection of reinforcement materials to ensure that bending, placing, splicing, spacing, and surface condition requirements are met is in accordance with the program described in Chapter 17 as is the extent of conformance to Regulatory Guide 1.94.

3.8B.7 FORMWORK AND CONSTRUCTION JOINTS

Formwork is designed and constructed in accordance with ACI 347. Such formwork maintains position and shape to keep deformations within limits established by the design requirements.

Prior to concrete placement, construction joints are cleaned to remove unsatisfactory concrete, laitance, coatings, debris, and other foreign material and to expose the aggregate. The joints are then saturated to produce a saturated surface dry condition. Horizontal construction joints then shall be covered with either approximately 1/4 in. of construction grout or a layer of starter mix which is approximately 4 to 6 in. deep.

Except as discussed below, concrete is placed in accordance with Regulatory Guide 1.55.

Regulatory positions 2 and 3 of the Regulatory Guide state the presumed functional responsibilities of the "Designer" and the "Constructor". Under the designer's role are listed the responsibilities for checking shop drawings and locations of construction joints. On this project, the former is fully delegated to the Bechtel field, although the design engineering office may check significant portions and may advise the field accordingly. The responsibility for construction joint location is partly delegated to the field in the sense that the field has to follow the guidelines set out in the design drawings and specifications prepared by the design engineering office. In interface areas, a delegation of the design engineering office's responsibility to the field office is within the definition of the terms "responsibility" and "delegated responsibility" as discussed in Paragraph 1.3 of the proposed ANSI N45.2.5. Delegation of the responsibilities for checking the reinforcing drawings to the field engineering group is justified by the following:

a) The Bechtel field engineering group is segregated from the field supervision group, although both are located at the jobsite and eventually report to the project construction manager.

b) The field engineering group is staffed, for the most part, by graduate engineers who have been trained in the use of the ACI code and understand the design implication of the proper location, splicing, and embedment of reinforcing steel.

c) The field inspection of the actual rebar as placed in the forms is conducted using the engineering drawings as the primary source document. This ensures a check on any errors which may have passed the critical review of the field engineer in checking the shop detail or erection drawings.

d) It is standard practice in the civil engineering profession that engineering requirement drawings for reinforcing be converted to shop detail and erection drawings in accordance with ACI standards applied by steel detailers at the reinforcing steel vendor's shop. Most contractors installing reinforcing steel rely upon their superintendent and foreman for correct interpretation of these detail drawings in erecting the reinforcing steel. While this is also true of Bechtel field operation, we do have the additional help and guidance of the field engineers both during the installation phase and finally at the inspection phase prior to final sign-off on the report card.

e) The field engineers have the added benefit of being able to plan and witness the actual installation and can, therefore, better foresee any difficulties in meeting the intended design requirements. Their assessment of the situation is further assisted by regular telephone communication with the design engineers who also periodically visit the jobsite.

The above procedure of delegation of the design engineering office's responsibility to the field personnel and periodic monitoring by the engineering office ensures correctness and conformance of the shop drawings to the design drawings and therefore meets the intent of Regulatory Guide 1.55.

APPENDIX 3.8C

CONCRETE UNIT MASONRY, MASONRY MATERIALS AND QUALITY CONTROL

Materials, workmanship and quality control are based on the applicable codes, standards, recommendations and specifications listed in Table 3.8-1. These documents are modified as required to suit the perticular conditions associated with nuclear power plant design and construction while maintaining structural adequacy.

3.8C.1 CONCRETE UNIT MASONRY AND MASONRY MATERIALS - QUALIFICATIONS

Concrete Unit Masonry

Concrete unit masonry conforms to either ASTM C90, Type I, Grade N for hollow masonry units or ASTM C145, Type I, Grade S for solid masonry units.

Masonry Mortar

Masonry mortar conforms to ASTM C270, Type M, and is of the following ingredients:

Portland cement conforming to ASTM C150, Type I or II.

Hydrated lime conforming to ASTM C207, Type S.

Aggregate conforming to ASTM C144.

Masonry Grout

Masonry grout conforms to ASTM C476.

Concrete Infill

Concrete infill conforms to the program and requirements described in Appendix 3.8B.

Reinforcing Steel

Reinforcing steel conforms to the program and requirements described in Appendix 3.8B.

Horizontal Joint Reinforcement

Horizontal joint reinforcement is made of wire conforming to ASTM A82. Certificates of compliance stating conformance to ASTM A82 are furnished for the joint reinforcement.

3.8C.2 CONCRETE UNIT MASONRY AND MASONRY MATERIALS - CONSTRUCTION AND ERECTION

Construction and erection of concrete unit masonry and masonry materials is in conformance with the requirements of the Uniform Building Code.

3.8C.3 CONCRETE UNIT MASONRY AND MASONRY MATERIALS - CONSTRUCTION TESTING

An independent testing laboratory has been established at the project site to monitor the quality of concrete unit masonry and masonry materials and to promptly report any deviations from specified conditions. Procedures and tests for accomplishing such work are reviewed and accepted by Bechtel prior to use.

Production testing for concrete unit masonry and masonry materials is as follows:

Concrete Unit Masonry

Tests of concrete unit masonry are performed at a frequency of six units randomly selected from each lot of 5000 units or fraction thereof delivered to the jobsite. Such units are tested in accordance with ASTM C140 to demonstrate compliance with ASTM C90 for hollow masonry units and with ASTM C145 for solid masonry units. Such tests are performed and acceptability determined, prior to use of that lot of masonry units.

Masonry Mortar

Tests of masonry mortar are performed prior to use initially and then for each 5000 concrete masonry units placed. Such tests are performed in accordance with and meet the acceptance standards of ASTM C270.

Masonry Grout

Tests of masonry grout are performed at a frequency of once for each 100 cubic yards of each class of masonry grout produced. Each test consists of 6 two inch cubes made, cured and tested in accordance with ASTM C109. Three cubes are tested at 7 days and three at 28 days.

Concrete Infill

Concrete infill is tested at the same frequency and by the methods described for Appendix 3.8B.

Materials that do not meet test requirements are not used in the construction.

Materials or portions thereof that do not meet the above criteria but may inadvertantly be used are handled as described in Appendix D and amendments to the PSAR.

3.11.5 ESTIMATED CHEMICAL, PHYSICAL, AND RADIATION ENVIRONMENT

3.11.5.1 Suppression Pool, Residual Heat Removal System, and Emergency Core Cooling System Water Quality

The water in these systems shall not be chemically inhibited. The maximum limits for the suppression pool have been established to be compatible with those of the primary coolant and are listed in Table 3.11-7 for comparison.

Observations made of suppression pool water quality over a period of several years in suppression pool with and without coatings, indicate that the feed and bleed to radwaste that occurs during normal system testing and level adjustments maintain the water quality well within the above limits.

During reactor shutdown cooling, the RHR system water mixes with reactor water. Therefore, to insure reactor water quality the shutdown cooling piping and equipment shall be flushed with water of the quality specified above for maximum limit with suspended solids concentration of 5 ppm or less. During layup the RHR system will be filled with water of the following limits.

Parameter	RHR System Maximum Limit
Conductivity	≤ 3 mho/cm at 25°C
Chlorides (as Cl)	≤ 0.05 ppm
pH	5.3 to 7.5 at 25°C

3.11.5.2 Physical Environment

Engineered safety feature (ESF) systems are designed to perform their safety related functions in the temperature, pressure, and humidity conditions described in Subsection 3.11.2, and Sections 6.2 and 6.3.

The containment atmosphere is maintained below 4 percent by volume hydrogen consistent with the recommendations of Regulatory Guide 1.7 as discussed in Subsection 6.2.5.

3.11.5.3 Radiation Environment

ESF systems and components are designed to perform their safety related functions after the normal operational exposure plus an accident exposure. The normal operational exposure is based on

the design source terms presented in Chapter 11 and Subsection 12.2.1 and the equipment and shielding configurations presented in Section 12.3.

Post-accident ESF system and component radiation exposures are dependent on equipment location. In the containment and control room area, exposures are due to a hypothesized LOCA. Source terms and other accident parameters are presented in Subsection 12.2.1, and Chapter 15, and are consistent with the recommendations of Regulatory Guides 1.3 and 1.7.

In the Reactor Building, exposures are based on the assumption that 50 percent of the core halogen inventory and 1 percent of the core solid fission products, as listed in Subsection 5.6.3.5, are recirculated by the ECCS systems.

Normal, accident, and design (normal plus accident) radiation exposures for plant areas, based on the above assumptions, are presented in Table 3.11-6.

Organic materials that exist within the containment are identified in Subsection 6.1.2. The design radiation exposures identified in Table 3.11-6 are based on gamma radiation exposure only except as noted. The effect of beta radiation is effectively attenuated by small amounts of shielding, such as conduits for cable and casings for equipment. For the organic coating materials used inside the containment (see Table 6.1-2), irradiation tests have been performed for a cumulative gamma dose up to 1×10^9 rads, which exceeds the design calculated value in Table 3.11-6, and therefore, conservatively accounts for the surface exposure due to beta radiation in the design basis accident environment.

3.11.6 REFERENCES

3.11-1. J.J. DiNunno, R.E. Baker, F.D. Anderson, and R.L. Waterfield, "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, Division of Licensing and Regulation, AEC, Washington, D.C. (1962).

3.11-2. J.F. Kircher and R.E. Bowman, Effects of Radiation on Materials and Components, Van Nostrand Reinhold, New York, 1964.

TABLE 3.11-6

NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS 18

AREA	KEY (10)	NORMAL OPERATING CONDITIONS PRESSURE	TEMP°F MAX/MIN	RELATIVE HUMIDITY MAX/MIN%	DOSE RATE (R/HR)(12)	INTEGRATED DOSE (RAD)	MAXIMUM CONDITIONS(6) PRESSURE	TEMP °F	RELATIVE HUMIDITY %	LOCA DOSE RATE (RADS/HR)	TOTAL INTEGR. DOSE (RADS)(4)
Primary Containment Drywell, No. RPV Shield	C2	.1 psig to 1.5 psig	150/100	90/20	Gamma $6.5X10^4$ Neutron(2) $6.3X10^7$	$2.3X10^{10}$ $7.9X10^{16}$	See Notes (1),(13) and (7)	See Notes (1),(13) and (7)	See Notes (1),(13) and (7)	$1.3X10^6$ -	$2.3X10^{10}$ $7.9X10^{16}$
With Vessel Shield:											
Zone 1 Above Core	C2a	.1 psig to 1.5 psig	150/100	90/20	Gamma 25 Neutron(2) $3X10^4$	$8.8X10^6$ $6.3X10^{13}$	As Above As Above	As Above As Above	As Above As Above	$1.3X10^6$ -	$3.4X10^7$ $6.3X10^{13}$
Zone 2 Core Region	C2b	.1 psig to 1.5 psig	150/100	90/20	Gamma 50 Neutron(2) $1.4X10^5$	$1.8X10^7$ $1.8X10^{14}$	As Above As Above	As Above As Above	As Above As Above	$1.3X10^6$ -	$4.4X10^7$ $1.8X10^{14}$
Zone 3 Under Vessel	C2c	.1 psig to 1.5 psig	185/100	90/20	Gamma 7.2 Neutron(2) <1	$2.5X10^6$ $<1.3X10^9$	As Above As Above	As Above As Above	As Above As Above	$1.3X10^6$ -	$2.8X10^7$ $<1.3X10^9$
Zone 4 Near Recirculation Pumps	C2d	.1 psig to 1.5 psig	135/100	90/20	Gamma 25 Neutron(2) $2X10^3$	$8.8X10^6$ $2.5X10^{12}$	As Above As Above	As Above As Above	As Above As Above	$1.3X10^6$ -	$3.4X10^7$ $2.5X10^{12}$

TABLE 3.11-6

NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS 18

AREA	KEY (10)	NORMAL OPERATING CONDITIONS PRESSURE	TEMP°F MAX/MIN	RELATIVE HUMIDITY MAX/MIN%	DOSE RATE (R/HR)(12)	INTEGRATED DOSE (RAD)	MAXIMUM CONDITIONS(6) PRESSURE	TEMP °F	RELATIVE HUMIDITY %	LOCA DOSE RATE (RADS/HR)	TOTAL INTEGR. DOSE (RADS)(4)
Zone 5 >15 ft. from Recirc Pumps	C2e	.1 psig to 1.5 psig	150/100	90/20	Gamma 4	$1.4X10^6$	As Above	As Above	As Above	$1.3X10^6$	$2.7X10^7$
					Neutron(2) $2X10^3$	$2.5X10^{12}$	As Above	As Above	As Above	-	$2.5X10^{12}$
Zone 6 Suppression Pool	C3	.1 psig to 1.5 psig	125/90	100/50	Gamma .1	$3.5X10^4$	See Note (8)	See Note (8)	See Note (8)	$1.3X10^6$	$2.6X10^7$
					Neutron(2) $2X10^2$	-				-	$7X10^7$
Core Spray Pump Rooms	R1a	-.375" wg	100/60	90/10	.015	$5.3X10^3$	-.25"wg	130	100/90 Note(15)	$1.6X10^2$	$5X10^4$
HPCI Pump Rooms & Penetration Room	R1b	-.375" wg	104/60	90/10	.015	$5.3X10^3$	6.1 psig [-.25" wg]	305 for 60 sec [130](3)	100/90 Note(15)	$1.6X10^2$	$5X10^4$
RHR Piping & Penetration Room At El. 683	R1c	-.375" wg	115/60	90/10	.015	$5.3X10^3$	2.48 psig [-.25" wg]	305 for 60 sec [130](3)	100/90 Note(15)	$1.6X10^2$	$5X10^4$
RWCU System Heat Exchanger Room	R1d	-.375" wg	110/60	90/10	15	$8.8X10^6$	2.4 psig [-.25" wg]	220 for 40 sec [130](3)	100/90 Note(15)	$6.5X10^2$	$8.9X10^6$

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TABLE 3.11-6

NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS

AREA	KEY (10)	NORMAL OPERATING CONDITIONS PRESSURE	TEMP°F MAX/MIN	RELATIVE HUMIDITY MAX/MIN%	DOSE RATE (R/HR)(12)	INTEGRATED DOSE (RAD)	MAXIMUM CONDITIONS(6) PRESSURE	TEMP °F	RELATIVE HUMIDITY %	LOCA DOSE RATE (RADS/HR)	TOTAL INTEGR. DOSE (RADS)(4)
RWCU System Recirculation Pump Room & Penetration Area	R1e	-.375" wg	104/60	90/10	<.05	$1.8X10^4$	3.5 psig [-.25" wg]	222 for 40 sec [130](3)	100/90 Note(15)	$6.5X10^2$	$1.9X10^5$
RWCU System Filters Tanks & Pump Rooms	R1f	-.375" wg	104/60	90/10	10	$3.6X10^6$	-.25" wg	104(11)	100/90 Note(15)	$6.5X10^2$	$3.8X10^6$
Reactor Building Steam Tunnel	R3	-.375" wg	130/40	90/10	5	$1.8X10^6$	8.2 psig [-.25" wg]	300°F for 15 sec [130](3)	100	$1.6X10^2$ [$>2.5X10^2$](5)	$1.8X10^6$
Refueling Floor	R5	-.25" wg	100/60	90/10	-	-	-.25" wg	104(9)	100/90 Note(15)	-	-
RHR Pump Rooms	R1g	-.375" wg	100/60	90/10	.015	$5.3X10^3$	1.84 psig [-.25" wg]	296 for 60 sec [130](3)	100/90 Note(15)	$1.6X10^2$	$5X10^4$
RCIC Pump Room & Penetration Room	R1h	-.375" wg	102/60	90/10	.015	$5.3X10^3$	2.2 psig [-.25" wg]	306 for 25 sec [130](3)	100/90 Note(15)	$1.6X10^2$	$5X10^4$

TABLE 3.11-6

NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS 18

AREA	KEY (10)	NORMAL OPERATING CONDITIONS PRESSURE	TEMP°F MAX/MIN	RELATIVE HUMIDITY MAX/MIN%	DOSE RATE (R/HR) (12)	INTEGRATED DOSE (RAD)	MAXIMUM CONDITIONS (6) PRESSURE	TEMP °F	RELATIVE HUMIDITY %	LOCA DOSE RATE (RADS/HR)	TOTAL INTEGR. DOSE (RADS) (4) 5
Reactor Building General Access Areas	R1n	-.25" wg	100/60	90/10	.0025	$8.8X10^2$	-.25" wg	104	90	$6.5X10^2$	$1.7X10^5$
Standby Liquid Control Area	R1n	-.25" wg	100/60	90/10	.0025	$8.8X10^2$	-.25" wg	104	90	$6.5X10^2$	$1.7X10^5$
Emergency Switchgear	R1i	-.125" wg	104/70	90/10	-	-	-25" wg	122	90	-	-
Penetration Rooms Not Otherwise Noted	R1j	-.375" wg	130/60	90/10	.0025	$8.8X10^2$	-.25" wg	130	90	-	$1.7X10^5$
CRD Hydraulic Area	R1k	-.25" wg	100/60	90/10	.0025	$8.8X10^2$	-.25" wg	104	90	-	$1.7X10^5$
Return Air Plenun, Recirc System	R2	-	-	-	.0025	$8.8X10^2$	-1.5" wg	104	90	-	$1.7X10^5$
Reactor Bldg. H&V Equipment Room	R4	-.25" wg	104/40	90/10	-	-	-.25" wg	104	90	-	-
SGTS Equipment Room	CS4	Atmos	104/40	100/10	.015	$5.3X10^3$	Atmos	104	100	$5.7X10^5$	$3.8X10^4$
Control Room	CS1	+.125" wg	80/70	55/45	.0005	$1.75X10^2$	+.125" wg	80	55	-	$1.78X10^2$

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TABLE 3.11-6

NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS 18

AREA	KEY (10)	NORMAL OPERATING CONDITIONS PRESSURE	TEMP°F MAX/MIN	RELATIVE HUMIDITY MAX/MIN%	DOSE RATE (R/HR) (12)	INTEGRATED DOSE (RAD)	MAXIMUM CONDITIONS (6) PRESSURE	TEMP °F	RELATIVE HUMIDITY %	LOCA DOSE RATE (RADS/HR)	TOTAL INTEGR. DOSE (RADS) (4)
Cable Spreading Rooms & HVAC Equipment Room, Relay Rooms Elect. Equip. Rooms	CS2	+.125" wg	80/10	60/10	.0005	$1.75X10^2$	+.125" wg	90	60	-	-
Battery Room	CS5	+.125" wg	80/60	60/10	-	-	+.125" wg	80	60	-	-
Computer Room	CS3	+.125" wg	85/65	60/40	-	-	+.125" wg	85	60	-	-
Radwaste Control Room	RW1	Atmos	75/50	80/10	.0025	$8.8X10^2$	Atmos	80	80	-	-
Radwaste Valve and Pump Rooms	RW2	-.125	104/40	90/10	.02	$7.0X10^3$	-.25" wg	120	100	-	-
Storage Tank Rooms (unprocessed)	RW3	-.125	120/40	90/10	20	$7.0X10^6$	-.25" wg	120	100	-	-
Diesel Generator Rooms (14)	G	Atmos	104/72	90/5	-	-	Atmos	120	50	-	-
ESW Pumphouse	SW	Atmos	104/40	100/5	-	-	Atmos	104	100	-	-
UPS Rooms	CS3	.125" wg	104/65	60/40	-	-	.125" wg	104	60	-	-

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TABLE 3.11-6

NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS 18

AREA	KEY (10)	NORMAL OPERATING CONDITIONS PRESSURE	TEMP°F MAX/MIN	RELATIVE HUMIDITY MAX/MIN%	DOSE RATE (R/HR)(12)	INTEGRATED DOSE (RAD)	MAXIMUM CONDITIONS(6) PRESSURE	TEMP °F	RELATIVE HUMIDITY %	LOCA DOSE RATE (RADS/HR)	TOTAL INTEGR. DOSE (RADS)(4)
Turbine Building Operating Floor	T2a	Atmos	104	90/10	.005-.020	$7.7X10^4$	-.125" wg	104	90	-	-
Turbine Building General Areas (shielded)	T1	-.125" wg	104	90/10	.001	$4X10^3$	-.125" wg	104	100	-	-
HP Turbine	T2b	-.125" wg	-	-	.5	$1.8X10^5$	-.125" wg	-	-	-	-
LP Turbine	T2c	-.125" wg	-	-	.1	$3.5X10^4$	-.125" wg	-	-	-	-
Feedwater Heaters Condensers	T3	-.125" wg	120	90/10	5	$1.8X10^6$	-.125" wg	120	100	-	-
Steam Jet Air Ejectors	T4	-.125" wg	120	90/10	15	$5.3X10^6$	-.125" wg	120	100	-	$5.3X10^6$
Condensate Treatment	T5	-.125" wg	120	90/10	10	$3.5X10^6$	-.125" wg	120	100	-	$5.3X10^6$

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(1) Temperatures are for small line break; pressures are for recirculation line break. Temperatures for recirculation line break are lower.

(2) Units for neutron flux are nutrons per cm^2-sec.

16

TABLE 3.11-6

NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS 18

(3) Pipe break outside containment results in short term peak temperature and pressure shown, until the break is isolated within the time noted. Shown in brackets is the long term temperature and pressure for the pipe break outside containment. The conditions in brackets are also those for the full duration of a pipe break inside containment. During the short term transient, capability to detect and isolate the pipe break and capability to shut down the reactor exists. 16 16 18 16

(4) Includes integrated accident and normal doses.

(5) Accident is rod drop not LOCA.

(6) Pressure, temperature, and humidity maximums are not simultaneous.

(7)	a) 0-45 sec.	44 psig	340°F	100% R.H.
	b) 45 sec.-3 hrs.	35	340°F	100% R.H.
	c) 3 hrs.-6 hrs.	35	320°F	100% R.H.
	d) 6 hrs.-24 hrs.	20	250°F	100% R.H.
	e) 24 hrs.-100 days	15	200°F	100% R.H.
(8)	a) 0-45 sec.	29 psig	130°F	100% R.H.
	b) 45 sec.-3 hrs.	30	200°F	100% R.H.
	c) 3 hrs.-6 hrs.	30	210°F	100% R.H.
	d) 6 hrs.-30 hrs.	15	200°F	100% R.H.
	e) 30 hrs.-150 hrs.	10	200°F	100% R.H.
	f) 150 hrs.-100 days	10	140°F	100% R.H.

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(9) Spent fuel pool boiling results in higher temperatures not exceeding 210°F.

(10) Key letter and number identifies a particular group of environmental parameters.

(11) Pipe breaks outside containment can result in higher temperatures and pressures in certain of these compartments, however, leak detection, isolation, and shutdown is accomplished from outside these compartments.

(12) If not otherwise noted, dose is Gamma. 16

TABLE 3.11-6

NORMAL AND MAXIMUM PLANT ENVIRONMENTAL CONDITIONS

(13) Minimum drywell pressure is -5 psig. 5

(14) For DG rooms: Normal operation means DG in Standby, maximum conditions means DG operating. 18

(15) Relative Humidity Maximum: 100%, 1-12 Hours; 90%, 12 Hours to 100 days.

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

6.5.1 ENGINEERED SAFETY FEATURE (ESF) FILTER SYSTEMS

6.5.1.1 Standby Gas Treatment System (SGTS)

6.5.1.1.1 Design Bases

The SGTS is designed to accomplish the following safety related objectives:

a) Exhaust sufficient filtered air from the reactor building to maintain a negative pressure of about 0.25 in. wg in the affected volumes following secondary containment isolation (see Subsection 9.4.2 for the secondary containment isolation signals) for the following design basis events: 3

(1) spent fuel handling accident in the refueling floor area

(2) LOCA

b) Filter the exhausted air to remove radioactive particulates and both radioactive and nonradioactive forms of iodine to limit the offsite dose to the guidelines of 10CFR100.

c) Filter and exhaust discharge from the main steam isolation valve leak control system

Nonsafety related objectives for design of the SGTS are as follows:

a) Filter and exhaust air from the primary containment for purging and ventilating

b) Filter and exhaust discharge from the HPCI barometric condenser

c) Filter and exhaust from the primary containment pressure relief line

d) Filter and exhaust nitrogen from the primary containment for nitrogen purging

The design bases employed for sizing the filters, fans, and associated ductwork are as follows:

a) Each train is sized and specified for treating incoming air-steam mixture at 180&F, and containing fission products and incoming particulates equivalent to 1.0 volume percent per day of the fission products available in the

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primary containment as determined in accordance with Regulatory Guide 1.3 and TID-14844.

b) System capacity to match the maximum air flow rate required for the primary containment purge.

c) The system capacity to be maintained with all filters fully loaded (dirty).

d) For HEPA filters, maximum free velocity not to exceed 300 fpm, with maximum airflow resistance of 1 in. wg when clean and 3 in. wg when dirty, and minimum efficiency of 99.97 percent by DOP test method.

e) For prefilters, maximum face velocity not to exceed 300 fpm, with maximum airflow resistance of 0.5 in. wg when clean, and 1.0 in. wg when dirty.

f) Associated ductwork is designed using the equal friction method at a rate of approximately .06 in. wg/100 ft.

g) Charcoal adsorber is rated for 99 percent trapping of radioactive iodine as elemental iodine (I), and 99 percent trapping of radioactive iodine as methyliodine (CH I) when passing through charcoal at 70 percent relative humidity and 25°C.

h) Each equipment train contains the amount of charcoal required to absorb the inventory of fission products leaking from the primary containment, based on a one unit LOCA.

i) Media cooling arrangement for each SGTS train is designed to remove heat generated by fission product decay on the HEPA filters and charcoal adsorbers during shutdown of the train.

j) Relative humidity at charcoal adsorber is limited to maximum of 70 percent by removing moisture entrained in the airstream and by preheating the air.

Failure of any component of the filtration train, assuming loss of offsite power, cannot impair the ability of the system to perform its safety function

The system remains intact and functional in the event of a Safe Shutdown Earthquake (SSE).

6.5.1.1.2 System Design

Each of the two redundant SGTS trains consists of a mist eliminator, an electric air heater, a bank of prefilters, two banks of HEPA filters, upstream and downstream of charcoal adsorber, and a vertical 8 in. deep charcoal adsorber bed with fire detection temperature sensors, water spray system for fire protection, and associated dampers, ducts, instruments, and controls. The airflow diagram for the SGTS is shown on Figure 9.4-5. The instruments and controls are shown on Figure 9.4-9. The system design parameters are provided in Table 6.5-1.

The work, equipment and materials conform to the applicable requirements and recommendations of the guides, codes, and standards listed in Section 3.2.

Compliance of the system design with Regulatory Guide 1.52, is described in Section 3.13. Also see Table 6.5-2.

Each redundant SGTS train has a controllable capacity of 3,000 cfm to 10,500 cfm, and each is capable of treating required amount of air from both Unit 1 and Unit 2 reactor building volumes. (see Subsection 6.5.3). Components for each SGTS are designed as explained in the following paragraphs.

The fan performance and motor selection is based on the maximum air density and the maximum system pressure drop, that is, 70°F air temperature at the fan (55°F air at the inlet of the SGTS train plus approximately 15°F constant temperature pickup across the heater), and the pressure drop is based on maximum pressure drops across dirty filters.

The charcoal adsorber is a gasketless, welded seam type, filled with KI, impregnated coconut shell charcoal. The bank holds a total of approximately 6,920 lb of charcoal of 28 lb/ft^3 density, having an ignition temperature of not less than 330°C. The charcoal adsorber is designed for a maximum loading capacity of 2.5 mg of total iodine (radioactive plus stable) per gram of active charcoal.

Six test canisters are provided for each adsorber. These canisters contain the same depth of the same charcoal that is in the adsorber. The canisters are mounted, so that a parallel flow path is created between each canister and the adsorber. Periodically one of the canisters is removed and laboratory tested to verify the adsorbent efficiency.

Thirty by fifty in. access doors into each filter compartment are provided in the equipment train housing. The doors have transparent portholes to allow inspection of components without violating the train integrity.

The housing is of all welded construction.

Gas tight interior lights with external light switches and fixture access are provided between all train filter banks to facilitate inspection, testing, and replacement of components.

Filter housings, including water drains, are in accordance with recommendations of Section 4.3 of Ref. 6.5-1.

Ductwork is designed in accordance with recommendations of Section 2.8 of Reference 6.5-1, except for sheet metal gages that are slightly less, and the round duct reinforcements. The ductwork, however, has been seismically qualified by analysis and testing of duct specimens.

Outdoor makeup air supplements low exhaust airflow rates for most of the SGTS operational modes to satisfy the SGTS fan minimum airflow requirement, which is approximately 3000 cfm. The outdoor makeup air is also used, at a rate of 3000 cfm, for charcoal bed cooling after a charcoal pre-ignition temperature is detected.

The mist eliminator is designed to prevent blinding of the HEPA filter when operated at 200°F with steam-air mixture containing 1 gal of water droplets actually entrained in the airstream per 1000 cfm airflow.

The electric heater reduces the relative humidity of the entering air to below 70 percent for charcoal adsorber operation, by maintaining a constant temperature rise across the heater. An analysis of heater capabilities for various entering saturated air conditions ranging from 55°F to 180°F yields a peak heating requirement of 180,000 Btu/hr, at maximum 10,500 cfm airflow. In addition, 55,400 Btu/hr heat loss is calculated from the section of SGTS housing between the heater and the charcoal bed. Overall required capacity is 235,400 Btu/hr. A 90 kW heater is provided.

The charcoal bed is provided with an integral water spray system connected to the station fire protection system. A deluge valve and Seismic Category I backup valve are mounted in series adjacent to the charcoal adsorber. The backup valve is provided to prevent charcoal flooding if the deluge valve fails in an open position. Fire protection for the SGTS filter trains is also discussed in Subsection 9.5.1.

A continuous type thermister is provided on the inlet and outlet of the charcoal bed.

The SGTS is actuated either automatically (safety related mode), or manually (nonsafety related mode). The automatic actuation is originated by the reactor building isolation signal, or by detection of pre-ignition temperature in the charcoal adsorber

bed, the latter for charcoal cooling purposes. The manual actuation is controlled by administrative procedures in such a way that the SGTS is started and airflow established (outdoor makeup air) prior to introduction of air or gas to be exhausted from a reactor building source.

The automatic or manual actuation will result in a start of a lead fan; then, associated controls will be activated to open or modulate appropriate dampers, so that the system function is accomplished.

The SGTS inlet header pressure is monitored and controlled to preclude the possibility of nonfiltered gas or air bypassing the filtration train through the outdoor air makeup duct. This pressure is maintained at approximately 1 in. wg negative, referenced to the static pressure at the outdoor makeup air intake during the system operation. The lead SGTS is started automatically and an alarm sounded in the control room, if this pressure rises to 2 in. wg positive when the system is not in operation and the negative pressure will be established and maintained. The system will be stopped manually, once the cause of the high inlet header pressure is identified and eliminated.

Outside air is used for either charcoal cooling or making up the total system flow to approximately 3,000 cfm to ensure each fan's stable operation. Once the exhaust air/gas flow from the reactor building reaches 3,000 cfm the makeup air dampers, two in parallel, will be modulated gradually to a closed position.

An inlet header pressure controller sets the system flow rate to maintain the inlet header negative pressure. In turn, a flow controller sequentially modulates the fan inlet vanes and the outdoor makeup air damper, thus maintaining the flow rate and the inlet header negative pressure.

Any section of the charcoal bed inlet or outlet thermisters sensing a temperature higher than preset charcoal pre-ignition or ignition temperatures will result in the following:

a) The pre-ignition temperature (set at 190°F) will actuate an alarm in the control room, and will automatically initiate the affected SGTS train's charcoal cooling mode of operation by establishing a flow of approximately 3,000 cfm of outdoor makeup air across the charcoal bed.

b) The ignition temperature (set at 450°F) will actuate an alarm in the control room and open the deluge valve and the backup valve, thus introducing the fire protection water to the charcoal spray system. Four drain valves provided to drain the deluge water will be opened automatically by the ignition temperature signal. The operation of the deluge system will continue until the

charcoal temperature falls below the ignition temperature. The deluge water flow will be controlled by the backup valve; the deluge valve will remain open after the initial actuation.

The SGTS is designed to Seismic Category I requirements.

The power supply meets IEEE-308 criteria and ensures uninterruptible operation in the event of loss of normal, onsite, ac power.

6.5.1.1.3 Design Evaluation

The SGTS is designed to preclude direct exfiltration of contaminated air from either reactor building, following an accident or abnormal occurrence which could have resulted in abnormally high airborne radiation in the secondary containment. Equipment is powered from essential buses and all power circuits will meet IEEE-279 and IEEE-308. Redundant components are provided where necessary to ensure that a single failure will not impair or preclude system operation. SGTS failure mode and effect analysis is presented in Table 6.5-3.

6.5.1.1.4 Tests and Inspections

Except for Item 5, all tests and inspections described in Table 9.4-1 apply to the SGTS.

The system will be preoperationally tested in accordance with the requirements of Chapter 14. Refer to Chapter 16 for periodic test requirements for the SGTS.

6.5.1.1.5 Instrument Requirements

The SGTS can be actuated manually from the control room. Each SGTS train is designed to function automatically upon receipt of an ESF system actuation signal. The status of system equipment, which is an indication of pertinent system pressure drops and flow rates, is displayed in the control room during both normal and accident operation.

Table 6.5-2 addresses the extent to which the recommendations of NRC Regulatory Guide 1.52 are followed with respect to instrumentation.

All instrumentation is qualified to Seismic Category I requirements.

Redundancy and separation of the instrumentation is maintained, and it follows the redundancy and separation of the equipment.

The following alarms are annunciated in the control room:

a) Fan failure

b) Heater failure (low temperature rise across the heater)

c) High or low pressure drop across the upstream HEPA

d) High pressure drop across any filter (a group alarm)

e) Pre-ignition charcoal temperature

f) Ignition charcoal temperature

g) Charcoal temperature detection system failure (include the deluge valve solenoid circuit discontinuity)

h) Low pressure differential, referenced to the outdoor ambient pressure, in the reactor building ventilation zones being isolated

i) High and low positive pressure in the SGTS header

j) Outside makeup air damper failed open

k) Outside charcoal cooling air damper failed open.

6.5.1.1.6 Materials

The materials of construction used in or on the filter systems are given in Tables 6.1-1 and 6.5-5. Each of the materials is compatible with the normal and accident environmental conditions.

Accident environments (ie, extreme temperature or radiation) that could potentially produce radiolytic or pyrolytic decomposition of filter materials are not applicable to the control structure where the SGTS is located. Thus, filter system decomposition products will not be present.

6.5.1.2 Control Room Emergency Outside Air Supply System (OV-101)

6.5.1.2.1 Design Bases

The control room emergency outside air supply system (CREOASS) is designed to accomplish the following objectives:

a) Filter particulate matter which may be radioactive and remove gaseous iodine

b) Recirculate and clean up room air when chlorine is present in the outside air

c) Maintain ventilation air supply for the control room and control structure envelope when radiation is detected in the outside air

d) Maintain a positive pressure above atmospheric to inhibit outside air infiltration into the control room during radiation isolation (0.25 in. wg. in the control room and 0.125 in. wg. in other control building areas)

e) Operate during and after design basis accident and reactor building isolation mode conditions without loss of function

f) Provide radiation monitoring and chlorine detection of outside air supply

The bases employed for sizing the filters, fans, heater, and associated ductwork are as follows:

a) System capacity (flow rate) to be based on required air changes for the control room, the air exhausted from the battery storage area, and additional air to slightly pressurize the control room

b) The system capacity to be maintained with all particulate filters fully loaded (dirty)

c) HEPA filters, maximum face velocity not to exceed 300 fpm with maximum airflow resistance of 1 in. wg. when clean and 3 in. wg. when dirty. A minimum efficiency to be 99.97 percent by DOP test method.

e) Prefilters, maximum face velocity not to exceed 300 fpm, with maximum airflow resistance 0.3 in. wg. when clean and 0.9 in. wg. when dirty.

f) Ductwork is designed using equal friction method at a rate of approximately .06 in. wg/100 ft.

g) Charcoal adsorber is rated for 99 percent trapping of radioactive iodine as elemental iodine (I_2), and 99 percent trapping of radioactive iodine as methyliodine (CH_3I) when passing through charcoal at 70 percent relative humidity and 25°C.

h) Maximum relative humidity for air entering the charcoal adsorber to be limited to 70 percent by appropriate air heating.

j) The CREOASS filter trains are designed to meet single failure criteria.

k) The CREOASS is designed to Seismic Category I requirements, so that it remains operable during and after a Safe Shutdown Earthquake (SSE).

m) The power supply to meet IEEE 308 criteria and ensure uninterrupted operation in the event of loss of normal ac power. The controls meet IEEE 279.

6.5.1.2.2 System Design

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Each of the two redundant CREOASS filter trains consists of an electric heater, a bank of prefilters, two banks of HEPA filters, one upstream and one downstream of the charcoal adsorber, and a vertical 4 in. deep charcoal adsorber bed with fire detector temperature sensors, associated dampers, instruments, controls, and water flooding system for fire protection. The CREOASS is shown on Figure 9.4-1. The instrument and controls are shown on Figure 9.4-2. The system design parameters are shown in Table 6.5-1.

The work, equipment and materials conform to the applicable requirements and recommendations of the guides, codes, and standards listed in Section 3.2.

The system design is consistent with recommendations of NRC Regulatory Guide 1.52, as described in Section 3.13, and shown in Table 6.5-2.

Each CREOASS filter train contains the following components listed in the direction of airflow:

a) A 30 kW electric heater to maintain relative humidity of the entering air below 70 percent. The heater is energized at the same time as the fan and provides

approximately 15°F temperature rise across the coil, ensuring that entering outside air ranging from -15°F to 100°F will enter the filters with a relative humidity of less than 70 percent.

b) A charcoal adsorber designed with six gasketless welded 4 in. vertical beds, containing a total of 2336 lb of coconut shell charcoal (30 lb/ft³) impregnated with KI_3 and triethylenediamine (TEDA). Six canisters are provided for each adsorber. The canisters contain the same depth of identical charcoal as the adsorber. The canisters are mounted, so that a parallel flow path is created between each canister and the adsorber. Periodically one of the canisters is removed and laboratory tested to verify the adsorbent efficiency.

c) The housing is constructed of carbon steel welded construction in accordance with Ref. 6.5-1. Stainless steel is used for filter support brackets. The housing is designed for -20 in. wg. and a +5 psig. Each housing is provided with five 20x50 in. access doors for servicing the heater and filter banks.

The access doors are provided with transparent portholes to allow inspection of components without violating the trains' integrity.

Filter housings, including water drains, are in accordance with recommendations of Section 4.3 of Ref. 6.5-1.

Interior lights with external light switches and outside access for bulb replacement are provided to facilitate inspection, testing, and replacement of components.

d) A centrifugal fan designed for a flow rate of 6,000 cfm. The fan performance and motor selection is based on the maximum air density and the maximum system pressure drop.

e) Ductwork is designed in accordance with recommendations of Section 2.8 of Ref. 6.5-1, except for sheet metal gages that are slightly less and round duct reinforcement. The ductwork, however, has been seismically qualified by analysis and testing of duct specimens.

A fire protection system, designed to extinguish a fire within the charcoal bed by flooding the housing, is provided. The fire protection system is designed to spray 36 gpm of water at 15 psi on the charcoal. A deluge valve and a backup valve are installed in series in the fire protection water connection adjacent to the

housing. The back-up valve is installed downstream of the deluge valve to prevent charcoal flooding in the event of a malfunction of the deluge valve. One pre-ignition (190°F setting) and one ignition (450°F setting) temperature switch are located in the discharge duct connection. Six pre-ignition and six ignition switches are evenly spaced across the downstream face of the charcoal adsorber. A 190°F or greater leaving air temperature will trip any of the seven temperature switches, and alarm in the control room. A 450°F or greater leaving air temperature will trip any of the seven temperature switches, alarm in the control room, stop the fan, energize the deluge valve and the back-up valve. An overflow is provided in the housing to allow water to drain once the housing is full. The water must be shut off manually. The housing is drained by opening five manual drain valves.

See Subsection 9.4.1.2.4 for additional details of the CREOASS operation.

The CREOASS is designed to Seismic Category I requirements.

The power supply meets the IEEE-308 criteria and ensures uninterruptible operation in the event of loss of normal, onsite, ac power.

6.5.1.2.3 Design Evaluation

The CREOASS work in conjunction with the control room and control structure HVAC systems to maintain habitability in the control room. The design evaluation is given in Subsection 9.4.1 including failure mode and effect analysis presented in Table 9.4-19.

6.5.1.2.4 Tests and Inspections

With the exception of Items 5, 6, and 7, all tests and inspections described in Table 9.4-1 apply to the CREOASS.

6.5.1.2.5 Instrumentation Requirements

The CREOASS can be actuated manually from the control room. Each CREOASS is designed to function automatically upon receipt of a radiation detection signal from detector elements located in the outside air intake plenum. In addition to starting the CREOASS, high radiation is annunciated in the control room.

When chlorine is detected in the outside air intake, the following automatic actions are initiated:

a) High chlorine is annunciated in the control room. All outside air intake and exhaust air dampers close and the structure is isolated before chlorine reaches the intake isolation dampers.

b) The battery room exhaust system is shut down. Recirculation dampers open and the CREOASS starts in the recirculation mode to clean up the air within the control room.

The reactor building isolation signal will cause the CREOASS to operate in exactly the same manner as a high radiation signal from the outside air intake.

The status of system equipment, indication of pertinent system pressure drops, and flow rates are displayed in the control room.

Table 6.5-2 addresses the extent to which the recommendations of NRC Regulatory Guide 1.52 are followed with respect to instrumentation.

All instrumentation is qualified to Seismic Category I requirements. Redundancy and separation of the instrumentation is maintained and follows the redundancy and separation of the equipment.

The following alarms are annunciated in the control room:

a) Fan failure

b) Heater failure (low temperature differential across the heater)

c) High pressure drop across the upstream HEPA

d) High charcoal temperature

e) High-high charcoal temperature.

6.5.1.2.6 Materials

The materials of construction used in or on the filter systems are given in Tables 6.1-1 and 6.5-6. Each of the materials is compatible with the normal and accident environments postulated in the control structure and the fuel handling building.

Accident environments (ie, extreme temperature or radiation) that could potentially produce radiolytic or pyrolytic decomposition of filter materials are not applicable to the control structure where the filter units are located. Thus, filter system decomposition products will not be present.

6.5.2 CONTAINMENT SPRAY SYSTEMS

The containment spray system is described in Subsection 6.2.2. The containment spray system is not required for fission product removal.

6.5.3 FISSION PRODUCT CONTROL SYSTEM

6.5.3.1 Primary Containment

The standby gas treatment system (SGTS) is used to control the release of fission products to the environment when purging the containment. This is described in detail in Subsection 6.5.1.1.

The Primary Containment is charged with nitrogen during plant start-up in accordance with the Technical Specifications. Gaseous nitrogen is used to reduce the concentration of oxygen, as discussed in Subsection 6.2.5.2. The containment is purged of nitrogen during reactor shutdown in accordance with the Technical Specifications with air from the Reactor Building Ventilation Supply Air System. The purge piping and valves are shown on Figure 6.2-55. The 24" diameter and 18" diameter piping can be used for purging during reactor power operation (as mentioned above), start-up and hot standby; otherwise, the purge supply and exhaust valves HV-15704, HV-15714, HV-15721, HV-15722, HV-15723, HV-15724 and HV-15725 remain closed. These valves cannot be manually overridden to open following containment isolation.

The 2" vent by-pass valves, HV-15711 and HV-15705, and the inner isolation valves, HV-15703 and HV-15713, on the purge exhaust lines will be used to relieve containment pressure increases caused by thermal expansion during normal operations. Containment isolation valves HV-15711, HV-15705, HV-15703 and HV-15713 cannot be opened until 60 minutes following containment isolation. The containment make-up line valves SV-15767 and SV-15737 also cannot be opened until 60 minutes following containment isolation, while valves SV-15776A and SV-15736A are isolated for a period of 10 minutes. After the isolation period has elapsed, these valves may be opened remote manually under administrative control for control of hydrogen, as discussed in Subsection 6.2.5.2.

Layout drawings of the primary containment are listed in Section 1.2.

Hydrogen recombiners and the hydrogen purge system are discussed in Subsection 6.2.5.

The primary containment leak rates are discussed in Section 6.2.

6.5.3.2 Secondary Containment

The following are provided to control fission products within the secondary containment following a design basis accident:

a) A secondary containment that completely surrounds each of the two primary containments

b) The Standby Gas Treatment System (SGTS) discussed in Subsection 6.5.1.1

c) A recirculation system.

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The secondary containment consists of a reinforced concrete structure up to the refueling floor (el 818 ft 1 in.) and of a metal sided superstructure above el 818 ft 1 in., both discussed in Subsection 3.8.4.

The secondary containment isolation is discussed in Subsection 9.4.2.1. This section also defines three ventilation zones (I, II, and III).

The SGTS is used to maintain the affected zone(s) of the secondary containment under approximately 0.25 in. wg. negative pressure and control the cleanup of the fission products from the primary containment following a design basis LOCA, or from the refueling floor following a refueling accident.

A common recirculation system is provided for Units 1 and 2 to perform the following functions:

a) Mix the atmosphere in the reactor building to obtain a lesser and more uniform concentration of radioactivity following a design basis LOCA and refueling accident

b) Prevent the spread of radioactivity by the heating-ventilating-cooling systems between:

   1) Zone I and Zone II

   2) Zone III and Zones I or II during and after a refueling accident

c) Provide mixing of the atmosphere within the reactor building. This may involve mixing the atmosphere of Zone I or Zone II and the refueling area (Zone III) or of Zone III above. See Subsection 9.4.2.1.3 for the secondary containment isolation modes. Also see Subsection 6.2.3 for the secondary containment analysis.

The recirculation flow diagram is shown on Figures 9.4-4 and 9.4-5. The instruments and controls are shown on Figure 9.4-7.

Estimated respective zone(s) recirculation flow rates and their volumes are listed in Table 6.5-7.

The recirculation system consists of two 100 percent redundant, vane-axial fans connected to the emergency power supply, associated ductwork, dampers, and controls.

The recirculation air is distributed to all areas and rooms through the existing normal ventilation ductwork.

Both fans, ductwork used in the recirculation mode, supports, and instruments and controls meet the Seismic Category I requirements.

The recirculation system starts automatically on receiving the reactor building isolation signal, which is defined in Subsection 9.4.2.1.3.

For the recirculation system failure mode and effect analysis see Table 6.5-4.

The tests and inspection described in items 1, 2, 3, 13 and 14 of Table 9.4.1 are applicable to the recirculation system.

6.5.4 ICE CONDENSER AS A FISSION PRODUCT CLEANUP SYSTEM

Not applicable.

6.5.5 REFERENCES

6.5-1 ORNL-NSIC-65

CHAPTER 8.0

ELECTRIC POWER

TABLE OF CONTENTS

8.1 INTRODUCTION

8.1.1 GENERAL

The electric power systems of the Susquehanna Steam Electric Station Units 1 and 2 are designed to generate and transmit electric power for the supply of PP&L customer needs utilizing the power network of the PP&L and of the Pennsylvania-New Jersey-Maryland (PJM) interconnection.

The two independent offsite electric connections to Susquehanna SES are designed to provide reliable power sources for plant auxiliary loads and the engineered safety features loads of both units such that any single failure can affect only one power supply and cannot propagate to the alternate source.

The onsite ac electric power system consists of Class IE and non-Class IE power systems. The two offsite power systems provide the preferred ac electric power to all Class IE loads through the Class IE distribution system. In the event of total loss of offsite power sources, four onsite independent diesel generators provide the standby power for all engineered safety features loads.

The non-Class IE ac loads are normally supplied through the unit auxiliary transformer or the startup transformer. However, during plant startup, shutdown, and post-shutdown, power is supplied from the offsite power through the startup transformers.

Onsite Class IE and non-Class IE dc systems supply all dc power requirements of the plant.

8.1.2 UTILITY POWER GRID AND OFFSITE POWER SYSTEMS

Unit 1 and 2 generators are connected by separate isophase buses to their respective main step-up transformer banks as shown on Figure 8.3-1. Unit 1 main step-up transformer bank, with two three-phase, half capacity power transformers, steps up the 24 kV generator voltage to 230 kV; the Unit 2 bank, with three single phase power transformers, steps up the 24 kV generator voltage to 500 kV. As shown on Figure 8.3-1, the step-up transformer for Unit 1 connects to the 230 kV substation and for Unit 2 to the 500 kV substation. The 230 kV substation uses a breaker and one-half scheme design, and the 500 kV substation a ring bus arrangement with provision for future expansion to a breaker and one-half operation. The substations are approximately 1.9 miles apart and are interconnected by a 500-230 kV bus tie transformer

and transmission line. Aerial transmission connects the 230 kV substation with two other generating stations, Sunbury and Montour, and with Stanton, Siegfried, Harwood, and Jenkins substations. Aerial transmission lines integrates the 500 KV substation into the 500 KV system with connections at Wescosville, Alburtis and Sunbury. Both the 500 kV substations and the 230 kV substations are tied into the PJM Interconnection.

The plant startup and preferred power for the engineered safety features systems is provided from two independent offsite power sources Figure 8.2-1.

a) A tap from the Montour-Mountain 230 kV line feeds the start-up transformer No. 10.

b) A 230 kV tap from the 500-230 kV tie line feeds the startup transformer No. 20.

The transmission system, including the 230 kV line to Unit 1 main transformers and the two offsite power lines to the two startup transformers, is operational before Unit 1 fuel load. The transmission line to Unit 2 main tranformers is operational before Unit 2 fuel load.

The offsite power systems and their interconnections are described in detail in Section 8.2.

8.1.3 ONSITE POWER SYSTEMS

The onsite power system for each unit is divided into two major categories:

a) Class IE Power System

The Class IE power system supplies all engineered safety features (ESF) loads, and other loads that are needed for safe and orderly shutdown, and for keeping the plant in a safe shutdown condition.

The Class IE power system for each unit consists of four independent load group channels, channels A, B, C, and D. Any combination of three out of four load group channels meets the design basis requirements. In addition, two divisionalized load groups are established for those ESF loads which require one out of two load groups to meet the design basis requirements. ESF load group division separation and channel separation are shown in Tables 3.12-1 and 3.12-2 respectively.

Physical separation is discussed fully in Subsection 3.12.3. 13

The Class IE power system distributes power at 4.16 KV, 480 V, ± 24 V dc, 125 V dc, 250 V dc voltage levels. 3

The Class IE power system is shown on Figures 8.3-3 thru 8.3-8.

b) Non-Class IE Power System

The non-Class IE ac portion of the onsite power system supplies electric power to all nonsafety related plant auxiliary loads. The non-Class IE ac auxiliary system distributes power at 13.8 kV, 4.16 kV, 480 V, and 208/120 V voltage levels. These distribution levels are grouped into two symmetrical bus systems emanating from the 13.8 kV level as shown in Figure 8.3-1.

Power transmitted to the utility grid is discussed in Subsection 8.1.2.

Non-Class IE dc power is discussed in Subsection 8.3.2.

A detailed description of the onsite power system is found in Subsections 8.3.1 and 8.3.2.

8.1.4 SAFETY RELATED LOADS

The Class IE loads supplied by the Class IE ac power system are listed in Tables 8.3-1 to 8.3-5. Class IE loads supplied by the Class IE dc system are listed in Tables 8.3-6 to 8.3-8.

8.1.5 DESIGN BASES

8.1.5.1 Safety Design Bases

The following principal design bases are applied to the design of the onsite and offsite power systems:

Offsite Power System

a) Electric power from the offsite power sources to the onsite distribution system is provided by two physically separated transmission lines designed and located to minimize the likelihood of simultaneous failure.

b) The loss of one or both generating units or the loss of the most critical unit on the power grid will not result in total loss of offsite power.

Onsite Power System

a) One unit auxiliary transformer per generating unit is provided to supply power to the plant electrical auxiliary distribution system.

b) Two startup transformers common to both units are provided to supply offsite power to the Class IE power system and common plant auxiliary power system and to supply power to the Unit Auxiliary loads during startup, shutdown, and in the event of loss of a unit auxiliary transformer.

c) Outage of one startup and/or one engineered safeguard transformer would not jeopardize continued plant operation except where the operation is limited as suggested by Regulatory Guide 1.93. See compliance statement to Regulatory Guide 1.93 in Subsection 8.1.6.1

d) Standby diesel generators are shared by two units. See Subsection 8.1.6 responses to Regulatory Guide 1.81, for diesel generator capability and compliance discussions.

e) Each generating unit has its own independent dc system.

f) The onsite Class IE electric power system for each unit is divided into four independent load groups. Besides the sharing of the diesel generator with the counterpart load group of the other unit, each load group has its own distribution buses and loads. Minimum engineered safety feature loads required to shut down the unit safely and maintain it in a safe shutdown condition are met by any three of the four load group channels.

g) The four Class IE load groups are subgrouped generally to form two divisions for meeting the design basis of one out of two ESF load requirements.

h) Automatic or manual transfers are not provided between redundant load groups except swing buses as discussed in Subsection 8.3.1.3.5.

i) The Class IE electric systems are designed to satisfy the single failure criterion in accordance with IEEE 379-1972.

j) The dc system battery banks are individually sized for four hours of operation under the maximum design loading without the support of the battery charger.

k) Raceways are not shared by Class IE and non-Class IE cables. However, the affiliated cables that are supplied from the Class IE buses are treated as Class IE cables with regard to redundant system separation and identification criteria.

l) Special identification criteria applies for Class IE equipment, cabling, raceways, and affiliated circuits. Affiliated circuits are uniquely identified.

m) Separation criteria apply which establish requirements for preserving the independence of redundant Class IE system and providing isolation between Class IE and non-Class IE equipment.

n) Class IE equipment has been designed with the capability for periodic testing.

8.1.6 Regulatory Guides and IEEE Standards

Codes and standards applicable to the onsite power system are listed in Table 3.2-1. Generally, the system is designed in accordance with IEEE Standards 308-1974, 317-1972, 323-1971, 334-1971, 344-1971, 382-1972, 384-1974, 387-1972, and 450-1972.

8.1.6.1 Compliance with Regulatory Guides

Compliance with General Design Criteria 17 and 18 of 10CFR50, Appendix A, is discussed in Subsections 8.3.1.11.1 and 8.3.2.2.1. Compliance with applicable Regulatory Guides 1.6, 1.9, 1.22, 1.29, 1.30, 1.32, 1.40, 1.41, 1.47, 1.53, 1.62, 1.63, 1.73, 1.75, 1.81, 1.89, 1.93, and 1.106 is discussed below.

a) Regulatory Guide 1.6 (3/71)

The design of the standby power system is in compliance with Regulatory Guide 1.6.

The standby power system consists of four independent load groups. All safety related loads are divided among these four load groups so that loss of any one group will not prevent the minimum safety functions from being

performed. Each load group consists of both standby ac and dc power systems.

Each ac load group has connections to two independent offsite power supplies and to a single onsite diesel generator. The power feeder breakers to each load group are interlocked so that only one of the power supplies can be connected at any one time except during diesel generator load test where the diesel generator is sychronized to one of the preferred offsite power sources. Only one diesel generator is tested at a time.

Each diesel generator is exclusively connected to the corresponding load group of the two units; ie, diesel generator A connects to load group channel A of both units, etc.

The diesel generator of one load group cannot be paralleled, either manually or automatically, with the diesel generator of the redundant load groups.

No provision exists for automatic transfer of loads between load groups except as discussed in Subsection 8.3.1.3.5.

The dc power system of each of the four load groups consists of a 125 V dc battery and a charger. The battery charger is supplied by its corresponding ac power system. The dc power system of any one load group is independent of any other dc power system.

Two independent 250 V divisionalized dc power systems are also provided for each unit to supply large dc loads. Loss of any one 250 V dc subsystem will not prevent the safety functions from being performed.

Physical separation of Class IE equipment is fully discussed in Section 3.12.

b) <u>Regulatory Guide 1.9 (3/71)</u>

The standby diesel generators comply with Regulatory Guide 1.9 except as noted in (5) and (6) of the following:

1) The continuous or the 2000 hr rating of the standby diesel generators is greater than the sum of conservatively estimated loads needed to be supplied following any design basis event within one of the two units. Load requirements are listed in Tables 8.3-1 to 8.3-5.

2) The standby diesel generators are capable of starting and accelerating all engineered safety features and forced shutdown loads to the rated speed in the time frame and sequence shown in Tables 8.3-1 to 8.3-5.

3) The standby diesel generators are capable of maintaining, during steady state and loading sequence, the frequency and voltage above a level that may degrade the performance of any of the loads.

4) The standby diesel generators are capable of recovering from transients caused by step load increase or resulting from the disconnection of partial or full load so that the speed does not damage any moving parts.

5) The suitability of each diesel generator is confirmed by factory qualification testing.

6) Power quality is in accordance with IEEE 308-1974, Section 4.3. At no time during the loading sequence will the frequency and/or voltage drop to a level that will degrade the performance of any of the loads below their minimum requirements. The power quality is confirmed by preoperational tests.

c) Regulatory Guide 1.22 (2/72)

Refer to Section 3.13 for compliance statement.

d) Regulatory Guide 1.29 (2/76)

Refer to Section 3.13 for compliance statement.

e) Regulatory Guide 1.30 (8/72)

Refer to Section 3.13 for compliance statement.

f) Regulatory Guide 1.32 (3/76)

All safety related electric systems are in compliance with Regulatory Guide 1.32. Compliance is discussed as follows:

The portions of Regulatory Guide 1.32 applying to dc power are discussed in Subsection 8.3.2.2.1(d).

The availability of the offsite power meets the criteria set forth in Regulatory Guide 1.32. The two offsite

circuits have immediate access to the transmission network. See response to Regulatory Guide 1.93 for operating restrictions when offsite power is not immediately available.

IEEE 308-1974is generally accepted by Regulatory Guide 1.32. Compliance with the Regulatory Guide is discussed as follows:

Class IE ac power systems are designed to ensure that any design basis event, as listed in Table 1 of IEEE 308, does not cause either (1) loss of electric power to more than one load group, surveillance device, or protection system to jeopardize the safety of the reactor unit, or (2) transients in the power supplies, which could degrade the performance of any system.

Controls and indicators for the Class IE 4.16 kV bus supply breakers are provided in the control room and on the switchgear. Controls and indicators for the standby ac power supplies are also provided in the control room and on the local diesel generator control panels. Control and indication for the standby power system is described in Subsection 8.3.1.

Class IE equipment and associated design, operating, and maintenance documents are distinctly identified as described in Subsection 8.3.1.3.

Each Class IE equipment is qualified by analysis, by successful use under required conditions, or by actual test to demonstrate its ability to perform its function under applicable design basis events.

The surveillance requirements of IEEE 308 are followed in design, installation, and operation of Class IE equipment and consist of the following:

1) Preoperational equipment and system tests and inspections are performed in accordance with the requirements described in Chapter 14.

2) Periodic equipment tests are performed in accordance with the requirements of Chapter 16.

The standby ac power supplies are shared by both units. The total standby capacity is sufficient to operate the engineered safety feature loads following a design basis accident on one unit and a concurrent forced shutdown of the other unit.

The two preferred offsite power supplies are also shared by both units. The capacity of each offsite power supply is sufficient to operate the engineered safety features of one unit and safe shutdown loads of the other unit.

Connection of non-Class IE equipment to Class IE systems is discussed in the response to Regulatory Guide 1.75.

Selection of diesel generator set is discussed in the response to Regulatory Guide 1.9.

g) Regulatory Guide 1.40 (3/73)

Refer to Subsection 3.11.2 for compliance statement.

h) Regulatory Guide 1.41 (3/73)

The preoperational testing program conforms to the general guidance provided by Regulatory Guide 1.41 as described in Chapter 14.

The onsite Class IE electric power system, designed in accordance with Regulatory Guides 1.6 and 1.32, is tested as part of the preoperational testing program and also after major modifications. The tests are performed in accordance with the requirements outlined in Chapter 14. Facilities are provided to test the independence between the redundant onsite power sources and their load groups.

The onsite Class IE electric power system can be tested functionally, one load group at a time, by allowing one load group to be powered only by its associated diesel generator while the bus is isolated from the preferred offsite power source. The isolation of the offsite power source can be done by direct actuation of undervoltage relays monitoring the Class IE system.

Each test may include injection of simulated accident signals, startup of diesel generators, and automatic load applications. Functional performance of the loads is checked. Each test is of sufficient duration to achieve stable operating conditions and thus permit the onset and detection of adverse conditions that could result from improper assignment of loads.

During test of one Class IE load group, the buses and loads of the redundant load group not under test are monitored to verify independance of load groups.

i) Regulatory Guide 1.47 (5/73)

Refer to Section 3.13 for compliance statement.

j) Regulatory Guide 1.53 (6/73)

Refer to Section 3.13 for compliance statement.

k) Regulatory Guide 1.62 (10/73)

Refer to Section 3.13 for compliance statement.

l) Regulatory Guide 1.63 (10/73)

The design of electric penetration assemblies is in compliance with Regulatory Guide 1.63.

In accordance with Regulatory Guide 1.63, the electrical penetration assemblies are designed to withstand, without loss of mechanical integrity, the maximum fault condition vs time conditions which could occur as a result of a single random failure of circuit overload devices. The following system features are provided to ensure compliance with the requirements of the Regulatory Guide.

1) Medium Voltage System

For medium voltage circuits feeding loads in the primary containment the circuit breaker associated with the load is backed up by the main bus feeder breaker to interrupt the circuit, should the load breaker fail to open during a fault.

2) 480 V System

The 480 V motor control centers feed all 480 V loads inside the primary containment. In addition to the primary feeder circuit breaker a back up breaker is provided to each circuit to provide back up protection of the penetration assemblies. The penetration withstands the available fault current and time duration for either primary or back up circuit breakers to interrupt the circuit.

3) 208 V and Lower Voltage Systems

The majority of low voltage control and power circuits are self limiting in that the circuit resistance and/or short circuit capability limits the fault current to a level that does not damage

the penetration assemblies. The remaining control and power circuits have primary and backup fuse or breaker to ensure fault isolation.

4) Instrument Systems

The overload and short circuit capability in the instrument systems are sufficiently low such that no damage can occur to the penetration assemblies.

m) Regulatory Guide 1.73 (1/74)

Selection of electric valve operators for use inside the containment is in compliance with Regulatory Guide 1.73.

The electric valve operators for service inside the containment are type tested in accordance with IEEE 382-1972 as modified by Regulatory Guide 1.73. The tests consist of (1) aging, (2) seismic, and (3) accident or other special environmental requirements. Test parameters are discussed in Subsections 3.11.2a.3 and 3.11.2b.2.

See Section 3.13 for compliance statement for GE furnished valve operators.

n) Regulatory Guide 1.75 (1/75)

The Regulatory Guide endorses the IEEE 384-1974, subject to the additions and clarifications delineated in Section C of the guide. Regulatory compliance for the NSSS scope of supply Power Generation Control Complex (PGCC), Advance Control Room system (ACR) and Nuclear Steam Supply Shutoff System (NSSSS) local panels are addressed in Section 3.13. All remaining balance of plant (BOP) circuits and equipment meet the requirements of the Regulatory Guide 1.75 except as discussed and clarified in items 4, 5, 7, 11, 13, 14, 15 and 16 below.

1) The electric power system has physical independence required by General Design Criterion 3, 17, and 21 of Appendix A of 10 CFR Part 50 to provide the minimum number of circuits and equipment to perform the required safety and protective functions assuming a single failure. 17

2) The separation of circuits and equipment (including Class IE from non-Class IE circuits) is achieved by structural design, distance, or barrier (as defined per IEEE 384-1974 Section 4 and 5), or any combination thereof. 11

Two basic circuit isolation schemes are used to isolate control circuits of two redundant load groups and Class IE from non-class IE control circuits. The first scheme consisting of an isolation type relay, P&B type MDR relay, is used to isolate interfacing control circuits. This relay has an internal physical separation between the coil and the electrical contacts. The relay coil motive power is transmitted through an extended rotary shaft which actuates a contact assembly. This relay is of Class IE category and is designed for metal plate (barrier) mounting so that the coil circuit is at one side of the plate while the contact circuits are on the other. In all applications of this relay, either the metal
17 11 plate is wide enough to provide a 6 inch minimal air space between the isolated circuits, or the relay is boxed so that the circuits have no common air space at all.

The second isolation scheme is applicable to non-interlocking control circuits of redundant separation groups (including non-class IE) that are housed in the same cabinet for operational expediency. In this case, the isolated circuit device is completely boxed, and all cabinet wiring to the device is either enclosed in a flexible metal conduit or is in a wireway with at least 6 inches of separation from the wiring and devices of the circuits it is isolated from. Isolation devices for power circuits are addressed in Paragraph 5 below.

3) The mechanical systems that are served by the electrical systems satisfy the physical independence requirements.

4) "Affiliated" circuits are non-Class IE circuits which satisfy at least one of the following conditions:

i. Supply power to non-Class IE loads from Class
17 IE power supplies.

ii Routed in a common raceway with Class IE circuits.

iii Share the same enclosure with Class IE circuits without a 6 inch minimum separation or a physical barrier.

"Affiliated" circuits are used in SSES in place of "associated" circuits which are defined in Section 4.5 of IEEE 384-1974. Affiliated circuits are same as associated circuits except the terminal equipment/devices are not subject to the requirements of Class IE equipment/devices. "Affiliated" circuits encompass the isolation methods described in paragraph 5).

The affiliated circuits are subject to the same requirements as Class IE circuits, such as unique identification, derating, environmental qualification, flame retardance, splicing restriction, raceway fill, and separation, except circuits located in the Turbine Building. All Class IE circuits (RPS) and affiliated circuits (control rod drive water pump motors, turbine building chillers, main condensate vacuum pump motors, and instrument air compressors), located in the Turbine Building, are routed in qualified Class IE raceways although they are supported from a non-Seismic Category I structure.

5) Reference: Section 4.5 and 4.6 of IEEE 384-1974. Affiliated circuits are avoided wherever possible, but where non-Class IE loads are connected to a Class IE power supply, isolation between the Class IE and non-Class IE equipment is accomplished by either of methods i through iv below. Method V is applicable to non-Class IE power supply feeding a non-Class IE circuit which becomes affiliated due to the circuits proximity to Class IE circuits/devices.

Isolation Methods:

(i) Shunt-tripping the Class IE circuit breaker or tripping of the motor contactor (Class IE) on a loss of coolant accident (LOCA) signal.

(ii) Shunt-tripping the Class IE circuit breaker or tripping of the motor contactor (Class IE) on a LOCA and total loss of offsite power (LOOP) signal.

(iii) An isolation system which consists of a Class IE circuit overcurrent interrupting device is placed in series with a non-Class IE circuit overcurrent interrupting device. The circuit between the two devices is affiliated. This method is used for a non-Class IE distribution bus.

(iv) A Class IE circuit interrupting device actuated by overcurrent is placed in series with a non-Class IE equipment. The circuit between the interrupting device and the non-Class IE equipment is affiliated.

(v) For non-Class IE circuit in proximity of Class IE circuits, an isolation system which trips on an overcurrent is placed in series with the non-Class IE circuit.

All non-Class 1E loads connected to Class 1E power supplies per isolation methods i through iv are summarized in table 8.1-2. Circuits using isolation method v are all Class IE equipment space heaters, utility, or lighting circuits where the minimum physical separation cannot be met (See Para. 16). An isolation system is defined as two separate overcurrent devices (isolation method iii and v) placed in series in a circuit to minimize any failure in the non-Class IE equipment from causing unacceptable influences in the Class IE system. The type of isolation devices used actuated by overcurrent are breakers and fuses. One of the overcurrent devices of the isolation scheme is Class IE and located in or adjacent to the Class IE equipment. The other is non-Class IE and located at or near the non-Class IE equipment. The basis for the selection of two devices in series are:

a) Both devices are of different type and different electrical characteristic to eliminate the possibility of a common mode failure due to a manufacturing defect.

b) The devices are selected to minimize the effects on the Class 1E power supply against faults in the non-Class 1E equipment.

c) The devices are coordinated to clear the fault in the non-Class 1E equipment, without tripping the Class 1E main source breaker.

d) During a seismic event, the Class 1E devices feeding to non-class 1E equipment will provide adequate circuit isolation in the event of a non-Class IE equipment failure.

e) The devices are selected to protect the Class IE circuits against faults at the non-Class IE

power circuit (isolation method v) such as short circuit and overvoltage.

6) Non-Class IE power and control circuits are separated from the Class IE and associated circuits by the minimum separation requirements specified in Section 5 of IEEE 384-1974.

Isolation devices are used where a non-Class IE control circuit and Class IE control circuits are interfaced. (See paragraph 2).

7) Reference: Position C.7 of Regulatory Guide 1.75 and Sections 5.1.3, 5.1.4 and 5.6.2 of IEEE 384-1974.

Exception to Section 5.1.3 of IEEE 384-1974: The 1" minimum separation requirement of totally enclosed raceway is not met due to space limitation in some areas. This is limited to instrument to instrument, instrument to control, and control to control, and non-Class 1E control to Class 1E power totally enclosed raceway only. For justification, refer to Wyle Lab. Test Report No. NE56719 dated November 20, 1980.

Non-Class 1E, low energy circuits for digital/analog information and instrumentation such as annunciators, data loggers, meters, recorders and transient monitoring system are permitted to be connected to Class 1E devices for required inputs. These non-Class 1E circuits are exempted from separation requirement only with the same channel/division which the circuits are connected for their inputs. The cabling of these non-Class 1E low energy circuits, with the exception of annunciators, are routed exclusively in non-Class 1E instrumentation raceways which do not contain control or power (high energy) circuits except 120 V AC.

All annunciator circuits are non-Class 1E. The cable runs of these circuits are separated from Class 1E circuits by the minimum separation requirements specified in Section 5 of IEEE 384-1974. However, annunciator cables are routed only in the non-Class 1E control raceways which contain cables of voltage level of 120 V AC, 125 V DC and 250 V DC.

All instrumentation and annunciator cables have fire retardant insulation (See Subsection 8.3.3).

The raceways are of fire retardant materials. Instrumentation cables have grounded shields.

Analysis:

Annunciator and instrumentation circuits are low energy circuits. The annunciator circuits operate in 125 V dc high impedance (60 KΩ) source. Most of the instrumentation systems operate on 1-5 V DC signals in high impedance circuits or 4-20 ma signals in low impedance circuits.

Since only low energy can be derived from instrumentation circuits, the probability of these non-Class 1E circuits providing a mechanism of failure to the Class 1E circuits inside Class 1E devices or enclosures is extremely low.

The worst credible event which could affect the Class 1E system through the non-Class 1E low energy circuits is a fire involving a control raceway containing annunciator cables. Assume in the worst case where annunciator cables from redundant class 1E equipment are both shorted to a 120 V AC, 125 V DC or 250 V DC cable due to the fire, further assume that the sensor contacts are both closed and that the overcurrent protective device of the 120 V AC, 125 V DC or 250 V DC cable does not trip. Then the class 1E devices could be damaged and therefore prevent the devices from performing their Class 1E function.

To summarize the above failure mode, the redundant Class 1E systems will fail only if all of the following conditions occur at the same time:

a. Annunciator cable from a Class 1E device is fused to the highest voltage circuit conductors (250 V DC).

b. Annunciator cable from a redundant Class 1E device is also fused to the highest voltage source (250 V DC).

c. The highest voltage (250 V DC) circuit conductors are not short circuited or grounded.

d. The highest voltage (250 V DC) circuit protective devices failed (breaker or fuse failed to perform its intended function)

e. Class IE device contact closed (alarm state)

f. Redundant Class 1E device contact closed (alarm state)

g. In order for the Class 1E protective system, as designed, to fail due to fire the above six independent low probability events must happen simultaneously. This is considered extremely unlikely. Thus, the low energy non-Class 1E circuits, which are not separated from the Class 1E circuits at the input devices do not provide a mechanism of failure of the Class 1E system.

8) In addition to the minimum separation requirements as outlined in items 6 and 7 above; (a) there are no cable splices in raceways, (b) cables and raceways are flame retardant, (c) cable trays are limited to 30 percent fill and are not filled above the side rails.

9) Raceway and cable identifications are in compliance with Regulatory Guide 1.75. Detailed description is given in Subsection 1.8.6.

10) Standby diesel generators are housed in separate rooms within a Seismic Category I structure with independent air supplies. The auxiliaries and local controls of each unit are also housed in the same room as the unit they serve.

11) Redundant Class IE batteries are located in separate rooms within a Seismic Category I structure; however, each battery room is exhausted by an individual ventilation duct to a common exhaust plenum. Two redundant Class IE centrifugal exhaust fans service the common exhaust ductwork.

Battery chargers of redundant load groups are physically separated in accordance with the requirements of Regulatory Guide 1.75.

12) All redundant Class IE switchgear, motor control centers, and distribution panels are physically separated in accordance with Regulatory Guide 1.75.

13) Redundant Class IE containment electrical penetrations are dispersed around the circumference of the containment and are physically separated in accordance with the requirements of Section 5.5 of IEEE 384-1974. Due to limited space, cable penetrations into the suppression pool contain both non-Class IE and Class IE circuits. These non-Class IE circuits are for instrumentation, annunciation, and computer inputs and are not treated as affiliated circuits.

The suppression pool area is serviced by three (3) electrical penetration assemblies: W300, W301, and W330B. Penetrations for Unit I, 1W300 and 1W301, each contains circuits of one division of the Class IE systems and non-class IE circuits. The third penetration, 1W330B, contains only non-Class IE circuits. The Unit II penetrations 2W300 and 2W301 contain only circuits of one of the redundant Class IE divisions and the third penetration 2W330B contains all the non-Class IE circuits to the suppression pool area. Penetrations W300 and W301 are located in opposite quardrents of the suppression pool for each unit.

The unit I penetrations 1W300 and 1W301 have both Class IE and non-Class IE power, control, and instrumentation circuits. The class IE power circuit conductors are routed through the penetration in a 2 inch flexible metalic conduit. The conduit extends beyond the penetration and the power cables are spliced to the plant cable in a junction box used only for the Class IE power circuit. These Class IE power circuits are separated from all others by the 2 inch conduit.

The non-Class IE power circuits service the portable lights and service equipment during personnel entrance to the suppression pool area. During plant operation the loads are removed.

Penetrations 1W300 and 1W301 also have non-Class IE instrument and control circuits. Three of the non-Class IE instrument circuits are for non-Class IE RTD inputs (Except on affiliated RTD cable, RM1I9804E, is routed together with non-Class IE circuits since it cannot be accommodated by another penetration module). These are low energy and do not degrade the Class IE circuits as discussed in Section 8.1.6.1.n 7). The non-Class IE control circuits are used for annunciator inputs only.

These annunciator circuits derive digital information from the same Class IE equipment as the Class IE control cables (ie PSV-15704A2, solenoid valve control and valve position annunciation). No other non-Class IE circuit cables are routed in the same raceway with the annunciator cables from the Class IE valve to the penetration inboard to the suppression pool. For further justification on annunciator circuits see Section 8.1.6.1.n-7). The remainder of non-Class IE instrument and control circuits are used for the Integrated Leak Rate Test (ILRT). This

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testing is performed only when the reactor is in the cold shutdown mode and personnel access to the suppression pool is permitted. After the ILRT test are completed these circuits are isolated from the rest of the plant as all test instruments and sensors are disconnected and removed from both the suppression pool and the reactor building areas. The segments of the ILRT circuits not disconnected after testing are run in separate plant raceways used only for the ILRT system.

All future non-Class IE circuits will be routed through the penetration 1W330B reserved for non-Class IE only.

14) References: Section 5.6.2 and 5.6.3 of IEEE 384-1974.

In general, circuits for redundant Class IE systems and circuits for non-Class IE systems are located in separate enclosures such as, boxes, racks, and panels. However, in cases where redundant channel/division Class IE circuits or Class IE and non-Class IE circuits, or RPS and other Class IE and non-Class IE circuits are located in the s e enclosure, physical separation is achieved either by minimum of 6" horizontal and vertical separation, steel barriers, metallic enclosure, or metallic flexible conduit (exception to this separation requirement is taken for non-Class IE low energy circuits discussed in paragraph 7 of this section). Where the above separation methods are not feasible, one of the separation group circuits are to be covered with one of the following qualified nonflammable material:

i. Haveq Industries, siltemp sleeving type S and woven tape type WT65.

ii. Carborundum, Fiberfrax sleeving type HP144T and woven tape type 3L144T.

These materials have been qualified to be used as separation barriers (Wyle Lab. Test Report No. 56669 dated May, 1980).

Applications of these materials are controlled and documented by the engineering office. Enclosures that contain wiring and devices for Class IE circuits are labeled distinctively to identify externally the separations system and grouping (see

Subsection 3.12.3.2). Internal to enclosures, terminal blocks and devices such as relays, switches and instruments are uniquely identified. In addition, external cables are color coded and marked to be readily identified (see Subsection 3.12.3.4.2). Wire bundles or cables internal to control boards are not distinctively or permanently identified.

15) Due to spatial limitation beneath the reactor vessel, the following is a description of electrical cable separation for the Neutron Monitoring System (NMS), Reactor Protection System (RPS), and Control Rod Drive System (CRD):

i. All Class IE cables are routed through enclosed raceway such as enclosed wireways, rigid and flexible conduits except as noted in paragraph iv.

ii. Non-Class IE cables are routed in open trays.

iii. Cables of different systems may be routed in some portion of raceway. But channel separation is maintained.

iv. Because of space limitation and need for flexibility, the flexible conduits end after the horizontal runs where cables drop down for connection to connectors.

v. The 1 inch minimum separation requirement of IEEE 384-1974 is not met for enclosed raceways beneath the reactor vessel. Also, the minimum separation requirements of IEEE 384-1974 Section 5.1.3 and 5.1.4 are not met for Class IE enclosed raceways and non-Class IE open trays.

All cables (Class IE and non-Class IE) beneath the reactor vessel are low energy instrumentations circuits. Fire hazard beneath the reactor vessel is described in Fire Protection Review Report Section 4.3.8 Fire Zone 1-1H.

16) Non-Class IE circuits inside a Class IE equipment, such as lighting, utility or space heater circuit, shall be considered affiliated unless a 6" minimum separation or physical barrier from the Class IE circuits is provided or unless analysis or test shows that the non-Class IE space heater circuits

will not affect the Class IE system. If power is supplied from a non-Class IE distribution panel, an isolation device or system (Isolation Method V) is installed at or near the equipment to prevent failures in the non-Class IE circuits from affecting redundant Class IE circuits. Alternatively, the non-Class IE supply cables may be routed in separate raceways such that no common mode failure could affect redundant Class IE circuits due to a single event.

o) Regulatory Guide 1.81 (1/75)

The design of the standby electric power systems meets Regulatory Guide 1.81.

The dc power systems are not shared between the two units.

The standby ac power supplies are shared between the two units. The standby ac power systems have the capability to concurrently supply the engineered safety feature loads of one unit and the safe shutdown loads of the other unit, assuming a total loss of offsite power and a single failure in the onsite power system, such as the loss of one diesel generator.

The standby ac power systems for the two units are designed with minimum interactions between each unit's safety feature circuit so that allowable combinations of maintenance and test operations in either or both units would not degrade the capability to perform the minimum required safety functions in any unit, assuming a total loss of offsite power.

p) Regulatory Guide 1.89 (11/74)

Refer to Section 3.13 for compliance statement.

q) Regulatory Guide 1.93 (12/74)

Redundant offsite and onsite power sources are provided to meet the "Limiting Conditions for Operation" as defined in Regulatory Guide 1.93. See Chapter 16 for plant operating restrictions after the loss of power sources.

r) Regulatory Guide 1.106 (11/75)

The requirements of Regulatory Guide 1.106 are met.

The thermal overload protection devices for all safety related motors on motor-operated valves (MOV) are bypassed except during testing.

This is accomplished by the use of an operate/test or normal/test type selector switch located in Panel OC697 at rear section of control room:

A. Operate/Test Type Switches

1. In the operate position, a set of normally closed (N.C.) contacts for each MOV is connected in parallel across the thermal overload trip contacts, thus bypassing the overload trip.

2. In the test position, the above set of contracts open thus permitting the overload trip contacts to trip the motor on closing or opening should an overload condition occur.

B. Normal/Test Type Switches

1. In the normal position, a set of normally open (N.O.) contacts in series with one or more relays (designated as 95) deenergizes the 95 relays. A set of normally closed relay contacts is paralleled across the thermal overload trip contacts thus bypassing the overload trip. Loss of power to the relays will cause the overloads to be bypassed.

2. In the test position the above, N.O. contacts close, energizing the 95 relays, and thus opens the contact across the MOV overload trip contacts. This permits a motor overload to trip the motor during a closing or opening test operation.

A by pass indication system is provided to alert the control room operator when a safeguard MOV is in a disabled condition. Loss of power supply, such as when the breaker is tripped for maintenance, or less of control power is indicated in the bypass indication panel C694 located behind the unit operating benchboard. A division I or II group alarm will then be made and this will be annunciated at the emergency care cooling benchboard C601.

Table 8.1-1 provides a listing of all MOV's with their thermal overload bypassed during plant operation (refer to Section 1.7 for changes).

8.1.6.2 Compliance with IEEE 338-1975, 344-1971, and 387-1972

a. IEEE 338-1971

See response to Regulatory Guide 1.118 in Section 3.13 for compliance statement.

b. IEEE 344-1971

Compliance with this standard is discussed in Section 3.10.

c. IEEE 387-1972

The following paragraphs analyze compliance with the design criteria of IEEE 387-1972.

Adequate cooling and ventilation equipment is provided to maintain an acceptable service environment within the diesel generator rooms during and after any design basis event, even without support from the preferred power supply.

Each diesel generator is capable of starting, accelerating, and accepting load as described in Subsection 8.3.1.4. The diesel generator automatically energizes its cooling equipment within an acceptable time after starting.

Frequency and voltage limits and the basis of the continuous rating of the diesel generator are discussed in the compliance statement to Regulatory Guide 1.9 in Subsection 8.1.6.1.

Mechanical and electric systems are designed so that a single failure affects the operation of only a single diesel generator.

Design conditions such as vibration, torsional vibration, and overspeed are considered in accordance with the requirements of IEEE 387-1972.

Each diesel governor can operate in the droop mode and the voltage regulator can operate in the paralleled mode during diesel generator testing. If an underfrequency condition occurs while the diesel generator is paralleled with the preferred (offsite) power supply, the diesel generator will be tripped automatically.

Each diesel generator is provided with control systems permitting automatic and manual control. The automatic start signal is functional except when the diesel generator is in the maintenance mode. Provision is made for controlling the diesel generator from the control room and from the diesel generator room. Subsection 8.3.1.4.10 provides further description of the control systems.

Voltage, current, frequency, and output power metering is provided in the control room to permit assessment of the operating condition of each diesel generator.

Surveillance instrumentation is provided in accordance with IEEE 387 as follows:

1) Starting System

   Starting air pressure low alarm

2) Lubrication System

   Lube oil pressure low trip and lube oil temperature high and low alarms. Lube oil pressure low trip is by coincident logic.

3) Fuel System

   Fuel oil level in day tank high and low, fuel oil pressure high and low, and fuel oil level in storage tank high and low alarms

4) Primary Cooling System

   Essential service water low pressure

5) Secondary Cooling System

   Jacket coolant temperature high and low, jacket coolant pressure low

6) Combustion Air Systems

   Failure alarm is provided

7) Exhaust System

Pyrometers located at diesel generator local control panel

8) Generator

Generator differential, ground overcurrent, and reverse-power, underfrequency, and overvoltage trip and alarm. Neutral overvoltage and overcurrent alarm.

9) Excitation System

Low field current and overexcitation relay trip and alarm

10) Voltage Regulation System

Diesel generator overvoltage alarm

11) Governor System

Diesel generator underfrequency alarm and trip, and engine overspeed trip

12) Auxiliary Electric System

4.16 kV bus undervoltage relays initiate bus transfer and alarm.

A detailed list of trip and alarm functions and testing of the diesel generator is discussed in Subsection 8.3.1.4.

CIRCUIT NUMBER	NON CLASS 1E LOAD	CLASS 1E POWER SUPPLY	METHOD OF ISOLATION (Ref.FSAR 8.1.6.1n.5)	23
71	480/277V Essential Lighting Panel 0EP01	Control structure H&V Room Eng. Div.I Safeguard MCC 0B136	iii	
72	480V/277V Essential Lighting Panel 0EP02	Control structure H&V Room Eng. Div.II Safeguard MCC 0B146	iii	
73	480V/277V Essential Lighting Panel 1EP05	Control structure H&V Room Eng. Div.II Safeguard MCC 0B146	iii	
74	Reactor Bldg. Chiller compressor 1K206A	Channel A/Div.I Emergency auxiliary Switchgear 1A201	iv	
75	Control Rod Drive Water pump 1P132A	Channel A/Div.I Emergency auxiliary Switchgear 1A201	iv	
76	Turbine Bldg. Chiller compressor 1K102A	Channel A/Div. I Emergency auxiliary Switchgear 1A201	iv	
77	Reactor Bldg. Chiller compressor 1K206B	Channel B/Div.II Emergency auxiliary Switchgear 1A202	iv	
78	Main condenser Mechanical vacuum pump 1P105	Channel B/Div.II Emergency auxiliary Swtichgear 1A202	iv	
79	Turbine Bldg. Chiller compressor 1K102B	Channel B/Div.II Emergency auxiliary Switchgear 1A202	iv	
80	Control Rod Drive Water pump 1P132B	Channel D/Div.II Emergency auxiliary Switchgear 1A204	iv	
81	Control Structure Passenger Elevator 0DS108	Control Structure H&V Room Div.I Engineered Safeguard MCC 0B136	iv	23
82	Engr. Safeguard Service Water Pumphouse 0LP16	Div.I Engr. Safeguard Service Water Pump house MCC 0B517	i	

CIRCUIT NUMBER	NON CLASS 1E LOAD	CLASS 1E POWER SUPPLY	METHOD OF ISOLATION (Ref.FSAR 8.1.6.1n.5)	23
94	Control Rod Drive Water Pump 2P132A	Channel A/Div. I Emergency Auxiliary Switchgear 2A201	iv	
95	Turbine Bldg. Chiller Compressor 2K102A	Channel A/Div. I Emergency Auxiliary Switchgear 2A201	iv	
96	Reactor Bldg. Chiller Compressor 2K206B	Channel B/Div. II Emergency Auxiliary Switchgear 2A202	iv	
97	Main Condenser Mechanical Vacuum Pump 2P105	Channel C/Div. I Emergency Auxiliary Switchgear 2A203	iv	
98	Turbine Bldg. Chiller Compressor 2K206A	Channel C/Div. I Emergency Auxiliary Switchgear 2A203	iv	
99.	Control Rod Drive Water Pump 2P132B	Channel D/Div. II Emergency Auxiliary Switchgear 2A204	iv	
100	Turbine Bldg. Chiller Compressor 2K102B	Channel D/Div. II Emergency Auxiliary Switchgear 2A204	iv	
101	Reactor Bldg. Closed Cooling Water Pump 2P210A	Reactor Area Div. II Engineered Safeguard MCC 2B247	i	
102	Reactor Bldg. Closed Cooling Water Pump 2P210B	Reactor Area Div. II Engineered Safeguard MCC 2B226	i	
103				23
104				
105	Process Radiation Monitoring Cabinet 2C604	Div. I 24VDC Distribution Panel 2D672	iv	

8.3 ONSITE POWER SYSTEMS

8.3.1 AC POWER SYSTEMS

8.3.1.1 Description

The onsite ac power systems are divided into Class IE and non-Class IE systems. Figure 8.3-1 shows the single line of both systems with the Class IE system identified by a dotted line enclosure.

The onsite ac power systems consist of main generators, main step-up transformers, unit auxiliary transformers, and diesel generators. The distribution system has nominal ratings of 13.8 kV, 4.16 kV, 480 V, and 208/120 V.

The off-site ac power system supplies power to plant systems through two start-up transformers.

8.3.1.2 Non-Class IE ac System

The non-Class IE portion of the onsite power systems provides ac power for non-nuclear safety related loads. A limited number of nonsafety related loads are important to the power generating equipment integrity and are fed from the Class IE distribution system through the isolation system as discussed in Subsection 8.1.6.1(n).

The non-Class IE ac power system distributes power at 13.8 kV, 4.16 kV, 480 V, and 208/120 V voltage levels. These distribution levels are grouped into two symmetrical distribution systems emanating from the 13.8 kV buses.

All non self-activated switchgears receive control power from the 125 Vdc control power sources. The 125 Vdc control power sources for the non-Class 1E 13.8 kV and 4 kV switchgear breakers and 480 V load center breakers are shown in Tables 8.3-17 and 8.3-18 respectively.

8.3.1.2.1 Operation

The unit auxiliary transformer supplies all the non-Class IE unit auxiliary loads except unit HVAC and Units 1 and 2 common loads, which are fed by the two startup transformers as shown on Figures 8.3-1 and 8.3-2.

The unit auxiliary transformer primary is connected to the main generator isolated phase bus duct tap (24 kV) while the secondary of the transformer is connected to two 13.8 kV unit auxiliary buses through a nonsegregated phase bus.

During plant startup, shutdown, and post shutdown, power is supplied from the off-site power sources through the two startup transformers. In addition, capability is provided to transfer the unit auxiliary buses to the startup power source to maintain continuity of power at the unit auxiliary distribution system.

In addition to the loading conditions mentioned in the above paragraph, the 13.8 kV startup buses also supply the preferred power supplies to the Class IE load groups through their respective 13.2 kV - 4.16 kV engineered safeguard transformers as discussed in Subsection 8.3.1.3 (Figure 8.3-1).

The auxiliary bus feeder breakers from the unit auxiliary transformers and the startup tie bus section are interlocked to prevent supplying power to the startup bus from the unit auxiliary transformer.

A 13.8 kV tie bus is provided for the two startup buses. A separate (not in switchgear line-up) bus tie breaker is located in the tie bus. In the event of a loss of startup power supply to the 13.8 kV startup bus, an alarm is initiated and, a time delay undervoltage relay initiates the tripping of the 13.8 kV incoming breaker and the closing of the tie breaker, resulting in a slow transfer. However, this transfer is prevented if either auxiliary 13.8 kV bus is being fed from the undervoltage tie bus section. This condition is sensed by the closure of two (2) auxiliary "b" contacts in series, one from each of the unit auxiliary bus to tie-bus circuit breakers connected to a common tie bus section. Manual initiation of the tie breaker is also provided. However, the use of this manual control is administratively limited as an overriding means only. Under automatic operating conditions of the tie breaker, auxiliary switch "b" contacts of the startup transformer incoming breakers are also utilized as a permissive to close the tie breaker to prevent tying of the two startup transformers.

At the 4 kV ESF power distribution subsystem a three-way transfer system is provided to enable the ESF loads to connect to either of the two off-site power sources or to the standby diesel

generators. Each ESF bus is normally connected to a preferred source which is one of two ES transformers connected respectively to the two startup buses. During loss of one off-site power source, that is, upstream of the startup bus, the startup bus undervoltage relay will trip the feeder breaker to the ES transformer, causing a transfer at the 4kV ESF bus. If power loss occurs between the 13kV startup bus and the 4kV ESF bus, a 4kV transfer will occur. The 4kV ESF bus transfer is initiated by the bus undervoltage relay, which trips the normal incoming breaker and subsequently closes the alternate incoming breaker. This is practically a dead bus transfer. If both off-site power sources are unavailable, the diesel generator breaker closes as soon as the diesel generator power is available. 23

The above transfer mechanism allows only one source breaker to be closed at any one time and to ensure this, breaker auxiliary switch contacts are used for interlocking. A manual live bus transfer is possible through a synchronizing device in which case an alternate source breaker is first closed and is followed by an automatic tripping of the preferred supply breaker. In this case the duration of the tie is merely a few cycles. Furthermore, the diesel generator can be tied with any one of the two off-site sources for an indefinite time under test condition but this does not in any way cause the two off site power systems to be tied together.

The plant security load center is double ended, each end being supplied from one of the 13 kV start-up buses through a stepdown transformer and is provided with a normally open tie breaker. Each bus is supplied from its own start-up source. Should one source be lost the undervoltage relay at the transformer secondary trips the bus incoming breaker. The bus undervoltage relay then initiates closure of the tie breaker provided the incoming breaker has successfully tripped. Upon return of the failed source the incoming breaker will not automatically close and can only be manually closed after the tie breaker has been tripped.

In all of the foregoing tie or transfer systems, there is no way that the two off-site power systems can be tied together at the on-site buses assuming loss of one off-site source.

Automatic bus transfer for the non-Class IE power system is provided at the 13.8 kV level only.

The 13.8 kV switchgear provide power for large auxiliary loads and 480 V load centers. The 13.8 kV switchgear feed double-ended 480 V load centers. A manual tie breaker is provided for each set of load centers to intertie the two load centers in the event of failure of one load center transformer. Load centers generally supply power to 480 V loads larger than 100 hp and power for their respective motor control centers. The motor control centers supply 480 V loads smaller than 100 hp while 480 V, 480/277 V, 208/120 V panels provide miscellaneous loads such as unit heaters, space heaters, lighting systems, etc.

8.3.1.2.2 Non-Class IE Equipment Capacities

Refer to Figure 8.3-1 for interconnections of the following equipment. Physical locations of each of the following equipment can be found in Section 1.2.

a) Unit Auxiliary Transformer

33/44/55 MVA, 3ø, OA/FA/FOA, 55°C
37/49.3/61.6 MVA, OA/FA/FOA, 65°C
23.0-13.8 kV Grd. Y/7.96 kV
Z = 9.0% @ 33 MVA

b) Startup Transformer

45/60/75 MVA 3ø, OA/FOA/FOA, 65°C
225/129.9 -- 13.8/7.97 kV
Z = 15.0% @ 45 MVA
LTC ± 15% in 15/16% steps

c) Engineered Safeguard Transformer

10.5/13.12 MVA, 3ø, OA/FA, 55°C
11.76/14.7 MVA, OA/FA, 65°C
13.2-4.16 kV Grd. Y/2.4 kV
Z = 6.8% @ 10.5 MVA

d) Unit Auxiliary 13.8 kV Switchgear

Buses	2000 A continuous rating, 750 MVA bracing
Incoming breakers	2000 A continuous rating, 750 MVA 3ø Class 28,000 A sym interrupting rating

Feeder breakers	1200 A continuous rating, 750 MVA 3ø Class 28,000 A sym interrupting rating

e) Startup 13.8 kV Switchgear

Buses	3000 A continuous rating, 750 MVA bracing
Incoming breakers	3000 A continuous rating, 750 MVA 3ø Class 28,000 A sym interrupting rating
Tie breaker	3000 A continuous rating, 750 MVA 3ø Class 28,000 A sym interrupting rating
Feeder breakers	1200 A continuous rating, 750 MVA 3ø Class 28,000 A sym interrupting rating

f) 4.16 kV Switchgear

Buses	1200 A continuous rating, 250 MVA bracing
Incoming breakers	1200 A continuous rating, 250 MVA 3ø Class 29,000 A sym interrupting rating
Feeder breakers	1200 A continuous rating, 250 MVA 3ø Class 29,000 A sym interrupting rating

g) 480 V Load Centers

Transformers	1500/2000 kVA, 3ø, AA/FA, 13200-480 V Grd. Y/277 V
Control structure and Administration and machine shop transformers only	1000/1333 kVA, 3ø, AA/FA, 13200-480 V Grd. Y/277 V
Buses	3000 A continuous; 65,000 A bracing (1500/2000 kVA)
	1600 A continuous; 50,000 A bracing (1000/1500 kVA)

3

Incoming breakers	3000 A continuous, 65,000 A sym interrupting rating (1500/2000 kVA)
	1600 A continuous, 50,000 A sym interrupting rating (1000/1500 kVA)
Feeder breakers	600 A continuous, 30,000 A sym interrupting rating
Tie breakers	1600 A continuous, 50,000 A sym interrupting rating

h) 480 V Motor Control Centers

Horizontal bus (main)	600 A continuous; 42,000 A bracing
Vertical bus	400 A continuous; 42,000 A bracing
Breakers (Molded Case)	
150 A frame	25,000 A symmetrical interrupting rating
250 A frame	22,000 A symmetrical interrupting rating

i) 480 V Distribution Panel

Bus	225 A rating, 14,000 A bracing
Branch breakers	100 A frame, 14,000 A interrupting rating

j) 208/120 V ac Instrument ac Distribution Panels

Main breaker (molded case)	225 A continuous 22,000 A sym interrupting rating
Buses	225 A continuous
Branch breakers (molded case)	100 A frame size 10,000 A sym interrupting rating

8.3.1.3 Class IE ac Power System

The Class IE ac portion of the onsite power system is shown on Figure 8.3-1.

The Class IE ac system distributes power at 4.16 kV, 480 V, and 208/120 V to the safety related loads. The safety related loads are divided into four load groups per generating unit and are tabulated in Table 8.3-1. Each load group has its own distribution system and power supplies.

The 4.16 kV bus of each Class IE load group channel is provided with connections to two offsite power sources designated as preferred and alternate power supplies. Diesel generators are provided as a standby power supply in the event of total loss of the preferred and alternate power supplies. Standby power supply is discussed in Subsection 8.3.1.4.

Preferred and alternate power supplies up to the 4.16 kV buses of the Class IE power system are considered as non-Class IE.

All non self-activated switchgears receive control power from the 125 Vdc control power sources. The 125 Vdc control power sources for the Class 1E 4.16 kV switchgear breakers and 480 V load center breakers are shown in Tables 8.3-19 and 8.3-20 respectively.

In order to achieve adequate separation between channelized load group and divisionalized load group, two 125 Vdc control power supplies are provided for each 4.16 kV switchgear (refer to Table 8.3-19).

8.3.1.3.1 Power Supply Feeders

Each Class IE 4.16 kV switchgear of a load group channel is provided with a preferred and an alternate (offsite) power supply feeder and one standby diesel generator feeder. Each bus is normally energized by the preferred power supply. If the preferred power source is not available at the 4.16 kV bus, automatic transfer is made to the alternate power source as described in Subsection 8.3.1.3.6. If both preferred and alternate power feeders become de-energized, the safety-related loads on each bus are picked up automatically by the standby diesel generator assigned to that bus as described in Subsection 8.3.1.4.

8.3.1.3.2 Power Feeder Cables

Power feeder cables for the 4.16 kV system are aluminum conductor, and are rated 5 kV, 90°C conductor temperature with high temperature Kerite insulation. The cables are provided with an overall flame resistant Kerite jacket covering. For the 480 V system, cables of size #4/0 AWG and larger are aluminum conductor; cables less than #4/0 AWG are copper conductor. Both types of cables are rated 600 V, 90°C conductor temperature with ethylene-propylene insulation with a flame-resistant hypalon jacket covering. The conductors are sized to carry the maximum available short circuit current for the time required for the circuit breaker to clear the fault. All Class IE cables have been designed for operation as discussed in Section 3.11.

The 4.16 kV switchgear, D.C. load centers, and D.C. Control centers are equipped with aluminum busses and silver-plated bolted connections. The 480 V load centers and motor control 11 centers are equipped with copper/aluminum busses and the bolted connections are also silver-plated. All circuit breaker terminals are copper. For power cable terminations, Burndy compression aluminum terminals (HYLUG) are used. These terminals 9 are of seamless tubular construction, tin-plated to resist corrosion, and factory filled with oxide inhibiting compound 11 penetrox A. Compression adapters MAC ADAPT MPT series or equivalent are used for equipment/vendor supplied components having mechanical lugs which cannot be converted to accept a Burndy compression lug due to physical or practical limitations. 11 A non-oxidizing lubricant such as D50H47 or equivalent will be applied on all contact surfaces at bolted joints to avoid damaging the silver-plated contact surfaces.

8.3.1.3.3 Bus Arrangements

The Class IE ac system is divided into four load group channels per unit (load group Channels A, B, C, and D). Power supplies for each load group are discussed in Subsection 8.3.1.3.1. All Class IE ac loads are divided among the four load groups so that any combination of three out of four load groups has the capability of supplying the minimum required safety loads.

The distribution system of each load group consists of one 4.16 kV bus, one 480 V load center, four or five motor control centers, and several low voltage distribution panels. The bus arrangements are shown on Figure 8.3-1, 8.3-3, 8.3-4, 8.3-7 and 8.3-8.

8.3.1.3.4 Loads Supplied from Each Bus

Table 8.3-1 provides a listing of all the loads supplied from each Class IE bus.

8.3.1.3.5 Class IE Isolated Swing Bus

Two redundant 480 V swing buses are provided for each unit for the RHR injection valve motor operators, recirculation loop bypass valve motor operators, and recirculation discharge valve motor operators. The single line of the swing bus is shown on Figure 8.3-9.

A Class IE 480 V load center of one load group channel supplies the preferred power to the swing bus through the electrical isolation of a motor-generator (M-G) set. The alternate power is supplied directly from another redundant Class IE 480 V load center. The M-G set is used for electrically isolating two redundant load groups. Faults at the swing bus cannot be propagated onto more than one load group.

The swing buses are Class IE motor control center constructions. An automatic transfer switch is provided for transferring the swing bus from the preferred to the alternate power source upon reduction or loss of voltage at the swing bus. If the undervoltage is caused by a fault at the swing bus, the transfer will be prevented even if the alternate power is available. The swing bus will be retransferred back to preferred power when the voltage is restored within acceptable limits.

The swing bus and transfer switch are designed so that for a loss of offsite power and any single failure, the minimum required ECCS flow to meet 10CFR50 Appendix K criteria is always available.

The following is a common mode-common cause failure analysis (CMCCFA) for the automatic transfer switches:

Figure 8.3-13 depicts a simplified single line diagram for the swing bus system to facilitate the analysis.

Table 8.3-24 provides a step-by-step CMCCFA of the auto transfer switch by postulating the various major common causative factors (events).

Normal conservatism in design and manufacturing margins, mandatory requirements of QA/QC procedures, Initial Test Program, Preoperational Tests, applicable administrative procedures and maintenance programs as well as operator actions contribute to minimize the susceptibility of the auto transfer switch to the various common causative factors as analyzed in Table 8.3-24.

This analysis demonstrates that the transfer switch, as a component of the swing bus system design, will not degrade the

independence and separation between the redundant Class IE channels (load center channels A and C or B and D).

The test program (Section 14.2 and Technical Specification 314.8) for the 480V swing bus system (Figure 8.3-13) consists of:

a) Periodic inspection of wiring, insulation, and connections etc. to assess the continuity of the components and system.

b) Periodic testing to verify the operability and functional performance of individual components in the system.

c) Periodic testing of operational sequence and operability of the system as a whole.

8.3.1.3.6 Manual and Automatic Interconnections Between Buses, Buses and Loads, and Buses and Supplies

No provision exists for automatically or manually connecting one Class IE load group to the redundant Class IE load group or for automatically transferring loads between load groups except the swing buses as discussed in Subsection 8.3.1.3.5.

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For each load group, one 4.16 kV feeder circuit breaker is provided for the normal incoming preferred power source, and another 4.16 kV feeder circuit breaker is connected to the alternate power source (see Subsection 8.3.1.3.1). The normal preferred power source to each bus is electrically interlocked with the alternate power source such that the bus can be connected to a single power source at any one time. In the event of loss of preferred power to the load group, undervoltage relays (less than or equal to 15 percent voltage) on the 4.16 kV switchgear will initiate an automatic transfer to the alternate power source if available. In the event of losing both preferred and alternate power supplies, the load group will be powered from the standby diesel generator.

Restoration of power from standby power to alternate power is manually initiated in the control room on panel OC653. After synchronizing the standby power source to the alternate power source, the alternate source incoming breaker is closed. Upon closing of this alternate source breaker, the standby source breaker will automatically trip. This tripping is initiated by the alternate source breaker auxiliary switch contact interlock.

A similar procedure is used to restore power from standby to the preferred source.

Restoration of power from standby power to alternate power is manually initiated in the control room on panel OC653. After synchronizing the standby power source to the alternate power source, the alternate source incoming breaker is closed. Upon closing of this alternate source breaker, the standby source breaker will automatically trip. This tripping is initiated by the alternate source breaker auxiliary switch contact interlock.

A similar procedure is used to restore power from standby to the preferred source.

8.3.1.3.7 Interconnections Between Safety Related and Nonsafety Related Buses, Nonsafety Related Loads, and Safety Related Buses

Discussion of interconnections between safety related and nonsafety related buses, nonsafety related loads, and safety related buses is presented in Subsections 3.12.2 and 8.1.6.1.

8.3.1.3.8 Redundant Bus Separation

The engineered safety features switchgear, load centers, and motor control centers for the redundant Class IE load groups are located in separate Seismic Category I rooms in the reactor building to ensure electrical and physical separation. Electrical equipment separation is discussed in Subsection 3.12.2. Equipment layout drawings can be found in Section 1.2.

8.3.1.3.9 Class IE Equipment Capacities

a) 4.16 kV Switchgear

Buses	1200 A continuous rating, 250 MVA bracing
Incoming breakers	1200 A continuous rating, 250 MVA 3ø Class 29,000 A sym interrupting rating
Feeder breakers	1200 A continuous rating, 250 MVA 3ø Class 29,000 A sym interrupting rating

b) 480 V Load Centers

Transformers (Unit 1)	750/1000 kVA, 3ø, AA/FA, 4160-480 V Grd. Y/277 V 3
Transformer (Unit 2)	750 kVA, 3ø, AA, 4160-480 V Grd. Y/277 V 3
Buses	1200 A continuous, 30,000 A bracing
Breakers	600 A frame size, 30,000 A sym interrupting rating

c) 480 V Motor Control Centers

Buses	
Horizontal (main)	600 A continuous, 42,000 A bracing
Vertical	400 A continuous, 42,000 A bracing
Breakers (molded case)	

	150 A frame	25,000 A sym interrupting rating
	250 A frame	22,000 A sym interrupting rating
d)	Automatic transfer switch	480 V, 3ø, 400 A continuous 31,000 A sym withstand capability
e)	208/120 V ac Instrument ac Distribution Panels	
	Buses	225 A continuous 10,000 A sym interrupting rating
	Branch breakers (molded case)	100 A frame size 10,000 A sym interrupting rating

8.3.1.3.10 Automatic Loading and Load Shedding

If preferred offsite power is available to the Class 1E 4.16 kV bus following a LOCA signal, the required ESF loads will start as shown in Tables 8.3-1 and 8.3-1b.

In the event of loss of preferred and alternate offsite power supplies, the Class 1E 4.16 kV buses will shed all loads except the 480 V load centers and connect the standby diesel generator to the Class 1E bus. The loading sequence is shown on Table 8.3-1.

However, if a slow bus transfer (bus voltage on transfer is less than 25%) at the Class 1E 4.16 kV bus is initiated to the alternate offsite power as a result of a loss of preferred
14 offsite power, all loads are shed except the 480 V load centers. Then the required ESF loads will start as shown on Tables 8.3-1 and 8.3-1b.

Emergency loads are also sequenced with offsite power because of the power system limitation (transformer capability). Load sequencing is desinged to minimize system disturbance and hence insure system stability.

Tables 8.3-1 and 8.3-1b show the anticipated starting time of all ESF loads. Both Unit 1 and Unit 2 busses for a given diesel generator are normally supplied by the same offsite power supply. An individual timing unit is provided for each of the ESF loads with automatic start function. Failure to start on one load will not affect the starting initiation of other loads.

The loading sequence for a simultaneous LOCA in one unit and a false LOCA in the other unit is shown in Table 8.3-1b. A false LOCA signal as used in this section refers to a non-mechanistic failure resulting in a LOCA signal in one reactor unit when a LOCA has not occurred in that unit.

The load starting transient n the diesel generator is reduced if the Unit 1, and Unit 2 load sequences do not start simultaneously.

If offsite power is available, the LOCA signal in one unit and false LOCA signal in the other will shed 2 RHR motors and 2 core spray motors of each unit and sequentially start 2 RHR and 2 core spray motors as shown in Table 8.3-1b. This is done in order not to exceed the loading limitations of the ES Transformers and to provide at least the minimum core cooling requirements of both units. Under the modified core cooling arrangement, 2 RHR pumps (one in each loop) and 2 core spray pumps (both in the same loop) will satisfy the minimum cooling requirements of each unit. Approximately ten minutes after the above event the operator will be able to determine which is the false-LOCA unit and shutdown non-essential loads in the non-LOCA unit. In case off-site power is not available, the loading is the same as discussed above, but the sequencing is slightly altered as shown in Table 8.3-16. 14

Under all conditions discussed in Subsections 8.3.1.3.10.1 and 8.3.1.3.10.2, safety function are met within the time limits shown in Table 6.3-1.

8.3.1.3.11 Safety Related Equipment Identification

Subsection 8.3.1.11.3 provides information regarding the physical identification of Class 1E equipment.

8.3.1.3.12 Instrumentation and Control Systems for the Applicable Power Systems with the Assigned Power Supply Identified

The dc power supplies for the control of the redundant Class 1E equipment are physically and electrically separate and independent. Refer to Subsection 8.3.2 for a detailed discussion of the dc system.

8.3.1.3.13 Electric Circuit Protection Systems

Protective relay schemes and direct-acting trip devices on primary and backup circuit breakers are provided throughout the onsite power system in order to:

a) Isolate faulted equipment and/or circuits from unfaulted equipment and/or circuits

b) Prevent damage to equipment

c) Protect personnel

d) Minimize system disturbances

e) Maintain continuity of the power supply

Major types of protection measures employed include the following:

a) Bus Differential Relaying

A bus differential relay is provided for each Class 1E 4.16 kV bus. This relay provides high speed disconnecting of bus supply breakers to prevent propagation of internal bus fault to another bus.

b) Overcurrent Relaying

Each Class 1E 4.16 kV bus feeder circuit breaker is equipped with three extremely inverse-time overcurrent relays to sense and to protect the bus from overcurrent condition.

The standby diesel generator feeder circuit breaker to the 4.16 kV bus is equipped with three voltage restrained overcurrent relays and one inverse-time ground fault relay for feeder circuit protection.

Each 4.16 kV motor feeder circuit breaker has three overcurrent relays, each with one long time and one instantaneous element for overload, locked rotor, and short-circuit protection. Each breaker is also equipped with an instantaneous ground current relay.

Each Class IE 4.16 kV supply circuit breaker to a 480 V load center transformer is protected by three overcurrent relays with long-time and instantaneous elements. An instantaneous overcurrent ground sensor relay provides sensitive ground fault protection.

for simultaneous operation when the engine is below 280 rpm. Refer to Subsections 9.5.5 and 9.5.7 for further description.

8.3.1.4.3 Alarm and Tripping Device

The protective and alarm logic diagrams for the diesel generator and its associated breakers are shown on Figures 8.3-11 and 8.3-12.

While supplying loads following an automatic start, each diesel engine and related generator circuit breaker are tripped by protective devices under the following conditions only:

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c) Under/Overvoltage Relaying

Each 4.16 kV Class IE bus is equipped with undervoltage relays for diesel generator starting and undervoltage annunciation. Each 480 V Class IE load center bus is equipped with under/overvoltage relays for annunciation.

d) Diesel Generator Differential Relaying

Each diesel generator is equipped with differential relaying protection. This circuitry provides high speed disconnection to prevent severe damage in case of diesel generator internal faults.

e) 480 V Load Center Protection

Each load center circuit breaker is equipped with integral, solid-state, dual magnetic, adjustable, direct-action trip devices providing inverse-time overcurrent protection. Motor feeders are equipped with long-time overcurrent and instantaneous short-circuit protection.

f) 480 V Motor Control Center Protection

Molded-case circuit breakers provide inverse-time overcurrent and/or instantaneous short circuit protection for all connected loads. For motor circuits, the molded-case circuit breakers are equipped with an adjustable instantaneous magnetic trip function only. Motor thermal overload protection is provided by the heater element trip unit in each phase of the motor feeder circuit. The molded-case breakers for nonmotor feeder circuits provide thermal inverse-time overcurrent protection and instantaneous short circuit protection. The thermal overload trip units for safety related motor-operated valves are normally bypassed except during maintenance tests.

The circuit protection system is designed so that fault isolation is secured with a minimum circuit interruption. 3

The combination of devices and settings applied affords the selectivity necessary to isolate a faulted area quickly with a minimum of disturbance to the rest of the system. 3

The protective devices are preoperationally tested in accordance with the requirements of Chapter 14. After the plant is in operation, periodic tests will be performed to verify the

protective device calibration, set points, and correct operation in accordance with the requirements of Chapter 16.

8.3.1.3.14 Testing of the ac System During Power Operation

All Class IE circuit breakers and motor starters, except for the electric equipment associated with Class IE loads identified in Subsection 8.3.1.3.15, are testable during reactor operation. During periodic Class IE system tests, subsystems of the ESF system such as safety injection, containment spray, and containment isolation are actuated, thereby causing appropriate circuit breaker or contactor operation. The 4.16 kV and 480 V circuit breakers and control circuits can also be tested independently while individual equipment is shut down. The circuit breakers can be placed in the test position and exercised without operation of the related equipment.

8.3.1.3.15 Class 1E Loads not Testable During Power Operations

A. Feedwater Line Isolation Valves

The feedwater line isolation valves (HV-F032 A/B) are of the motor operated check valve type and are not testable with the feedwater flow present. Motor operation is not required for isolation. Only the outermost isolation valve is Class 1E powered and would be motor operated for long term isolation after isolation of the feedwater line.

Conformance with Regulatory Guide 1.22 Section D.4:

1. The feedwater isolation is not designed for isolation with feedwater flow present as the loss of flow would adversely affect operability of the plant.

2. Motor operation is not required for isolation.

3. The motor operator of the outermost isolation valve is fully testable during shutdown.

B. Main Steam Isolation Valves

The main steam isolation valves can be tested individually to the 90% open position at full power with the slow acting test solenoid valve. A fully closed test using the two fast acting main solenoids would require a reduction in reactor power.

Conformance with Regulatory Guide 1.22: See sections 7.3.2a.2.2.1.2 and 5.4.5.4.

C. ADS System - Safety/Relief Valves

The active components of the ADS system except the safety/relief valves and their associated solenoid valves are designed so they may be tested during plant power operation. The relief valve and associated solenoid are not tested during reactor power operation.

Conformance with Regulatory Guide 1.22:

1. The safety/relief valves are not tested during power operation because of resulting adverse affect on plant operation.

2. Because of low failure rates of valve actuation, the probability of failure is acceptably low without testing.

3. The safety/relief valves and associated solenoid valves can be tested during startup following shutdown. 15

D. Recirculation Loop Isolation Valves

The recirculation pump isolation valves are not tested during reactor power operations.

Conformance with Regulatory Guide 1.22 Section D.4:

1. Operation of a recirculation loop isolation valve would result in a reduction of circulation which would adversely affect the safety and operability of the plant.

2. The probability of failure is acceptably low without testing the valve motor during operation.

3. The valve and motor are fully testable during reactor shutdown.

8.3.1.4 Standby Power Supply

The standby power supply for each safety related load group consists of one diesel generator complete with its accessories and fuel storage and transfer systems. Each diesel generator is rated 4000 kw at 0.8 pf for continuous operation and 4700 kw for 2000 hr operation. The ratings for each diesel generator are calculated in accordance with the recommendation of Regulatory Guide 1.9 (discussed in Subsection 8.1.6.1). The diesel-generators can operate at loads of from 50 to 100 percent for unlimited periods without harm. Any diesel generator continuously operated at loads of less than 50 percent will be 11

loaded to 75-100 percent for 15-30 minutes approximately every six hours and immediately prior to shutdown. Such operation will enhance engine performance and reliability. 11 18

The four diesel generators are shared by the two units. Each diesel generator is connected to the 4.16 kV bus of the assigned load group per unit. The capacity of the diesel generators (assuming one diesel fails) is sufficient to operate the engineered safety features loads of one unit and those systems required for concurrent safe shutdown of the second unit.

No provisions are provided for parallel operation of the diesel generator of one load group with the diesel generator of the redundant load group. The diesel generator circuit breaker and the offsite power incoming circuit breakers are interlocked to prevent feedback into the offsite power system. These interlocks are bypassed during diesel generator load tests; however, only one unit is tested at any one time. During the test period, the diesel generator under test is manually synchronized to the preferred offsite power system. Upon receipt of a LOCA signal under the test condition, the diesel generator breaker is tripped but the diesel generator continues to run.

The diesel generators are physically and electrically isolated from each other. Physical separation for fire and missile protection is provided between diesel generators by separate rooms within a Seismic Category I structure. Power and control cable for each of the diesel generators and associated switchgear are routed in separate raceways. Physical electrical equipment layout of the diesel generator rooms is shown on Figure 8.3-10.

Auxiliaries required for starting and continuous operation of each diesel generator are fed by the Class IE power load group associated with that diesel generator.

Control power for each diesel generator is provided by its corresponding 125 V dc systems from both Unit 1 and Unit 2. These two power feeders are not redundant, but have been provided for ease of maintenance. Indication of which unit is supplying the dc control power is not provided in the control room. Manual switches are installed at the local panel to select the preferred power feeder. Since each diesel generator is shared by both units, either source of DC control power is adequate. Loss of DC power to the Diesel Generator is indicated on the BIS panels as a group trouble alarm on panel OC653 in the main control room. 11

Each diesel generator is provided with a local engine control panel, a generator-exciter control panel, a local 4.16 kV

distribution panel, and a 480 V motor control center in the diesel generator room.

a) Local Engine Control Panel - consists of a local annunciator, engine control devices, gages, and control for diesel generator auxiliary equipment such as fuel oil transfer pump, standby jacket water pump, etc.

The diesel generator control system is designed in such a manner that some control devices are mounted in the free standing control panel separate from the engine, while others are mounted directly on the engine, as required for reliable service. All devices that are essential to the start-up or power output of the diesel-generator set have been seismically qualified by analysis or test to acceleration levels consistent with their mounting location. 18

b) Generator-Exciter Control Panel - consists of generator excitation control equipment, generator protective relays and devices, etc.

c) 4.16 kV Distribution Panel - provides connections for diesel generator feeders to Unit 1 and 2. Also houses potential transformers and current transformer, etc.

d) 480 V Motor Control Center - provides power to all 480 V auxiliary equipment related with that diesel generator. This MCC is equipped with an automatic transfer switch for connection to either Unit 1 or 2 480 V Class IE load center. These two load centers belong to the same load group channel as the diesel generator.

Physical separation of standby power system is discussed in Section 3.12.

8.3.1.4.1 Automatic Starting Initiating Circuits

The diesel generators are automatically started by any of the following conditions:

a) Total loss of power at the 4.16 kV Class IE bus of either unit to which the diesel generator is connected

b) Safety injection signal - low water level in the reactor, high drywell pressure, or manual actuation.

Two redundant control/starting circuits are provided for each diesel generator. Failure of one circuit would not prevent the respective diesel generator from starting or from continuous operation.

The diesel generators are ready to accept loads within 10 sec after the initiation of the start circuit.

8.3.1.4.2 Diesel Starting Mechanism and System

The diesel generator start system is described in Subsection 9.5.6. To ensure fast and reliable starting, each diesel engine is provided with immersion heaters in the engine jacket water and the lube oil system to maintain the engine coolant and lube oil temperature at an operable level. The electric jacket water immersion heater and the water circulating pump are interlocked for simultaneous operation when the jacket water temperature drops below the preset temperature. The electric lube oil immersion heater and the prelube circulating pump are interlocked for simultaneous operation when the engine is below 280 rpm. Refer to Subsections 9.5.5 and 9.5.7 for further description.

8.3.1.4.3 Alarm and Tripping Device

The protective and alarm logic diagrams for the diesel generator and its associated breakers are shown on Figures 8.3-11 and 8.3-12.

While supplying loads following an automatic start, each diesel engine and related generator circuit breaker are tripped by protective devices under the following conditions only:

a) Engine overspeed

b) Lube oil low pressure

c) Generator differential

To prevent spurious tripping of the diesel generator due to malfunction of the engine lube oil low pressure trip device, four independent sensors are provided and connected in a coincidence one-out-of-two taken twice tripping logic. An individual tripping alarm is provided by the annunciator at each local control panel.

The starting circuit is also equipped with a "fail to start" relay operator that interrupts the starting of the diesel generator if a predetermined speed is not reached within a limited time following a start initiation.

In addition to the above-listed trips, each generator circuit breaker is tripped by the following protective relays to disconnect the generator from a faulty bus (the diesel generator continues to run):

a) Voltage restrained overcurrent

b) 4 kV bus differential.

Following a manual start, a diesel generator is in the test mode and ready for a load test. When so operated, in addition to the above-listed trips, each diesel engine and related generator circuit breaker are automatically tripped by the following protective devices:

a) Generator loss of field

b) Generator overexcitation

c) Antimotoring

d) Generator underfrequency

e) Generator overvoltage

f) Generator high bearing temperature

g) High jacket water temperature

h) Turbo lube oil pressure low

i) Main and connecting rod bearing temperature high

j) Engine vibration

k) Turbo thrust bearing failure.

An individual alarm is also provided for each of these abnormal conditions at the local control panel. A group alarm is provided in the main control room as a high priority alarm.

Other relays and devices are provided to annunciate abnormal diesel engine and generator conditions at the local control panel as following. These conditions are annunciated in the main control room as a low priority alarm.

a) Generator field ground

b) Generator voltage unbalance

c) Generator neutral overvoltage

d) Engine lube oil pressure high

e) Crankcase pressure high

f) Engine lube oil temp off normal

g) Engine crankcase level low

h) Auxiliary standby pump on

i) Jacket water temperature off normal

j) Jacket water low pressure

k) Fuel oil pressure high

l) Fuel oil pressure low

m) Fuel strainer high differential pressure

n) Fuel filter high differential pressure

o) Lube oil filter high differential pressure

p) Starting air system low pressure or malfunction

q) Voltage regulator transfer to standby

r) Jacket water standpipe level high

s) Jacket water standpipe level low

t) Fuel oil day tank level high

u) Fuel oil day tank level low

v) Fuel storage tank level high

w) Fuel storage tank level low

x) Motor control center not proper for automatic operation (actuated by blown control fuse, etc.)

y) Control switches not proper for remote automatic operation (diesel generator auxiliaries)

z) Lube oil circulating pump malfunction

aa) Lube oil heater malfunction

bb) Jacket water heater malfunction

cc) Jacket water circulating pump malfunction

The following alarms are provided in the main control room annunciator:

a) Diesel generator tripped

b) High priority alarm (all trip conditions listed previously

c) Low priority alarm (all abnormal conditions listed previously

d) Diesel generator breaker tripped

e) Diesel generator fail to start

f) Diesel generator near full load

g) Diesel generator not in automatic mode.

8.3.1.4.4 Breaker Interlocks

Interlocks have been provided in the closing and tripping of the 4.16 kV Class IE circuit breakers to protect against the following conditions:

a) Automatic energizing of electric devices or loads during maintenance

b) Automatic closing of the diesel generator breaker to any energized or faulted bus

c) Connecting two sources out of synchronism

8.3.1.4.5 Control Permissive

A single key-operated switch at the local control panel is provided for each diesel generator to block automatic start signals when the diesel is out of service for maintenance. An annunciator alarm in the main control room and an indication at the bypass-indication-system panel indicate when the switch is not in automatic position.

A pushbutton in the control room and a local pushbutton at the local control panel in the diesel generator room are provided to allow manual start of the diesel when all protective systems are permissive. During periodic diesel generator tests, permissives and interlocks are designed to permit manual synchronizing and loading of the diesel generator with either offsite power source.

8.3.1.4.6 Loading Circuits

Upon automatic starting of the diesel (emergency mode), connection of the diesel generator to the 4.16 kV bus is not made unless both offsite power sources are lost. As the generator reaches the predetermined voltage and frequency levels, control relays provide a permissive signal for the closing of the respective diesel generator breaker to the corresponding 4.16 kV bus. The diesel generator circuit breaker is closed within 10 sec after the receipt of the starting signal. The required safety related loads are connected in sequential order to the Class IE buses as shown in Table 8.3-1. This prevents diesel generator instability and ensures voltage recovery thereby minimizing motor accelerating time. A fast-responding exciter and voltage regulator ensures voltage recovery of the diesel generator after each load step.

8.3.1.4.7 Testing

Preoperational Test

Each diesel generator is tested at the site prior to reactor fuel loading in accordance with requirements of Chapter 14.

Periodic Testing

After being placed in service, the standby power system is tested periodically to demonstrate continued ability to perform its intended function, in accordance with the requirements of Chapter 16.

8.3.1.4.8 Fuel Oil Storage and Transfer System

The diesel generator fuel oil system is described in Subsection 9.5.4.

8.3.1.4.9 Diesel Generator Cooling and Heating

The diesel generator cooling system is described in Subsection 9.5.5.

8.3.1.4.10 Instrumentation and Control Systems for Standby Power Supply

The instrumentation and control circuit of each diesel generator is provided with a manual selector switch for connection to either Unit 1 or 2 125 V dc power supply. These two power supplies belong to the same load group channel to which the diesel generator is connected.

Control hardware is provided in the control room for each diesel generator for the following operations:

a) Starting and stopping

b) Synchronization

c) Frequency and voltage adjustment

d) Manual or automatic voltage regulator selection

e) Isochronous and droop selection.

Control hardware is provided at each local control panel for the following operations:

a) Starting and stopping

b) Frequency and voltage adjustment

c) Manual or automatic diesel generator mode (key lock selector switch)

d) Automatic or manual voltage regulator selection

e) Normal or standby voltage regulator selection

f) Units 1 or 2 dc control power supply selection.

Electrical metering instruments are provided in the control room for surveillance of the diesel generator:

a) Voltage

b) Current

c) Frequency

d) Power output.

Electrical metering instruments are provided at the local control panel for surveillance of the diesel generator:

a) Voltage

b) Current

c) Frequency

d) Power (watt) output

e) Reactive power (var) output.

8.3.1.4.11 Qualification Test Program

8.3.1.4.11.1 Class IE Equipment Identification

The diesel-generator sets are designated Class IE since they perform essential safety-related functions. Therefore, the equipment was qualified per IEEE 323-1971 and documented in Cooper Energy Services (CES) Report #CE-0188-1. The diesel engine, synchronous generator, and auxiliaries, such as heat exchangers, air receivers, and fuel tanks were qualified.

8.3.1.4.11.2 Qualification Techniques and Documentation

All testing conducted by CES for the Susquehanna SES diesel-generator sets provides the basis for data evaluation of future, ongoing, periodic, jobsite testing. Periodic exercising of the diesel-generator sets shows availability and reliability. Data taken during those tests will be compared to data taken under corresponding load conditions during factory testing. By comparison, trends which may indicate equipment degradation are developed and utilized to predict maintenance intervals.

Testing and analyses completed to verify equipment performance capability are as follows:

a) Testing performed on the first generator of this contract included the following parameters, with testing procedures as outlined in IEEE 115. Refer to Electric Products test report for generator serial number 17402243-200 dated 5-20-76 for documentation of test results.

1. Synchronous impedance curve.

2. Zero power factor saturation curve.

3. Losses (for efficiency calculation).

4. Direct-axis synchronous reactance.

5. Negative sequence reactance.

6. Direct-axis transient reactance.

7. Direct-axis transient open circuit time constant.

8. Open circuit saturation curve.

9. Start circuit test.

b) Testing performed on each generator furnished under this contract included the following parameters with testing procedure as outlined in IEEE 115. Refer to Electric Products test report for generator serial numbers 17402244/246-200 dated 6/22/76 for documentation of test results.

1. Insulation resistance.

2. High potential tests.

3. Winding resistance.

4. Overspeed.

5. Phase sequence rotation.

6. Mechanical balance.

c) Testing was performed on each assembled engine-generator set per IEEE 387 and included the following. Refer to CES test procedure T1-T5 and to CES reports for engine serial numbers 7157-60 for documentation of test results.

1. High potential testing of control wiring.

2. Measurement of engine vibration.

3. Fast start capability.

4. Transient performance evaluation.

5. Steady state load capability.

6. Load rejection.

7. Number of starts from a single air receiver.

8. Performance evaluation of power factor discriminator and standby voltage regulator.

d) Functional auxiliaries, such as lube oil pumps, jacket water pumps, heaters, and coolers were evaluated to ensure proper operation during the assembled engine-generator set testing described in c above. The functional capability of the auxiliaries is documented in the test log section of the CES reports for engine serial numbers 7157-60. The establishment of adequate pressures and temperatures in the lube oil, cooling water, and fuel oil systems confirms correct operation of auxiliaries.

e) Engine and generator control panels were assembled and tested with their respective engine-generator sets and evaluated for proper control and monitoring. Refer to CES reports for engine serial numbers 7157-60 for test results. The achievement of engine-generator transient and steady state performance confirms correct operation of control panels.

f) To evaluate the seismic effects on the safe shutdown capability some tests have been performed, but most evaluations were achieved by analysis. Both CES and vendor furnished equipment, which are essential to the power output capability of the generator, have been seismically evaluated and determined adequate to meet the specified response spectra with no loss of functional or structural integrity. Refer to CES seismic reports numbered CES-1 through CES-49 for documentation of seismic analyses and tests.

8.3.1.4.11.3 Performance In Service Environment

Actual performance requirements and service conditions are achievable in the field installation only. Simulation of performance is attained through computer techniques which comparatively analyze motor starting data taken during factory testing with motor load starting characteristics predicted for the essential pumps-motors to be started at the jobsite. Simulation of service environments, such as the predicted diesel generator room ambient temperature, would require an environmental chamber large enough to store the entire engine-generator set. In order to ascertain the ability of this equipment to perform in the predicted environment, operating experience and design experience are used. The varied types of engines designed, the varied installation applications, and the resultant experience gained have determined the capabilities of this equipment to perform under different service conditions. This experience is augmented by previous and ongoing R&D testing of a similar CES Type KSV engine where specific data may be needed relative to particular performance requirements. However, much of this data is proprietary.

As a result of this experience and testing, it is concluded the service conditions described in Section 3.11 can be accommodated while fulfilling the performance requirements. For example, installation elevations of up to 1500 feet are accommodated without any derating or design modification. The 676 feet elevation for the Susquehanna SES diesel-generator sets falls well within this range. To accommodate variance in combustion air temperature, coolers/heaters are supplied which either add heat to or take heat from combustion air as needed to provide the necessary manifold air temperature. The range of -19°F to +105°F air temperature is therefore accommodated.

In addition, all service water heat exchangers are designed with fouling factors incorporated permitting the buildup of specified amounts of dirt or sludge while maintaining the necessary heat tansfer characteristics under the most adverse load and cooling water temperature conditions. Particles or minerals in the service water are therefore accommodated in heat exchanger design.

Seismic effects are taken into account analytically and by test for all essential components and systems of the diesel-generator sets.

8.3.1.4.12 Control and Alarm Logic

The control and alarm logic for the diesel generators is shown on Fig. 8.3-12. Conditions which render the diesel generator incapable of responding to an automatic emergency start are shown on Table 8.3-16. The following is an item by item analysis of each of these conditions:

General Note

The diesel generator will be tripped by (1) generator differential relay, (2) engine overspeed, and (3) low engine lube oil pressure (one-out-of-two taken twice logic) under emergency operation. For test operation, the diesel generator will be tripped by all conditions listed under "Diesel Generator High Priority" alarm as shown on Figure 8.3-12. Following a manual stop, no reset is necessary for subsequent emergency or test operation except the mode selector switch must be returned to "Remote" position. This condition is annunciated locally and in the control room. Following a trip, control circuit must be reset. Diesel generator trip is also alarmed locally and in the control room. 2

There are two engine starting circuits for each diesel generator for added reliability. Each circuit is supplied from the same 125V battery system but through separate circuit breakers. Only

one circuit is required for starting and keeping the diesel generator in a running mode. Therefore, any single component
13 failure (as listed in Column B of Table 8.3-16) cannot prevent the diesel generator from starting.

1) ID-B.1 Generator Differential Relay activated

A generator differential relay is provided for each diesel generator for internal fault protection. This relay will trip the diesel generator under any mode of operation. The diesel generator differential alarm is annunciated locally and repeated as a group alarm "Diesel Generator High Priority Trouble" in the main control room.

2) ID-B.2 Engine Overspeed Relay activated

An independent overspeed sensor is provided for each diesel generator starting circuit. Activation of any one sensor is alarmed, but will not prevent the diesel generator from starting or running.

3) ID-B.3 Engine Lube Oil Low Pressure Relay activated

2 Each of the control circuits have two independent engine lube oil low pressure switches arranged in a one out of two logic. Pressure switches are bypassed during engine starting. Therefore, alarm is initiated for any one pressure switch (or relay) activation. Disabling of the diesel generator can only be accomplished with one engine lube oil low pressure relay activated in each control circuit. 4) ID-B.4 Operating Mode Switch in "Local"

Operating mode switch (key locked) is put on "Local" for local testing and maintenance services only. "Local position" is annunciated in the main control room as "Diesel Generator not in Auto." Alarm is also indicated in the Bypass Indication System (BIS) on "Diesel Generator Switch in Local" (also in the main control room). Automatic bypass of the "Local" operating mode under emergency condition is not provided. Only one diesel generation will be tested or taken out for service at any one time.

5) ID-B.5 Loss of 125 VDC Engine Control Power

As discussed above, two separate control circuits are provided for each diesel generator. Alarm is indicated locally and annunciated in the main control room as "Diesel Generator High Priority." Indication is also provided at the BIS panel. Loss of either circuit will not prevent the diesel generator from starting or operating.

6) ID-B.6 Control Relay Malfunction

Control relays can fail in either contact open or closed state. Since there are two circuits provided, assuming a single relay failure, the diesel generator will not be prevented from starting or operating.

7) ID-B.7 Engine & Generator Mechanical Trouble

Low priority and high priority trouble alarms are provided for engine and generator mechanical trouble as shown on Figure 8.3-12 and Table 8.3-16. 13 13

8) ID-B.8 Starting Air Control Solenoid Valve Failure

There are two starting air solenoid valves for each of the two starting circuits for each D/G. Loss of any three starting solenoids will not prevent the diesel generator from starting.

9) ID-B.9 Starting Air System Trouble

See (9) ID-B.9 and Section 9.5.6 for a complete starting air system discussion. The starting air pressure is monitored at all times with annunciation provided locally and in the main control room.

10) ID-B.10 Fuel Oil Control Solenoid Failure

One fuel oil control solenoid is provided in each of the two control circuits for each diesel-generator. A failure of either fuel oil control solenoid will not preent the diesel generator from starting. 2

11) ID-B.11 Loss of 125 VDC Generator Control Power

Loss of the generator control power will prevent the operation of the excitation system. Indication is provided at the Bypass Indication System as "Excitation Control Power Loss" (Main Control Room).

12) ID-B.12 Disabling of Engine and Generator Mechanical Parts During Maintenance Services

Before the diesel generator is taken out of automatic mode for maintenance services, the operating mode selector switch must be in "Local" position as required by maintenance procedures. This will result in an alarm in the main control room as "Diesel generator not in auto" ("Diesel generator control switch in LOCAL" in BIS panel)

No alarms are specifically provided for monitoring of engine and generator mechanical parts under the subject condition.

Conclusion

No modifications are necessary as a result of this evaluation because adequate alarms and indications are provided in addition to the alarm redundancy of the control circuits.

8.3.1.5 Electrical Equipment Layout

Class IE switchgear, load centers, motor control centers, and distribution panels of redundant load groups are in separate rooms of the reactor building and the control structure.

Standby diesel generators and associated equipment are in separate rooms of the Seismic Category I diesel generator building. Each room is provided with a separate ventilation system.

Plant layout drawings are included in Section 1.2.

8.3.1.6 Reactor Protection System Power Supply

The reactor protection system (RPS) power supply is a non-Class IE system. The normal 120 V ac power to each of the two reactor protection systems is supplied, via a separate bus, by its own high inertia motor generator set. The drive motor is supplied from a 480 V non-Class IE motor control center. High inertia is provided by a flywheel. The inertia is sufficient to maintain voltage and frequency within 5 percent of rated values for at least 1.0 sec following a loss of power to the drive motor.

The alternate 120 V ac power for each of the reactor protection systems is supplied by a non-Class IE motor control center through a 480-120 V, 1ø transformer. A selector switch is provided for the selection of the two power supplies. The switch also prevents paralleling the motor generator set with the alternate supply.

The electrical protective assembly (EPA), consisting of Class 1E protective circuitry is installed between the RPS and each of the power sources. The EPA provides redundant protection to the RPS and other systems which receive power from the RPS busses by acting to disconnect the RPS from the power source circuits.

The EPA consists of a circuit breaker with a trip coil driven by logic circuitry which senses line voltage and freqency and trips the circuit breaker open on the conditions of overvoltage, undervoltage and underfrequency. Provision is made for setpoint verification, calibration and adjustment under administrative

control. After tripping, the circuit breaker must be reset manually. Trip setpoints are based on providing 115 VAC, 60 Hz power at the RPS logic cabinets. The protective circuit functional range is ± 10% of nominal AC voltage and -5% of nominal frequency.

The EPA assemblies are packaged in an enclosure designed to be wall mounted. The enclosures are mounted on a seismic Category I structure separately from the motor generator sets and separate from each other. Two EPAs are installed in series between each of the two RPS motor-generator sets and the RPS busses and between the auxiliary power sources and the RPS busses. The block diagram in Figure 7.2-9 provides an overview of the EPA units and their connections between the power sources and the RPS busses. The EPA is designed as a Class 1E electrical component to meet the qualification requirements of IEEE 323-1974 and IEEE 344-1975. It is designed and fabricated to meet the quality assurance requirements of 10CFR50, Appendix B.

The enclosures containing the EPA assemblies are located in an area where the ambient temperature is between 40°F and 122°F. The circuits within the enclosure are qualified to operate under accident conditions from 40°F to 137°F, at 10% to 95% relative humidity and survive a total integrated radiation dose of $2x10^5$ rads. The assemblies are seismically qualified per IEEE 344-1975, to the Safe Shutdown Earthquake (SSE) and Operating Base Earthquake acceleration response spectra and environmentally qualified to the requirement of IEEE 323-1974. The enclosure dimensions are approximately 16x24x8 inches and accommodate power cable sizes from 7 AWG to 250 MCM.

8.3.1.7 Class IE 120 V ac Instrumentation and Control Power Supply

Four independent Class IE 120 V ac instrumentation and control power supplies are provided to supply the four channels of engineered safety features load groups. The four bus arrangement provides a separate single-phase electric power supply to each of the four protection channels that are electrically and physically isolated from the other protection channels. Each

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power supply consists of a 480-120 V transformer and a distribution panel. The 480 V power supply is provided by the corresponding 480 V Class IE motor control center.

There is no manual or automatic transfer between the four 120 V ac Class IE panels.

There is no automatic loading or load shedding of the panels.

8.3.1.8 Non-Class IE Instrument and Vital ac Power Supply

Non-Class IE Instrument ac Power Supply

Two 208/120 V non-Class IE instrument ac power supplies per unit furnish reliable power to non-Class IE miscellaneous instrumentation systems.

The non-Class IE instrument ac power supply for each unit consists of two subsystems, each with a regulating transformer, an automatic transfer switch, and a 208/120 V distribution panel. Each distribution panel is supplied as an associated circuit from two Class IE motor control centers.

The transfer switch maintains separation between the two Class IE power supplies, and the redundant breakers act as an isolation system between the Class IE power supply and the non-Class IE load.

Vital ac Power Supply

Two 208/120 V non-Class IE vital ac power supplies (uninterruptible power supplies) per unit supply essential non-Class IE equipment such as the plant computer. Each vital ac power supply consists of one inverter, automatic transfer switch, manual bypass switch, and distribution panel(s). Normally, the distribution panel is supplied by the inverter.

Each inverter is supplied by a separate Class IE 250 V dc subsystem as described in Subsection 8.3.2. If the inverter is inoperable or is to be removed from service for maintenance or testing, a transfer to the backup supply is made through the manual bypass switch. The backup supply is a regulating type transformer from a 480 V Class IE motor control center. A transfer switch provides the automatic switch-over in case of inverter failure.

The supply from the Class IE 480 V MCC is an associated circuit. Redundant breakers act as an isolation system between the Class IE power supply and non-Class IE load.

8.3.1.9 Design Criteria for Class IE Equipment

The following design criteria are applied to the Class IE equipment.

MOTOR SIZE - Motor size (horsepower capability) is equal to or greater than the maximum horsepower required by the driven load under normal running, runout, or discharge valve (or damper) closed condition.

MINIMUM MOTOR ACCELERATING VOLTAGE - The electrical system is designed so that the total voltage drop on the Class IE motor circuits is less than 20 percent of the nominal motor voltage. The Class IE motors are specified with accelerating capability at 80 percent nominal voltage at their terminals.

MOTOR STARTING TORQUE - The motor starting torque is capable of starting and accelerating the connected load to normal speed within sufficient time to perform its safety function for all expected operating conditions, including the design minimum terminal voltage.

MINIMUM MOTOR TORQUE MARGIN OVER PUMP TORQUE THROUGH ACCELERATING PERIOD - The minimum motor torque margin over pump torque through the accelerating period is determined by using actual pump torque curve and calculated motor torque curves at 80 and 100 percent terminal voltage. The minimum torque margin (accelerating torque) is such that the pump-motor assembly reaches nominal speed in less than five seconds. This margin is usually not less than 10 percent of the pump torque.

MOTOR INSULATION - Insulation systems are selected on the basis of the ambient conditions to which the insulation is exposed. For Class I motors located within the containment, the insulation system is selected to withstand the postulated accident environment.

TEMPERATURE MONITORING DEVICES PROVIDED IN LARGE HORSEPOWER MOTORS - Six resistance temperature detectors (RTD) are provided in the motor stator slots, two per phase, for motors larger than 1500 hp. In normal operation, the RTD at the hottest location (selected by test) monitors the motor temperature and provides an alarm on high temperature. RTDs are provided for motors from 250 to 1500 hp. Each bearing that is not antifriction type has a chromel-constantan ISA Type E thermocouple bearing temperature device to alarm on high temperature.

INTERRUPTING CAPACITIES - The interrupting capacities of the protective equipment are determined as follows:

a) Switchgear

Switchgear interrupting capacities are greater than the maximum short circuit current available at the point of application. The magnitude of short circuit currents in medium voltage systems is determined in accordance with ANSI C37.010-1972. The offsite power system, a single operating diesel generator, and running motor contributions are considered in determining the fault level. High voltage power circuit breaker interrupting capacity ratings are selected in accordance with ANSI C37.06-1971.

b) Load Centers, Motor Control Centers, and Distribution Panels

Load center, motor control center, and distribution panel interrupting capacities are greater than the maximum short circuit current available at the point of application. The magnitude of short circuit currents in low-voltage systems is determined in accordance with ANSI C37.13-1973, and NEMA AB1. Low-voltage power circuit breaker interrupting capacity ratings are selected in accordance with ANSI C37.16-1970. Molded case circuit breaker interrupting capacities are determined in accordance with NEMA AB1.

ELECTRIC CIRCUIT PROTECTION - Electric circuit protection criteria are discussed in Subsection 8.3.1.3.13.

GROUNDING REQUIREMENTS - Equipment and system grounding are designed in accordance with IEEE 80-1961 and 142-1972.

8.3.1.10 Safety-related Logic and Schematic Diagrams

Safety-related logic and schematic diagrams are provided as listed in Section 1.7.

8.3.1.11 Analysis

A failure mode effects analysis for the ac power system is presented in Table 8.3-9.

8.3.1.11.1 General Design Criteria and Regulatory Guide Compliance

The following paragraphs analyze compliance with General Design Criteria 17 and 18. All Regulatory Guides are discussed in Subsections 3.13 and 8.1.6.1.

GENERAL DESIGN CRITERION 17, ELECTRIC POWER SYSTEMS

An onsite electric power system is provided to permit functioning of structures, systems, and components important to safety. With total loss of offsite power, the onsite power system provides sufficient capacity and capability to ensure that:

a) Specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences

b) The core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Tables 8.3-1 to 8.3-5 list those loads important to safety under design conditions.

The onsite electric power system includes four load groups. The load groups are redundant in that three load groups are capable of ensuring (a) and (b) above. Sufficient independence is provided between redundant load groups to ensure that postulated single failures affect only a single load group and are limited to the extent of total loss of that load group. The redundant load groups remain intact to provide for the measures specified in (a) and (b) above.

During a loss of offsite power, the Class IE system is automatically isolated from the offsite power system. This minimizes the probability of losing electric power from the onsite power supplies as a result of the loss of power from the transmission system.

Protection, such as voltage restraint overcurrent and 4.16 kV bus differential relays, is provided to trip the diesel generator circuit breaker, if abnormal conditions occur. This protection prevents damage to or shutdown of the diesel generator.

The turbine generator is automatically isolated from the switchyard following a turbine or reactor trip. Therefore, its loss does not affect the ability of either the transmission network or the onsite power supplies to provide power to the

Class IE system. Transmission system stability studies indicate that the trip of the most critical fully loaded generating unit does not impair the ability of the system to supply plant station service. Further discussion is provided in Subsection 8.2.2.

GENERAL DESIGN CRITERION 18, INSPECTION AND TESTING OF ELECTRICAL POWER SYSTEMS

The Class IE system is designed to permit:

a) Periodic inspection and testing, during equipment shutdown, of wiring, insulation, connections, and relays to assess the continuity of the systems and the condition of components

b) During normal plant operation, periodic testing of the operability and functional performance of onsite power supplies, circuit breakers and associated control circuits, relays, and buses

c) During plant shutdown, testing of the operability of the Class IE system as a whole, including the system's operational sequence, operation of signals of the engineered safety features actuation system and the transfer of power between the offsite and the onsite power system.

8.3.1.11.2 Safety Related Equipment Exposed to Accident Environment

The detailed information on all Class IE equipment that must operate in an accident environment during and/or subsequent to an accident is furnished in Section 3.11.

8.3.1.11.3 Physical Identification of Safety Related Equipment

Each circuit and raceway is given a unique alphanumeric identification, which distinguishes a circuit or raceway related to a particular voltage, function, channel, or load group. One alpha character of the identification is assigned to a load group on the basis of the following criteria:

SEPARATION GROUP CHANNEL A (Red Color Code) - Class IE instrumentation, controls, and power cables, raceways, and equipment related to Channel A loads, dc subsystem A, 120 V ac instrumentation and control channel A, Division I raceways.

SEPARATION GROUP CHANNEL B (Green Color Code) - Class IE instrumentation, controls, and power cables, raceways, and equipment related to Channel B loads, dc subsystem B, 120 V ac instrumentation and control channel B, Division II raceways.

SEPARATION GROUP CHANNEL C (Orange Color Code) - Class IE instrumentation, controls, and power cables, raceways, and equipment related to Channel C loads, dc subsystem C, 120 V ac instrumentation and control channel C.

SEPARATION GROUP CHANNEL D (Blue Color Code) - Class IE instrumentation, controls, and power cables, raceways, and equipment related to Channel D loads, 120 V ac instrumentation and control channel D.

SEPARATION GROUP N (Black Color Code) - Non-Class IE instrumentation, controls, and power cables, raceways, and related equipment.

SEPARATION GROUP DIVISION I (Red/Brown Color Code) - Class IE instrumentation, control, and power cables.

SEPARATION GROUP DIVISION II (Green/Brown Color Code) - Class IE instrumentation, control, and power cables.

The associated power cables are routed with the separation groups they are associated with. The associated power cables are identified as follows:

a) Red/Brown - associated with separation group channel A or division I.

b) Green/Brown - associated with separation group channel B or division II.

c) Orange/Brown - associated with separation group channel C.

d) Blue/Brown - associated with separation group channel D.

Cable and raceway separation groups are summarized in Table 8.3-10 .

For identification of raceways and Class IE cables refer to Section 3.12.

Design drawings provide distinct identification of Class IE equipment. The applicable separation group or load group designation is also identified.

Electrical component identification is discussed in Subsection 1.8.6.

8.3.1.11.4 Independence of Redundant Systems

Separation Criteria

This subsection establishes the criteria and the bases for preserving the independence of redundant Class IE power systems. (For PGCC see Section 3.12).

Raceway and Cable Routing

Wherever possible, cable trays are arranged from top to bottom, with trays containing the highest voltage cables at the top. A raceway designated for one voltage category of cables contains only those cables. Voltage categories are:

a) 480 V ac, 120 V ac, 125 V dc and 250 V dc power | 12

b) 120 V ac, 125 V dc, and 250 V dc control and digital signal

c) Low level signal.

The 480 VAC power, 120 VAC control, and digital alarm signal cables originated from the same 480 VAC motor control center (MCC) are routed through a common shuttle tray and riser above the MCC. The shuttle tray covers the length of the MCC, and it is used to connect the MCC to the main raceway system via vertical tray risers. The cables are routed in accordance with the above raceway categories once they leave the shuttle tray and vertical tray risers. | 22 | 19

15 kV and 5 kV class cables are routed in conduits only.

Cables corresponding with each separation group, as defined in Subsection 8.3.1.3, are run in separate conduits, cable trays, ducts, and penetrations.

Refer to Subsection 3.12.3.4.2 for description of physical separation of raceway and cable routing. | 19

8.3.1.11.5 Administrative Responsibilities and Controls for Ensuring Separation Criteria

The separation group identification described in Subsection 8.3.1.11.3 facilitates and ensures the maintenance of separation in the routing of cables and the connections. At the time of the cable routing assignment during design, those persons responsible

for cable and raceway scheduling ensure that the separation group designation on the scheme to be routed is compatible with a single-line-diagram load group designation and other schemes previously routed. Extensive use of computer facilities assists in ensuring separation correctness. Each cable and raceway is identified in the computer program, and the identification includes the applicable separation group designation. Auxiliary programs are made available specifically to ensure that cables of a particular separation group are routed through the appropriate raceways. The routing is also confirmed by quality control personnel during installation to be consistent with the design document. Color identification of equipment and cabling (discussed in Subsection 8.3.1.11.3 and Section 3.12) assists field personnel in this effort.

8.3.2 DC POWER SYSTEMS

8.3.2.1 Description

The dc power systems are divided into Class IE and non-Class IE systems.

8.3.2.1.1 Class IE dc Power System

The Class IE dc system is shown on Figures 8.3-5 and 8.3-6 . The dc system for each generating unit consists of four 125 V dc subsystems, two 250 V dc subsystems, and two ±24 V dc subsystems.

8.3.2.1.1.1 125 V dc Subsystems

Four Class IE 125 V dc power subsystems provided for each unit are located in separate rooms in the control structure. These four subsystems are identified as channels A, B, C, and D. Each subsystem provides the control power for its associated Class IE ac power load group channel: 4.16 kV switchgear, 480 V load centers, and standby diesel generator as dicussed in Subsection 8.3.1. Also these dc subsystems provide dc power to the engineered safety feature valve actuation, diesel generator auxiliaries, plant alarm and indication circuits, and emergency lighting system.

Each 125 V dc subsystem consists of one load center, one Class IE and one non-Class IE distribution panels, one 125 V battery bank, and one battery charger. The non-Class IE distribution panel is connected to the Class IE dc power supply through an isolation

system. The isolation system is defined in Subsection 8.1.6.1. The battery charger of each system is supplied with 480 V Class IE ac power from the motor control center associated with the same load group channel. One spare 125 V battery charger is provided for both generating units.

The charger output voltage can be regulated at two different control points. One is a variable resistor located inside the cabinet and is used for rough voltage settings. The other is a screwdriver adjusted potentiometer located on the front of the cabinet, and is used for fine adjustments. By setting both 12

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controls at their maximum positions, the charger output voltage would be 145.2 volts. All equipment or devices connected to the 125 V DC supply are rated 105 V to 144 V DC. Maximum output voltage resulting from a failure of charger voltage control circuit is not available at the present time. 12

There are no overvoltage protection devices provided for the 125 vdc subsystem. The 125 V dc power is distributed through circuit breaker type distribution panels. The 125 V dc loads are shown in Table 8.3-6.

The failure mode and effect analysis for the 125 Vdc susbystem is shown in Table 8.3-21. 15

8.3.2.1.1.2 250 V dc Subsystems

Two Class IE 250 V dc subsystems are provided for each unit and identified as Divisions I and II as shown on Figure 8.3-5. The 250 V dc subsystems are located in separate rooms in the control structure. The two subsystems supply the dc power required for larger loads such as dc motor driven pumps and valves, inverters for plant computer and vital 120 V ac power supplies. The 250 V dc loads are shown in Table 8.3-7.

A 2,000 amp fuse is provided at each pole of the 250 Vdc battery output for short circuit protection. These fuses are also used to disconnect the load center from the battery during battery discharge and service tests. 15

The Division I 250 V dc subsystem is provided with one 250 V battery bank, one load center, two equal capacity chargers, and motor control centers. The Division II 250 V dc subsystem is provided with one 250 V battery bank, one distribution load center, one battery charger, and motor control centers.

The 250 V dc battery chargers are supplied by 480 V Class IE ac motor control centers. 12

One spare 250 V battery charger is provided for both generating units.

There is no load shedding provided for any of these non-Class 1E loads.

All 250 Vdc motor control centers (MCC), including non-Class 1E, are seismically qualified. However, the Class 1E MCC's are located in a seismic Category I structure while the non-Class 1E MCC's are located in a non-seismic Category I structure (Turbine Building). 15

The charger output voltage can be regulated at two different control points. One is a variable resistor located inside the cabinet and is used for rough voltage settings. The other is a screwdriver adjusted potentiometer located on the front of the cabinet, and is used for fine adjustments. By setting both
12 controls at their maximum positions, the charger output voltage would be 290.4 volts. All equipment or devices connected to the 250 V DC supply are rated 210 V to 288 VDC. Maximum output voltage resulting from a failure of charger voltage control circuit is not available at the present time. There are no overvoltage protective devices provided for the 250 V DC subsystem.

The 250 V dc power is distributed through dc motor control centers except the inverters, which are fed directly from the distribution load centers.

The non-Class IE 250 V dc loads are supplied by a non-Class IE dc motor control center. The non-Class IE dc motor control center is connected to the Class IE dc distribution load center through an isolation system as defined in Subsection 8.1.6.1(n). The
15 non-Class IE 250 V dc loads consist mainly of emergency turbine generator auxiliaries.

15 The failure mode and effect analysis for the 250 Vdc subsystem is shown in Table 8.3-22.

8.3.2.1.1.3 ±24 V dc Subsystems

Two ± 24 V dc subsystems are provided for each unit for radiation monitoring circuits. These two subsystems are located in separate rooms in the control structure and are identified as Divisions I and II. Each ±24 V dc subsystem consists of two 24 V battery banks, two chargers, and a circuit breaker type distribution panel.

The 24 V dc chargers are supplied by 120 V Class IE instrument ac power panels. The ±24 V dc loads are shown in Table 8.3-8.

One spare 24 V dc battery charger is provided for both generating units.

The 24 vdc subsystem is equipped with under/overvoltage relays
12 for tripping of the chargers and annunciation. All 24 V dc equipment and devices in Susquehanna SES are rated for 20 to 28 vdc.

8.3.2.1.1.4 Class IE Station Batteries and Battery Chargers

Refer to Subsection 8.3.2.1.1.5 for all Class IE dc system equipment ratings.

The battery chargers are full wave, silicon controlled rectifiers. The housings are freestanding, NEMA Type I, and are ventilated. The chargers are suitable for equalizing the batteries. The chargers are in compliance with all applicable NEMA and ANSI standards. 15

The capacity of each battery charger, or the combined capacity of both chargers in the case of Division I 250 V dc subsystem, is based on the largest combined demand of all the steady-state loads and the charger current required to restore the battery from the design minimum charged state to the fully charged state within 12 hr.

The battery chargers are constant voltage type with capability of operating as battery eliminators, and would function properly with battery disconnection being a normal condition. The battery eliminator feature is incorporated as a precautional measure to protect against inadvertant disconnection of the battery. There is no planned modes of operation which would require battery disconnection. Variation of the charger output voltage has been determined by testing to be less than 1% with or without the battery connected. Maximum output ripple for the 24 V and 125 V dc chargers is 30 millivolts RMS with or without the battery, and 200 millivolts for the 250 V chargers. 12

The failure mode and effect analysis for the ± 24 Vdc subsystem is shown in Table 8.3-23. 15

Each 125 V, 250 V, and ±24 V battery bank has sufficient capacity without its charger to independently supply the required loads for 4 hr as shown in Tables 8.3-6, 8.3-7, and 8.3-8 respectively.

In accordance with IEEE 450-1972 initial rated battery capacity is 25 percent greater than required. This margin allows replacement of the battery to be made when its capacity has decreased to 80 percent of its rated capacity (100 percent of design load).

8.3.2.1.1.5 Class 1E DC System Equipment Ratings

a) 125 V dc Subsystems

Battery	60 lead-calcium cells 720 amp-hr (8 hrs to

	1.75 V per cell @ 77°F)
Charger	ac input - 480 V, 3Ø dc output - 100 A continuous rating
Load Center	
Main bus (horizontal)	1600 A continuous rating, 25,000 A short circuit bracing
Vertical bus	1200 A continuous rating, 25,000 A short circuit bracing
Breakers	600 A frame size, 2 poles 25,000 A interrupting rating
Distribution Panel	
Main bus	225 A continuous rating, 50,000 A short circuit bracing
Breakers (molded case)	100 A frame size, 2 poles 10,000 A interrupting rating

b) 250 V dc Subsystems

Battery	120 lead - calcium cells 1800 amp-hr (8 hrs to 1.75 V per cell @ 77°F)
Chargers	ac input - 480 V, 3Ø dc output - 300 A continuous
Load Center	
Main bus (horizontal)	1600 A continuous rating 25,000 A short circuit bracing
Vertical bus	1,200 A continuous rating 25,000 A short circuit bracing
Breakers	600 A continuous rating 25,000 A interrupting rating
Control Center	
Main bus (horizontal)	600 A continuous rating 10,000 A short circuit bracing
Vertical bus	600 A continuous rating 10,000 A short circuit bracing

Breakers (molded case)	100 A, 225 A and 600 A frame rating sizes, 2 poles, 10,000 A interrupting

c) ±24 Volt Subsystems

Battery	2 groups of 12 lead-calcium cells. 75 amp-hr (8 hrs to 1.75 V per cell @ 77°F)
Chargers	ac input - 120 V, 1Ø dc output - 25 amp continuous
Distribution Panels	
Main bus	100 A continuous 5,000 A short circuit bracing
Breakers (molded case)	100 A frame size, 2 poles, 5,000 A interrupting rating

12

8.3.2.1.1.6 Inspection, Maintenance, and Testing

Testing of the dc power systems are performed prior to plant operation in accordance with the requirements of Chapter 14.

In-service tests and inspections of the dc power systems including batteries, chargers, and auxiliaries are specified in Chapter 16.

8.3.2.1.1.7 Separation and Ventilation

For each Class IE dc subsystem, the battery bank, chargers, and dc switchgear are located in separate rooms of the Seismic Category I control structure. The battery rooms are ventilated by a system that is designed to preclude the possibility of hydrogen accumulation. Section 9.4 contains a description of the battery room ventilation system.

8.3.2.1.1.8 Non-Class IE dc System

Generally, non-Class IE dc loads are connected to a Class IE dc system through a non-Class IE dc distribution panel. These cases are discussed in Subsections 8.3.2.1.1.1 and 8.3.2.1.1.2.

A non-Class IE 125 V dc system is provided for the remote river water intake pump house 4.16 kV switchgear control. This 125 V dc system consists of a distribution panel, two 25A chargers, 60 lead-calcium cells and is rated 50 Ah at 8 hr discharge rate based on a terminal voltage of 1.75 V per cell when discharged.

8.3.2.2 Analysis

8.3.2.2.1 Compliance with General Design Criteria, Regulatory Guides, and IEEE Standards

The following paragraphs analyze compliance of the Class IE dc power systems with General Design Criteria 17 and 18, Regulatory Guides 1.6, 1.32, 1.41, 1.81, and 1.93, and IEEE 308-1974 and 450-1972.

a) General Design Criterion 17, Electric Power Systems

Consideration of Criterion 17 leads to the inclusion of the following factors in the design of the dc power systems:

1) Separate Class IE 125 V dc subsystems supply control power for each of the Class IE ac load groups.

2) The ac power for the battery chargers in each of these dc subsystems is supplied from the same ac load group for which the dc subsystem supplies the control power.

3) Two independent 250 V dc subsystems are provided to ensure the availability of the dc power system for maintaining the reactor integrity during postulated accidents.

4) The Class IE dc subsystems including batteries, chargers, dc switchgear, and distribution equipment are physically separate and independent.

5) Sufficient capacity, capability, independence, redundancy, and testability are provided in the Class IE dc subsystems, ensuring the performance of safety functions assuming a single failure.

b) General Design Criterion 18, Inspection and Testing of Electric Power Systems

Each of the Class IE subsystem is designed to permit:

1) Inspection and testing of wiring, insulation, and connections during equipment shutdown to assess the continuity of the subsystem and the condition of its components.

2) Periodic testing of the operability and functional performance of the components of the subsystems during normal plant operation.

The Class IE dc subsystems are periodically inspected and tested to assess the condition of the battery cells, charger, and other components in accordance with Chapter 16. Preoperational testing is discussed below in assessment of compliance with Regulatory Guide 1.41.

c) Regulatory Guide 1.6 (1/71)

The design of the dc system complies with Regulatory Guide 1.6.

Separate Class IE 125 V dc subsystems supply control power for each of the four Class IE load groups. Loss of any one of the subsystems does not prevent the minimum safety function from being performed. The 125 V dc subsystem chargers are supplied from the same ac load group for which the dc subsystem supplies the control power. Each of the four 125 V dc subsystems, including battery bank, charger, and distribution system, is independent of other 125 V dc subsystems. Thus, sufficient independence and redundancy exist between the 125 V dc subsystems to ensure performance of minimum safety functions, assuming a single failure.

Two independent Class IE 250 V dc subsystems are provided. Each subsystem is independent of the other. Sufficient independence and redundancy exist in these subsystems so that a single failure in the 250 V dc subsystems does not prevent the performance of minimum safety functions.

Two independent Class IE ±24 V dc subsystems are provided. Each subsystem is independent of the other. Sufficient independence and redundancy exist in these subsystems so that a single failure in the ±24 V dc subsystems does not prevent the performance of minimum safety functions.

d) Regulatory Guide 1.32 (8/72)

The battery charger capacity for each of the Class IE dc subsystems complies with this Regulatory Guide.

Each Class IE battery charger has sufficient capacity to supply the largest combined demand of the various steady-state loads and the charging current required to restore the battery from the design minimum charge state to the fully charged state irrespective of the status of the plant during which these demands occur.

e) Regulatory Guide 1.41(3/73)

The Class IE dc subsystems have been designed in accordance with Regulatory Guides 1.6 and 1.32 and testing capabilities are provided in accordance with the guidance of Regulatory Guide 1.41 and will be preoperationally tested as described in Chapter 14.

f) Regulatory Guide 1.81 (1/75)

The requirements of the Regulatory Guide are met. Each generating unit is provided with separate and independent onsite dc electric power systems capable of supplying power to the control systems of engineered safety features loads and loads such as valves, and actuators, required for attaining a safe and orderly cold shutdown of the unit, assuming a single failure.

g) Regulatory Guide 1.93 (12/74)

Compliance is discussed in Subsection 8.1.6.1 (q).

h) IEEE Standard 308-1974

The Class IE dc systems provide power to Class IE loads and for control and switching of Class IE systems. Physical separation and electrical isolation are provided to prevent the occurrence of common mode failures. The design of the Class IE dc systems includes the following:

1) The 125 V dc system is separated into four subsystems

2) The 250 V dc and ± 24 V dc systems are each separated into two subsystems

3) The safety action by each group of loads are independent of the safety actions provided by their redundant counterparts

4) Each dc subsystem includes power supplies that consist of one battery bank and one or two chargers as required for capacity as shown on Figures 8.3-5 and 8.3-6.

5) The batteries are not interconnected.

Each Class IE distribution circuit is capable of transmitting sufficient energy to start and operate all required loads in that circuit. Distribution circuits to redundant equipment are independent of each other. The distribution system is monitored to the extent that it is shown to be ready to perform its intended function. The dc auxiliary devices required to operate equipment of a specific ac load group are supplied from the same load group.

Each battery supply is continuously available during normal operations and following the loss of power from the ac system to start and operate all required loads.

The 125 V dc and 250 V dc subsystems are ungrounded; thus, a single ground fault does not cause immediate loss of the faulted system. Ground detection and alarm is provided for each dc subsystem so that ground faults can be located and removed. The ±24 V dc subsystem is grounded.

Equipment of the Class IE dc system is protected and isolated by fuses or circuit breakers for short circuit or overload protection. The following instrumentation is provided to monitor the status of each of the dc subsystems:

1) 125 V dc and 250 V dc subsystems:

System undervoltage

System ground

Battery availability

Battery charger trouble - ac undervoltage; charger failure; charger output breaker trip

Load center breaker trip (250 V dc subsystem only)

All above alarms are annunciated as a group alarm in the main control room.

2) ± 24 V dc subsystems:

Positive bus low voltage

Negative bus low voltage

Positive bus high voltage

Negative bus high voltage

Battery availability

Battery charger trouble - ac failure; charger failure; charger output breaker trip

All above alarms are annunciated in the main control room as ± 24 V dc system trouble, a group alarm for each battery bank and its associated system.

The batteries are maintained in a fully charged condition and have sufficient stored energy to operate all necessary circuit breakers and to provide an adequate amount of energy for all required emergency loads for four hours after loss of ac power.

Each battery charger has an input ac and output dc circuit breaker for isolation of the charger. Each battery charger power supply is designed to prevent the ac supply from becoming a load on the battery due to a power feedback as the result of the loss of ac power to the chargers.

The battery charger ac supply breaker can be periodically opened to verify the load carrying ability of the battery.

The batteries, battery chargers, and other components of the dc subsystems are housed in the control structure, which is a Seismic Category I structure.

The periodic testing and surveillance requirements for the Class IE batteries are detailed in Chapter 16.

i) IEEE Standard 450-1972

The recommended practices of IEEE 450 for maintenance, testing, and replacement of batteries are followed for the Class IE batteries and are discussed in Chapter 16.

8.3.2.2.2 Physical Identification of Safety Related Equipment

Physical identification of Class IE equipment is discussed in Subsection 8.3.1.3.

8.3.2.2.3 Independence of Redundant Systems

The general considerations for the independence of Class IE dc power subsystems are described in Subsection 8.1.6.1(n). The physical separation criterion is discussed in Section 3.12.

8.3.3 FIRE PROTECTION FOR CABLE SYSTEMS

8.3.3.1 Cable Derating and Cable Tray Fill

The power and control cable insulation is designed for a conductor temperature of 90°C. Allowable current carrying capacity of the cable is based on not exceeding the insulation design temperature while the surrounding air is at an ambient temperature of 65.5°C for the primary containment and 40°C for all other areas. The design operating conditions of all Class IE cables are discussed in Section 3.11.

The power cable ampacities are established in accordance with IPCEA Publications P-54-440 and P-46-426 and are shown in Tables 8.3-11 through 8.3-15.

For control circuits, minimum #14 AWG conductors are generally used.

Instrumentation cable is also designed for a conductor temperature of 90°C. Operating currents of these cables are low (usually mA or mV) and will not cause the design temperature to be exceeded at maximum design ambient temperature.

In general, cable tray fill is limited to 30 percent fill by cross-sectional area. In cases where the limitation is exceeded, a review will be performed for each case for the adequacy of the design.

Conduit fill is in compliance with Tables I and II, Chapter 9, National Electrical Code, 1975.

Power cables, control cables, and instrumentation cables are defined as follows:

Power Cables

Power cables are those cables that provide electrical energy for motive power or heating to all 13.8 kV ac, 4.16 kV ac, 480 V ac, 120 V ac, 250 V dc, and 125 V dc loads.

Control Cables

Control cables, for the purpose of derating, are generally 120 V ac, 250 V dc, 125 V dc, and 24 V dc circuits between components responsible for the automatic or manual initiation of auxiliary electrical functions and the electrical indication of the state of auxiliary components.

Instrumentation Cables

Instrumentation cables are those cables conducting low-level instrumentation and control signals. These signals can be analog or digital. Typically, these cables carry signals from thermocouples, resistance temperature detectors, transducers, neutron monitors, etc.

8.3.3.2 Fire Detection for Cable Systems

Fire detection systems are discussed in Subsection 9.5.1.

8.3.3.3 Fire Barriers and Separation Between Redundant Trays

Electrical equipment and cabling has been arranged to minimize the propagation of fire from one separation group to another. Physical separation of cabling systems is discussed in Subsection 3.12.2.

Where the minimum physical separation cannot be met as specified in Subsection 3.12.2, and a fire barrier is selected as the alternative, a 1/4 in. Haysite ETR-FR-C is installed. The bolts and hardware used to secure the Haysite panel to the tray support are coated after installation with 1/8 in. of fireproofing material Dynatherm's Flamemastic 71A compound.

8.3.3.4 Fire Stops

Fire stops and seals are provided for cable penetrations in the floor for vertical runs of raceways, at each access opening in ceilings and at fire-rated wall penetrations. The fire stops are furnished to provide a method of sealing off air spaces around cable penetrations. The properties of materials and qualification tests are discussed in Subsection 9.5.1.

TABLE 8.3-1

ASSIGNMENT OF ESF AND SELECTED NON-ESF
LOADS TO DIESEL GENERATORS AND DIESEL RATINGS

Equipment #	Description	Rating Each, hp	Operating kW Each	Number Connected: Diesel Gen A Unit 1 Bus	Diesel Gen A Unit 2 Bus	Diesel Gen C Unit 1 Bus	Diesel Gen C Unit 2 Bus	Diesel Gen B Unit 1 Bus	Diesel Gen B Unit 2 Bus	Diesel Gen D Unit 1 Bus	Diesel Gen D Unit 2 Bus	Loading Sequence (3) (Note 3): Unit 1 - DBA Required Number	Unit 1 - DBA Time From DBA	Unit 2-Shutdown Required Number	Unit 2-Shutdown Time
	ESF Loads														
1P 206 A,B,C,D,	Reactor Core Spray Pumps	700	555	1	1	1	1	1	1	1	1	3	20 sec	-	-
1P 202 A,B,C,D	RHR Pumps	2000	1425	1	1	1	1	1	1	1	1	3	10 sec	1	30 min*
1P 506 A,B	RHR Service Water Pumps	600	460	-	1	1	-	-	1	1	-	1	10 min*	1	31 min*
1V 211 A,B,C,D	Core Spray Pump Room Unit Coolers	2	1.7	1	1	1	1	1	1	1	1	3	20 sec	-	-
	Motor Operated Valves (Note 1)	Set										Set	10 sec	Set	10 sec
1V 222 A,B	Engineered Safeguards Switchgear & L.C. Room Unit Coolers	15	12	1	1	-	-	1	1	-	-	2	70 sec	1	70 sec
OV116 A,B,	Control Structure Battery Room Exhaust Fans.	5	4	-	-	1	-	-	-	1	-	1	70 sec	-	-
1V 210 A,B,C,D	RHR Pump Room Unit Coolers	10	8	1	1	1	1	1	1	1	1	3	10 sec	1	30 min
1V 208 A,B	RCIC Pump Room Unit Coolers	1.5	1.2	1	1	1	1	-	-	-	-	1	60 sec	1	60 sec

*Manual Initiation

TABLE 8.3-1 (cont'd)

Equipment #	Description	Rating Each, hp	Operating kW Each	Number Connected: Diesel Gen A Unit 1 Bus	Diesel Gen A Unit 2 Bus	Diesel Gen C Unit 1 Bus	Diesel Gen C Unit 2 Bus	Diesel Gen B Unit 1 Bus	Diesel Gen B Unit 2 Bus	Diesel Gen D Unit 1 Bus	Diesel Gen D Unit 2 Bus	Loading Sequence (3) (Note 3): Unit 1 - DBA Required Number	Unit 1 - DBA Time From DBA	Unit 2-Shutdown Required Number	Unit 2-Shutdown Time
	ESF Loads (con't)														
1V 209 A,B	HPCI Pump Room Unit Coolers	1.5	1.2	-	-	-	-	1	1	1	1	1	60 sec	-	-
1V 613 to 643	Battery Chargers, 125V D.C.	-	25	1	1	1	1	1	1	1	1	3	10 sec	3	10 sec
OV 512 A,B,C,D	Diesel Generator Room Ventilation Supply Fans	40	32	1	-	1	-	1	-	1	-	3	10 sec	-	-
OP 514 A,B,C,D	Diesel Generator Diesel Oil Transfer Pumps	-	2.5	1	-	1	-	1	-	1	-	3	60 min & beyond	-	-
OV 201 A,B	Reactor Building Recirc Fans	75	60	1	-	-	-	1	-	-	-	1	10 sec	-	-
OP 504 A,B,C,D	Emergency Service Water Pumps	450	360	1	-	1	-	1	-	1	-	2	65 sec	-	-
OV 109 A,B	Standby Gas Treatment System Exhaust Fans	50	40	-	-	1	-	-	-	1	-	1	10 sec	-	-
OV 115 A,B & OV 117 A,B	Control and Computer Rooms Air Cond. Unit Fans.	40	32	-	-	2	-	-	-	2	-	2	70 sec	-	-
OK 507 A,B,C,D	Diesel Generator Starting Air Compressors.	10	8	2	-	2	-	2	-	2	-	6	10 sec	-	-

TABLE 8.3-1 (cont'd)

Equipment #	Description	Rating Each, hp	Operating kW Each	Number Connected: Diesel Gen A Unit 1 Bus	Diesel Gen A Unit 2 Bus	Diesel Gen C Unit 1 Bus	Diesel Gen C Unit 2 Bus	Diesel Gen B Unit 1 Bus	Diesel Gen B Unit 2 Bus	Diesel Gen D Unit 1 Bus	Diesel Gen D Unit 2 Bus	Loading Sequence (3) (Note 3): Unit 1 - DBA Required Number	Unit 1 - DBA Time From DBA	Unit 2-Shutdown Required Number	Unit 2-Shutdown Time
	ESF Loads (con't)														
1V 216 to 246	120 V Instrument A.C. Dist. Panels	-	25	1	1	1	1	1	1	1	1	3	10 sec	3	10 sec
OV 521 A,B,C,D	Engineered Safeguards Service Water Pump House Ventilation Fans (ESWP)	5	4	2	-	-	-	2	-	-	-	2	65 sec	-	-
OP 162 A,B	Control Structure Chilled Water Circulating Pumps	30	24	-	-	1	-	-	-	1	-	1	2 min	-	-
OV 101 A,B	Control Structure Emergency Outside Air Supply Fans	20	16	-	-	1	-	-	-	1	-	1	70 sec	-	-
OK 112 A,B	Control Structure Water Chiller Compressors	351	306	-	-	1	-	-	-	1	-	1	3 min	-	-
OE 145 A,B	Control Structure Air Cond. Unit Heating Coils	-	130	-	-	1	-	-	-	1	-	1	70 sec	-	-
OV 118 A,B	Standby Gas Treatment System Equip. Room Exhaust Fans	5	4	-	-	1	-	-	-	1	-	1	10 sec	-	-
1E 219/1E 220	Standby Liquid Cont. Tank Heater	-	40	-	-	1	1	-	-	-	-	1	55 sec	1	55 sec.
1P 208 A,B	Standby Liquid Cont. Pumps	40	32	1	-	1	-	-	1	-	1	-	-	-	-

TABLE 8.3-1 (cont'd)

Equipment #	Description	Rating Each, hp	Operating kW Each	Number Connected: Diesel Gen A Unit 1 Bus	Diesel Gen A Unit 2 Bus	Diesel Gen C Unit 1 Bus	Diesel Gen C Unit 2 Bus	Diesel Gen B Unit 1 Bus	Diesel Gen B Unit 2 Bus	Diesel Gen D Unit 1 Bus	Diesel Gen D Unit 2 Bus	Loading Sequence (3) (Note 3): Unit 1 - DBA Required Number	Unit 1 - DBA Time From DBA	Unit 2-Shutdown Required Number	Unit 2-Shutdown Time
	ESF Loads (con't)														
1D 653 A,B 1D 663	Battery Chargers - 250V D.C.	-	75	1	1	1	1	1	1	-	-	1	10 sec	1	13 sec
OV 144 A,B	Standby Gas Treatment System Equip. Room Heating Unit Fans	5	4	-	-	1	-	-	-	1	-	1	10 sec	-	-
OV 103 A,B	Control Structure Air Cond. Unit Fans	50	40	-	-	1	-	-	-	1	-	1	70 sec	-	-
OP 122 A,B	Control Structure Chiller Comp Oil Pumps	1.5	1.2	-	-	1	-	-	-	1	-	1	60 sec	-	-
OC 866 A,B,	Standby Gas Treatment System Heaters	-	90	-	-	1	-	-	-	1	-	1	10 sec	-	-
OP 171 A,B	Control Structure Chiller Condenser Water Circ Pumps	20	16	-	-	1	-	-	-	1	-	1	60 sec	-	-
1E 440 A,B,C,D	Containment Hydrogen Recombiners	-	75	1	1	1	1	1	1	1	1	2	61 min	-	-
OE 143 A,B	Control Structure Emergency Outside Air Supply Unit Heating Coils	-	30	-	-	1	-	-	-	1	-	1	60 sec	-	-
1V 506 A,B	Engineered Safeguards Service Water Pump House Ventilation System (RHRSWP)	5	4	1	1	-	-	1	1	-	-	1	70 sec	1	70 sec.

TABLE 8.3-10

ROUTING TABLE

Cable Separation Group	Raceway Separation Group						
Cables Permitted in Selected Raceways	Non-Class IE	Div I	Div II	Chan A	Chan B	Chan C	Chan D
Non-Class IE	Yes	No	No	No	No	No	No
Div I	No	Yes	No	Yes	No	No	No
Div I Associated	No	Yes	No	Yes	No	No	No
Div II	No	No	Yes	No	Yes	No	No
Div II Associated	No	No	Yes	No	Yes	No	No
Chan. A	No	Yes	No	Yes	No	No	No
Chan. A Associated	No	Yes	No	Yes	No	No	No
Chan. B	No	No	Yes	No	Yes	No	No
Chan. B Associated	No	No	Yes	No	Yes	No	No
Chan. C	No	No	No	No	No	Yes	No
Chan. C Associated	No	No	No	No	No	Yes	No
Chan. D	No	No	No	No	No	No	Yes
Chan. D Associated	No	No	No	No	No	No	Yes

	RPS A1	RPS A2	RPS B1	RPS B2
RPS A1	Yes	No	No	No
RPS A2	No	Yes	No	No
RPS B1	No	No	Yes	No
RPS B2	No	No	No	Yes

Note: To determine raceways in which cable may be routed, read across from selected cable until "yes" appears. Column heading is raceway required.

TABLE 8.3-11

CABLE AMPACITIES - 15 kV CABLES (ALUMINUM)

Conductor Size	Amps in Duct and Embedded Conduit 40°C Ambient	Amps in Conduit in Air 40°C Ambient	Amps in Conduit in Air 65.5°C Ambient
3-1/c *	1/c	1/c	1/c
#4/0 AWG	195	232	162
350 KCMIL	248	306	214
500 KCMIL	301	380	266
750 KCMIL	372	469	328
1000 KCMIL	428	552	386
6-1/c	2/c	2/c	2/c
#4/0 AWG	353	436	305
350 KCMIL	450	575	403
500 KCMIL	541	714	500
750 KCMIL	666	882	617
1000 KCMIL	764	1038	727
9-1/c	3/c	3/c	3/c
#4/0 AWG	472	633	443
350 KCMIL	601	835	585
500 KCMIL	720	1037	726
750 KCMIL	882	1280	896
1000 KCMIL	1006	1507	1055

* 3-1/c indicates single conductor per phase, and 6-1/c indicates two conductors per phase, etc.

TABLE 8.3-12

CABLE AMPACITIES - 5 kV CABLES (ALUMINUM)

Conductor Size	Amps in Duct and Embedded Conduit 40°C Ambient	Amps in Conduit in Air 40°C Ambient		65.5°C Ambient
3-1/c	1/c		1/c	1/c
#4/0 AWG	194	226	158	
350 KCMIL		248	298	209
500 KCMIL		301	368	258
750 KCMIL		373	473	331
1000 KCMIL		429	542	391
6-1/c	2/c	2/c	2/c	
#4/0 AWG	351	425	298	
350 KCMIL		449	560	392
500 KCMIL		543	692	484
750 KCMIL		669	889	622
1000 KCMIL		768	1019	713
9-1/c	3/c	3/c	3/c	
#4/0 AWG	472	617	432	
350 KCMIL		603	814	570
500 KCMIL		728	1005	704
750 KCMIL		890	1291	904
1000 KCMIL		1017	1480	1036

TABLE 8.3-13

CABLE AMPACITIES IN TRAY - 600 V CABLES

Material	Conductor Size	Amps 40°C Ambient 1-3/c	Amps 40°C Ambient 3-1/c	Amps 65.5°C Ambient 1-3/c	Amps 65.5°C Ambient 3-1/c
Copper	#10 AWG	21	13	15	9
	# 8 AWG	36	24	25	17
	# 6 AWG	52	33	36	23
	# 2 AWG	104	69	73	48
Aluminum	#4/0 AWG	-	165	-	116
	350 KCMIL	-	274	-	192
	500 KCMIL	-	371	-	260
	750 KCMIL	-	528	-	370
	CONDUCTOR: 6-1/c (2-1/c Per Phase)				
	350 KCMIL	-	548	-	384
	500 KCMIL	-	742	-	520
	750 KCMIL	-	1056	-	740
	CONDUCTOR: 9-1/c (3-1/c Per Phase)				
	350 KCMIL	-	822	-	576
	500 KCMIL	-	1113	-	780
	750 KCMIL	-	1584	-	1110

TABLE 8.3-14

CABLE AMPACITIES IN DUCT OR EMBEDDED CONDUIT
600 V CABLES

Material	Conductor Size	Amps 40°C Ambient	Amps 65.5°C Ambient
	CONDUCTOR:	1-3/c or 3-1/c (One Conduit)	
Copper	#10 AWG	- 36	25
	#8 AWG	- 50	35
	#6 AWG	- 66	46
	#2 AWG	- 112	79
Aluminum	#4/0 AWG	- 174	122
	350 KCMIL	- 235	165
	500 KCMIL	- 287	201
	750 KCMIL	- 360	252
	CONDUCTOR:	6-1/c (2-1/c Per Phase) (Two Conduit)	
	350 KCMIL	- 430	301
	500 KCMIL	- 524	367
	750 KCMIL	- 653	458
	CONDUCTOR:	9-1/c (3-1/c Per Phase) (Three Conduits)	
	350 KCMIL	- 586	410
	500 KCMIL	- 710	497
	750 KCMIL	- 880	616

TABLE 8.3-15

CABLE AMPACITIES IN CONDUIT IN AIR
600 V CABLES

Material	Conductor Size	Amps 40°C Ambient 1-3/c	Amps 40°C Ambient 3-1/c	Amps 65.5°C Ambient 1-3/c	Amps 65.5°C Ambient 3-1/c
Copper	#10 AWG	36	36	25	25
	#8 AWG	52	52	36	36
	#6 AWG	69	69	48	48
	#2 AWG	123	123	86	86
Aluminum	#4/0 AWG	-	200	-	140
	350 KCMIL	-	274	-	192
	500 KCMIL	-	341	-	239
	750 KCMIL	-	432	-	302

CHAPTER 9.0

AUXILIARY SYSTEMS

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9.1 FUEL STORAGE AND HANDLING

9.1.1 NEW FUEL STORAGE

9.1.1.1 Design Bases

9.1.1.1.1 Safety Design Bases

9.1.1.1.1.1 Safety Design Bases - Structural

a) The new fuel storage racks containing a full complement of fuel assemblies are designed to withstand all credible static and dynamic loadings to prevent damage to the structure of the racks, and therefore the contained fuel, and to minimize distortion of the racks arrangement. (See Table 3.9-2(s)).

b) The racks are designed to protect the fuel assemblies from excessive physical damage which may cause the release of radioactive materials in excess of 10CFR20 requirements under normal conditions.

c) The racks are constructed in accordance with the Quality Assurance Requirements of 10CFR50, Appendix B.

d) The new fuel storage racks are categorized as Safety Class 2 and Seismic Category I.

9.1.1.1.1.2 Safety Design Bases - Nuclear

a) The new fuel storage racks are designed and maintained with sufficient spacing between the new fuel assemblies to assure that the array, when racks are fully loaded, shall be subcritical, by at least 5% ΔK including allowance for calculational biases and uncertainties. In the calculations performed to assure that keff $\leq$0.95, the standard lattice methods (Ref 9.1-1) used at General Electric are employed. Under conditions where Diffusion theory is valid, it is used in calculations (i.e., conditions where the fuel is flooded with water at a density of between 0.7 and 1.0 g/cc).

The new fuel storage vault is covered by leak tight, metal, removable covers. The movement of these covers 23

23 will be administratively controlled by approved plant procedures.

It is assumed that the storage array is infinite in all directions. Since no credit is taken for leakage, the values reported as effective neutron multiplication factors are in reality infinite neutron multiplication factors.

The biases between the calculated results and experimental results as well as the uncertainty involved in the calculations are taken into account as part of the calculational procedure to assure that the specified keff limits are met.

9.1.1.1.2 Power Generation Design Bases

a) New fuel storage racks are supplied for 30% of the full core fuel load in each unit.

b) New fuel storage racks are designed and arranged so that the fuel assemblies can be handled efficiently during refueling operations.

9.1.1.2 Facilities Description

The location of the new fuel storage facility within the station complex is shown in Section 1.2. Each new fuel storage rack (Figure 9.1-1) holds up to 10 channeled or unchanneled assemblies in a row. Fuel spacing (7 inches nominal center-to-center within a rack, 12 inches nominal center-to-center between adjacent racks) within the rack and from rack-to-rack will limit the effective multiplication factor of the array (keff) to not more than 0.95. The fuel assemblies are loaded into the rack through the top. Each hole for a fuel assembly has adequate clearance for inserting or withdrawing the assembly channeled or unchanneled. Sufficient guidance is provided to preclude damage to the fuel assemblies. The upper tie plate of the fuel element rests against the rack to provide lateral support. The design of the racks prevents accidental insertion of the fuel assembly in a position not intended for the fuel. This is achieved by abutting the sides of each casting to the adjacently installed casting. In this way, the only spaces in the new fuel racks are those into which it is intended to insert fuel. The weight of the fuel assembly is supported by the lower tie plate which is seated in a chamfered hole in the base casting.

The floor of the new fuel storage vault is sloped to a drain located at the low point. This drain removes any water that may be accidentally and unknowingly introduced into the vault. The drain is part of the liquid radwaste collection system.

The radiation monitoring equipment for the new fuel storage area is described in Subsection 12.3.4.

9.1.1.3 Safety Evaluation

9.1.1.3.1 Criticality Control

The calculations of keff are based upon an infinite geometrical arrangement of the fuel array. The arrangement of fuel assemblies in the fuel storage racks results in keff below 0.95 in a dry condition or completely flooded with water which has a density of 1 g per cc. General Design Criterion 62 requirements (Prevention of Criticality in Fuel Storage and Handling), are met if fuel is stored in the dry condition or if the abnormal condition of flooding (water with a density of 1 g/cc) occurs. New fuel storage vault covers prevent optimum moderation in the new fuel vault. 23

9.1.1.3.2 New Fuel Rack Design

a) The new fuel storage vault contains 23 sets of castings each of which may contain up to 10 fuel assemblies; therefore a maximum of 230 fuel assemblies may be stored in the fuel vault. 23

b) There are three tiers of castings which are positioned by fixed box beams. This holds the fuel assemblies in a vertical position and supported at the lower and upper tie plate with additional lateral support at the center of gravity of the fuel assembly.

c) The lower casting supports the weight of the fuel assembly and restricts the lateral movement; the center and top casting restricts lateral movement only of the fuel assembly.

d) The new fuel storage racks are made from aluminum. Materials used for construction are specified in accordance with ASTM specifications in effect in 1970. The material choice is based on a consideration of the susceptibility of various metal combinations to electrochemical reaction. When considering the

susceptibility of metals to galvanic corrosion, aluminum and stainless steel are relatively close together insofar as their coupled potential is concerned. The use of stainless steel fasteners in aluminum to avoid detrimental galvanic corrosion in a predominantly air environment, is a recommended practice and has been used successfully for many years by the aluminum industry. 23

e) The minimum center-to-center spacing for the fuel assembly between rows is 11.875 inches. The minimum center-to-center spacing within the rows is 6.535 inches. Fuel assembly placement between rows is not possible.

f) Lead-in and lead-out of the casting, in the rack, provides guidance of the fuel assembly during insertion or withdrawal.

g) The rack is designed to withstand the impact force of 4000 ftlbs while maintaining the safety design basis. This impact force could be generated by the vertical free fall of a fuel assembly from the height of 53 feet.

h) The storage rack is designed to withstand the pull-up force of 4000 lbs. and a horizontal force of 1000 lbs. There are no readily available forces in excess of 1000 lbs. 23

i) The storage rack is designed to withstand horizontal combined loads up to 222,000 lbs, well in excess of expected loads.

j) The maximum stress in the fully loaded rack in a faulted condition is 26 Kips. (See Table 3.9-2(s)). This is lower than the allowable stress. 23

k) The fuel storage rack is designed to handle non-irradiated, low emission radioactive fuel assemblies. The expected radiation levels are well below the design levels.

l) The fuel storage rack is designed using non-combustible materials. Plant procedures and inspections assure that combustible materials are restricted from this area. Fire prevention by elimination of combustible materials and fluids is regarded as the prudent approach rather than fire accomodation and the need for fire suppressant materials which could negate criticality control assurances. Therefore, fire accommodation is not considered necessary. 23

m) The new fuel vault covers, which are carbon steel, are illustrated in Figure 9.1-1a. The covers overlap the curb and have a protective lip that prevents direct impingment of water into the vault. The modified I-beams that span the vault provide mechanical support and direct water run-off from the covers. 23

9.1.2 SPENT FUEL STORAGE

9.1.2.1 Design Bases

9.1.2.1.1 Safety Design Bases

9.1.2.1.1.1 Safety Design Bases - Structural

a) The high density spent fuel storage racks containing a storage space sufficient for approximately 372% of one full core of fuel assemblies are designed to withstand all credible static and dynamic loadings to prevent excessive damage to the structure of the racks, and therefore the contained fuel. (See Table 3.9-2(s)).

b) The racks are designed to protect the fuel assemblies from excessive physical damage which may cause the release of radioactive materials in excess of 10CFR20 requirements under normal or abnormal conditions.

c) The racks are constructed in accordance with the Quality Assurance Requirements of 10CFR50, Appendix B.

d) The spent fuel storage racks are categorized as Safety Class 2 and Seismic Category I.

e) The spent fuel pool structure and the anchorage system to the fuel storage racks are categorized as Seismic Category I.

9.1.2.1.1.2 Safety Design Bases - Nuclear

The effective neutron multiplication factor (Keff) of the fuel array in any combination of any stored positions up to and including the fully loaded condition, is less than 0.95. The positioning of the neutron poisoning material (boral) between each adjoining fuel assemblies assures subcriticallity by at least 5% ΔK under all normal and abnormal conditions. 23 Consideration has been given to the geometry of the racks, possible abnormal loading, and the density of the coolant/moderator.

9.1.2.1.2 Power Generation Design Bases

The spent fuel storage pool and fuel storage racks are designed to assure:

a) subcriticality, by at least 5% ΔK

b. decay heat from fuel assemblies/bundles will not adversely affect the fuel, racks, or pool walls.

c) radiation levels will be "As Low As Reasonably Achieveable".

9.1.2.1.3 Storage Capacity Design Bases

Each reactor unit has a spent fuel pool which has high density fuel storage racks providing a maximum storage capacity of 2840 fuel assemblies. These fuel locations also provide storage of used fuel channels, if needed. In addition, the fuel storage rack design provides storage of 10 various reactor internal components, such as:

a. control rods
b. control rod guide tubes
c. defective fuel storage containers
d. "out-of-core" shipping containers

23

This capacity provides each reactor unit storage space for off loading one-quarter(1/4) of a core for approximately ten (10) years, plus one complete core load of fuel.

Each reactor unit's spent fuel pool is interconnected, via a transfer canal. Spent fuel may be transferred safely, through 23 this transfer canal, to the other pool. This capability provides greater flexibility for the stations storage of spent fuel, if the need ever arises.

Each reactor unit's spent fuel pool walls also have storage hangers for one hundred and thirty control rods. These hangers, empty or full of control rods, do not interfere with the storage of fuel or the other mentioned reactor internal components in this section. 23

9.1.2.2 Facilities Description

The location of the spent fuel storage facility within the station complex is shown in Figure 1.2-23 and 1.2-33. The racks are connected to wall embedments on the pool walls and shown in Figure 9.1-2b. Each pool has 24 racks for a storage capacity of 2840 fuel assemblies plus 10 multipurpose cavities for storage of control rods, control rod guide tubes, and defective fuel containers.

The spent fuel liner plate is not a structural element (i.e. it is not load bearing). The fuel racks are attached to the pool walls by embeds and anchors, which are designed for all credible loads (See Table 9.1-7a). The liner plate is welded to these embeds. In addition, the liner plate is attached to the pool walls by a system of stiffeners and anchors. The racks, embeds and fuel pool walls and liner plate (including anchor system) are designed for all credible loads.

A leak detection system is provided for the collection of possible leakage through the pools' liner plate. The liner leakage detection system is segregated into sections that collect leakage at independent locations below the pools. Drainage paths are formed by welded channels behind the liner weld joints and are designed to permit free gravity flow to manual telltale valves.

This system is provided to:

a) Prevent pressure buildup behind the liner plate

b) Prevent the uncontrolled loss of contaminated pool water to other cleaner locations within the secondary containment, and

c) Provide expedient liner leak detection and measurement.

Both Units 1 and 2 share a common cask pit that accepts the spent fuel shipping cask and accommodates underwater fuel transfer to the cask from either unit through its respective transfer canals. Movements of the cask on the refueling floor are restricted as shown on Figures 9.1-16A and 9.1-16B.

The evaluations of the consequences of a postulated accidental drop of a spent fuel assembly and the shipping cask are discussed in Chapter 15. The capability of the spent fuel pool storage facility to prevent missiles generated by high winds from contacting the fuel is discussed in Subsection 3.5.2.

The rack arrangement is designed to prevent accidental insertion of fuel bundles between adjacent racks.

The six (6) foot thick spent fuel pool walls provide radiation shielding to 25 MRem/Hr measured on the outside of the spent fuel pool walls. Normal water shielding over the stored fuel in the racks is approximately 23 feet and is sufficient to provide shielding for required building occupancy. Under the normal water level conditions, 9' of water is above the active fuel when moved through the refueling channel. This depth of water provides shielding to assure less than 2.5 MRem/Hr to the operators on the refueling platform.

Accidental droppage of heavy objects into the fuel pool is precluded by the use of electrical interlocks to limit the reactor building crane travel over the spent fuel pool, and the use of guardrails and curbs around all pools and the reactor wells to prevent fuel handling and servicing equipment from falling into the pools.

The spent fuel pools, reactor wells, dryer-separator storage pools, and common shipping cask pool including all gates are designed to Seismic Category I requirements. All pools and wells are lined with stainless steel to minimize leakage and reduce corrosion product formation. The spent fuel pools are further designed so that they cannot be drained to a level that uncovers the top of the stored fuel. The normal water shielding over the stored fuel in the racks is approximately 23 ft. However, in the unlikely event that the pool gates fail to contain the pool water, the fuel racks and their contained fuel are assured of maintaining water coverage at all times. Cooling water supply lines enter the spent fuel pool from above the normal water level and are provided with high point siphon breaking vent lines to
23 prevent siphoning of water from the pools.

The superstructure of the reactor building serves as a low leakage barrier to provide atmospheric isolation of the spent fuel storage pool and associated fuel handling area.

The superstructure is composed of structural steel framing, metal siding and metal roof decking. The superstructure is designed to Seismic Category I criteria.

Features to limit potential offsite exposures in the event of significant release of radioactivity from the spent fuel have been provided.

These include a ventilation exhaust system, isolation of the secondary containment on high radiation, air mixing, and a standby gas treatment system capable of maintaining the secondary containment at 1/4-in. water column negative pressure with respect to the outside ambient pressure. These features are discussed in Subsection 9.4.2.

The radiological considerations for the spent fuel storage arrangement are described in Chapter 12.

9.1.2.3 Safety Evaluation

9.1.2.3.1 Criticality Control

9.1.2.3.1.1 Description of Fuel Racks

The high density fuel storage racks are a modular design. Each fuel pool contains 24 modules. Twenty-two (22) of the modules provide 120 positions for fuel on a rectangular 10 x 12 array. This array is made up of 60 square poison cans placed in a checkerboard pattern. Each square can is constructed by placing a slab of "Boral" between a square tube-within-a-tube. These tubes are seal welded and anodized. The nominal width of the "Boral" is 5.25 inches and the length continues so it overlaps the fuel pellet stack length in the fuel assemblies by one inch at both the top and bottom. 23

Two of the 24 modules provide fuel storage spaces for 100 fuel assemblies in a 10 X 10 array plus five (5) storage positions for other components, each. The fuel array is made up of 50 poison cans placed in a checkerboard pattern. The five other storage positions are made up of 11.5 inch ID, right circular anodized aluminum tubing.

The twenty-four (24) modules are placed in the spent fuel storage pool in such a manner to assure there is a "Boral" slab between each adjoining fuel storage position.

9.1.2.3.1.2 Basic Assumptions of Criticality Analysis

The geometry of the spent fuel storage array is such that Keff will be ≤ 0.95. To ensure that the design criteria are met, the following normal and abnormal spent fuel storage conditions were analyzed (See Figure 9.1-3).

a) normal positioning in the spent fuel storage array,

b) fuel stored in multi-purpose storage positions, 23

c) pool water temperature increases to 212°F,

d) normal storage array of ruptured fuel,

e) abnormal positioning in the spent fuel storage array,

23 f) moving fuel bundle adjacent to storage area,

g) dropped fuel bundle

To ensure that the analysis followed a conservative approach and conformed to the general guidelines of criticality safety analysis, the calculations were performed using the following criteria:

1. A uniform 3.25 w/o enriched U-235 distribution in an 8 x 8 bundle
2. Fresh fuel, no burnable poison
3. Minor structural members replaced by water
23 4. Fresh water @ 68°F
5. Two water rods
6. Supercell calculations include the water gap and channel surrounding the assembly.

Cross sections for all other material regions of the reference storage rack cell of Figure 2 come from the approprite CHEETAH-B calculation.

a) CORC-BLADE

1. Two mesh intervals
2. Boron density adjusted (densified) to accommodate area lost in cylindricizing boral to preserve thickness of poison control blades.

3. The fast and thermal flux spectra generated by peripheral pin cell calculation in CHEETAH-B are input to CORC-BLADE which in turn yields the effective Boron cross-sections.

b) PDQ-7

1. The use of a four group conventional model

2. Arbitrary mesh selections include two mesh intervals per fuel pin, two mesh intervals for outer water gap and three mesh intervals for homogenized Aluminum can and void gap.

3. The use of infinite water cross sections for the pool water in the dropped assembly analysis.

9.1.2.3.1.3 Calculational Models

a) The Principal Analytical Model

The analysis of the Susquehanna spent fuel storage rack design uses the CHEETAH-B/CORC-Blade/PDQ-7 calculational model. CHEETAH-B is the BWR lattice version of Nuclear Associates International's (NAI's) CHEETAH code which is a modified version of the original LEOPARD code and uses a modified ENDF/B-II cross section library. CORC-Blade generates diffusion theory cross-sections for the control poison in boiling water reactors and utilizes in its calculations an input neutron spectrum from the cell code, CHEETAH-B. The PDQ-7 program is the well-known few-group spatial diffusion theory code. The CHEETAH-B/PDQ-7 model which is also a part of the LEAHS (Lifetime Evaluations and Analysis of Heterogeneous Systems) nuclear analysis series of Control Data Corporation, has been extensively tested by means of benchmarking calculations of measured criticals as well as core physics calculations for several operating power reactors.

CHEETAH-B determines a multi-group neutron spectrum for a given homogeneous mixture of materials and uses this spectrum to weigh the cross sections and provides average few group cross-sections. The generation of macroscopic edits over user defined sub-extra regions facilitates preparation of constants for nondepleting regions of the assembly as input to subsequent PDQ calcualtions.

The reference case storage rack cell is shown in Figs. 9.1-4a & 4b. A zero neutron current boundary condition is applied to the four sides of this cell to produce an infinite array effect. The two dimensional PDQ-7 reference case calculations are made for four neutron energy groups, two mesh blocks per fuel pin and a zero axial buckling to account for no axial leakage.

b) The Verification Model

The verification calculation employs the KENO-IV/AMPX model. The basic neutron cross section data comes from the master library of AMPX - a 123 group GAM-THERMOS neutron library prepared from ENDF/B version II data. The NITAWL module of the AMPX program is used to perform a Nordheim integral treatment of the U-238 resonances accounting for the self-shielding effect. The working library produced by the NITAWL/AMPX module retains the 123 group energy structure and is used directly by KENO-IV.

In the KENO-IV calculation, each fuel and water rod cell is represented discretely. The array option of KENO-IV is

applied to arrange the box types into a matrix representing the fuel assembly. Then a water reflection region is added to the outside of this matrix followed by the zircaloy channel, water, and the aluminum canister. Void and Boral slab regions complete the reference case storage rack cell. 23

To simulate the arrangement of large number of storage rack units, and for a non-leakage condition in the axial directions, a specular reflective condition is applied to all six sides of this storage rack cell.

9.1.2.3.1.4 Reference Case Calculations

a) Physical Parameters and Basic Storage Rack Cell Geometry

The reference storage rack cell is 6.625 inches square (See Figure 9.1-4b). The Aluminum canister has an inside dimension of 6.156 inches and is 0.125 inches thick to accommodate an 8 x 8 fuel assembly of dimension 5.12 inches. The fuel is channeled in 0.120 inch thick Zircaloy-4. Sheets of Boral, clad with Aluminum hang between adjoining canisters. The Boral sandwich is 0.125 inches thick and 5.25 inches in width. The Boral Core is assumed to have a thickness of 80 mils and a minimum B-10 density of $0.0233 g/cm^2$. Between the Aluminum Canister and the Boral sheets is a void space of 0.047 inches. 23

b) Results of the Reference Case Calculations

The reference case for this study is a CHEETAH-B/CORC-BLADE/PDQ-7 calculation for the configuration at a temperature of 68°F and zero voids. This calculation results in a keff value of 0.8931. 23

The KENO-IV results gave an average Keff of 0.9124 ± 0.0043 with a 95% confidence interval ranging from 0.9038 to 0.9210.

c) Channel Effect

Since the rack must accommodate both channel and unchanneled fuel, studies reveal that the channeled fuel in the rack is more reactive than the unchanneled fuel. Taking the conservative approach, the study here involves channeled fuel except in the accident condition where unchanneled fuel is dropped. The decrease in Δk from channeled to unchanneled fuel is 0.002 in the reference case rack. 23

9.1.2.3.1.5 Susquehanna SES Bias Calculations 23

a) Temperature Effect

Using the reference storage rack cell geometry, the temperature of the fuel and pool water was varied. Intermediate steps at 32°F, 95°F, 120°F, 150°F, 180°F, 212°F were studied, and the results of the CHEETAH-B/CORC-BLADE/PDQ-7 runs show reactivity decreased continuously as temperature increased from 32°F.

23

b) Void Effect

The effect of boiling (assuming equal voids inside and outside fo the rack) is studied by varying the voids from 0% to 20% at a temperature of 212°F with the reference geometry and CHEETAH-B/CORC-BLADE/PDQ-7 calculations. The keff variation due to voids shows a continuous decrease.

23

c) Enrichment Sensitivity

The basic analysis was performed for an average enrichment of 3.25 w/o of U-235, which gives an average U-235 fuel loading of 15.7521 grams per axial centimeter of the active section of the assembly. For the purpose of determining the effect of fuel loading change, the analysis found that, in the range of interest to the Susquehanna spent fuel pool facility, an enrichment reactivity coefficient of 0.008 $\Delta k/0.1$ w/o U-235 can be applied. 23

d) Effect of Boron

The Boral slab which separates two adjacent fuel assemblies has a nominal thickness of 0.125 inches, nominal width of 5.250 inches, and an overall length or height of 12 feet 8 inches. The nominal Boral core thickness is 80 mils. The minimum B-10 loading in the Boral core is guaranteed by the manufacturer to be 0.0233 g/cm² which yields a B-10 number density of 0.006905×10^{24} atoms/cm³ (based on 80 mils thickness).

Although the analysis was based on a minimum B-10 loading, a study of boron density revealed an increase in reactivity of $0.005\Delta k$ if the boron density was lowered from 70% to 65% but only a $0.003\Delta k$ decrease if it was raised from 70% to 75%. The effect of reducing the Boral width was also examined. The results yield a reactivity variation of approximately $0.003\Delta k/0.125$ inch of Boral in a study of widths from 5.12 inches to 5.455 inches. The change due to the -0.03 inch minimum tolerance on Boral width is $+0.001\Delta k$.

23 e) Boral Acceptance Criterion

The criticality calculations are based on the assumption that the core of the Boral slabs contains a homogeneous mixture of fine B C and aluminum powder. The computer codes used for these calculations contain no provision to take into account directly the self-shielding effect due to the random distribution of B C grains of finite sizes in the mixture. Based on a homogeneous mixture, the neutron attenuation factor for a Boral slab of 80 mil thickness with 0.0233 g/cm² B-10 loading was calculated to be 0.969.

Neutron attenuation studies for Boral slabs identical to those furnished on the Susquehanna project were made. They have determined that using an attenuation factor of 0.963 minimum assures a B-10 areal density of 0.0233 gm/cm² minimum.

Using calculational techniques utilizing a B-10 loading of 0.0233 gm/cm² producing an attentuation factor of 0.963 yielded a Boral core thickness of 0.055 ± 0.003 inches.

Analysis shows the difference between an attentuation factor of 0.969 (which was calculated from the reference case Boral core thickness of 0.080"), and 0.963 yields a $\Delta K = 0.003$ for the Susquehanna spend fuel storage racks. Therefore, the Boral acceptance criteria is either a minimum B-10 loading of 0.0233 gm/cm² or a minimum neutron attentuation factor of 0.963.

f) Change in Pitch (pitch sensitivity)

The reactivity effect of mechanical tolerance changes caused by structural, fabrication, installation, and seismic conditions can be conservatively determined by the pitch sensitivity study. The calcualtions were carried out on a 0.125 inch increment. The pitch reactivity coefficient was determined to be about $+0.007\Delta k/0.125$ inch pitch reduction in the range from 6.875 inch pitch to 6.375 inch pitch. Based on a total dimensional average positional tolerance of
23 0.256", a bias $\Delta K = -0.014$ is included in the final reactivity summary.

g) Boral Width Reduction

The nominal width of the Boral slab is 5.25". The change in reactivity due to the -0.03" minimum tolerance on this width is $\Delta K = -0.001$.

9.1.2.3.1.6 Abnormal Configuration 23

a) 6 x 9 - One Boral Slab Missing

To simulate the effect of one missing Boral slab, a 6 x 9 array was described in a PDQ-7 model using reference geometry and cross sections. With 1 internal boral slab missing the Keff was 0.8981 or an increase of 0.005 Δk from the reference case. 23

b) Off Center Positioning

The reactivity effect of off-center positioning due to improper loading or some natural phenomenon can be conservatively determined. The four assembly cluster with assemblies loaded off center in their cavities and preferentially leaning toward the center of the cluster was analyzed. The zero neutron current boundary condition applied to the outer boundaries of this 2 x 2 array produces the effect of an infinite array of these four off-center assembly clusters. With a reduced pitch of 6.5 inches and modified cross sections which reflect the changing spectral effect of additional water or no water adjacent to the channel the keff was 0.8897. An off-centered loading of assemblies decreases the reactivity so no adverse effect occurs.

c) Rack Module Junction (no Boral)

At the junction of four rack modules, four fuel assemblies in the corner locations could face each other without being separated by Boral slabs, depending on the rack orientation. The structural design provides a minimum separation of 9.375 inches for these four assemblies. Study shows that this configuration will cause no adverse reactivity effect.

d) Storage of Fresh Fuel Under Mist Conditions

The storage of fresh fuel under mist or foam conditions was considered. A series of CHEETAH-B, CORC-BLADE, PDQ-7 calculations was made for the base case geometry at 68°F by varying values of water density (.15 to .75 gm/cm³). Two KENO-IV verification runs were made for water densities of .15 and .25 gm/cm³; the resulting Keff's are 0.5301±0.0023 and 0.5958 ± 0.0029 respectively. The results indicate a negative reactivity coefficient for decreasing the water density. Thus it may be concluded that fresh fuel may be stored under dry or mist conditions with no further safety implications. 23

e) Assembly Drop Accident

The consequences of dropping an assembly outside the rack and parallel to an assembly located in the outermost row of the storage array were analyzed. The assembly dropped during handling is assumed to lodge parallel to an assembly in an outer cavity with no Boral slab separating the parallel assemblies. The analysis produced an increase in reactivity less than +0.5% Δk in a 4 x 4 array with 8 inches of pool water in the outer cavity.

9.1.2.3.1.7 SUMMARY AND CONCLUSIONS

Diffusion Theory Results

The results of the criticality analysis of the proposed storage rack for Susquehanna Spent Fuel are summarized below:

PDQ Results:

Keff Reference Case	0.893
Dimensional and Positional Tolerance, ΔK	0.014
Boron Width Effect, ΔK	0.001
Temperature Effect, ΔK	0.004
Variation on Boral Accept.	0.003
Void Effect, ΔK	Negative
23 Adjusted Keff for Susquehanna	0.915

Monte Carlo Results and the Calculational Bias

The KENO-IV verification calculation for the nominal case yields a Keff value of 0.912±0.004 with 95% confidence interval ranging from 0.904 to 0.920. The KENO model has a slightly negative bias (ΔK = −0.001) deduced from NAI's benchmarking calculations. Comparing the upper bound 95% confidence interval result of the KENO nominal case and the PDQ runs, and based on the KENO benchmarking results, a calculational bias of 0.026 in ΔK should be applied to the adjusted PDQ Keff.

Summary of Results

Keff (PDQ) adjusted	Keff	0.915
Calculational bias	ΔK	0.026
Final Keff	Keff	0.941
23 Design Limit, Keff	Keff	0.950
Calculational Margin for Susquehanna SES	ΔK	0.009

The final Keff value (0.941) includes all the design specification tolerances, model bias, and the 95% confidence interval from the KENO calculations. However, the negative

reactivity effect (-0.5% Δk) due to the presence of U-234 and the parasitic structure materials (i.e. spacer grids) in each assembly, and the positive reactivity effects due to the possible absence of a Boral plate and an assembly drop accident are not included.

9.1.2.3.2 High Density Fuel Storage Rack Design

Spent fuel storage racks provide a place in the spent fuel pool for storing new and spent fuel. The high density spent fuel racks contain a neutron-absorbing medium of natural boron carbide (B_4C) in an aluminum matrix core clad with 1100 series aluminum. This neutron absorber is marketed under the trade name of Boral.

Boral slabs are manufactured under a proprietary qualified process. This process assures a uniform minimum B-10 density of 0.0233 gm/cm² in the Boral slabs utilized in the construction of the Susquehanna Racks. Benchmark measurements of those slabs yield a neutron attenuation factor of 0.963 minimum. 23

The rack manufacturer, assures that correct Boral locations and quantities were present in accordance with the design and procurement documents through a rigorous quality assurance program evaluated and approved by the AE. The construction of the rack assures that all adjacent storage cavities are separated by a Boral slab. The Boral is sealed within two concentric square aluminum tubes referred to as poison cans.

Figure 9.1-2a shows the structural design. Each rack module consists of six basic components:

1) top grid casting

2) bottom grid casting

3) poison cans

4) side plates

5) corner angle clips

6) adjustable foot assemblies

Each component is anodized separately.

The top and bottom grid casting are machined to maintain a nominal fuel pitch (center-to-center spacing) of 6.625 inches. Within these machined areas, in a checkerboard pattern, Boral poison cans are nested. This ensures smooth entry and removal of fuel assemblies in each fuel cavity. This design also assures Boral is between each stored fuel assembly. To complete the

module, the grids are bolted and/or riveted together by four corner angles and four side shear panels. Adjustable foot assemblies are located at the four corners of each module to allow adjustment for variations of the pool floor level of ±0.75 inches. To maintain a flat, uniform contact area, the leveling screw bearing pads are free to pivot.

Each model is level with each other module at the top. There is nominally seven inches of clearance from the bottom of the module to the pool floor. This assures adequate clearance for cooling water to enter each fuel cell, and through natural convection, keep each fuel assembly cool.

Each module is bolted to each other. The perimeter modules have seismic bracing to embedments in the pool wall assuring
23 structural integrity through all anticipated dynamic loads. The weight of the fuel assembly is supported in the chamfered hole in the bottom casting. Nominal center-to-center fuel spacing between modules is 9.375 inches.

a) The square poison cans are positioned in a top and bottom grid in a checkerboard pattern. Each poison can is pressure and vacuum leak tested for integrity.

b) The seismic restraints from the racks to the wall embedments consist entirely of a welded stainless steel construction. To reduce any galvanic corrosion, inconel pins are used
23 between the wall seismic restraints and racks. The only interface of each module with the pool floor are four stainless steel pads attached to the rack leveling screws. A 1/4 inch ABS plastic material is volumetrically captured between this pad and the aluminum leveling screw to prevent galvanic corrosion with the pool floor stainless steel liner plate.

c) All materials used for construction are specified in accordance with the 1972 issue of the ASTM specifications, as applicable.

Traceability of major rack components to a heat lot are maintained.

In addition, the suppliers' quality assurance-quality are control program audited by the AE and user, in effect to ensure that the Boral has the required minimum B4C density and uniform B4C distribution in each sheet. Boral traceability is maintained.

d) A dimensional, visual, and functional (including testing with a dummy fuel assembly) inspection of the racks is performed prior to shipment by the rack manufacturer.

e) The rack materials have no significant degradation due to the total radiation doses expected in the spent fuel pool over the design life.

f) The minimum fuel spacing within a rack assembly is 6.500". The minimum fuel spacing between racks is 9.125". Fuel assembly placement between modules or cavities of a module are not possible. 23

g) The racks are designed to withstand the loading under the following loading conditions: dead, live, jammed fuel assembly, dropped fuel assembly, thermal, OBE and DBE seismic, SRV, and LOCA or Chugging.

h) The racks are installed in the pool on four tension and compression quadrants to eliminate thermal loads resulting from confined expansion.

i) An inservice inspection (ISI) program will be in effect throughout the life of the racks to assure quality of the poisoned racks is maintained; as described in Section 9.1.2.3.3.

9.1.2.3.3 Inservice Inspection

Sixteen test coupons are to be provided for an on-going in-service inspection program. Two coupons, one of which is vented and the other sealed, would be removed and analyzed at intervals of 1, 3, 5, 10, 15, 20, 30, and 40 years after installation.

9.1.2.3.3.1 Test Coupon Description and Installation

A typical test coupon is a shortened production-type can similar to the spent fuel rack. Four sheets of BORAL neutron poison are encapsulated between the inner and outer cans. After assembly, the entire coupon is anodized.

The sealed cans are pressure-checked through a hole in the outer can. This hole is then welded to prevent water from contacting the BORAL. The unsealed cans will also have a 13/64 inch hole which will not be welded. 23

Two test coupons, one vented and the other unvented, are tied together with a hanger. This hanger contains a handling eye so that they can be hung on the perimeter of the spent fuel rack.

9.1.2.3.3.2 Test Coupon Inspection

a) The test coupon assembly will be removed from the spent fuel pool.

b) The test coupon will be drained of the entrapped water from the vented coupon into a beaker and the pH determined.

c) The vented coupon will be disassembled sufficiently so that the neutron absorber can be extracted.

d) Upon disassembly, note whether there is water in the sealed coupon. If so, perform step #B above.

e) Visual inspection of the neutron absorber plates will be noted and any discoloration, corrosion damage or physical damage will be recorded. If corrosion or physical damage is noted, record depth and extent of damage.

f) The plates will be washed in a mild abrasive and detergent solution, then rinsed in clean water and alcohol. The plates will be dried in a 250° F oven for 3 hours, followed by 3 hours in a 600°F oven.

g) Each plate will be weighed and determine weight change.

h) Reperform step #e.

i) All data will be recorded, including pH values, for future comparison.

9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

9.1.3.1 Design Bases

The Fuel Pool Cooling and Cleanup System (FPCCS) is designed with the following considerations:

a) Maintaining the fuel pool water temperature below 125°F. The heat load is based upon filling the pool with 2840 fuel assemblies from normal refueling discharges and transferred to the fuel pool within 160 hours after shutdown. Tables 9.1-2a and 9.1-2b show one discharge schedule and heat load for this condition.

b) During an emergency, heat load (EHL) condition, one RHR pump and heat exchanger are available for fuel pool cooling. The EHL condition occurs when the spent fuel racks contain 2840 fuel assemblies including a full core discharged to the pool within 250 hours after shutdown (control rods inserted). Tables 9.1-2c and 9.1-2d show the discharge schedule and heat load for this condition for Units 1 and 2. The RHR cooling system will maintain the fuel pool water temperature,

23

(with the heat load of 3.26 x 10 BTU/hr) at or below 125°F with or without assistance from the FPCCS. When the decay heat load of the spent fuel drops to the level for which the FPCCS is designed, the RHR system may be disengaged. If, under EHL conditions, the RHR cooling system and the FPCCS are not available for cooling, the water in the fuel pool will begin to boil in about eight (8) hours. 23

c) Redundant Seismic Category I Emergency Service Water connection to each pool are provided to allow for makeup of evaporative losses in the event of failure of the FPCCS. This event is discussed in detail in Appendix 9-A. The pool will begin to boil 25 hours after loss of cooling. The makeup line provides sufficient flow to maintain the fuel pool water level approximately 23 feet above the top of the fuel storage racks.

d) To maintain the water clarity and quality in the pools as follows to facilitate underwater handling of fuel assemblies and to minimize fission and corrosion product buildup that pose a radiological hazard to operating personnel:

Conductivity	≤ 3 mircromho/cm at 25°C
pH	5.3 - 7.5 at 25°C
Chloride (as Cl)	≤ 0.5 ppm
Heavy elements (Fe,Cu,Hg,Ni)	< 0.1 ppm
Total insolubles	< 1 ppm

23

9.1.3.2 System Description

Each reactor unit is provided with its own FPCCS as shown on Figures 9.1-5 and 9.1-6.

The system cools the fuel storage pool water by transferring the decay heat of the irradiated fuel through heat exchangers to the service water system.

Water clarity and quality in the fuel storage pools, transfer canals, reactor wells, dryer-separator pools, and shipping cask pit is maintained by filtering and demineralizing.

The FPCCS consists of fuel pool cooling pumps, heat exchangers, skimmer surge tanks, filter demineralizers, associated piping, valves, and instrumentation.

Equipment Description

Table 9.1-1 shows the design parameters of the FPCCS equipment. The seismic and quality group classifications of the FPCCS components are listed in Section 3.2.

One skimmer surge tank for each unit collects overflow water from skimmer drain openings with adjustable weirs at the water surface elevation of each pool and well. The common shipping cask pit water overflows to both units' skimmer surge tanks.

Wave suppression scuppers along the working side of the fuel pools also drain to the skimmer surge tanks. The skimmer openings in the pool liners are protected with a wire mesh screen to prevent floating objects such as the surface breaker viewing aids from entering the surge tanks. The adjustable weir plates are set according to the required cooling flow, desired flow pattern, and water shielding needs.

The skimmer surge tank provides a suction head for the fuel pool cooling pumps and the RHR pumps, and a buffer volume during transient flows in the normally closed loop FPCCS. It provides sufficient live capacity for three days' normal evaporative loss from the fuel pool without makeup from the demineralized water system. A removable object retention screen in the tank is accessible through the flanged tank top. Tank level indication and alarms on a control panel on the refueling floor and/or the vicinity of the fuel pool cooling pumps announce when the remote manual makeup valves must be opened or water drained from the system.

The fuel pool cooling pumps are stopped upon a low tank level signal.

Three fuel pool heat exchangers piped in parallel are located in the reactor building below the surge tank bottom elevation. The shell side is subjected to the static head of the skimmer surge tank level only. This is a minimum of 5 psi lower than the tube side service water pressure, thus minimizing the possibility of radioactive contamination of the service water system (see Subsection 9.2.1) from a tube leak.

The number of heat exchangers in service depends on the decay heat load from irradiated fuel in the spent fuel pool. The common inlet and each heat exchanger outlet temperature is recorded and high temperature alarmed on a local control panel.

Three fuel pool cooling pumps piped in parallel are placed in service in conjunction with the heat exchangers. They take suction from the heat exchangers and develop sufficient head to process a partial system flow through the filter demineralizers

and transfer it combined with the bypass flow to the diffuser pipes at the bottom of the pools.

The pump controls, discharge pressure indicators, flow indicator, and alarms for low flow and low discharge pressure are provided on a local control panel.

The pumps trip individually upon low NPSH. Three fuel pool filter demineralizers are piped in parallel. One fuel pool filter demineralizer is normally associated with each FPCCS with the third one in standby. The design flow per filter demineralizer is less than the total system flow. Part of the cooled water is therefore bypassing at a manually adjustable rate.

If the inlet temperature should exceed 150°F, the filter demineralizer must be manually bypassed to prevent degradation of the ion exchange resin.

The filter demineralizer units are designed to operate with water flowing at nominal 2 gpm/sq ft filter area. Powdered ion-exchange resin or resin mixed with Solka-Floc is used as a filter medium. The filter elements are stainless steel mesh, mounted vertically in a tube sheet and replaceable as a unit. Venting is possible from the upper head of the filter vessel to the reactor building ventilation system. The upper head is removable for installation and replacement of the filter elements. The filter demineralizer units are located separately in shielded cells. Sufficient clearance is provided to permit removal of the filter elements from the vessels. Each cell contains only the filter demineralizer and connecting piping. All inlet, outlet, recycle, vent, drain valves, and the holding pumps are located in a separate shielded room, together with necessary piping and headers, instrument elements, and controls. Penetrations through shielding walls are located so that shielding requirements are not compromised.

A post-strainer is provided in the effluent stream of each filter demineralizer to limit the migration of the filter material. The post-strainer element is capable of withstanding a differential pressure greater than the shut-off head for the system.

The ion exchange resin is a mixture of finely ground, 300 mesh or less, cation and anion resins in proportions determined by service. The cation resin is a strongly acidic polystyrene with a divinylbenzene cross-linkage. The resin is supplied in fully regenerated hydrogen form. The anion resin is a strongly basic, Type I, quaternary ammonium polystyrene with a divinylbenzene cross-linkage. The resin is supplied in a fully regenerated hydroxide form.

The resin is replaced when the pressure drop is excessive or the ion exchange resin is exhausted. Backwashing and precoating operations are controlled from a local control panel in the reactor building. The spent filter medium is backwashed from the elements with instrument air and condensate and transferred via a receiving tank to the waste sludge phase separator in the radwaste building.

New ion exchange resin is mixed in a resin tank and transferred as a slurry by a precoat pump to the filter where it is deposited on the filter elements. A separate precoat tank is provided to allow precoating of the filter elements with Solka-Floc only or prior to depositing ion exchange resins. Both tanks are furnished with an agitator for mixing the filter medium slurries. The precoat subsystem is common to both FPCS and may also be used for chemical cleaning of the filter demineralizers.

The holding pump associated with each filter demineralizer maintains circulation through the filter in the interval between the precoating operation and the return to normal system operation, or upon decrease in process flow below a point where the precoating may fall off the filter elements.

The filter demineralizers are controlled from a panel in the reactor building of Unit 1. Differential pressure and inlet and outlet pressure instrumentation are provided for each filter demineralizer unit to indicate when backwash is required. Suitable alarms, differential pressure indicators, and flow indicating controllers are provided to monitor the condition of the filter demineralizer and the post effluent strainers.

The backwash and precoat operations are push-button initiated, automatically sequenced operations. The filter demineralizer inlet and outlet conductivity is recorded and 0.1 micromho/cm in the outlet is alarmed on the reactor building sample station cabinet.

Fuel pool high and low level alarms with adjustable set points over the skimmer weir range and temperature indication and high alarms are provided on a refueling floor control panel.

A high rate of leakage through the refueling bellows assemblies, drywell to reactor well seals, or the fuel pool and shipping cask pit double gates is alarmed on a refueling floor control panel.

All local alarms are duplicated individually or as group alarms in the main control room.

Operational Description

During normal plant operation, the fuel pools are isolated from the reactor wells and the common shipping cask pit. The fuel pool cooling pumps circulate the pool water in a closed loop, taking suction from the skimmer surge tank through the heat exchangers and discharging a partial flow through the filter demineralizer, the balance through a bypass line back to the fuel pool diffusers.

After the reactor has been shut down, the vessel head and one refueling gate is removed. Two refueling water pumps (see Subsection 9.2.10) transfer condensate from the refueling water storage tank through diffusers into the reactor well and dryer-separator pool. The water level rises from the RPV flange elevation to the fuel pool water level in approximately 4 hr. The second refueling gate is then removed and refueling operations continued.

As the heat load increases with additional spent fuel elements being transferred from the reactor core to the spent fuel pool, additional pumps and heat exchangers of the FPCCS are put into service to meet the design objectives. Part of the cooled water can be diverted to the reactor well through the filling diffusers assisting the RHR system in removing decay heat rising from the core to the water surface. At this time two fuel pool filter demineralizers may be used in conjunction with the reactor water cleanup system to maintain required water quality in the reactor, reactor well, dryer-separator pool, and fuel pool. After refueling has been completed, the refueling water pumps transfer the water from the reactor well and dryer-separator pool through a condensate demineralizer back to the refueling water storage tank. This is accomplished in approximately 4 hr. Gravity draining of the refueling water to the refueling water storage tank is possible at a lower flow rate.

As the decay heat from the spent fuel decreases with time, the number of operating pumps and heat exchangers may be reduced to keep the fuel pool below the maximum normal design temperature.

The shipping cask storage pit is filled and drained in the same manner as the reactor well within approximately one hour with one refueling water transfer pump. The shipping cask storage pit is interconnected with the fuel pool during cask loading operations of spent fuel for offsite disposal. A small stream of fuel pool cooling water may be diverted from the fuel pool cooling pumps to the filling diffuser of the shipping cask pit to remove decay heat and water impurities during cask loading operations. This water returns over a skimmer weir to the skimmer surge tanks.

During periods of higher than MNHL generation in the fuel pool, eg, storing of a full core of irradiated fuel shortly after

shutdown, the RHR system is used to assist the FPCCS in dissipating the decay heat. One RHR pump takes suction from an intertie line to the skimmer surge tank and discharges through one RHR heat exchanger to two independent diffusers at the fuel pool bottom.

Makeup water to replenish evaporative and small leakage losses from the pools is provided from the demineralized water storage tank into the skimmer surge tank by opening a remote manual valve.

A Seismic Category I line from each of the two emergency service water loops is connected to the RHR intertie diffuser lines of each fuel pool, allowing for emergency makeup during boiling of the pool water. The manual supply valves in these emergency makeup lines are accessible apart from the refueling floor.

9.1.3.3 Safety Evaluation

The FPCCS is designed to maintain the fuel pool water at 125°F under normal refueling conditions (discharge schedule Table 9.1-2a) with all three heat exchangers and pumps in operation.

Under this condition an analysis was made of the natural circulation cooling of a maximum power spent fuel assembly in the most restrictive natural circulation flow loop in the spent fuel pool. The maximum coolant temperature at the outlet of the fuel assembly was calculated to be 169°F while the maximum clad temperature was calculated to be 188°F. Under these conditions there is no boiling in any fuel assembly.

Under full core unload conditions, (discharge schedule Table 9.1-2c) the bulk water temperature cannot be maintained below the desired maximum value of 125°F by the spent fuel pool cooling system alone. It is therefore necessary to connect the RHR system to the spent fuel pool. When this is done the pool temperature can be maintained well below 125°F.

All piping and equipment shared with or connecting to the RHR intertie loop are Seismic Category I, Quality Group C, and can be isolated from any piping associated with the non-Seismic Category I Quality Group C fuel pool cooling system.

Provisions to minimize and monitor leakage from the fuel pool are described in Subsection 9.1.2.3.

Makeup for evaporative and small leakage losses from the fuel pool is normally supplied from the demineralized water system to the skimmer surge tanks of each unit. The intermittent flow rate is approximately 50 gpm to each surge tank.

A Seismic Category I makeup of 30 gpm is provided by a 2 in. line from each emergency service water (ESW) loop to the RHR fuel pool diffusers, thus providing redundant flow paths from a reliable source of water. The design makeup rate from each ESW loop is based on replenishing the boil-off from the MNHL in each fuel pool for 30 days following the loss of the FPCCS capacity. The time required to reach boiling after loss of loading is approximately 25 hours.

The water level in the spent fuel storage pool is maintained at a height which is sufficient to provide shielding for required building occupancy. Radioactive particulates removed from the fuel pool are collected in filter demineralizer units in shielded cells. For these reasons, the exposure of station personnel to radiation from the spent fuel pool cooling and cleanup system is normally minimal. Further details of radiological considerations are described in Chapter 12.

An evaluation of the radiological effect of a boiling fuel pool is presented in Appendix 9A.

9.1.3.4 Inspection and Testing Requirements

No special tests are required because at least one pump, heat exchanger, and filter demineralizer are continuously in operation while fuel is stored in the pool. The remaining components are periodically operated to handle increased heat loads during refueling.

The pool liner leak detection drain valves are periodically opened and the leak rate estimated by the volumetric method. Gas or dye pressure testing from behind the liner plate may be performed to locate a liner plate leak.

Routine visual inspection of the system components, instrumentation, and trouble alarms is provided to verify system operability. Components and piping of the FPCCS designed per ASME Boiler and Pressure Vessel Code, Section III, Class 3 are in-service inspected as described in Section 6.6.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.1.4 FUEL HANDLING SYSTEM

9.1.4.1 Design Bases

The fuel-handling system is designed to provide a safe and effective means for transporting and handling fuel from the time it reaches the plant until it leaves the plant after post-irradiation cooling. Safe handling of fuel includes design considerations for maintaining occupational radiation exposures as low as practicable during transportation and handling.

Design criteria for major fuel handling system equipment is provided in Tables 9.1-2 through 9.1-4 which list the safety class, quality group, and seismic category. Where applicable, the appropriate ASME, ANSI, Industrial and Electrical Codes are identified. Additional design criteria is shown below and expanded further in Subsection 9.1.4.2.

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The transfer of new fuel assemblies between the uncrating area and the new fuel inspection stand and/or the new fuel storage vault is accomplished using the reactor building crane or the refueling floor jib cranes equipped with a general purpose grapple.

The reactor building crane auxiliary hoist or a refueling floor jib crane is used with a general purpose grapple to transfer new fuel from the fuel inspection stand or the new fuel vault to the fuel storage pool. From this point on, the fuel will be handled by the telescoping grapple on the refueling platform.

The refueling platform including refueling platform rails, clamps, and clips are Safety Class 2 and Seismic Class 1 from a structural standpoint in accordance with 10CFR50, Appendix A and B. Allowable stress due to safe shutdown earthquake loading is 120 percent of yield or 70 percent of ultimate, whichever is least. A dynamic analysis is performed on the structures using the response spectrum method with load contributions resulting from each of three earthquakes being combined by the RMS procedure.

Working loads of the platform structures are in accordance with the AISC Manual of Steel Construction. All parts of the hoist systems are designed to have a safety factor of five based on the ultimate strength of the material. A redundant load path is incorporated in the fuel hoists so that no single component failure could result in a fuel bundle drop. Maximum deflection limitations are imposed on the main structures to maintain relative stiffness of the platform. Welding of the platform is in accordance with AWS D14-1 or ASME Boiler and Pressure Vessel Code Section 9. Gears and bearings meet AGMA Gear Classification Manual and ANSI B3.5. Materials used in construction of load bearing members are to ASTM specifications. For personnel safety, OSHA Part 1910-179 is applied. Electrical equipment and controls meet ANSI CI, National Electric Code, and NEMA Publication No. IC1, MG1.

The general purpose grapple and the main telescoping fuel grapple have redundant hooks. The fuel grapple has an indicator which confirms positive grapple engagement.

The fuel grapple is used for lifting and transporting fuel bundles. It is designed as a telescoping grapple that can extend to the proper work level and in its fully retracted state still maintains adequate shielding over fuel.

To preclude the possibility of raising radioactive material out of the water, the cables on the auxiliary hoists incorporate an adjustable, removal stop that will jam the hoist cable against some part of the platform structure to prevent hoisting when the free end of the cable is at a preset distance below water level.

In addition, redundant electrical interlocks are a part of the grapple.

Provision of a separate cask loading pool, capable of being isolated from the fuel storage pool, will eliminate the potential accident of dropping the cask and rupturing the fuel storage pool. Refer to Chapter 15 for accident considerations.

9.1.4.2 System Description

Table 9.1-5 is a listing of typical tools and servicing equipment supplied with the nuclear system. The following paragraphs describe the use of some of the major tools and servicing equipment and address safety aspects of the design where applicable.

9.1.4.2.1 Spent Fuel Cask

The spent fuel cask is used to transfer spent reactor fuel assemblies from the spent fuel pool via the cask pit to a fuel storage or fuel reprocessing facility. The cask may also be used for offsite shipment of irradiated reactor components such as control rod blades, in-core monitors, etc.

The maximum loaded weight and, hence, the capacity of the cask is determined by the 125 tons lifting capacity of a reactor building crane. The maximum loading height, ie, height of the open cask in the storage pit, is determined by the depth of the shipping cask pit from the gate bottom. This allows for a constant water depth over the fuel in transit from the reactor to the fuel pool and into the shipping cask.

The cask is designed to dissipate the maximum allowable heat load from contained irradiated fuel by natural convection at least from the time the cask pit is drained until the cooling system on the transport vehicle is connected.

It further allows underwater replacement of the lid and other operations that may pose unacceptable radiation hazards to personnel. Considerations facilitating decontamination of the cask are given in the design. The design of the cask meets all applicable regulations of the Department of Transportation and 10CFR71 with respect to shipping of large quantities of fissile materials.

At present, no specific type of cask has been chosen. Over the lifetime of the plant, several different sizes and models may be used which the fuel handling facilities can accommodate.

9.1.4.2.2 Cask Crane

See Subsection 9.1.5 for discussion of reactor building cranes.

9.1.4.2.3 Fuel Servicing Equipment

The fuel servicing equipment described below has been designed in accordance with the criteria listed in Table 9.1-2.

9.1.4.2.3.1 Fuel Prep Machine

The fuel preparation machine, Figure 9.1-7, is mounted on the wall of the fuel storage pool and is used for stripping resuable channels from the spent fuel and for rechanneling of the new fuel. The machine is also used with the fuel inspection fixture to provide an underwater inspection capability, and with the defective fuel storage container to contain a defective fuel assembly for stripping of the channel.

The fuel preparation machine consists of a work platform, a frame, and a moveable carriage. The frame and moveable carriage are located below the normal water level in the fuel storage pool, thus providing a water shield for the fuel assemblies being handled. The fuel preparation machine carriage has a permanently installed up-travel-stop to prevent raising fuel above the safe water shield level. The moveable carriage is operated by a foot pedal controlled air hoist.

9.1.4.2.3.2 New Fuel Inspection Stand

The new fuel inspection stand, Figure 9.1-8, serves as a support for the new fuel bundles undergoing receiving inspection and provides a working platform for technicians engaged in performing the inspection.

The new fuel inspection stand consists of a vertical guide column, a lift unit to position the work platform at any desired level, bearing seats and upper clamps to hold the fuel bundles in position.

9.1.4.2.3.3 Channel Bolt Wrench

The channel bolt wrench, Figure 9.1-9, is a manually operated device approximately 12 feet in overall length. The wrench is used for removing and installing the channel fastener assembly while the fuel assembly is held in the fuel preparation machine.

The channel bolt wrench has a socket which mates and captures the channel fastener capscrew.

9.1.4.2.3.4 Channel Handling Tool

The channel handling tool, Figure 9.1-10, is used in conjunction with the fuel preparation machine to remove, install, and transport fuel channels in the fuel storage pool.

The tool is composed of a handling bail, a lock/release knob, extension shaft, angle guides, and clamp arms which engage the fuel channel. The clamps are actuated (extended or retracted) by manually rotating lock/ release knob.

The channel handling tool is suspended by its bail from a spring balancer on the channel handling boom located on the fuel pool periphery.

9.1.4.2.3.5 Fuel Pool Sipper

The fuel pool sipper, Figure 9.1-11, provides a means of isolating a fuel assembly in demineralized water in order to concentrate fission products in relation to a controlled background.

The fuel pool sipper consists of a control panel assembly and a sipping container cover to the tank.

9.1.4.2.3.6 Fuel Inspection Fixture

The fuel inspection fixture, Figure 9.1-12, is used in conjunction with the fuel preparation machine to permit remote inspection of fuel elements. The fixture consists of two parts: (1) a lower bearing assembly, and (2) a guide assembly at the upper end of the carriage. The fuel inspection fixture permits the rotation of the fuel assembly in the carriage, and, in conjunction with the vertical movement of the carriage, provides complete access for inspection.

9.1.4.2.3.7 Channel Gauging Fixture

The channel gauging fixture, Figure 9.1-13, is a go/no-go gauge used to evaluate the condition of a fuel channel, prior to rechanneling or when one is difficult to install.

The channel gauging fixture consists basically of a frame, gauging plate and gauging block. The gauging plate is shimmed to correspond to the outside dimension of a usable fuel channel. The gauging block conforms to the inside dimension of lower end of a usable fuel channel.

The channel gauging fixture is installed in the vertical position, between the two fuel preparation machines and hangs from the fuel storage pool curb.

9.1.4.2.3.8 General Purpose Grapple

The general purpose grapple, Figure 9.1-14, is a handling tool used generally with the fuel. The grapple can be attached to the reactor building auxiliary hoist, jib crane, and the auxiliary hoists on the refueling platforms. The general purpose grapple is used to remove new fuel from the vault, place it in the inspection stand, and transfer it to the fuel pool. It can be used to handle fuel during channeling.

9.1.4.2.3.9 Fuel Grapple

The fuel grapple is a telescoping mast with a double hook grapple head used to lift and orient fuel bundles for core and storage rack placement. It is a triangular, open sectioned mast constructed of tubular stainless steel.

Mast section-to-section guidance is provided by nylon bearing pads. Vertical motion is supplied by a dual wire rope cable hoist, which provides a redundant load path, and is mounted on the Refueling Platform Main Trolley. Hoist cable attachment to the inner-most grapple section is achieved through a rocker arm/clevis assembly which allows for load equilization in the hoist wire ropes. A redundant hook grapple head featuring individual hook engage switches and air cylinders consists of engage switches wired in series and interlocked in a manner to prevent raising a fuel bundle with either hook disengaged. Figure 9.1-20 outlines the main fuel grapple.

9.1.4.2.4 Servicing Aids

General area underwater lights are provided with a suitable reflector for illumination. Suitable light support brackets are furnished to support the lights in the reactor vessel to allow the light to be positioned over the area being serviced independent of the platform. Local area underwater lights are small diameter lights for additional illumination. Drop lights are used for illumination where needed.

A radiation hardened designed portable underwater closed circuit television camera is provided. The camera may be lowered into the reactor vessel and/or fuel storage pool to assist in the inspection and/or maintenance of these areas. The camera is capable of pitching ninety degrees which allows infinite scanning of three hundred and sixty degrees, solid angle.

A general purpose, plastic viewing aid is provided to float on the water surface to provide better visibility. The sides of the viewing aid are brightly colored to allow the operator to observe it in the event of filling with water and sinking. A portable, submersible type, underwater vacuum cleaner is provided to assist in removing crud and miscellaneous particulate matter from the pool floors, or the reactor vessel. The pump and the filter unit are completely submersible for extended periods. The filter "package" is capable of being remotely changed, and the filters will fit into a standard shipping container for off-site burial. Fuel pool tool accessories are also provided to meet servicing requirements. A fuel sipper is provided. This is to be used to detect defective fuel assemblies during open vessel periods while the fuel is in the core. The fuel sipper head isolates individual fuel assemblies by sealing the top of the fuel channel and pumping water from the bottom of the fuel assembly, through the fuel channel, to a sampling station, and return to the primary coolant system. After a "soaking" period a water sample is obtained and is radio-chemically analyzed.

9.1.4.2.5 Reactor Vessel Servicing Equipment

The essentiality and safety classifications, the quality group, and the seismic category for this equipment are listed in Table 9.1-3. Following is a description of the equipment designs in reference to that table.

9.1.4.2.5.1 Reactor Vessel Service Tools

These tools are used when the reactor is shut down and the reactor vessel head is being removed or reinstalled. Tools in this group are:

Stud Handling Tool
Stud Wrench
Nut Runner
Stud Thread Protector
Thread Protector Mandrel
Bushing Wrench
Seal Surface Protector
Stud Elongation Measuring Rod
Dial Indicator Elongation Measuring Device
Heat Guide Cap

These tools are designed for a 40 year life in the specified environment. Lifting tools are designed for a safety factor of 5 or better with respect to the ultimate strength of the material used. When carbon steel is used, it is either hard chrome plated, parkerized, or coated with an acceptable paint.

9.1.4.2.5.2 Steam Line Plug

The steam line plugs are used during reactor refueling or servicing; they are inserted in the steam outlet nozzles from inside of the reactor vessel to prevent a flow of water from the reactor well into the main steam lines during servicing of safety relief valves, main isolation valves, or other components of the main steam lines, while the reactor water level is raised to the refueling level.

The steam line plug design provides two seals of different types. Each one is independently capable of holding full head pressure. The equipment is constructed of non-corrosive materials. All calculated safety factors are 5 or greater. The plug body is designed in accordance with the "Aluminum Construction Manual" by the Aluminum Association.

9.1.4.2.5.3 Shroud Head Bolt Wrench

This is a hand held tool for operation of shroud head bolts. It is deisgned for a 40 year life, it is made of aluminum to be easy to handle and to resist corrosion. Testing has been performed to confirm the design.

9.1.4.2.5.4 Head Holding Pedestal

Three pedestals are provided for mounting on the refueling floor for supporting the reactor vessel head. The pedestals have studs which engage three evenly spaced stud holes in the head flange. The flange surface rests on replaceable wear pads made of aluminum. When resting on the pedestals, the head flange is approximately 3 feet above the floor to allow access to the seal surface for inspection and O-ring replacement.

The pedestal structure is a carbon steel weldment, coated with an approved paint. It has a base with bolt holes for mounting it to the concrete floor. The structure is designed in accordance with "The Manual of Steel Construction" by AISC.

9.1.4.2.5.5 Head Nut and Washer Rack

The RPV head nut and washer rack is used for transporting and storing up to 6 nuts and washers. The rack is a box shaped aluminum structure with dividers to provide individual compartments for each nut and washer. Each corner has a lug and shackle for attaching a 4-leg lifting sling.

The rack is designed in accordance with the "Aluminum Construction Manual" by the Aluminum Association, and for a safety factor of 5.

9.1.4.2.5.6 Head Stud Rack

The head stud rack is used for transporting and storage of 8 reactor pressure vessel studs. It is suspended from the auxiliary building crane hook when lifting studs from the reactor well to the operating floor.

The rack is made of aluminum to resist corrosion.

9.1.4.2.5.7 Dryer and Separator Sling

The dryer and separator sling is a lifting device used for transporting the steam dryer or the shroud head with the steam separators between the reactor vessel and the storage pools. The sling consists of a cruciform shaped structure which is suspended from a hook box with four wire ropes and turnbuckles. The hook box, with two hook pins, engages the reactor building crane sister hook. On the end of each arm of the cruciform is a socket

with a pneumatically operated pin for engaging the four lift eyes on the steam dryer or shroud head.

The sling has been designed such that one hook pin and one main beam of the cruciform is capable of carrying the total load and so that no single component failure will cause the load to drop or swing uncontrollably out of an essentially level attitude.

The safety factor of ll lifting members is 5 or better in reference to the ultimate breaking strength of the material. The structure is designed in accordance with "The Manual of Steel Construction" by AISC. The completed assembly is proof tested at 125 percent or greater of rated load and all strucutral welds are magnetic particle inspected after load test.

9.1.4.2.5.8 Head Strongback

The RPV head strongback is used for lifting the pressure vessel head. It is a cruciform shape with four equally spaced lifting points on the ends of the arms. In the center it has a hook box which engages with two pins to the reactor building crane sister hook.

The strongback is designed such that one leg of the cruciform will support the rated load and such that no single component failure will cause the load to drop or swing uncontrollably out of an essentially level attitude. The strucutre is designed in accordance with "The Manual of Steel Construction" by AISC. All welding is in accordance with the ASME Boiler and Pressure Vessel Code Section IX. A safety factor of 5 or greater in reference to the ultimate material strength is used for the design. The completed assembly is proof tested at 125 percent rated load. After the load test, all structural welds are magnetic particle inspected.

9.1.4.2.5.9 Service Platform

The service platform is provided to facilitate maintenance work on reactor internals. It provides a working platform for people and hand guided tools, and it also has provision for supporting a jib crane. The service platform is supported by four wheels which run on a circular track resting on the vessel flange and confined by the vessel closure studs.

The service platform is non-Seismic Class I equipment, and it has been designed for 0.75 g horizontal and 0.00 g vertical. The physical size of the device is such that it cannot enter the reactor pressure vessel.

The structure design is in accordance with "The Manual of Steel Construction" by AISC. Materials are in accordance with ASTM Standards. Welding is in accordance with ASME Section IX or AWS D1.1 structural welding.

The electrical system is in accordance with ANSI-ANS C1 National Electrical Code, and NEMA Publications No. IC1 and MG1.

Painting and surface preparation is in conformance with SSPC and in compliance with Req. Guide 1.54.

9.1.4.2.5.10 Service Platform Support

The service platform support serves as a sealing surface protector for the reactor vessel flange, and as a track for the service platform. It has continuous vertical support on the vessel flange, and horizontally it is confined by the vessel studs by strapping to the outer edge of the flange. The service platform support is made from aluminum and all welding is done in accordance with AWS Code D1.0.

9.1.4.2.5.11 Steam Line Plug Installation Tool

The steam line plug installation tool is suspended from the building crane auxiliary hook for transporting and installing the steam line plugs in the steam line nozzles of the reactor vessel. This tool is made of aluminum, it is designed for a safety factor of 5, and in accordance with "Aluminum Construction Manual" by the Aluminum Association.

9.1.4.2.6 In-Vessel Servicing Equipment

The instrument strongback attached to the Reactor Building crane auxiliary hoist is used for servicing neutron monitor dry tubes should they require replacement. The strongback supports the dry tube during transfer to the vessel. The in-core dry tube is then decoupled from the strongback and is guided into place while being supported by the Instrument Handling Tool. Final in-core insertion is accomplished from below the reactor vessel. The instrument handling tool is attached to the refueling platform auxiliary hoist and is used for removing and installing fixed in-core dry tubes as well as handling neutrons source holders and the Source Range Monitor/Intermediate Range Monitor dry tubes.

Each in-core instrumentation guide tube is sealed by an O-ring on the flange and in the event that the seal needs replacing, an in-

core guide tube sealing tool is provided. The tool is inserted into an empty guide tube and sits on the beveled guide tube entry in the vessel. When the drain on the Water Seal Cap is opened, hydrostatic pressure seats the tool. The flange can then be removed for seal replacement.

The auxiliary hoist on the refueling platform is used with appropriate grapples to handle control rods, flux monitor dry tubes, sources, and other internals of the reactor. Interlocks on both the grapple hoists and auxiliary hoist are provided for safety purposes; the refueling interlocks are described and evaluated in Section 7.6.

9.1.4.2.7 Refueling Equipment

Fuel movement and reactor servicing operations are performed from a platform which spans the refueling, servicing, and storage cavities.

9.1.4.2.7.1 Refueling Platform

The refueling platform is a gantry crane which is used to transport fuel and reactor components to and from pool storage and the reactor vessel. The platform spans the fuel storage and vessel pools on rails bedded in the refueling floor. A telescoping mast and grapple suspended from a trolley system is used to transport and orient fuel bundles, for core, storage rack, or shipping cask placement. Control of the platform is from an operator station on the main trolley with a position indicating system provided to position the grapple over core locations. The platform control system includes interlocks to verify hook engagement and grapple load, prevent unsafe operation over the vessel during control rod movements, and limit vertical travel of the grapple. Two 1000 pound capacity auxiliary hoists, one main trolley mounted and one auxiliary trolley mounted, are provided for servicing such as LPRM replacement, fuel support replacement, jet pump servicing, and control rod replacement. The grapple in its fully retracted position provides 8 feet 6 inches minimum water shielding over the active fuel during transit.

9.1.4.2.8 Storage Equipment

Specially designed equipment storage racks are provided. Additional storage equipment is listed on Table 9.1-5. For fuel storage racks description and fuel arrangement, see Subsections 9.1.1 and 9.1.2.

Defective fuel assemblies are placed in defective fuel storage containers, as necessary. These containers are stored in the multi-purpose storage container which is a part of the high density spent fuel racks. These may be used to isolate leaking or defective fuel while in the fuel pool and during shipping.

Defective fuel storage containers can be picked up and moved with a fuel bundle in them. Channels can also be removed from the fuel bundle while in a defective fuel storage container.

The Fuel Pool Sipper may be used for out-of-core wet sipping at any time. They are used to detect a defective fuel bundle while circulating water through the fuel bundle in a closed system. The containers cannot be used for transporting a fuel bundle. The bail on the container head is designed not to fit into the fuel grapple.

9.1.4.2.9 Under Reactor Vessel Servicing Equipment

The primary functions of the under reactor vessel servicing equipment are to: (1) remove and install control rod drives, (2) service thermal sleeve and control rod guide tube, (3) install and remove the neutron detectors. Table 9.1-4 lists the equipment and tools required for servicing.

The control rod drive handling equipment, which is powered electrically, is used for the removal and installation of the control rod drives from their housings. This equipment is designed in accordance with the requirements of National Electrical Manufacturers Association (NEMA, MG-1, Motor and Generator Standards), American National Standards Institute Standards (ANSI C-1, National Electric Code), Occupational Safety and Health Act (OSHA - 1910.179), American Institute of Steel Construction (AISC - Manual of Steel Construction). All lifting components are equipped with adequate brakes or gearing to prevent uncontrolled movement upon loss of power or component failure.

The equipment handling platform is also powered electrically and provides a working surface for equipment and personnel performing work in the under vessel area. It is a polar platform capable of 360° rotation. This equipment is designed in accordance with the

applicable requirements of OSHA (Vol. 37, No. 202, Part 191 ON), AISC, ANSI-C-1, (National Electric Code).

The spring reel is used to pull the in-core guide tube seal or in-core detector into the in-core guide tube during in-core servicing.

The thermal sleeve installation tool locks, unlocks, and lowers the thermal sleeve from the control rod drive guide tube.

The in-core flange seal test plug is used to determine the pressure integrity of the in-core flange O-ring seal. It is constructed of non-corrosive material. The key bender is designed to install and remove the antirotation key that is used on the thermal sleeve.

9.1.4.2.10 Fuel Transfer Description

9.1.4.2.10.1 Arrival of Fuel on Site

New fuel arrives in the railway bay of the reactor building Unit 1 either by railcar or truck. The access doors are closed to maintain the secondary containment as required by Technical Specifications. Unloading of the metal shipping containers is done by the auxiliary hoist of the reactor building crane.

9.1.4.2.10.2 Refueling Procedure

The plant refueling and servicing sequence diagram is shown in Figure 9.1-15. Fuel handling procedures are described below and shown visually in Figure 9.1-16 through Figure 9.1-19.

The Refueling Floor Layout is shown in Figure 9.1-4 and component drawings of the principal fuel handling equipment are shown in Figures 9.1-7 through 9.1-14 and Figure 9.1-20.

The fuel handling process takes place primarily on the refueling floor above the reactor. The principal locations and equipment are shown on Figure 9.1-16. The reactor, fuel pool, and shipping cask pool are connected to each other by slots, as shown at (A) and (B). Slot (A) is open during reactor refueling, and slot (B) is open during spent fuel shipping. At other times the slots are closed by means of blocks and gates, which make water-tight barriers.

The handling of new fuel on the refueling floor is illustrated in Figure 9.1-17. The transfer of the bundles between the crate (C)

and the new fuel inspection stand (D) and/or the new fuel storage vault (E) is accomplished using 5-ton auxiliary hoist of the reactor building crane or a half-ton floor mounted refueling jib crane equipped with a general-purpose grapple. The fuel bundle cannot be handled horizontally without support, so the crate is placed in an almost vertical position before being opened. The top and front of the crate are opened, and the bundles removed in a vertical position.

The auxiliary hoist of the reactor building crane or the jib crane are also used with a general-purpose grapple to transfer new fuel from the new fuel vault or inspection stand to a storage rack position in the fuel pool. From this point on, the fuel is handled by the telescoping grapple on the refueling platform.

The storage racks in both the vault and the fuel pool hold the fuel bundles or assemblies vertical, in an array which is subcritical under all possible conditions.

The new fuel inspection stand holds one or two bundles in vertical position. The Inspector(s) ride up and down on a platform, and the bundles are manually rotated on their axes. Thus the inspectors can see all visible surfaces on the bundles.

The general-purpose grapples and the fuel grapple of the refueling platform have redundant hooks, and an indicator which confirms positive grapple engagement.

The refueling platform uses a grapple on a telescoping mast for lifting and transporting fuel bundles or assemblies. The telescoping mast can extend to the proper work level; and, in its fully retracted state, maintains adequate water shielding over the fuel being handled.

The reactor refueling procedure is shown schematically in Figure 9.1-18. The refueling platform (G) moves over the fuel pool, lowers the grapple on the telescoping mast (H), and engages the bail on a new fuel assembly which is in the fuel storage rack. The assembly is lifted clear of the rack, and moved through slot (A) and over the appropriate empty fuel location in the core (J). The mast then lowers the assembly into the location, and the grapple releases the bail.

The operator then moves the platform until the grapple is over a spent fuel assembly which is to be discharged from the core. The assembly is grappled, lifted, and moved through slot (A) to the fuel pool. Here it is placed in one of the fuel prep machines (K).

An operator, using a long-handled wrench, removes the screws and springs from the top of the channel. The channel is then held, while a carriage lowers the fuel bundle out of the channel. The

channel is then moved aside, and the refueling platform grapple carries the bundle and places it in a storage rack. The channel handling boom hoist, (L), moves the channel to storage, if appropriate.

In actual practice, channeling and dechanneling may be performed in many sequences, depending on whether a new channel is to be used, or a used channel is to be installed on a new bundle and returned to the core. A channel rack is conveniently located near to the fuel prep machines, for temporary storage of channels which are to be reused.

To preclude the possibility of raising radioactive material out of the water, redundant electrical limit switches are incorporated in the auxiliary hoists of the refueling platform and the jib crane hoist, and interlocked to prevent hoisting above the preset limit. In addition, the cables on the hoists incorporate adjustable stops that will jam the hoist cable against the hoist structure, which prevents hoisting if the limit switch interlock system should fail.

When spent fuel is to be shipped, it is placed in a cask, as shown in Figure 9.1-19. The refueling platform grapples a fuel bundle from the storage rack in the fuel pools, lifts it, carries it through slot (B) into the shipping cask pool, and lowers it into the cask, (M). When the cask is loaded, the reactor building crane sets the cask cover (N) on the cask. After draining the shipping cask pool, the cask is decontaminated and lowered through the open hatchways, (P), onto the truck or railcar in the railway bay at grade level.

Provision of a separate cask loading pool, capable of being isolated from the fuel storage pool, eliminates the potential accident of dropping the cask and rupturing the fuel storage pool.

Additional detailed information is provided below.

9.1.4.2.10.2.1 New Fuel Preparation

9.1.4.2.10.2.1.1 Receipt and Inspection of New Fuel

The incoming new fuel will be delivered to a receiving station . The crates should be unloaded from the transport vehicle and examined for damage during shipment. The crate dimensions are approximately 32" x 32" x 18 feet long. Each crate contains two fuel bundles supported by an inner metal container. Shipping weight of each unit is approximately 3000 pounds. The receiving station includes a separate area where the crate covers can be

removed. The crates are then moved to the reactor building where the metal inner containers are removed and lifted to the refueling floor. Both inner and outer shipping containers are reusable. Handling during uncrating is to be accomplished by use of the reactor building crane extending down from the refueling floor through the equipment hatch.

9.1.4.2.10.2.1.2 Channeling New Fuel

The initial core for both units will be channeled as each new fuel bundle is inspected in the fuel inspection stand. This process will be repeated whenever new fuel channels are to be placed on new fuel bundles. Usually channeling new fuel is done concurrently with de-channeling spent fuel. Two fuel preparation machines are located in the fuel pool; one used for de-channeling spent fuel and the other to channel new fuel. The procedure is as follows: Using a jib crane and the general purpose grapple, a new fuel bundle is transported to one fuel prep machine if it had been residing in the fuel storage vault. Otherwise it is moved from a spent fuel pool storage rack to the fuel preparation machine using the refueling bridge. A spent fuel bundle is moved from a spent fuel pool storage rack to the other fuel prep machine. The channel is unbolted from the spent fuel bundle using the channel bolt wrench. The channel handling tool is fastened to the top of the channel and the fuel prep machine carriage is lowered removing the fuel from the channel. The channel is then positioned over a new fuel bundle located in the first fuel prep machine #2 and the process reversed. The channeled new fuel is then stored in the pool storage racks ready for insertion into the reactor.

9.1.4.2.10.2.1.3 Equipment Preparation

Prior to the plant shutdown for refueling, all equipment must be placed in readiness. All tools, grapples, slings, strongbacks, stud tensioners, etc. should be given a thorough check and any defective (or well worn) parts should be replaced. Air hoses on grapples should be routinely leak tested. Crane cables should be routinely inspected. All necessary maintenance and interlock checks should be performed to assure no extended outage due to equipment failure.

The in-core flux monitors, in their shipping container, should be on the refueling floor. The channeled new fuel and the replacement control rods should be ready in the storage pool.

9.1.4.2.10.2.2 Reactor Shutdown

The reactor is shut down according to a prescribed procedure. During cool down the reactor pressure vessel is vented and filled to above flange level to equalize cooling. The drywell and suppression chamber are de-inerted. The eight reactor well shield plugs can be removed. This is accomplished with the reactor building crane and the supplied slings.

This operation can be immediately followed by removal of the three canal plugs and the three slot plugs. Thus, a total of 14 separate plugs must be removed and placed on the refueling floor. A "Refueling Equipment Storage and Crane Clearance" arrangement drawing is issued to locate placement of these plugs on the refueling floor. The outer fuel pool gate is also removed at this time. The gate sling is attached to the gate lifting lugs and the reactor building crane lifts the gate and places it on the fuel pool gate storage lugs.

9.1.4.2.10.2.2.1 Drywell Head Removal

Immediately after removal of the reactor well shield plugs, the work to unbolt the drywell head can begin. The drywell head is attached by removable bolts protruding from the lower drywell flange. The nuts on top are merely loosened and the bolt heads swing outward. The bolts are then pulled upwards and supported with the nuts on a slotted lip of the head.

The sister hook of the reactor building crane is attached to the hook box on top of the unbolted drywell head and lifted to its appointed storage space on the refueling floor. The drywell seal surface protector is installed before any other activity proceeds in the reactor well area.

9.1.4.2.10.2.2.2 Reactor Well Servicing

When the drywell head has been removed, an array of piping is exposed that must be serviced. Various vent piping penetrations through the reactor well must be removed and the penetrations made water tight. Vessel head piping and head insulation must be removed and transported to storage on the refueling floor.

Water level in the vessel is now brought to flange level in preparation for head removal.

9.1.4.2.10.2.3 Reactor Vessel Opening

9.1.4.2.10.2.3.1 Vessel Head Removal

The stud tensioner is transported by the reactor building crane and positioned on the reactor vessel head. Each stud is tensioned and its nut loosened in a series of 2-3 passes. When the nuts are loose, they are backed off using a nut runner until only a few threads engage. The vessel nut handling tool is engaged in the upper part of the nut and the nut is rotated free from the stud. The nuts and washers are placed in the racks provided for them and transported to the refueling floor for storage. With the nuts and washers removed, the vessel stud protectors and vessel head guide caps are installed.

The head strongback, transported by the reactor building crane, is attached to the vessel head and the head transported to the head holding pedestals on the refueling floor. The head holding pedestals keep the vessel head elevated to facilitate inspection and "O" ring replacement.

The six studs in line with the fuel transfer canal are removed from the vessel flange and placed in the rack provided. The loaded rack is transported to the refueling floor for storage.

9.1.4.2.10.2.3.2 Dryer Removal

The dryer-separator sling is lowered by the reactor building crane and attached to the dryer lifting lugs. The dryer is lifted from the reactor vessel and transported to its storage location in the dryer-separator storage pool adjacent to the reactor well. The dryer is transported in air. However, if the dryer should become highly contaminated, the reactor well and storage pool can be flooded and a wet transfer effected.

9.1.4.2.10.2.3.3 Separator Removal

In preparation for separator removal, the service platform and service platform support are installed on the vessel flange. From the service platform work area, the four main steam lines are plugged from inside the vessel using the furnished plugs for this duty. Servicing of the safety and relief valves can thus be accomplished without adding to the critical refueling path time. Working from the service platform, the separator is unbolted using the shroud head bolt wrenches furnished.

When the unbolting is accomplished, the service platform is removed and stored on the refueling floor. The service platform support remains on the vessel flange during the remainder of the refueling outage and acts as the flange seal surface protector.

The dryer-separator sling is lowered into the vessel and attached to the separator lifting lugs. The water in the reactor well and in the dryer-separator storage is raised to fuel pool water level and the separator is transferred underwater to its allotted storage place in the adjacent pool.

9.1.4.2.10.2.3.4 Fuel Bundle Sampling

During reactor operation, the core off-gas radiation level is monitored. If a rise in off-gas activity has been noted, the reactor core will be sampled during shutdown to locate any leaking fuel assemblies. The fuel sampler or sipper rests on the channels of a four bundle array in the core. An air bubble is pumped into the top of the 4 fuel bundles and allowed to stay about 10 minutes. This stops water circulation through the bundles and allows fission products to concentrate if a bundle is defective. After 10 minutes, a water sample is taken for fission product analysis. If a defective bundle is found, it is taken to the fuel pool and if required, may be stored in a special defective fuel storage container to prevent the spread of contamination in the pool.

9.1.4.2.10.2.4 Refueling and Reactor Servicing

The remaining gate isolating the fuel pool from the reactor well is now removed thereby interconnecting the fuel pool, the reactor well, and the dryer-separator storage pool. The actual refueling of the reactor can now begin.

9.1.4.2.10.2.4.1 Refueling

During a normal equilibrium outage, approximately 25% of the fuel is removed from the reactor vessel, 25% of the fuel is shuffled in the core (generally from peripherial to center locations) and 25% new fuel is installed. The actual fuel handling is done with the fuel grapple which is an integral part of the refueling platform. The platform runs on rails over the fuel pool and the reactor well. In addition to the fuel grapple, the refueling platform is equipped with two auxiliary hoists which can be used with various grapples to service other reactor internals.

To move fuel, the fuel grapple is aligned over the fuel assembly, lowered and attached to the fuel bundle bail. The fuel bundle is raised out of the core, moved through the refueling slot to the fuel pool, positioned over the storage rack and lowered to storage. Fuel is shuffled and new fuel is moved from the storage pool to the reactor vessel in the same manner.

9.1.4.2.10.2.5 Vessel Closure

The following steps, when performed, will return the reactor to operating condition. The procedures are the reverse of those described in the proceeding sections: Many steps are performed in parallel and not as listed.

a) Install inner fuel pool gate.

b) Core verification. The core position of each fuel assembly must be verified to assure the desired core configuration has been attained.

c) Control rod drive tests. The control rod drive timing, friction and scram tests are performed.

d) Replace separator.

e) Drain dryer-separator storage pool and reactor well.

f) Decontaminate reactor well.

g) Install service platform, bolt separator, and remove the four steam line plugs. Return the service platform and platform support to storage on refueling floor.

h) Remove drywell seal surface covering.

i) Open drywell vents, install vent piping.

j) Replace fuel pool outer gate.

k) Replace steam dryer.

l) Decontaminate dryer-separator storage pool.

m) Replace vessel studs.

n) Replace slot plugs.

o) Install reactor vessel head.

p) Install vessel head piping and insulation.

q) Replace dryer-separator canal plugs.

r) Hydro-test vessel, if necessary.

s) Install drywell head.

t) Inert reactor drywell and suppression chamber.

u) Install reactor well shield plugs.

v) Startup tests. The reactor is returned to full power operation. Power is increased gradually in a series of steps until the reactor is operating at rated power. At specific steps during the approach to power, the in-core flux monitors are calibrated.

9.1.4.2.10.3 Departure of Spent Fuel from Site

The spent fuel shipping cask arrives by railcar or truck in the railway bay of the reactor building Unit 1. It is lifted from there by the 125 ton hook of a reactor building crane through the floor hatches to the refueling floor and placed into the empty shipping cask pit between the fuel pools of Units 1 and 2.

The cask outside is decontaminated from road dirt and the lid removed by the reactor building crane. One of the inner gates of the shipping cask pit is removed. After filling of the shipping cask pool, the second gate to one of the fuel pools is removed and loading of the cask with irradiated fuel commences. The refueling platform is used to transfer fuel bundles of sufficiently low decay heat level from the spent fuel storage racks underwater into the shipping cask.

Following replacement of the cask lid, the gates to the fuel pool are inserted, the shipping cask pit drained and the cask outside decontaminated. The reactor building crane then transfers the cask from the storage pit onto the shipping vehicle where a cooling system dissipates the remaining decay heat of the fuel during transport.

9.1.4.3 Safety Evaluation

9.1.4.3.1 Spent Fuel Cask

The spent fuel cask is equipped with dual sets of lifting lugs and yokes compatible with the reactor building crane main hook, thus preventing a cask drop due to a single failure. An analysis of the spent fuel cask drop is therefore not required.

9.1.4.3.2 Reactor Building Crane

See Subsection 9.1.5.3 for the reactor building crane safety evaluation.

9.1.4.3.3 Fuel Servicing Equipment

Failure of any fuel servicing equipment listed in Table 9.1-2 poses no hazard beyond the effect of the refueling accident analyzed in Chapter 15.

Safety aspects (evaluation) of the fuel servicing equipment are discussed in Subsection 9.1.4.2.3.

9.1.4.3.4 Servicing Aids

The small manual devices listed in Table 9.1-5 facilitate underwater viewing and handling of fuel. Failure of any servicing aid does not pose any hazard beyond the effect of the refueling accident.

9.1.4.3.5 Reactor Vessel Servicing Equipment

The dryer-separator sling and the reactor vessel head strongback are both of a cruciform design providing two redundant sets of lifting points compatible with the single failure proof reactor building crane main hoist and hook. Therefore accident analysis is not required.

9.1.4.3.6 In-Vessel Servicing Equipment

Failure of any in-vessel servicing equipment listed in Table 9.1-5 poses no hazard beyond the effect of the refueling accident analyzed in Chapter 15.

9.1.4.3.7 Refueling Equipment

The most severe failure of the refueling platform and associated grapple and hoists results in the dropping of a fuel assembly onto the reactor core. This refueling accident is analyzed in Chapter 15.

Safety aspects of the refueling equipment are discussed in Subsection 9.1.4.2.7. A description of fuel transfer, including appropriate safety features, is provided in Subsection 9.1.4.2.10. In addition, the following summary safety evaluation of the fuel handling system is provided below.

The fuel prep machine removes and installs channels with all parts remaining under water. Mechanical stops prevent the carriage from lifting the fuel bundle or assembly to a height where water shielding is less than 8 feet. Irradiated channels, as well as small parts such as bolts and springs, are stored underwater. The spaces in the channel storage rack have center posts which prevent the loading of fuel bundles into this rack.

There are no nuclear safety problems associated with the handling of new fuel bundles, singly or in pairs. Equipment and procedures prevent an accumulation of more than two bundles in any location.

The refueling platform is designed to prevent it from toppling into the pools during a SSE. Redundant safety interlocks are provided as well as limit switches to prevent accidentally running the grapple into the pool walls. The grapple utilized for fuel movement is on the end of a telescoping mast. At full retraction of the mast, the grapple is eight feet below water surface, so there is no chance of raising a fuel assembly to the point where it is inadequately shielded by water. The grapple is hoisted by redundant cables inside of the mast; and is lowered by gravity. A digital readout is displayed to the operator, showing him the exact coordinates of the grapple over the core.

The mast is suspended and gimbaled from the trolley, near its top, so that the mast can be swung about the axis of platform travel, in order to remove the grapple from the water for servicing and for storage.

The grapple has two independent hooks, each operated by an air cylinder. Engagement is indicated to the operator. Interlocks prevent grapple disengagement until a "slack cable" signal from the lifting cables indicates that the fuel assembly is seated.

In addition to the main hoist on the trolley, there is an auxiliary hoist on the trolley, and another hoist on its own monorail. These three hoists are precluded from operating simultaneously, because control power is available to only one of them at a time. The two auxiliary hoists have load cells with interlocks which prevent the hoists from moving anything as heavy as a fuel bundle.

The two auxiliary hoists have electrical interlocks which prevent the lifting of their loads higher than 8 feet under water. Adjustable mechanical jam-stops on the cables back up these interlocks.

In summary, the fuel handling system complies with Regulatory Guide 1.13 (3/71), General Design Criteria 2, 3, 4, 5, 61, 62, and 63, and applicable portions of 10CFR50.

A system-level, qualitative-type failure mode and effects analyis relative to this system is discussed in Subsection 15A.6.5.

9.1.4.3.8 Storage Equipment

The safety evaluation of the new and spent fuel storage is presented in Subsections 9.1.1.3 and 9.1.2.3.

9.1.4.3.9 Under Reactor Vessel Servicing Equipment

Failure of any under reactor vessel servicing equipment poses no hazard in excess of the effects of accidents analyzed in Chapter 15.

9.1.4.4 Inspection and Testing Requirements

9.1.4.4.1 Inspection

Refueling and servicing equipment provided by the NSSS supplier is subject to the strict controls of quality assurance, incorporating the requirements of federal regulation 10CFR50, Appendix B. Components defined as essential to safety, such as the fuel storage racks and refueling platform have an additional

set of engineering specified, "quality requirements" that identify safety-related features which require specific QA verification of compliance to drawing requirements.

For components classified as American Society of Mechanical Engineers (ASME) Section III, the shop operation must secure and maintain an ASME "N" stamp, which requires the submittal of an acceptable ASME quality plan and a corresponding procedural manual.

Additionally, the shop operation must submit to frequent ASME audits and component inspections by resident state code inspectors.

Prior to shipment, every component inspection item is reviewed by QA supervisory personnel and combined into a summary product quality checklist (PQL). By issuance of the PQL, verification is made that all quality requirements have been confirmed and are on record in the product's historical file.

9.1.4.4.2 Testing

Prior to multi-unit fabrication, major pieces of refueling or servicing equipment are fabricated and tested as prototype units. These units are tested to specifications defined by the responsible design engineer and implemented by a test engineering organization. In many cases, a full design review of the product is conducted before and after the testing cycle.

Any design changes affecting function, that are made after the design review of the qualification testing has been completed, are reverified by test or calculation.

When the unit is received at the site, it is inspected by quality assurance personnel to ensure that no damage has occurred during transit or storage. Prior to site operation, the refueling or servicing equipment must undergo a sequence of preoperational functional tests, as defined by a site preoperational test specification.

There is an operation and maintenance instruction manual for each tool, that additionally requires a series of functional checks each time the unit is operated for reactor refueling or servicing.

Fuel handling and vessel servicing equipment is preoperationally tested in accordance with Chapter 14.

Tools and servicing equipment used for refueling are inspected and preoperationally performance tested prior to the plant outage.

9.1.4.5 Instrumentation Requirements

The majority of the refueling and servicing equipment is manually operated and controlled by the operator's visual observations. This type of operation does not necessitate the need for a dynamic instrumentation system.

However, there are several components that are essential to prudent operation that do have instrumentation and control systems.

9.1.4.5.1 Refueling Platform

The refueling platform has a non-safety related X-Y-Z position indicator system that informs the operator which core fuel cell the fuel grapple is accessing. Interlocks and control room monitor are provided to prevent the fuel grapple from operating in a fuel cell where the control rod is not in the proper orientation for refueling. Refer to Subsection 7.6.1.1 for discussion of refueling interlocks.

Additionally, there are a series of mechanically activated switches and relays that provide monitor indications on the operator's console for grapple limits, hoist and cable load conditions, and confirmation that the grapple's hook is either engaged or released.

A series of load cells are installed to provide automatic shutdown whenever threshold limits are exceeded on either the fuel grapple or the auxiliary hoist units.

9.1.4.5.2 Fuel Support Grapple

Although the Fuel Support Grapple is not essential to safety, it has an instrumentation system consisting of mechanical switches and indicator lights. This system provides the operator with a positive indication that the grapple is properly aligned and oriented and that the grappling mechanism is either extended or retracted.

9.1.4.5.3 Other

Refer to Table 9.1-5 for additional refueling and servicing equipment not requiring instrumentation.

9.1.4.5.4 Radiation Monitoring

The area radiation monitoring equipment for the refueling area is described in Subsection 12.3.4.

9.1.5 REACTOR BUILDING CRANES

Two reactor building cranes are provided for the Susquehanna SES. Unit 1 crane is a single failure-proof crane and is designed to handle the spent fuel cask. The Unit 2 crane is not single failure-proof and is designed to handle construction loads and all normal plant operation loads except the spent fuel cask.

The Unit 2 reactor building crane, rated 125 tons (main hoist), 5 tons (auxiliary hoist), is potentially capable of carrying any loads within its rated capacity, but not over or within restricted areas of the refueling floor. Limits of the restricted areas are shown on Figures 9.1-16 A & B.

Administrative controls are used to preclude the Unit 2 reactor building crane from bing used for handling the spend fuel cask when stored in the spent fuel shipping cask storage pit.

The following description will address the Unit 1 crane only, which will be referred to as the reactor building crane or the crane.

9.1.5.1 Design Bases

The main purpose of the reactor building crane is to handle the spent fuel cask between the cask transport vehicle, the cask storage pit, and the wash-down area in the reactor building. Secondary purposes of the reactor building crane include:

a) Handling loads related to maintenance and replacement of equipment from the reactor building which are received or shipped through the railroad access doors

b) Handling of shield plugs, reactor vessel and drywell heads, steam dryer and separator, etc, during refueling operations.

The reactor building crane is designed for the following ratings:

Main hoist capacity	125 tons
Auxiliary hoist capacity	5 tons
Speed of main hoist (at rated load)	5 fpm (see Note 1)
Speed of auxiliary hoist (at rated load)	20 fpm (see Note 1)
Speed of trolley (using main hoist)	10 fpm
Speed of trolley (using aux hoist)	50 fpm
Speed of bridge	50 fpm
Lift of main hook (see Note 2)	173 ft
Lift of auxiliary hook	173 ft
Crane span	130 ft
Length of runway (between stops)	323 ft
Uncontrolled drop	
Main hoist	0.5 in. (max.)
Auxiliary hoist	8.55 in. (max.)

Note 1: Minimum speed at rated load is less than 2 percent of rated speed

Note 2: Unit 2 reactor building crane ratings are identical to those of the Unit 1 crane, except for the main hook lift, which is 68 ft. This, in addition to administrative controls, precludes inadvertent use of the Unit 2 crane for spent fuel cask handling, since the main hook does not reach the spent fuel cask plant entry level.

The use of the crane main hoist is restricted over the reactor wells and prevented over the spent fuel pools.

The auxiliary hooks of both cranes are designed for use underwater, up to 50 ft depth.

9.1.5.2 Equipment Design

a) General

The reactor building crane is designed, fabricated, installed, and tested in accordance with ANSI B30.2.0, CMMA-70, and OSHA regulations.

b) Structural

The structural portions of the crane bridge and trolley are designed for (1) dead load plus rated lift load plus impact load of 15 percent of the total dead plus rated live loads, not to exceed allowable stresses; (2) dead load plus rated lift load plus a lateral load of 10 percent of the total dead plus rated live loads, not to exceed allowable stresses; (3) the operating basis earthquake (OBE) while lifting the rated load, the working stresses not to exceed 125 percent of the allowable stress; (4) the design basis earthquake (DBE) while lifting the rated load, the allowable stresses to be less than 90 percent in bending, 85 percent in axial tension, and 50 percent in shear of the material minimum yield stresses; (5) a tornado loading of 300 psf, without live load, the allowable stresses to be the same as for (4) above.

The structure of the crane bridge consists of welded box type girders with truck saddles and truck frames of welded steel construction. The trolley side frames, sheave frames, and truck frames are of structural steel welded construction.

c) Mechanical

The crane is of a single trolley top running electric overhead travelling bridge design. The general arrangement of the crane in the reactor building is shown on Figure 9.1-4.

The main hoist is provided with the following dual components preventing a single failure to result in a drop of the spent fuel shipping cask:

1) Dual sister hook (hook within a hook)

2) Dual reeving systems complete with redundant wire ropes, upper, lower, and equalizing sheaves

3) Dual main hoist gear boxes with individual braking systems.

Each wire rope has a safety factor of five against breaking while lifting the rated capacity. In case of failure of one of the two reeving systems, the dynamic load transfer to the other system will not cause the rope load to exceed one-third of the rope breaking strength.

The following holding brakes are provided:

Main hoist	Three, rated for 150 percent of the motor torque, with provision for manual operation to allow lowering of the load after a power failure
Trolley	Two, rated for 50 percent of motor torque
Bridge	One, rated for 100 percent of motor torque.

All holding brakes are ac magnet operated. In addition, the bridge is provided with a hydraulic foot operated brake.

d) Controls

Bridge and trolley	ac static stepless speed control with reversing plugging control
Hoists	dc static reversing stepless speed control including regenerative braking, with a minimum speed of less than 2 percent of the rated speed.

Operation of the crane is from the bridge mounted cab or floor. The floor operation is by pendant or radio control. Control at any one time is from one point only.

9.1.5.3 Safety Evaluation

As described in Subsection 9.1.5.2, the main hoist is provided with dual main hoist components capable of holding the load in the event of a single failure.

The reactor building crane is provided with limit switches to prevent overtravel of the bridge and trolley and stop the main and auxiliary hooks in their highest and lowest safe positions.

Two limit switches, each of different design, are provided to limit the upward movement of the main and auxiliary hoist.

Two geared limit switches are provided for the main hoist, and one for the auxiliary hoist to limit the downward movement of the respective hoists.

When the 125-ton hook is loaded or unloaded but not in parked upper position, movement of the crane bridge and/or trolley will be stopped when entering the restricted areas shown on Figures 9.1-16A & B. The following means are provided to accomplish the above:

a) A series of proximity switches mounted on the crane, adjacent to the crane and trolley runways.

b) A series of trip bars mounted on the bridge and trolley runways are positioned to trip respective proximity switches.

c) Relays and logic systems to trip power supplies to affected drive motors, when a proximity switch is tripped. This will result in the setting of respective holding brakes and cessation of bridge or trolley movement. "Memory logic" will then allow the bridge or trolley to move in the opposite direction away from the restricted area.

The crane cannot enter the area above the spent fuel pools with any load on the main hoist. A key locked bypass switch is provided in the cab to allow the use of the main hoist over the RPV area for handling shield plugs, RPV and drywell heads, steam dryer/separator etc.

Crane overload protection is provided by an electrical cut-out on the hoist drive motor. In addition, two vane switches are provided on the equalizer to prevent the crane from lifting loads in excess of its rated capacity.

An overspeed switch activating all spring set motor brakes in the lowering direction holds the load in suspension.

See Section 3.13 for discussion of compliance with Regulatory Guide 1.104.

See Appendix 9B for a discussion of compliance with BTP ASB9-1.

The results of a failure mode and effect analysis are presented in Table 9.1-6.

The crane is safety related and a quality assurance program has been established and implemented in the design, fabrication, erection, and testing.

The crane is designed to remain on the runway in a parked and restrained position (by tornado locks) with no load attached under the following tornado wind loadings:

a) 300 psf on the windward crane girder

b) ±200 psf on the leeward crane girder.

The crane mechanical and structural components are qualified to Seismic Category I requirements. The crane, however, may become and remain inoperational after the operating basis earthquake, but no parts or the load will dislodge or fall. Manual towering of the main hoist load is provided.

9.1.5.4 Inspection and Testing Requirements

Crane components tests are performed during the crane fabrication as follows:

a)	Each hook:	Ultrasonic tests
		200 percent load test followed by dimensional check
		Dry powder magnetic particle test
b)	Wire rope:	Rope sample destructive breaking test
c)	Gears, gear pinions, swivels, load block frames, hook trunnions, seismic restraints, and tornado locks:	Magnetic particle tests
d)	Major structural welds:	100 percent magnetic particle testing.

The crane hoists, trolley, and bridge drives are operated in the shop to demonstrate their operability and the trolley tracking.

After the crane is erected, it is thoroughly tested, including the crane rating test in accordance with ANSI B30.2.0.

The crane periodic operational tests are in accordance with applicable OSHA regulations, local codes, and ANSI B30.2.0.

9.1.5.5 Instrumentation Requirements

The crane is furnished with dual devices and controls, as described in Subsection 9.1.5.3, to prevent or detect a single crane failure and thus preclude dropping of the spent fuel cask.

9.1.6 REFERENCES

9.1-2 "CHEETAH-B Manual, LEAHS Nuclear Fuel Management and Analysis Package, "Control Data Corporation, Publication Number 84006200, Minneapolis, Minnesota, (1974).

9.1-3 R. F. Barry, "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094, "WCAP-3741, Westinghouse Electric Corporation (1963).

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9.1-5 W. R. Cadwell, "PDQ-7 Reference Manual", WAPD-TM-678, January, 1967.

9.1-6 L. M. Petrie and N. F. Cross, "KENO-IV - An Improved Monte Carlo Criticality Program," ORNL-4938, November, 1975.

9.1-7 N. M. Greene, J. L. Lucius, W. E. Ford, III, J. E. White, R. Q. Wright, and L. M. Petrie, "AMPX - A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B", ORNL-TM-3706, 1974.

9.1-8 Design and Fabrication Criteria for Susquehanna FSAR.

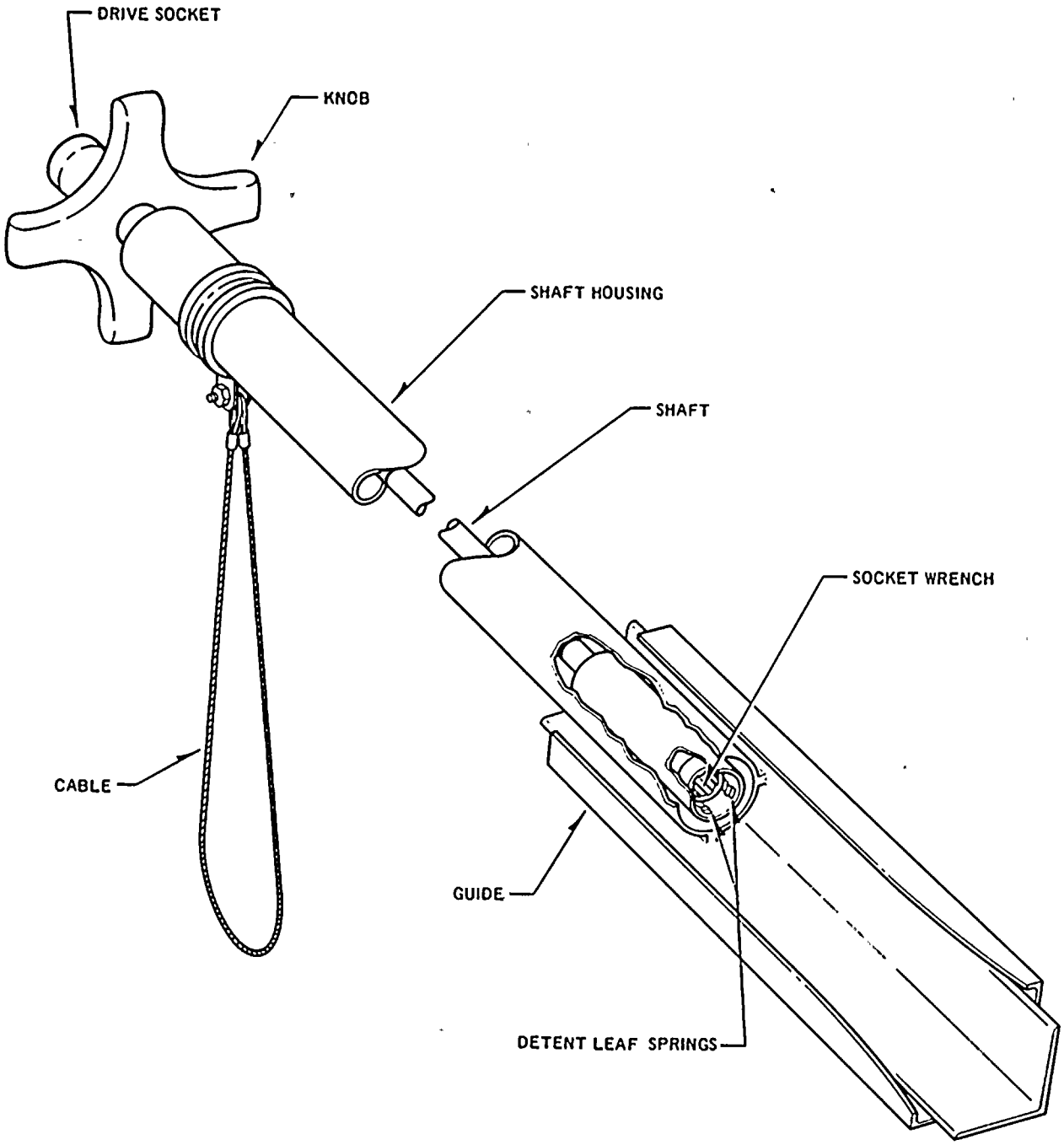
9.1-9 PARSP/3157, P. 7-1 and Appendix I.

9.1-10 Summary Report, Nuclear Criticality Analysis for the Spent Fuel Racks of the Susquehanna Power Plant; Nuclear Associates International, DR-3157-3, Report NAI78-15, May 15, 1979.

TABLE 9.1-1

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM COMPONENT DESCRIPTION

COMPONENT	EQUIPMENT NOS.	TYPE	QUAN-TITY	SIZE, EACH	MATERIAL	FLOW EACH	TDH, FT	PUMP POWER HX CAPACITY EACH	DESIGN PRESSURE/ TEMP. PSIG/°F
Fuel Pool Cooling Pumps	1P-211A,B,C	Horiz. Centr.	3	-	SS	600 gpm	200	60 hp	150/155
Fuel Pool Cooling Pumps	2P-211A,B,C	Horiz. Centr.	3	-	SS	600 gpm	200	60 hp	150/155
Fuel Pool F/D Holding Pump	OP,1P,2P-205	Horiz. Centr.	3	-	SS	160 gpm	45	5 hp	150/200
Fuel Pool F/D Precoat Pump	OP-201	Horiz. Centr.	1	-	SS	475 gpm	65	15 hp	150/200
Fuel Pool Skimmer Surge Tank	1T-208	Vert. Cyl.	1	7850/5050 gal	SS	-	-	-	15/200
Fuel Pool Skimmer Surge Tank	2T-208	Vert. Cyl.	1	7850/5050 gal	SS	-	-	-	15/200
Fuel Pool F/D Resin Feed Tank	OT-202	Vert. Cyl.	1	235/188 gal	SS	-	-	-	Atm/150
Fuel Pool F/D Precoat Tank	OT-201	Vert. Cyl.	1	500/360 gal	SS	-	-	-	Atm/150
Fuel Pool Filter Demineralizer	OF,1F,2F-202	Vert. Cyl. Pressure Precoat	3	325 ft^2	SS	650 gpm	-	-	150/200
Fuel Pool Heat Exch.	1E-202A,B,C	Shell and Straight	3	1310 ft^2	Shell & channels:CS	Shell: 296000 lb/hr	-	$4.4x10^6$ Btu/hr at	150/220
Fuel Pool Heat Exch.	2E-202A,B,C	Tubes, Fixed Tube Sheets, Counter Flow	3	1310 ft^2	Tubes & Tube-Sheets: SS	Tubes: 496000 lb/hr	-	125/110°F Shell 95/104°F Tubes	150/200



SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

CHANNEL BOLT WRENCH

FIGURE 9.1-9

APPENDIX 9A

ANALYSIS FOR NON SEISMIC SPENT FUEL POOL COOLING SYSTEMS

As described in Subsection 9.1.3 the Spent Fuel Pool (SFP) Cooling Systems are designed as non-seismic Category I, Quality Group C systems. The following analysis examines the consequences of a loss of SFP cooling.

Since the cooling systems for both units are cross-connected and in close proximity it was assumed that a seismic event causes the loss of cooling to both spent fuel pools. In addition, in order to maximize both the heat loads and the iodine inventories in the pools, sequential refuelings were postulated. The loss of cooling was assumed during the second refueling, just after the refueling cavity water level was lowered and the seismically qualified RHR system would not be available for cooling the cavity and SFP. The analysis involved an evaluation of the time to pool boiling, the capability to add makeup water if the pool boils, and the thyroid dose consequences at the LPZ boundary due to iodine releases from the boiling pools.

The assumptions used in this analysis were consistently chosen to be the "worst case" design basis assumptions, similar to those in Regulatory Guides for design basis accidents (e.g. RG 1.3, 1.25, etc.). The combination of all of these design basis assumptions occurring at the same time would be extremely unlikely, making this accident as analyzed, one of very low probability. Many of the assumptions are considered to be overly conservative. For example, operating experience with present BWR fuels (Reference 9A-1) indicates that the assumption of 1% of the fuel with cladding failures is at least a factor of 100 too conservative for 8 X 8 fuel bundles. Spiking factors are yet to be observed for a temperature rise in S.F.P.s. The assumption of 10% of the activity in the fuel gaps is at least 30 times the expected gap activity as discussed in Chapter 15, and 5 times the gap activity values used in the Rasmussen Report (Wash 1400). A more realistic evaluation of this accident would result in releases of radioactivity, if any, many orders of magnetude below the calculated values. The realistic releases would be well below the Appendix I technical specifications, indicating that such an incident is of little or no consequence.

The conservative results showed that the pools would not boil until at least 25 hours after the loss of cooling. If cooling is not restored before the pool boils, then makeup water from the Category I Emergency Service Water System can be added to the pool to keep the fuel covered at all times.

3

3 As shown in Table 9A-1 the thyroid dose consequences of the boiling pool are well below the guideline values of 10CFR100 and the 1.5 REM thyroid guideline of Regulatory Guide 1.29.

The following assumptions were used to calculate the heat generation and boiling rates in the two spent fuel pools.

1. Each pool contains the maximum fuel inventory of 15 quarter cores. Fourteen of the quarter cores were unloaded yearly from 1 to 14 years earlier. The last quarter cores are from the just completed sequential refuelings. For Unit 1 the fuel has decayed for 10.5 days, the length of time from shutdown until the water level in the refueling pool has been lowered and the RHR system would not be able to cool the refueling and spent fuel pools. For Unit 2 the decay time is 13.5+10.5 = 24 days. The 13.5 days is the minimum time to complete a fuel unloading and loading. Actual refuelings indicate that both of these times will result in the maximum heat generation rates and maximum evaporation rates at times when the RHR will not be available.

2. The decay heat was calculated using the decay heat curves from the proposed standard ANS-5.1 (10/73), corrected for a finite operating time. An uncertainty factor of 25% was applied to the calculated decay heat for all times > 10^7 sec, and 10% for $10^3 < t < 10^7$ sec. The SRP 9.2.5 methodology of applying the uncertainty factor only to the first term of the fission product decay equations (SPR Eq. 2) was not extrapolated to decay times beyond 10^7 sec because it gives unrealistic results. For example, the SPR equation says that 14 yr old spent fuel generates over 50% of the decay heat that 1 yr old fuel generates, whereas using no uncertainty factor results in the 14 yr old fuel generating about 5%. Thus the 25% uncertainity factor results in a factor of 10 conservatism. The decay heat generation rate for each pool is given in Table 9A-2 for various times after the postulated loss of cooling. 3

3. All heat generated by the fuel is assumed to be absorbed by the water in order to minimize the time to boiling. No heat is lost to the surroundings by conduction through the concrete and steel, or by evaporation. The temperature gradients from the fuel at the bottom of the pool to the cooler water at the top will create convective water and heat currents which should thoroughly mix the water, and promote an even distribution of heat rather than localized points of surface boiling.

3

The following assumptions were used to calculate the offsite doses for the loss of cooling to the spent fuel pools.

a. The saturation inventory of I-131 in the 3440 MWt core is 8.66×10^7 Ci.

b. During refueling 184 fuel elements (approximately 1/4 core) are removed and transferred to the SFP. Iodine in fuel from past refuelings will be negligible due to the long decay times.

c. It is assumed that 1% of the fuel rods in the core are defective and that this 1% is in the 1/4 core transferred to the SFP.

d. The iodine activity in the SFP water at the initiation of boiling is assumed to be negligible compared to the activity released from the fuel during pool boiling. Activity in the core coolant or from a shutdown spike would have been cleaned up to acceptably low levels by the RWCU and SFP Cleanup Systems before fuel transfer began.

e. The SFP cooling systems are assumed to fail 24 days after shutdown from the first reactor and 10.5 days after shutdown for the second. The 10.5 days is the time after a complete fuel transfer when the water level in the refueling pool will be lowered, and the RHR system would not be able to cool the refueling and spent fuel pools in case of an accident.

f. The gap activity, or 10% of the rod activity is available for leakage from the defective 1% of the rods. The leakage rate was assumed to be 8.1×10^{9} Ci/sec, which corresponds to a release rate of 700 Ci/sec for I-131. This is the full power design fuel leak rate. It should be noted that the available activity in the gaps of the defective fuel rods may have already been significantly depleted by the shutdown spike.

g. A constant spike factor of various magnitudes up to 100 was applied to the I-131 leakage rate from the fuel to account for the potential spiking effects during the temperature transient. The leakage rate returns to the normal full power unspiked rate of 8.1×10 Ci/sec when boiling begins, since the fuel should now be close to its new steady state temperature.

A comparison with Reference 9A-2 shows that the measured I-131 release rate at 9 days after shutdown is approximately .2 to .3 of the at power release rate. Since the temperature of the fuel during boiling is expected to be well below reactor operating temperature, the use of the "at power" leakage rate is considered to be extremely conservative.

h. The activity released from the fuel is assumed to be uniformily mixed in the 45,300 ft^3 (2.83×10^6 lb mass) of water in each SFP.

i. The activity release rate from the pool depends on the evaporation (boiling) rate. No evaporation was assumed during the heatup period until the pool water reaches 212°F. All heat generated by the fuel was assumed to be absorbed by the water and no losses were assumed through the concrete and steel. This results in the shortest time to boiling. The heat generation and evaporation rates after boiling begins are given as a function of time in Table 9A-2.

j. The iodine partition factor at the pool surface was varied between .1 and .01.

k. No credit was taken for iodine plateout on walls and equipment or washout by condensing water vapor in the refueling area.

3

l. The atmospheric dispersion factors for dilution of the radioactive releases are the same as those used in the Chapter 15 accident analysis. These 5th percentile ground level x/Q's are given below for the LPZ boundary distance. The time zero is assumed to be the start of the accident when pool cooling is lost.

Time	X/Q (Sec/M^3)
0-8 hrs	2.2×10^{-5}
8-24 hrs	2.8×10^{-6}
1-4 days	1.43×10^{-6}
4-30 days	1.08×10^{-6}

m. The thyroid dose models and breathing rates given in Regulatory Guide 1.3 were used. 3

The following model was used to calculate the offsite thyroid doses from the release of I-131 from the fuel.

1. The activity in the fuel available for leakage at the loss of cooling, S(o) was calculated using the reactor inventory equation from TID-14844 with the appropriate decay from shutdown until the loss of cooling, and the fractions of iodine available for release. During the pool heatup and boiling phases the activity in the fuel gaps available for leakage, S(t), was adjusted for decay and losses by leakage to the pool.

$$S(t) + S(o)\ e^{-(\lambda_d + \lambda_e)t} \qquad \text{(Eq. 9A-1)}$$

Where λ_d = Decay Lambda (1/sec)

λ_e = Leakage rate from the fuel (1/sec)

t = Time (sec)

2. The activity in the SFP as a function of time, A(t), is given by the solution to the following differential equation

$$\frac{dA(t)}{dt} = \lambda_e\ S(t) - (\lambda_d + \lambda_{ev})\ A(t)\ \text{(Ci/Sec)} \qquad \text{(Eq. 9A-2)}$$

Where λ_{ev} = Evaporation Lambda from the pool (1/sec)

Since SFP makeup water will be available, the evaporation lambda is found by dividing the evaporation rate (lb/sec) by the constant pool water mass (lb)

3. The activity released to the atmosphere over a time interval t_1 to t_2, $R(t_1\ t_2)$, was found by the solution to the following equation.

$$R(t_1\ t_2) = (PF) \int_{t_1}^{t_2} \lambda_e\ A(t) \quad \text{(Ci)} \qquad \text{(Eq. 9A-3)}$$ 3

Where:

PF = Iodine partition factor at the pool surface

3

For any time interval where all the parameters are kept constant the release in curies is given by:

3

$$R(t_1\ t_2) = \frac{(1/PF)\ \lambda_e \lambda_{ev}\ S(t_1)}{\lambda_d+\lambda_{ev}) - (\lambda_d + \lambda_e)}$$

$$\frac{1-e^{-(\lambda_d+\lambda_e)(t_2-t_1)}}{(\lambda_d+\lambda_e)} - \frac{1-e^{-(\lambda_d+\lambda_{ev})(t_2-t_1)}}{(\lambda_d+\lambda_{ev})} \qquad \text{(Eq. 9A-4)}$$

4. The thyroid dose at the LPZ is calculated using the equations and models from Regulatory Guide 1.3.

10.3 MAIN STEAM SUPPLY SYSTEM

The main steam supply system for this BWR cycle extends from the outermost containment isolation valve up to but not including the turbine stop valves and includes connected piping of 2 1/2 inches nominal diameter or larger up to and including the first valve that is either normally closed or is capable of automatic closure during all modes of reactor operation.

10.3.1 DESIGN BASES

The main steam supply system has no safety-related function and is designed to:

1) Deliver the required steam flow from the reactor to the turbine generator, at rated temperature and pressure, over the full range of operation from turbine warm up to valves wide open (VWO).

2) Provide motive steam to the steam jet air ejectors.

3) Provide steam for the steam seal evaporator and driving steam for reactor feed pump turbines.

4) Provide steam for the off gas recombiner.

5) Bypass reactor steam to the condensers during startup and any time the quantity of steam produced by the reactor is more than is required by the turbine generator.

10.3.2 DESCRIPTION

The design pressure/temperature rating of the main steam piping is 1230 psig at 585 degrees F. The piping is designed and tested according to ASME Section III, Class 2, and it is fabricated of seamless carbon steel (the 24 inch lines are SA106 Grade C, all other sizes are SA106 Grade B).

There are four 24 inch nominal main steamlines supplying steam to the turbine generator. Each line is provided with a drain downstream of the outermost containment isolation valve. The drains are routed to the condenser through a common 3 inch header. These drains and connections to each main steamline betweem the inboard and outboard isolation valves also tie into the main steam isolation valve leakage control system (see Section 6.7). Each main steamline is also provided with low point

drains consisting of a drip leg which, under normal operation, collects moisture and drains it to the condenser through a normally open valve and a restricting orifice. Each drip pot is provided with high and low level switches which operate another motorized drain valve that is normally closed and is installed in parallel to the normally open valve described above. On high level the level switch opens the motorized valve and drains the moisture directly to the condenser. When the level in the drip leg has been lowered sufficiently the low level switch closes the valve.

Pressure equalizing lines, 24 inch nominal size, branch from each main steamline and connect to a 24 inch nominal header which ties into the bypass valve chest through two 18 inch nominal lines. The 24 inch header is provided with a drip pot similar to that described for the main steamlines. The main steam supply to the reactor feed pump turbines originates from this 24 inch header.

See Figure 10.4-1 for details of the above description. For details of piping downstream of the turbine stop and control valves see Section 10.2.

During normal plant operation the turbine control valves and bypass valves are controlled by the two pressure regulators furnished by the turbine vendor. These two regulators are essentially identical and are installed in one of the four main steamlines in accordance with the turbine vendor's instructions. The regulator with the lowest set point will be the controlling regulator until it fails, then the other regulator which is biased approximately 10 psi higher will take over. A pressure transmitter is installed in one of the main steam lines, the readings from which are recorded in the control room.

10.3.3 EVALUATION

The main steamlines (MSL) from the outer isolation valves up to and including the turbine stop valves and all branch lines 2-1/2 inches in diameter and larger, up to and including the first valve (including their restraints) are designed by the use of an appropriate dynamic seismic-system analysis to withstand the Operating Bases Earthquake (OBE) and Safe Shutdown Earthquake (SSE) design loads in combination with other appropriate loads, within the limits specified for Class 2 pipe in the ASME Section III. The mathematical model for the dynamic seismic analyses of the MSL and branch line piping includes the turbine stop valves and piping beyond the stop valves including the piping to the turbine casing. The dynamic input loads for design of the main steamlines are derived from a time history model analysis or an equivalent method, as described in Section 3.9, of the Containment, Reactor Building, Turbine Building and turbine

pedestal. The Turbine Building, housing the main steamlines, may undergo some plastic deformation under the SSE, however, the plastic deformation is limited to a ductility factor of 2 and an elastic multi-degree-of-freedom system analysis is used to determine the input to the main steamlines. The stress allowable and associated deformation limits for piping are in accordance with ASME Section III, Class 2 requirements for the OBE and SSE loading combinations. The main steamline supporting structures (those portions of the Turbine Building) are such that the main steamlines and their supports can maintain their integrity within the ASME Section III, Class 2 requirements under the Seismic Category I loading conditions. The pipe supports for the main streamline meet the requirements of ASME Section III 1971 Edition thru winter 1972 Addenda, for materials, fabrication and inspection. 17

Between the outermost isolation valves and the turbine stop valves the four main steamlines are routed within the confines of a tunnel. Temperature elements are located at each end of this tunnel and the readings from these are fed into a temperature differential switch. The purpose of these temperature elements is to detect a failure of any of the main steamlines. This would be indicated by an increase in the temperature differential which would be sensed and an alarm initiated.

For details of the analysis of postulated high energy lines failure refer to Section 3.6.

10.3.4 INSPECTION AND TESTING REQUIREMENTS

The main steamlines are fabricated, examined and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 2.

During normal operation each turbine stop valve is tested daily to verify that it functions correctly. Similarly each bypass valve is tested weekly. The preoperational and inservice inspection of the main steamlines is described in Section 6.6. The preoperational and inservice inspection of the steamline isolation valves is described in Subsection 5.2.4.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

10.3.5 WATER CHEMISTRY (PWR)

Not applicable

10.3.6 STEAM AND FEEDWATER SYSTEM MATERIALS

10.3.6.1 Fracture Toughness

The main steam and feedwater piping are not impact tested. The penetrations of these lines through the primary containment, from the isolation valves outside the containment, are charpy, V-notch, or drop weight tested (See Subsection 3.1.2.5.4).

10.3.6.2 Material Selection and Fabrication

1) Materials used in the main steam and feedwater systems, SA-155 KC70 and SA-106, Grade C, are listed in Appendix I to Section III of the ASME Code.

2) There are no austenitic stainless steel components in these systems.

3) The cleaning and handling Class 2 and 3 components will be performed in accordance with cleanliness Specification 8850-M-167 which complies with the requirements of Regulatory Guide 1.37, March 16, 1973 and ANSI N45.2.1-73.

4) There is no low alloy steel in these systems.

5) Exceptions to Regulatory Guide 1.71 are described in Section 3.13

11.1 SOURCE TERMS

General Electric has evaluated radioactive material sources (activation products and fission product release from fuel) in operating boiling water reactors (BWRs) over the past decade. These source terms are reviewed and periodically revised to incorporate up-to-date information. Release of radioactive material from operating BWRs has generally resulted in doses to offsite persons which have been only a small fraction of permissible doses, or of natural background dose.

The information provided in this section defines the design basis radioactive material levels in the reactor water, steam and off-gas. The various radioisotopes listed have been grouped as coolant activation products, non-coolant activation products, and fission products. The fission product levels are based on measurements of BWR reactor water and off-gas at several stations through mid-1971. Emphasis was placed on observations made at KRB and Dresden 2. The design basis radioactive material levels do not necessarily include all the radioisotopes observed or predicted theoretically to be present. The radioisotopes included are considered significant to one or more of the following criteria:

(1) plant equipment design,

(2) shielding design,

(3) understanding system operation and performance,

(4) measurement practicability, and

(5) evaluating radioactive material releases to the environment.

For halogens, radioisotopes with half-lives less than 3 minutes were omitted. For other fission product radioisotopes in reactor water, radioisotopes with half-lives less than 10 min. were not considered.

11.1.1 FISSION PRODUCTS

11.1.1.1 Noble Radiogas Fission Products

The noble radiogas fission product source terms observed in operating BWRs are generally complex mixtures whose sources vary from miniscule defects in cladding to "tramp" uranium on external cladding surfaces. The relative concentrations or amounts of noble radiogas isotopes can be described as follows:

Equilibrium: $R_g \sim K_1 Y$ (11.1-1)

Recoil: $R_g \sim K_2 Y \lambda$ (11.1-2)

The nomenclature in Subsection 11.1.1.4 defines the terms in these and succeeding equations. The constants k_1 and k_2 describe the fractions of the total fissions that are involved in each of the releases. The equilibrium and recoil mixtures are the two extremes of the mixture spectrum that are physically possible. When a sufficient time delay occurs between the fission event and the time of release of the radiogases from the fuel to the coolant, the radiogases approach equilibrium levels in the fuel and the equilibrium mixture results. When there is no delay or impedance between the fission event and the release of the radiogases, the recoil mixture is observed.

Prior to Vallecitos Boiling Water Reactor (VBWR) and Dresden 1 experience, it was assumed that noble radiogas leakage from the fuel would be the equilibrium mixture of the noble radiogases present in the fuel.

VBWR and early Dresden 1 experience indicated that the actual mixture most often observed approached a distribution which was intermediate in character to the two extremes (Reference 11.1-1). This intermediate decay mixture was termed the "diffusion" mixture. It must be emphasized that this "diffusion" mixture is merely one possible point on the mixture spectrum ranging from the equilibrium to the recoil mixture and does not have the absolute mathematical and mechanistic basis for the calculational methods possible for equilibrium and recoil mixtures. However, the "diffusion" distribution pattern which has been described is as follows:

Diffusion: $R_g \sim K_3 Y \lambda^{0.5}$ (11.1-3)

The constant k_3 describes the fraction of total fissions that are involved in the release. The value of the exponent of the decay constant, λ , is midway between the values for equilibrium, 0, and recoil, 1. The "diffusion" pattern value of 0.5 was originally derived from diffusion theory.

Although the previously described "diffusion" mixture was used by GE as a basis for design since 1963, the design basis release magnitude used has varied from 0.5 Ci/sec to 0.1 Ci/sec as measured after 30-min decay (t = 30 min). The noble radiogas source-term rate after 30-min decay has been used as a conventional measure of the design basis fuel leakage rate since it is conveniently measurable and was consistent with the nominal design basis 30-min off-gas holdup system used on a number of plants. Since about 1967, the design basis release magnitude used (including the 1971 source terms) was established at an annual average of 0.1 Ci/sec (t = 30 min). This design basis is

considered as an annual average with some time above and some time below this value. This design value was selected on the basis of operating experience rather than predictive assumptions. Several judgment factors, including the significance of environmental release, reactor water radioisotope concentrations, liquid waste handling and effluent disposal criteria, building air contamination, shielding design, and turbine and other component contamination affecting maintenance, have been considered in establishing this level.

Noble radiogas source terms from fuel above 0.1 Ci/sec (t = 30 min) can be tolerated for reasonable periods of time. Continual assessment of these values is made on the basis of actual operating experience in BWRs (References 11.1-2 and 11.1-3).

While the noble radiogas source-term magnitude was established at 0.1 Ci/sec (t = 30 min), it was recognized that there may be a more statistically applicable distribution for the noble radiogas mixture. Sufficient data were available from KRB operations from 1967 to mid-1971 along with Dresden 2 data from operation in 1970 and several months in 1971 to more accurately characterize the noble radiogas mixture pattern for an operating BWR.

The basic equation for each radioisotope used to analyze the collected data is:

$$R_g = K_g Y \lambda^{m(1-e^{-\lambda T})(e^{-\lambda t})} \qquad (11.1\text{-}4)$$

With the exception of Kr-85 with a half-life of 10.74 yr, the noble radiogas fission products in the fuel are essentially at an equilibrium condition after an irradiation period of several months (rate of formation is equal to the rate of decay). So for practical purposes the term (1 - e- T) approaches 1 and can be neglected when the reactor has been operating at steady-state for long periods of time. The term (e- T) is used to adjust the releases from the fuel (t = 0) to the decay time for which values are needed. Historically t = 30 min has been used. When discussing long steady-state operation and leakage from the fuel (t = 0), the following simplified form of Equation 11.1-4 can be used to describe the leakage of each noble radiogas:

$$R_g = K_g Y \lambda^m \qquad (11.1\text{-}5)$$

The constant, K_g, describes the magnitude of leakage. The relative rates of leakage of the different noble radiogas isotopes is accounted for by the variable, m, the exponent of the decay constant, λ .

Dividing both sides of Equation 11.1-5 by y, the fission yield, and taking the logarithm of both sides results in the following equation:

$$\log(R_g/Y) = m \log(\lambda) + \log(K_g) \qquad (11.1\text{-}6)$$

Equation 11.1-6 represents a straight line when log R_g/y is plotted versus log (λ); m is the slope of the line. This straight line is obtained by plotting (R_g/y) versus (λ) on logarithmic graph paper. By fitting actual data from KRB and Dresden 2 (using least squares techniques) to the equation the slope, m, can be obtained. This can be estimated on the plotted graph. With radiogas leakage at KRB over the nearly 5-yr period varying from 0.001 to 0.056 Ci/sec (t = 30 min) and with radiogas leakage at Dresen 2 varying from 0.001 to 0.169 Ci/sec (t = 30 min), the average value of m was determined. The value for m is 0.4 with a standard deviation of $\pm$0.07. This is illustrated in Figure 11.1-1 as a frequency histogram. As can be seen from this figure, variations in m were observed in the range m = 0.1 to m = 0.6. After establishing the value of $\bar{m}$ = 0.4, the value of K_g can be calculated by selecting a value for R_g, or as has been done historically, the design basis is set by the total design basis source-term magnitude at t = 30 min. With R_g at 30 min = 100,000 Ci/sec, K_g can be calculated as being 2.6 x 10^7 and Equation 11.1-4 becomes:

$$R_g = 2.6 \times 10^7 \, Y\lambda^{0.4} \, (1-e^{-\lambda T})(e^{-\lambda t}) \qquad (11.1\text{-}7)$$

This updated noble radiogas source-term mixture has been termed the "1971 Mixture" to differentiate it from the "diffusion mixture." The noble gas source term for each radioisotope can be calculated from Equation 11.1-7. The resultant source terms are presented in Table 11.1-1 as leakage from fuel (t = 0) and after 30 min decay. While Kr-85 can be calculated using Equation 11.1-7, the number of confirming experimental observations was limited by the difficulty of measuring very low release rates of this isotope. Therefore, the table provides an estimated range for Kr-85 based on a few actual measurements.

11.1.1.2 Radiohalogen Fission Products

Historically, the radiohalogen design basis source term was established by the same equation as that used for noble radiogases. In a fashion similar to that used with gases, a simplified equation can be shown to describe the release of each halogen radioisotope:

$$R_h = K_h Y \lambda^n \qquad (11.1\text{-}8)$$

The constant, K_h, describes the magnitude of leakage from fuel. The relative rates of halogen radioisotope leakage is expressed in terms of n, the exponent of the decay constant, λ. As was done with the noble radiogases, the average value was determined for n. The value for $\bar{n}$ is 0.5 with a standard deviation of $\pm$0.19. This is illustrated in Figure 11.1-2 as a frequency

histogram. As can be seen from this figure, variations in n were observed in the range of n = 0.1 to n = 0.9.

It appeared that the use of the previous method of calculating radio-halogen leakage from fuel was overly conservative. Figure 11.1-3 relates KRB and Dresden 2 noble radiogas versus I-131 leakage. While it can be seen from Dresden 2 data during the period August 1970 to January 1971 that there is a relationship between noble radiogas and I-131 leakage under one fuel condition, there was no simple relationship for all fuel conditions experienced. Also, it can be seen that during this period, high radiogas leakages were not accompanied by high radioiodine leakage from the fuel. Except for one KRB datum point, all steady-state I-131 leakages observed at KRB or Dresden 2 were equal to or less than 505 μCi/sec. Even at Dresden 1 in March 1965, when severe defects were experienced in stainless-steel-clad fuel, I-131 leakages greater than 500 μCi/sec I-131 were not experienced. Figure 11.1-3 shows that these higher radioiodine leakages from the fuel were related to noble radiogas source terms of less than the design basis value of 0.1 Ci/sec (t = 30 min). This may be partially explained by inherent limitations due to internal plant operational problems that caused plant derating.

In general, it would not be anticipated that operation at full power would continue for any significant time period with fuel cladding defects which would be indicated by I-131 leakage from the fuel in excess of 700 μCi/sec. When high radiohalogen leakages are observed, other fission products will be present in greater amounts. This may increase potential radiation exposure to operating and maintenance personnel during plant outages following such operation.

Using these judgment factors and experience to date, the design basis radiohalogen source terms from fuel were established based on I-131 leakage of 700 μCi/sec. This value, as seen in Figure 11.1-3, accommodates the experience data and the design basis noble radiogas source term of 0.1 Ci/sec (t = 30 min). With the I-131 design basis source term established, K_h can be calculated as being 2.4×10^7 and halogen radioisotope release can be expressed by the following equation:

$$R_h = 2.4 \times 10^7 \; Y\lambda^{0.5} \; (1-e^{\lambda T})(e^{-\lambda T}) \qquad (11.1\text{-}9)$$

Concentrations of radiohalogens in reactor water can be calculated using the following equation:

$$C_h = \frac{R_h}{(\lambda+B+V)M} \qquad (11.1\text{-}10)$$

Although carryover of most soluble radioisotopes from reactor water to steam is observed to be 0.1% (0.001 fraction), the observed "carryover" for radiohalogens has varied from 0.1% to about 2% on newer plants. The average of observed radiohalogen carryover measurements has been 1.2% by weight of reactor water in steam with a standard deviation of ±0.9. In the present source-term definition, a radiohalogen carryover of 2% (0.02 fraction) was used.

The halogen release rate from the fuel can be calculated from Equation 11.1-9. Concentrations in reactor water can be calculated from Equation 11.1-10. The resultant concentrations are presented in Table 11.1-2.

11.1.1.3 Other Fission Products

The observations of other fission products (and transuranic nuclides, including Np-239) in operating BWRs are not adequately correlated by simple equations. For these radioisotopes, design basis concentrations in reactor water have been estimated conservatively from experience data and are presented in Table 11.1-3. Carryover of these radio-isotopes from the reactor water to the steam is estimated to be < 0.1% (< 0.001 fraction). In addition to carryover, however, decay of noble radiogases in the steam leaving the reactor will result in production of noble gas daughter radioisotopes in the steam and condensate systems.

Some daughter radioisotopes (e.g., yttrium and lanthanum), were not listed as being in reactor water. Their independent leakage to the coolant is negligible; however, these radioisotopes may be observed in some samples in equilibrium or approaching equilibrium with the parent radioisotope.

Except for Np-239, trace concentrations of transuranic isotopes have been observed in only a few samples where extensive and complex analyses were carried out. The predominant alpha emitter present in reactor water is Cm-242 at an estimated concentration of 10^{-6} μCi/g or less, which is below the maximum permissible concentration in drinking water applicable to continuous use by the general public. The concentration of alpha-emitting plutonium radioisotopes is more than one order of magnitude lower than that of Cm-242.

Plutonium-241 (a beta emitter) may also be present in concentrations comparable to the Cm-242 level.

11.1.1.4 Nomenclature

The following list of nomenclature defines the terms used in equations for source-term calculations:

R_g = leakage rate of a noble gas radioisotope (μCi/sec)
R_h = leakage rate of a halogen radioisotope (μCi/sec)
y = fission yield of a radioisotope (atoms/fission)
λ = decay constant of a radioisotope (sec^{-1})
T = fuel irradiation time (sec)
t = decay time following leakage from fuel (sec)
m = noble radiogas decay constant exponent (dimensionless)
n = radiohalogen decay constant exponent (dimensionless)
K_g = a constant establishing the level of noble radiogas leakage from fuel
K_h = a constant establishing the level of radiohalogen leakage from fuel
C_h = concentration of a halogen radioisotope in reactor water (μCi/g)
M = mass of water in the operating reactor (g)
= cleanup system removal constant (sec^{-1})
g = grams mass

$$\beta = \frac{\text{cleanup system flowrate (g/sec)}}{M}$$

γ = halogen steam carryover removal constant (sec^{-1})

$$\gamma = \frac{\text{concentration of halogen radioisotope in steam (}\mu\text{Ci/g)}_{\text{Steamflow}}}{C_h / M}$$

11.1.2 ACTIVATION PRODUCTS

11.1.2.1 Coolant Activation Products

The coolant activation products are not adequately correlated by simple equations. Design basis concentrations in reactor water and steam have been estimated conservatively from experience data. The resultant concentrations are presented in Table 11.1-4.

11.1.2.2 Noncoolant Activation Products

The activation products formed by activation of impurities in the coolant or by corrosion of irradiated system materials are not adequately correlated by simple equations. The design basis source terms of noncoolant activation products have been estimated conservatively from experience data. The resultant concentrations are presented in Table 11.1-5. Carryover of these isotopes from the reactor water to the steam is estimated to be <0.1% (< 0.001 fraction).

11.1.3 TRITIUM

In a BWR, tritium is produced by three principal methods:

(1) activation of naturally occurring deuterium in the primary coolant,

(2) nuclear fission of UO_2 fuel, and

(3) neutron reactions with boron used in reactivity control rods.

The tritium, formed in control rods, which may be released from a BWR in liquid or gaseous effluents, is believed to be negligible. A prime source of tritium available for release from a BWR is that produced from activation of deuterium in the primary coolant. Some fission product tritium may also transfer from fuel to primary coolant. This discussion is limited to the uncertainties associated with estimating the amounts of tritium generated in a BWR which are available for release.

All of the tritium produced by activation of deuterium in the primary coolant is available for release in liquid or gaseous effluents. The tritium formed in a BWR can be calculated using the equation:

$$R_{act} = \frac{\Sigma \phi V \lambda}{3.7 \times 10^4 \, P} \qquad (11.1\text{-}11)$$

where

R_{act} = tritium formation rate by deuterium activation (μCi/sec/MWt)
Σ = macroscopic thermal neutron cross section (cm^{-1})
ϕ = thermal neutron flux (neutrons/(cm^2) (sec))
V = coolant volume in core (cm^3)
λ = tritium radioactive decay constant (1.78×10^{-9} sec^{-1})
P = reactor power level (MWt)

For recent BWR designs, R_{act} is calculated to be $1.3 \pm 0.4 \times 10^{4}$ μCi/ sec/MWt. The uncertainty indicated is derived from the estimated errors in selecting values for the coolant volume in the core, coolant density in the core, abundance of deuterium in light water (some additional deuterium will be present because of the H (n, γ) D reaction, thermal neutron flux, and microscopic cross section for deuterium).

The fraction of tritium produced by fission which may transfer from fuel to the coolant (which will then be available for release in liquid and gaseous effluents) is much more difficult to estimate. However, since zircaloy-clad fuel rods are used in BWRs, essentially all fission product tritium will remain in the fuel rods unless defects are present in the cladding material (Reference 11.1-4).

The study made at Dresden 1 in 1968 by the U.S. Public Health Service suggests that essentially all of the tritium released from the plant could be accounted for by the deuterium activation source (Reference 11.1-3). For purposes of estimating the leakage of tritium from defected fuel, it can be assumed that it leaks in a manner similar to the leakage of noble radiogases. Thus, use can be made of the empirical relationship described as the "diffusion mixture" used for predicting the source term of individual noble gas radioisotopes as a function of the total noble gas source term. The equation which describes this relationship is:

$$R_{dif} = Ky\lambda \qquad (11.1\text{-}12)$$

where,

R_{dif} = leakage rate of tritium from fuel (μCi/sec)

y = fission yield fraction (atoms/fission)

λ = radioactive decay constant (sec^{-1})

K = a constant related to total tritium leakage rate

If the total noble radiogas source term is 10^5 μCi/sec after 30-min decay, leakage from fuel can be calculated to be about 0.24 μCi/sec of tritium. To place this value in perspective in the USPHS study, the observed rate of Kr-85 (which has a half-life similar to that of tritium) was 0.06 to 0.4 times that calculated using the "diffusion mixture" relationship. This would suggest that the actual tritium leakage rate might range from 0.015 to 0.10 μCi/sec. Since the annual average noble radiogas leakage from a BWR is expected to be less than 0.1 Ci/sec (t = 30 min),

the annual average tritium release rate from the fission source can be conservatively estimated at 0.12 ± 0.12 μCi/sec, or 0.0 to 0.24 μCi/sec.

For this reactor, the estimated total tritium appearance rate in reactor coolant and release rate in the effluent is about 19 Ci/yr.

Tritium formed in the reactor is generally present as tritiated oxide (HTO) and to a lesser degree as tritiated gas (HT). Tritium concentration in the steam formed in the reactor will be the same as in the reactor water at any given time. This tritium concentration will also be present in condensate and feedwater. Since radioactive effluents generally originate from the reactor and power cycle equipment, radioactive effluents will also have this tritium concentration. Condensate storage receives treated water from the radioactive waste system and reject water from the condensate system. Thus, all plant process water will have a common tritium concentration.

Off-gases released from the plant will contain tritium, which is present as tritiated gas (HT) resulting from reactor water radiolysis as well as tritiated water vapor (HTO). In addition, water vapor from the turbine gland seal steam packing exhauster and a lesser amount present in ventilation air due to process steam leaks or evaporation from sumps, tanks, and spills on floors will also contain tritium. The remainder of the tritium will leave the plant in liquid effluents or with solid wastes.

Recombination of radiolysis gases in the air ejector off-gas system will form water, which is condensed and returned to the main condenser. This tends to reduce the amount of tritium leaving in gaseous effluents. Reducing the gaseous tritium release will result in a slightly higher tritium concentration in the plant process water. Reducing the amount of liquid effluent discharged will also result in a higher process coolant equilibrium tritium concentration.

Essentially, all tritium entering the primary coolant will eventually be released to the environs, either as water vapor and gas to the atmosphere, or as liquid effluent to the plant discharge or as solid waste. Reduction due to radioactive decay is negligible due to the 12-yr. half-life of tritium.

The USPHS study at Dresden 1 estimated that approximately 90% of the tritium release was observed in liquid effluent, with the remaining 10% leaving as gaseous effluent (Reference 11.1-5). Efforts to reduce the volume of liquid effluent discharges may change this distribution so that a greater amount of tritium will leave as gaseous effluent. From a practical standpoint, the fraction of tritium leaving as liquid effluent may vary between 60 and 90% with the remainder leaving in gaseous effluent.

11.1.4 FUEL FISSION PRODUCTION INVENTORY AND FUEL EXPERIENCE

11.1.4.1 Fuel Fission Product Inventory

Fuel fission product inventory information is used in establishing fission product source terms for accident analysis and is, therefore, discussed in Chapter 15.

11.1.4.2 Fuel Experience

A discussion of fuel experience gained for BWR fuel including failure experience, burnup experience, and thermal conditions under which the experience was gained is available in three GE topical reports (References 11.1-2, 11.1-3 and 11.1-6).

11.1.5 PROCESS LEAKAGE SOURCES

Process leakage results in potential release paths for noble gases and other volatile fission products via ventilation systems. Liquid from process leaks are collected and routed to the liquid-solid radwaste system. Radionuclide releases via ventilation paths are at extremely low levels and have been insignificant compared to process off-gas from operating BWR plants. However, because the implementation of improved process off-gas treatment systems make the ventilation release relatively significant, General Electric has conducted measurements to identify and qualify these low-level release paths. General Electric has maintained an awareness of other measurements by the Electric Power Research Institute and other organizations; and routine measurements by utilities with operating BWRs.

Leakage of fluids from the process system will result in the release of radionuclides into plant buildings. In general, the noble radiogases will remain airborne and will be released to the atmosphere with little delay via the building ventilation exhaust ducts. The radionuclides will partition between air and water, and airborne radioiodines may "plateout" on metal surfaces, concrete, and paint. A significant amount of radioiodine remains in air or is desorbed from surfaces. Radioiodines are found in ventilation air as methyl iodide and as inorganic iodine which is here defined as particulate, elemental, and hypoiodous acid forms of iodine. Particulates will also be present in the ventilation exhaust air.

Experience with the airborne radiological releases from BWR building heating, ventilating, and air conditioning and the main condenser mechanical vacuum pump have been compiled and evaluated in NEDO-21159, "Airborne Releases from BWRs for Environmental Impact Evaluations", March 1976, Licensing Topical Report

(Reference 11.1-7). This report is periodically updated to incorporate the most recent data on airborne emission. The results of these evaluations are based on data obtained by utility personnel and special in-plant studies of operating BWR plants by independent organizations and the General Electric Company. An evaluation of the radioactive releases from ventilation systems, for compliance with Appendix I to 10CFR50, is given in Section 11.3. An evaluation of important exposure to airborne activity is given in Subsection 12.2.2.

11.1.6 OTHER RELEASES

All other releases are covered in Section 11.3.

11.1.7 REFERENCES

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CHAPTER 12

RADIATION PROTECTION

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CHAPTER 12

TABLES

12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE (ALARA)

12.1.1 POLICY CONSIDERATIONS

12.1.1.1 Management Policy

It is the policy of PP&L to maintain occupational radiation exposure As Low As Reasonably Achievable (ALARA) at the Susquehanna SES. This includes maintaining the annual dose to individuals working at the station ALARA, and keeping the annual integrated dose to station personnel ALARA. The management of this Company is firmly committed to performing all reasonable actions to ensure that radiation exposures are maintained ALARA.

Subsection 12.1.2 and Section 12.3 discuss the ALARA considerations that have been incorporated into the design of the Susquehanna SES.

The Susquehanna SES will be operated and maintained in such a manner as to ensure occupational radiation exposures (ORE) are ALARA. The operational ALARA program is described in Section 12.5. Training programs will be established to assure personnel understand both why and how occupational radiation exposures will be maintained ALARA. A corporate ALARA Review Committee has been established to ensure implementation of ALARA policy by various program reviews.

12.1.1.2 Management Responsibilities

Figures 17.2-2 and 13.1-1 exhibit the management organizational structure for the Susquehanna SES.

The Vice President - System Power and Engineering has the corporate responsibility for the ALARA Program. The responsibility for coordination and administration of the ALARA Program is assigned through the Manager-Power Producton to the Manager-Nuclear Support. This individual is responsible to determine that the policies and commitments contained in the PP&L ALARA Program are being properly implemented.

During the design and construction phase, the Susquehanna SES Project Manager is responsible to ensure that the design and construction of the facility is such that occupational exposures will be ALARA. This will include ensuring that, to the extent practicable:

a. Design concepts and station features reflect consideration of the activities of station personnel that might be anticipated and that might lead to personnel exposure to substantial sources of radiation and that station design features have been provided to reduce the anticipated exposures of station personnel to these sources of radiation.

b. Specifications for equipment reflect the objectives of ALARA, including among others, considerations of reliability, serviceability and limitations of internal accumulations of radioactive material.

During the startup and operation phase, the Superintendent of Plant is responsible for controlling radiation exposure in a manner consistent with ALARA requirements and is specifically responsible for the onsite radiation protection program. His responsibilities with respect to the ALARA Program include: ensuring support from all station personnel, participating in the selection of specific goals and objectives for the station, supporting the Health Physics Supervisor in formulating and implementing the station ALARA Program, and expediting the collection and dissemination of data and information concerning the program to the corporate management.

The ALARA responsibilities of the Superintendent of Plant are implemented through the Health Physics Supervisor who, in accordance with ALARA principles, develops the Health Physics Program and Procedures, reviews other applicable station procedures, and estimates and monitors personnel exposures.

Major ALARA responsibilities of the Health Physics Supervisor include the following:

a. Participating in reviews of design changes for facilities and equipment that can affect potential radiation exposures;

b. Identifying locations, operations, and conditions, that have the potential for causing significant exposures to radiation;

c. Initiating and implementing and exposure control program which includes the establishment of manrem goals,

d. Developing plans, procedures, and methods for keeping radiation exposures of station personnel ALARA:

e. Reviewing, commenting on, and recommending changes in applicable procedures to maintain exposures ALARA;

f. Developing or participating in the development of appropriate Health Physics training programs related to work in radiation areas or involving radioactive material;

g. Supervising the radiation surveillance program to maintain data on exposures of and doses to station personnel by specific job functions and type of work;

h. Supervising the collection, analysis, and evaluation of data and information attained from radiological surveys and monitoring activities;

i. Supervising, training, and qualifying the radiation protection staff of the station; and

j. Ensuring that adequate radiation protection coverage is provided for station personnel during all working hours.

Chapter 13 provides additional information concerning responsibilities and reporting relationships at the Susquehanna SES.

12.1.1.3 Policy Implementation

The management ALARA policy is implemented at the Susquehanna SES by the Health Physics Staff under the direction of the Superintendent of Plant and Health Physics Supervisor. The policy implementation is formalized by the incorporation of ALARA philosophy and considerations into permanent plant procedures dealing specifically with ALARA concerns. The operational ALARA considerations identified in Subsections 12.1.3 and 12.5.3.2 are implemented by these procedures.

Subsection 12.5.3.7 describes the training program established to give appropriate station personnel the necessary knowledge to understand why and how they should maintain their ORE ALARA.

The ALARA Review Committee has been established to review the implementation of the Company ALARA Program. Specific responsibilities of the ALARA review Committee include:

a. Ensuring that the corporate ALARA program integrates management philosophy and regulatory requiremetns and is maintained with specific goals and objectives for implementation;

b. Ensuring that an effective measurement system is established and used to determine the degree of success achieved by

station operations with regard to the ALARA goals and specific objectives;

c. Ensuring that the measurement system results are reviewed on a periodic basis and that corrective action is taken when attainment of the specific objectives appears to be jeopardized;

d. Ensuring that the authority for providing procedures and practices by which the specific goals and objectives will be achieved is delegated;

e. Ensuring that the resources needed to achieve ALARA goals and objectives are made available; and

f. Periodically review a sampling of permanent plant procedures concerning ALARA.

12.1.2 DESIGN CONSIDERATIONS

This subsection discusses the methods and features by which the policy considerations of Subsection 12.1.1 are applied. Provisions and designs for maintaining personnel exposures as low as reasonably achievable are presented in Subsections 12.3.1, 12.3.2 and 12.5.3.

Experiences and data from operating plants are evaluated to decide if and how equipment or facility designs could be improved to reduce overall plant personnel exposures. During plant design, operating reports and data such as that given in WASH 1311, NUREG-75/032, NUREG-109 and Compilation and Analysis of Data on occupational Radiation Exposure Experienced at Operating Nuclear Power Plants, AIP, September 1974, References 12.1-1, thru 12.1-4 respectively, are reviewed to determine which operations, procedures or types of equipment were most significant in producing personnel exposures. Methods to mitigate such exposures are implemented wherever possible and practicable.

12.1.2.1 General Design Considerations for ALARA Exposures

General design considerations and methods employed to keep in-plant radiation exposures ALARA have two objectives:

a) Minimizing the necessity for and amount of personnel time spent in radiation areas; and

b) Minimizing radiation levels in routinely occupied plant areas and in the vicinity of plant equipment expected to require personnel attention.

Both equipment and facility designs are considered in keeping exposures ALARA during plant operations including normal operation, maintenance and repairs, refueling operations and fuel storage, in-service inspection and calibrations, radioactive waste handling and disposal, and other events of moderate frequency. The actual design features used are described in Subsection 12.3.1.

12.1.2.2 Equipment General Design Considerations for ALARA

The following equipment general design considerations to minimize the necessity for and amount of personnel time spent in a radiation area include, where practicable:

a) Reliability, durability, construction, and design features of equipment, components, and materials to reduce or eliminate the need for repair or preventive maintenance;

b) Servicing convenience including ease of disassembly and modularization of components for replacement or removal to a lower radiation area for repair;

c) Provisions, where practicable, to remotely or mechanically operate, repair, service, monitor, or inspect equipment; and

d) Redundancy of equipment or components to reduce the need for immediate repair when radiation levels may be high and when no feasible method is available to reduce radiation levels.

The following equipment general design considerations directed toward minimizing radiation levels proximate to equipment or components requiring personnel attention include, where practicable:

a) Provision for draining, flushing, or, if necessary, remote cleaning of equipment containing radioactive material;

b) Design of equipment, to minimize the buildup of radioactive material and to facilitate flushing of crud traps;

c) Utilization of high quality valves, valve packings, and gaskets to minimize leakage and spillage of radioactive materials;

d) Provisions for minimizing the spread of contamination into equipment service areas; and

e) Provisions for isolating equipment from radioactive process fluids.

12.1.2.3 Facility Layout General Design Considerations for ALARA

The following facility general design considerations to minimize the amount of personnel time spent in a radiation area include where practicable:

a) Locating equipment and instruments, which will require routine maintenance, calibration, or inspection for ease of access and a minimum of required occupancy time in radiation fields;

b) Arranging plant areas to allow remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment; and

c) Providing, for transportation of equipment or components requiring service to a lower radiation area.

Facility general design considerations directed toward minimizing radiation levels in plant access areas and in the vicinity of equipment requiring personnel attention include, where practicable:

a) Separating radiation sources and occupied areas (eg, pipes containing potentially highly radioactive fluids do not pass through normally occupied areas);

b) Providing adequate shielding between radiation sources and access and service areas;

c) Locating appropriate equipment, instruments, and sampling sites in the lowest practicable radiation zone;

d) Providing means and adequate space for using movable shielding for sources within the service area when required; and

e) Providing means (eg. curbing, drains and flush) to control contamination and to facilitate decontamination of potentially contaminated areas.

12.1.2.4 ALARA Design Review

Bechtel Power Corporation as agents for PP&L have been given the basic responsibility for the performance of the ALARA design review. PP&L provides overall coordination and input as described below.

The ALARA Design Review is conducted in accordance with the Susquehanna SES procedure for ALARA specific review. This project-unique procedure defines the purpose of the review, establishes the project ALARA review team, describes the discipline ALARA review process, the extent and format of ALARA review meetings and the method of noting and resolving ALARA design changes.

The Bechtel Reactor-Plant (Nuclear) Group is responsible for the overall coordination of the project ALARA review and interfacing between the various Bechtel project disciplines, the Bechtel Radiation Protection Staff and PP&L. One engineer from the Bechtel Nuclear Group serves as the project ALARA coordinator. An engineer is assigned from each Bechtel discipline to serve as that discipline's ALARA coordinator. The discipline ALARA coordinators interface with the project ALARA coordinator.

Each discipline coordinator ensures that cognizant discipline engineers for each system are familiar with the available guidelines provided by his discipline chief. Guidelines cover ALARA items such as valve actuators and activating devices, radioactive pipe classification systems, design of spent resin handling systems and radioactive system component equipment specifications. Other related discipline documents are provided by the Bechtel project ALARA coordinator, including design standards, data letters and information bulletins.

The Bechtel project ALARA Coordinator obtains input and expertise from other Bechtel project groups such as Civil, Cost Engineering, and Construction and from the Staff Radiation Protection and Shielding groups as required.

The project ALARA coordinator informs the discipline ALARA coordinators of the system or area to be reviewed and the review schedule. In preparation for the review team meeting the discipline ALARA coordinator has the cognizant engineers of his group review the area/system for ALARA design coordinations for this discipline and complete a form documenting the review.

At the Bechtel ALARA review meeting, which is attended by all discipline ALARA coordinators, all potential problem areas are discussed and a resolution proposed. If necessary a dose assessment is made. Resolutions can take the form of a justification of the current design. The project ALARA coordinator is responsible for expediting close out of all items and documenting the review meeting.

Where resolution of any consideration cannot be reasonably achieved within the present plant layout, schedule, and scope of work, PP&L is notified with a recommended course of action.

All discipline ALARA coordinators review design documents on a continuing basis and inform the project ALARA coordinator of any new criteria, operating experience, or direction received from the discipline chief.

In coordination with the design reviews, the project ALARA coordinator schedules and leads site visits accompanied by cognizant engineers. The site visits serve to confirm the findings of the design review and identify problems that may not have been apparent on the drawings.

The major tool used in the design review is the SSES ALARA-Specific Review Considerations Matrix and Check Matrix which identify design features which have been judged to be cost effective with respect to maintaining (Occupational Radiation Exposures) ORE-ALARA in most applications. The review consideration matrix identifies the discipline(s) responsible for the ALARA consideration and the check matrix documents review of the design by the responsible discipline(s).

At the time when the SSES design was formulated, insufficient data such as radiation levels, exposure frequency and duration was available to utilize a dose assessment as a primary design tool. The inconsistent nature of the available data also limited the use of dose assessments as a design tool. The dose assessment is a part of the ALARA review in that the components of the dose calculation (radiation level, exposure time, exposure frequency) are considered in developing the Susquehanna SES ALARA Specific Review Considerations Matrix and in the actual review itself. A formal quantification of any potential dose reduction and its cost effectiveness is not performed as part of the review due to the late stage of design and construction, the large

variation in the benefit of any potential dose reduction, and inadequate quantitative data on the dose reduction effectiveness of selected design features.

ALARA Design Reviews to date have resulted in the following significant design modifications:

(1) The radwaste evaporator compact skid has been modified to allow the highly radioactive evaporator bottoms, concentrate pump and associated piping to be shielded from the remaining components.

(2) Rack mounting solenoid valves, pressure regulators and filters associated with valves located in phase separator tank cells outside in a low radiation zone and removing resin inlet and flush valves from the tank areas.

In addition to intensive system/area ALARA design review, field routed small piping drawings are continually reviewed, often resulting in changes in routing, valve and operator types, and connection points.

12.1.3 OPERATIONAL CONSIDERATIONS

To assure that occupational radiation exposures are maintained as low as reasonably achieveable (ALARA) during the operation of Susquehanna SES specific activities will be implemented.

12.1.3.1 Procedure Development

Station procedures will be prepared, reviewed, and approved in accordance with Section 13.5.

12.1.3.1.1 ALARA Procedures

To assure adequate emphasis on the necessity to minimize personnel exposures, ALARA procedures will be prepared as a sub category of Health Physics procedures. These procedures implement considerations of such topics as ALARA Training, ALARA review of applicable Radiation Work Permits (RWP), worker feedback, special task training and evaluation of proposed changes in applicable facilities or equipment. ALARA procedures will provide the necessary basis for instruction of station

personnel in the mechanisms available to minimize personnel exposures.

12.1.3.1.2 Station Procedures

Administrative requirements will be implemented to assure that applicable procedures developed by other plant disciplines have adequately incorporated the principle of minimizing personnel exposure. Station administrative documents will describe the criteria of selection of those procedures and revisions that will be reviewed by Health Physics. Recommendations made by Health Physics will normally be resolved with the appropriate plant discipline prior to submission for final review and approval.

12.1.3.2 Station Organization

As described in Subsection 12.5.1, the Station organization provides the Health Physics Supervisor direct access to the Superintendent of Plant to assure uniform support of Health Physics and ALARA requirements. This organization will allow the Superintendent of Plant direct involvement in the review and approval of specific ALARA goals and objectives as well as review of data and dissemination of information related to the ALARA program.

The organization also provides a Health Physics Engineer who is normally free from routine Health Physics activities to implement the Station ALARA program. This individual is primarily responsible for coordination of Station ALARA activities and will routinely interface with first line supervision in radiation work planning and post job review.

12.1.3.3 Operating Experience

The Radiation Work Permit process described in Subsection 12.5.3.2 will provide a mechanism for collection and evaluation of data relating to personnel exposure. Information collated by systems and/or components and job function will assist in evaluating design or procedure changes intended to minimize future radiaiton exposures.

12.1.3.4 Exposure Reduction

Specific exposure reduction techniques that will be employed at Susquehanna SES are described in Subsection 12.5.3.2. Procedures will assure that applicable station activities are completed with adequate preparation and planning; work is performed with appropriate Health Physics recommendations and support; and results of post job data evaluation are applied to implement improvements.

In addition, the Health Physics staff, will at all times be vigilant for ways to reduce exposures by soliciting employee suggestions, evaluating orgins of plant exposures, investigating unusual exposures, and assuring that adequate supplies and instrumentation are available.

PP&L management will perform periodic reviews of station programs to assure workers are receiving adequate instruction in ALARA and Health Physics requirements. Implementation of the Health Physics program, selected procedures, and past exposure records will also be reviewed. Management will perform formal reviews of the Susquehanna SES Health Physics program at least once every three years and results will be forwarded to the Superintendent of Plant, ALARA Review Committee and appropriate members of corporate management. The results of management reviews may also include recommendations on mechanisms which may reduce personnel exposure. The Superintendent of Plant will respond to noted recommendations or deficiencies and corrective action or improvements will be verified during subsequent reviews.

12.1.4 REFERENCES

12.1-1 T.D. Murphy, WASH-1311, UC-78, A Compilation of Occupational Radiation Exposure from Light Water Cooled Nuclear Power Plants 1969-1973, USNRC Radiological Assessment Branch, May 1974.

12.1-2 T.D. Murphy, et.al., NUREG-75/032, Occupational Radiation Exposure at Light Water Cooled Power Reactors 1969-1974, USNRC Radiological Assessment Branch, June 1975.

12.1-3 T.D. Murphy, et. al., NUREG-0109, Occupational Radiation Exposure at Light Water Cooled Power Reactors 1969-1975, USNRC Radiological Assessment Branch, August 1976.

12.1-4 C.A. Pelletier, et. al., National Environmental Studies Project, Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plants, Atomic Industrial Forum, September 1974.

12.2 RADIATION SOURCES

In this section the sources of radiation that form the basis for shield design calculations and the sources of airborne radioactivity required for the design of personnel protective measures and for dose assessment are discussed and identified.

12.2.1 CONTAINED SOURCES

The shielding design source terms are based on a noble gas fission product release rate of 0.1 Ci/sec (after 30 minutes decay) and the corresponding fission, activation, and corrosion product concentrations in the primary coolant. The sources in the primary coolant are discussed in Section 11.1 and listed in Tables 11.1-1 through 11.1-5. Throughout most of the primary coolant system, activation products, principally nitrogen-16, are the primary radiation sources for shielding design. For all systems transporting radioactive materials, conservative allowance is made for transit decay, while at the same time providing for daughter product formation.

Basic reactor data and core region description used for this section are listed in Tables 12.2-1 through 12.2-5.

In this subsection the design sources are presented by building location and system. General locations of the equipment discussed in this section are shown on the shielding and zoning drawings, Figures 12.3-8 through 12.3-27. Detailed data on source descriptions for each shielded plant area are presented in Tables 12.2-38 through 12.2-40.

Shielding source terms presented in this section and associated tables are based on conservative assumptions regarding system and equipment operations and characteristics to provide reasonably conservative radioactivity concentrations for shielding design. Therefore, the shielding source terms are not intended to approximate the actual system design radioactivity concentrations.

12.2.1.1 Drywell

12.2.1.1.1 Reactor Core

The primary radiations within the drywell during full power operation are neutron and gamma radiation resulting from the fission process in the core. Tables 12.2-4 and 12.2-5 list the multigroup neutron and gamma ray fluxes at the outside surfaces of the reactor pressure vessel and the primary shield at the core midplane. The gamma fluxes include those resulting from capture or inelastic scattering of neutrons within the reactor pressure vessel and core shroud and the gamma radiation resulting from prompt fission and fission product decay.

The largest radiation sources after reactor shutdown are the decaying fission products in the fuel. Table 12.2-9 lists the core gamma sources as a function of shutdown time. Secondary sources are the structural material activation of the RPV, its internals, and the piping and equipment located in the primary containment and also the activated corrosion products accumulated or deposited in the internals of the RPV, the primary coolant piping, and other process system piping in the primary containment.

12.2.1.1.2 Reactor Coolant System

Sources of radiation in the reactor coolant system are fission products estimated to be released from fuel and activation and corrosion products that are circulated in the reactor coolant.
15| These sources are listed in Tables 11.1-1 thru 11.1-5 and their bases are discussed in Section 11.1. The nitrogen-16 concentration in the reactor coolant is assumed to be 61μ Ci/gm of coolant at the reactor recirculation outlet nozzle.

12.2.1.1.3 Primary Steam System

Radiation sources in the primary steam system piping include activation gases, principally nitrogen-16, and the corrosion and fission products carried over to the steam system.

1| The nitrogen-16 concentration in the main steam is assumed to be 100μ Ci/gm of steam leaving the reactor vessel at the main steam
1| outlet nozzle. Fission product activity corresponds to an offgas release rate of 100,000μ Ci/sec at 30 minutes delay from the

reactor steam nozzle. Partition fractions for activity into the steam system are 100 percent for

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gases, 2 percent by weight for halogens, and 0.1 percent by weight for particulates. These partition factors are applied to the reactor water concentrations as given in Table 11.1-2 through 11.1-5.

12.2.1.2 Reactor Building

12.2.1.2.1 Reactor Water Cleanup System

Radiation sources in the RWCU system consist of those radioisotopes carried in the reactor water. Nitrogen-16 is the predominant radiation source in the regenerative and nonregenerative heat exchangers and RWCU pumps and piping. The inventory of N-16 is based upon component transit times, as shown in Table 12.2-6. The main sources for the RWCU filter demineralizers, holding pumps, and the RWCU backwash receiving tank are the accumulated corrosion and fission products, based on the inlet reactor water concentrations given in Section 11.1. Table 12.2-7 provides the inventory of the accumulated isotopes in the filter demineralizer, and Table 12.2-8 provides the inventory of isotopes in the RWCU backwash receiving tank.

12.2.1.2.2 Spent Fuel Handling and Transfer

The spent fuel assemblies are the predominant source of radiation in the containment after plant shutdown for refueling. A reactor operating time necessary to establish near fission product buildup equilibrium for the reactor at rated power is used in determining the source strength. Shielding requirements for spent fuel transfer are based on the fission product activity present 72 hours after shutdown to conservatively take credit for the time elapsed prior to the initiation of refueling operations. Source terms for spent fuel are discussed in Subsection 12.2.1.3.1 and are listed in Table 12.2-9.

12.2.1.2.3 Residual Heat Removal System

The pumps, heat exchangers, and associated piping of the Residual Heat Removal (RHR) System are potential carriers of radioactive materials. For plant shutdown, the RHR pumps and heat exchanger sources result from the radioactive isotopes carried in the reactor coolant, discussed in Subsection 12.2.1.1.2, after 4 hours of decay following shutdown. The radioactive isotopic concentrations are listed in Table 12.2-10.

12.2.1.2.4 Reactor Core Isolation Cooling System

Components of the Reactor Core Isolation Cooling (RCIC) System that are potential radiation sources are the RCIC turbine and steam inlet and exhaust piping. Radioactivity in the turbine and piping is that present in the driving steam that has been extracted from the main steam system. The steam activity as discussed in Subsection 12.2.1.1.3, decayed for the appropriate transit time to the RCIC turbine, is used for the shielding calculations for this system, and is listed in Table 12.2-11.

12.2.1.2.5 High Pressure Coolant Injection System

The radiation sources for the High Pressure Coolant Injection System are the HPCI turbine and the steam inlet and exhaust piping. The steam activity, as discussed in Subsections 12.2.1.1.3, decayed for the appropriate transit times is used for the shielding of this system as shown in Table 12.2-11.

12.2.1.2.6 Core Spray Systems

Because the core spray, when testing, uses condensate from the condensate storage tank with very low radioactivity concentrations, no shielding is required.

12.2.1.3 Refueling Facilities

12.2.1.3.1 Spent Fuel Storage and Transfer

The predominant radiation sources in the spent fuel storage and transfer areas are the spent fuel assemblies. Spent fuel assembly sources are discussed in Subsection 12.2.1.2.2. For shielding design, the spent fuel pool is assumed to contain the design maximum of 2472 fuel assemblies (Section 9.1). Of these, 764 spent fuel assemblies are assumed to be from unloading an entire core with 72 hours decay; 184 assemblies are assumed to be from previous refueling operations with 360 days decay; the remaining 1524 assemblies are assumed to be from previous refuelings with 720 days decay. Fission product gamma source strengths for these decay periods are shown in Table 12.2-9.

12.2.1.3.2 Spent Fuel Pool Cooling and Cleanup System

Sources in the Spent Fuel Pool Cooling and Cleanup (SFPCC) System are primarily a result of transfer of radioactive isotopes from the reactor coolant into the spent fuel pool during refueling operations. The reactor coolant activities for fission, corrosion, and activation products (Tables 11.1-1 through 11.1-5) are decayed for the amount of time required to remove the reactor vessel head following shutdown, are reduced by operation of the RWCU system filter demineralizers following shutdown, and are diluted by the total volumes of the water in the reactor vessel, refueling pool, and spent fuel pool (see Table 12.2-12). This activity then undergoes subsequent decay and accumulation on the SFPCC filter demineralizers (see Table 12.2-13). The SFPCC filter demineralizer resins are back washed periodically into a backwash receiving tank. Shielding source terms for the backwash receiving tank are shown in Table 12.2-14.

12.2.1.4 Turbine Building

12.2.1.4.1 Primary Steam and Power Conversion Systems

Radiation sources for piping and equipment which contain primary steam are based on the radioactivity carried over into the steam from the reactor coolant and include fission product gases and halogens, corrosion and fission products, and gaseous activation products as discussed in Subsection 12.2.1.1.3. Steam density variations and the steam transit times through equipment and pipes are factored into the source term evaluation to account for volumetric dilution effects, radiological decay, and daughter product generation.

12.2.1.4.2 Condensate System

The sources in the condensate system are based on decayed main steam activities (Subsection 12.2.1.1.3). Eighty percent of the N-16 and 100 percent of the noble gases are assumed to be removed from the condensate system by the main condenser evacuation system. The gaseous activities are minor in the hotwell and negligible in the remainder of the condensate system. The hotwell is designed for a two minute holding of condensate and therefore N-16 activity at the condenser outlet is negligible. Fission products, activated corrosion products, and the daughter products from the decay of fission product gases in transit through the turbine are the inlet sources to the

condensate system. These sources, as shown in Table 12.2-15, are present in the condensate pumps and piping and accumulate on the condensate filter demineralizers. Table 12.2-16 provides the isotopic inventory for the condensate demineralizer.

12.2.1.4.3 Offgas System Recombiner

Radioactive sources in the gas treatment system originate with the noble gases and noncondensible gases removed from the main condenser, and the activity entering with the extraction driving steam to the main condenser evacuation system. The activity removed from the main condenser is based on the primary steam activity as described in Subsection 12.2.1.1.3, decayed for the total transit time to the steam jet air ejector. Eighty percent of the N-16 and 100 percent of the noble gases are assumed to be removed by the air ejector. Activity in the extraction driving steam to the air ejector is the primary steam activity as described in Subsection 12.2.1.1.3, decayed by the transit time to the air ejector. The total quantity of activity in the offgas pipe and recombiner and source term assumptions are shown in Tables 12.2-17 and 12.2-18.

12.2.1.5 Radwaste Building

12.2.1.5.1 Liquid Radwaste Systems

The radwaste system sources are radioisotopes, including fission and activation products, present in the reactor coolant. The components of the radwaste systems contain varying degrees of activity depending on the detailed system and equipment design.

The concentrations of radionuclides present in the process fluids at various locations in the radwaste systems such as pipes, tanks, filters, demineralizers, and evaporators are discussed in Section 11.2 and are listed in Tables 11.2-5 through 11.2-7. These nuclide concentrations were used in the final shielding design. Shielding for each component of the radwaste systems is based on design activity conditions as are given in Sections 11.1 and 11.2.

12.2.1.5.2 Solid Radwaste System

The liquid and solid radwastes are collected, treated, and stored in the solid radwaste facilities as discussed in Section 11.4. The radwaste volumes may be treated by evaporation, filtration, decanting, and ion-exchange treatment. The resultant volume reduced products (e.g. evaporator bottoms, filter cakes, depleted resins) are solidified, normally with concrete, for storage and offsite shipment. Liquid products (eg. evaporator distillate) may be analyzed for reuse as condensate make-up, processed as radioactive waste, or diluted and discharged.

The radwaste is solidified in either 50 cu ft cylindrical containers or 200 cu ft cubical containers, then washed to minimize external surface contaminants, and shipped or stored in concrete shielded compartments. The aforementioned operations may be accomplished utilizing remote container loading, transfer, capping facilities, and an overhead crane. Shielding of the solid radwaste areas based on the maximum activity sources at zero decay described in Table 11.2-6 and 11.4-6. Wall and slab shielding requirements in the solid radwaste area are based on radwaste containers without any external container shielding credit.

12.2.1.5.3 Ambient Charcoal Offgas Treatment System

The charcoal offgas system as described in Section 11.3 is located in the radwaste building and primarily adsorbs the noble gases and daughter products remaining in the noncondensible gases removed from the main condenser after treatment in the recombiner offgas system.

The shielding of the components is based on the transit times for formation and accumulation of noble gas daughter products collected on the particulate filters and the remaining xenon and krypton gases on the carbon beds. The gases, after charcoal treatment, pass through a post HEPA filter where remaining particulates are trapped prior to exhausting. The concentration of the activity on the piping, equipment, and particulate and charcoal filters for shield design is shown in Tables 12.2-19 through 12.2-24.

12.2.1.6 Sources Resulting from Design Basis Accidents

The radiation sources from design basis accidents are discussed and evaluated in Section 15.7. Control room shielding considers radiation sources from two locations inside the reactor building (the primary containment, and the secondary containment) and the

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SGTS filters. Post LOCA sources for those areas are given in Tables 12.2-25 through 12.2-27.

12.2.1.7 Site Boundary N-16 Shine Dose

The N-16 present in the reactor steam in the primary steam lines, turbines, and moisture separator can contribute to the site boundary dose as a result of high energy gamma emission. The turbine shielding was designed to minimize shine dose. The N-16 shine dose rate at the site boundary was calculated based on the final turbine shielding design. The turbine operating floor component N-16 inventories are listed in Table 12.2-28.

12.2.1.8 Stored Radioactivity

Normally the only sources of activity not stored inside the plant structures are the refueling water storage tank (RWST) and the condensate storage tank (CST). Under normal conditions the condensate storage tank contains concentrations of radionuclides that yield a surface exposure rate of less than 0.5 mr/hr. The condensate storage tank isotopic inventory is shown in Table 12.2-29.

The refueling water storage tank is also expected to have a maximum contact exposure rate of less than 0.5 mr/hr when water is returned from the refueling pool. Maximum activity is based on Table 12.2-12 isotopic inventories reduced by a factor of approximately 10^{-10} as a result of continued cleanup during refueling operations.

Provisions have been made to recycle the water from both the condensate and refueling water storage tanks to the condensate demineralizer.

No other radioactive wastes are normally stored outside the plant structures. All spent fuel is stored in the spent fuel pool until it is placed in the spent fuel shipping cask for offsite transport. Storage space is provided in the radwaste for storage solidified material shielding for Radioactive wastes stored inside the plant structures is designed such that there is normally Zone I access outside the structure.

12.2.1.9 Special Sources

Special materials used in the radiochemistry laboratory and sealed sources used for calibration purposes are of the low activity level and are handled in accordance with station health physics procedures. Unsealed sources and radiochemistry samples are handled in hoods that exhaust to the ventilation system.

The radiation source for the Transverse Incore Probe System (TIP) is provided in Tables 12.2-41 through 12.2-44. The radiation source is based upon location within the core and residence time. As indicated in the tables, the TIP system consists of three components for shielding calculations, the fissionable material, non fissionable material, and the cable. Sources are provided for each component as a function of irradiation and decay times.

The reactor startup source is shipped to the site in a special cask designed for shielding. The source is transferred under water while in the cask and loaded into Beryllium containers. This is then loaded into the reactor while remaining under water. The source remains within the reactor for its lifetime. Thus, no unique shielding requirements after reactor operation are required.

12.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

12.2.2.1 Sources of Airborne Radioactivity

The sources of airborne radioactivity are found in the various confined areas of the plant facility and are primarily from the process leakage of the systems carrying radioactive gases, steam, and liquids. Depending on the type of the system and its physical condition, such as system pressures and temperatures, the leakage will be as a gas, steam, liquid, or a mixture of these.

12.2.2.2 Production of Airborne Materials

Radioactive materials become airborne through a number of mechanisms. The primary production mechanisms are spraying, splashing, flashing, evaporation, and diffusion.

12.2.2.3 Locations of Sources of Airborne Radioactivity

The primary sources of airborne radioactivity are found in the reactor, turbine, and radwaste buildings. Within these structures, the radioactivity may be released in equipment cubicles, system compartments, valve and piping galleries, sampling stations, radwaste handling areas, cleaning and decontamination areas and repair shops.

12.2.2.4 Control of Airborne Radioactivity

Ventilation is an effective means of controlling airborne radioactive materials. Ventilation flow paths are designed such that air from low potential airborne areas flows toward the higher potential airborne areas. This flow pattern will ensure that activity released in the above mentioned source locations, which usually have low personnel access requirements, will have little chance to escape to areas with a high personnel occupancy such as corridors, working aisles and operating floors.

12.2.2.5 Methodology For Estimating the Expected Concentration of Airborne Radioactive Material Within the Plant

In order to estimate the expected airborne radioactive material concentrations at locations within the plant, the following methodology was used:

(1) Estimate the total airborne releases (in Curies per year) for each of the buildings of the plant;

(2) Estimate a distribution for these releases among the various equipment areas of each building based on operating data and engineering judgement;

(3) Determine the annual exhaust flow from each equipment area;

(4) Calculate the resultant airborne radionuclide concentration (μCi/cc) in each equipment area based on the release distribution (Ci/yr) and exhaust flow rate (cc/yr).

The following subsections discuss each step in the above procedure in more detail.

12.2.2.6 Estimation of Total Airborne Releases Within the Plant

The estimated quantities of airborne radioactive material produced in the buildings of the plant are given in Table 12.2-30. These releases were based upon NUREG-0016, "Calculation of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors". The quantities in Table 12.2-30 were generated from NUREG-0016 as follows:

- All turbine building releases in NUREG-0016 were reduced by a factor of five to take credit for the leakage collection system installed for valves in lines 2 1/2" and larger (see Subsection 11.3.2.4.3).

  (NOTE: Releases assigned to the turbine building are assumed to include any control structure releases).

- The Susquehanna reactor building releases were taken to be the sum of the releases listed in NUREG-0016 for the auxiliary building and containment building.

- The radwaste building releases in NUREG-0016 are "per reactor" and consequently were doubled for Susquehanna SES.

- Tritium releases were divided equally between the reactor building and the turbine building.

12.2.2.7 Distribution of Airborne Releases Within the Plant

The approach taken to determine the anticipated distribution of gaseous effluents assumed that all airborne radioactive material originates only within the equipment areas of the plant. It was further assumed that a major percentage of the release is generated within a few specific areas of each building with the remainder coming from all other equipment areas. For the purposes of the estimate, 80 percent of each building's release was distributed as described below among the major contributing areas and 20 percent was assigned to the "all other equipment areas" category. Releases were assumed to be generated continuously throughout the year except for the drywell where a 30 day release period was used.

The basis for the selection and relative contributions of the major areas was an interim report for Electric Power Research Institute Research Project 274-1 entitled "Sources of Radioiodine at Boiling Water Reactors". This report provided data on the important sources of Iodine-131 at operating BWR's and used measured data to determine the relative release rate from each source. The relative release rates for all airborne radionuclides except for reactor building tritium were then assumed to be directly proportional to the Iodine-131 release rates. Since the spent fuel pool and the reactor vessel (when it is open during refueling) are the major sources of airborne tritium in the reactor building, tritium releases for that buidling were assigned entirely to the refueling area.

Table 12.2-31 lists the major airborne contributors in each building and the percentage of the total building release assigned to each. Tables 12.2-32 through 12.2-34 provide the specific equipment areas of the plant associated with the major contributors and the applicable exhaust air flow rates. Note that only those equipment areas which have a significant potential for airborne radioactive material releases were included in the "other equipment areas" category.

12.2.2.8 Estimated Airborne Radioactive Material Concentrations Within the Plant

The airborne radionuclide concentrations for each equipment area was calcualted using the following methodology. For a specific area, the appropriate building release (Table 12.2-30) was multiplied by the applicable release percentage for the area (Table 12.2-31) and divided by the area annual exhaust flow (Table 12.2-32, 12.2-33, or 12.2-34). The resultant concentrations are presented in Tables 12.2-35 through 12.2-37 which also include the fractions of the maximum permissible concentrations in air as defined in 10CFR20 Appendix B, Table I.

12.2.2.9 Changes to Source Data Since PSAR

Airborne radioactive material sources were not specified in the Susquehanna SES PSAR. Subsection 12.2.2 has been added in compliance with the "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants", Regulatory Guide 1.70.

TABLE 12.2-30

ESTIMATED AIRBORNE RADIOACTIVE RELEASES (CURIES/YEAR)(1)

Nuclide	Turbine(2) Building Releases (per Unit)	Reactor Building Releases (per Unit)	Radwaste Building Releases (per Plant)
H-3	8.0+0	4.0+1(3)	-
Kr-83m	-	-	-
Kr-85m	1.4+1(3)	6.0+0	-
Kr-85	-	-	-
Kr-87	2.6+1	6.0+0	-
Kr-88	4.6+1	6.0+0	-
Kr-89	-	-	-
Xe-131m	-	-	-
Xe-133m	-	-	-
Xe-133	5.0+1	1.3+2	2.0+1
Xe-135m	1.3+2	9.2+1	-
Xe-135	1.3+2	6.8+1	9.0+1
Xe-137	-	-	-
Xe-138	2.9+2	1.4+1	-
I-131	3.8-2	3.4-1	1.0-1
I-133	1.5-1	1.4+0	3.6-1
Co-60	4.0-4	2.0-2	1.8-1
Co-58	1.2-4	1.2-3	9.0-3
Cr-51	2.6-3	6.0-4	1.8-2
Mn-54	1.2-4	6.0-3	6.0-2
Fe-59	1.0-4	8.0-4	3.0-2
Zn-65	4.0-5	4.0-3	3.0-3
Zr-95	2.0-5	8.0-4	1.0-4
Sr-89	1.2-3	1.8-4	9.0-4
Sr-90	4.0-6	1.0-5	6.0-4
Sb-124	6.0-5	4.0-4	1.0-4
Cs-134	6.0-5	8.0-3	9.0-3
Cs-136	1.0-5	6.0-4	9.0-4
Cs-137	1.2-4	1.1-2	1.8-2
Ba-140	2.2-3	8.0-4	2.0-4
Ce-141	1.2-4	2.0-4	5.2-3

(1) Based on NUREG-0016
(2) Includes control structure releases
(3) 4.0+1 = 4.0×10^1

TABLE 12.2-35

ESTIMATED AIRBORNE CONCENTRATIONS IN THE TURBINE BUILDING

NUCLIDE	MPC μ Ci/cc	CONDENSER AREAS Concen. μ ci/cc	CONDENSER AREAS Fract. of MPC	SJAE AREAS Concen. μ Ci/cc	SJAE AREAS Fract. of MPC	MECH. VACUUM PUMP AREAS Concen. μ Ci/cc	MECH. VACUUM PUMP AREAS Fract. of MPC	TURBINE HALL AREAS Concen. μ Ci/cc	TURBINE HALL AREAS Fract. of MPC	OTHER EQUIPMENT AREAS Concen. μ Ci/cc	OTHER EQUIPMENT AREAS Fract. of MPC
H-3	5.0-6	2.2-8	4.4-3	2.4-8	4.8-3	9.6-9	1.9-3	6.6-10	1.3-4	1.9-9	3.7-4
Kr-83m	1.0-6	--	--	--	--	--	--	--	--	--	--
Kr-85m	6.0-6	4.0-8	6.7-3	4.3-8	7.1-3	1.7-8	2.8-3	1.2-9	2.0-4	3.3-9	5.4-4
Kr-85	1.0-5	--	--	--	--	--	--	--	--	--	--
Kr-87	1.0-6	7.5-8	7.5-2	7.9-8	7.9-2	3.1-8	3.1-2	2.2-9	2.2-3	6.0-9	6.0-3
Kr-88	1.0-6	1.3-7	1.3-1	1.4-7	1.4-1	5.6-8	5.6-2	3.8-9	3.8-3	1.1-8	1.1-2
Kr-89	1.0-6	--	--	--	--	--	--	--	--	--	--
Xe-131m	2.0-5	--	--	--	--	--	--	--	--	--	--
Xe-133m	1.0-5	--	--	--	--	--	--	--	--	--	--
Xe-133	1.0-5	1.4-7	1.4-2	1.5-7	1.5-2	6.0-8	6.0-3	4.2-9	4.2-4	1.2-8	1.2-3
Xe-135m	1.0-6	3.7-7	3.7-1	4.0-7	4.0-1	1.6-7	1.6-1	1.1-8	1.1-2	3.0-8	3.0-2
Xe-135	4.0-6	3.7-7	9.3-2	4.0-7	9.9-2	1.6-7	3.9-2	1.1-8	2.7-3	3.0-8	7.6-3
Xe-137	1.0-6	--	--	--	--	--	--	--	--	--	--
Xe-138	1.0-6	8.3-7	8.3-1	8.8-7	8.8-1	3.5-7	3.5-1	2.4-8	2.4-2	6.7-8	6.7-2
I-131	9.0-9	1.1-10	1.2-2	1.2-10	1.3-2	4.6-11	5.1-3	3.2-12	3.5-4	8.8-12	9.8-4
I-133	3.0-8	4.3-10	1.4-2	4.6-10	1.5-2	1.8-10	6.0-3	1.3-11	4.2-4	3.5-11	1.2-3
Co-60	9.0-9	1.1-12	1.3-4	1.2-12	1.4-4	4.8-13	5.4-5	3.3-14	3.7-6	9.3-14	1.0-5
Co-58	5.0-8	3.4-13	6.9-6	3.7-13	7.3-6	1.4-13	2.9-6	1.0-14	2.0-7	2.8-14	5.6-7
Cr-51	2.0-6	7.5-12	3.7-6	7.9-12	4.0-6	3.1-12	1.6-6	2.2-13	1.1-7	6.0-13	3:0-7
Mn-54	4.0-8	3.4-13	8.6-6	3.7-13	9.1-6	1.4-13	3.6-6	1.0-14	2.5-7	2.8-14	7.0-7
Fe-59	5.0-8	2.9-13	5.7-6	3.0-13	6.1-6	1.2-13	2.4-6	8.4-15	1.7-7	2.3-14	4.7-7
Zn-65	6.0-8	1.1-13	1.9-6	1.2-13	2.0-6	4.8-14	8.0-7	3.3-15	5.6-8	9.3-15	1.6-7
Zr-95	3.0-8	5.7-14	1.9-6	6.1-14	2.0-6	2.4-14	8.0-7	1.7-15	5.6-8	4.7-15	1.6-7
Sr-89	3.0-8	3.4-12	1.1-4	3.7-12	1.2-4	1.4-12	4.8-5	1.0-13	3.3-6	2.8-13	9.3-6
Sr-90	1.0-9	1.1-14	1.1-5	1.2-14	1.2-5	4.8-15	4.8-6	3.3-16	3.3-7	9.3-16	9.3-7
Sb-124	2.0-8	1.7-13	8.6-6	1.8-13	9.1-6	7.2-14	3.6-6	5.0-15	2.5-7	1.4-14	7.0-7
Cs-134	1.0-8	1.7-13	1.7-5	1.8-13	1.8-5	7.2-14	7.2-6	5.0-15	5.0-7	1.4-14	1.4-6
Cs-136	2.0-7	2.9-14	1.4-7	3.0-14	1.5-7	1.2-14	6.0-8	8.4-16	4.2-9	2.3-15	1.2-8
Cs-137	1.0-8	3.4-13	3.4-5	3.7-13	3.7-5	1.4-13	1.4-5	1.0-14	1.0-6	2.8-14	2.8-6
Ba-140	4.0-8	6.3-12	1.6-4	6.7-12	1.7-4	2.7-12	6.6-5	1.8-13	4.6-6	5.1-13	1.3-5
Ce-141	2.0-7	3.4-13	1.7-6	3.7-13	1.8-6	1.4-13	7.2-7	1.0-14	5.0-8	2.8-14	1.4-7

12.4 DOSE ASSESSMENT

This section discusses the estimated radiation exposures both in-plant and at locations outside the plant structures. Subsections 12.4.1 and 12.4.2 discuss direct radiation and airborne radiation exposures within the plant; Subsection 12.4.3 is concerned with exposures outside the plant structures; and Subsection 12.4.4 estimates the exposure to Unit 2 construction workers from the operation of Unit 1.

12.4.1 DIRECT RADIATION DOSE ESTIMATES FOR EXPOSURES WITHIN THE PLANT

To estimate the total annual man-rem dose from direct radiation to personnel within the plant, seven broad categories or job functions were defined and the annual man-rem dose for each category was evaluated. Where the functions and expected radiation levels were predictable or clearly defined, analytical methods were employed for the man-rem estimates. In other cases, the estimate basis was historical exposure data from operating BWR power plants. Subsection 12.4.1.1 provides the definitions and components of each of the seven broad categories while Subsection 12.4.1.2 describes briefly the estimation techniques used.

The resultant dose estimates are contained in Subsection 12.4.1.3, along with further discussion of the factors involved and the methodology used for each category and its related components.

12.4.1.1 Definition of Categories Used in Exposure Estimates

Seven broad categories were used in estimating the total annual man-rem dose. These categories are:

Routine Operations: This category is composed of three components or subcategories.

a) Routine patrols and surveillances of the reactor building, turbine building and control structure, and radwaste building

b) Periodic tests and checks in the reactor building, turbine building and control structure, and radwaste building

c) Control room operations, specifically, the dose received by operators in the main and radwaste control rooms.

Routine Maintenance: All maintenance that is scheduled. This does not imply that a particular date has been established, but rather that the maintenance is planned and will occur at least annually. This category also includes the preventative maintenance performed in the radiation areas of the turbine, reactor, and radwaste buildings.

In-service Inspections: These are inspections normally performed by quality assurance, NDT personnel, and outside contractors. Such inspections normally occur during outages on piping and systems that cannot be checked while at power.

Special Maintenance: All maintenance that has not been scheduled. This maintenance will not have been planned in advance and normally cannot be predicted.

Waste Processing: Includes any work with solid or liquid radwaste: movement of casks and liners; radwaste, condensate system, or fuel pool filter changes; resin moving; compacting of low level radwaste. Maintenance of radwaste equipment is covered by the maintenance categories and is not included in this job function.

Refueling: All work with fuel or reactor components performed in the reactor and pool area.

Health Physics: This covers all health physics activities.

12.4.1.2 Exposure Estimate Methodology

The analytical method used for man-rem estimation is based upon the product of estimated exposure time and estimated ambient dose rate. Initially, a review of equipment in plant radiation areas is performed. Estimates of the occupancy time requirements for operations associated with that equipment (e.g., maintenance time or surveillance time) are developed. An applicable frequency of occurrence is then factored in to provide the exposure time for that operation. For areas with no significant radiation sources, an estimated dose rate of 0.25 mRem/hr is used. Where radiation sources are present, 2.5 mRem/hr is assumed for Zone II and 15.0 mRem/hr for most Zone III areas. All other estimated dose rates are based on either calculations or actual radiation levels encountered at operating plants. The analytical method was used in determining the exposure estimates for the routine maintenance and routine operations categories.

In the historical method, the annual man-rem is estimated from the exposures received at operating BWR power plants. This method was used for all other categories (special maintenance, inservice inspection, waste processing, refueling, health physics). The data sources are the annual and semi-annual BWR operating reports and plant correspondence with regulatory agencies. Included are a total of sixty-one (61) reactor years of operation for sixteen (16) nuclear units. The average licensed power level of these units is 747 MWe with the smallest rated at 514 MWe, see Table 12.4-1. The data was collected and assembled using the following guidelines:

a) No data before the first calendar year which contained less than nine (9) months of commercial operation was used.

b) In multiple unit plants, each unit was assumed to contribute equally to the annual exposures.

c) If exposure contributions from two or more job functions could not be separated, a conservative approach was taken by assigning all the exposure to one function and having no entry in the data base for the other.

Table 12.4-2 contains the results of the historical data compilation and includes both the number of reactor years contributing and the standard deviations associated with each job function. The large standard deviations, which range from about 60 to 160 percent of the mean values, are indicative of the wide spread of data that has been reported within each exposure category.

12.4.1.3 Results of Annual Direct Radiation Dose Estimates

The annual man-rem estimates for each category and subcategory are detailed below in Subsections 12.4.1.3.1 through 12.4.1.3.7. The methods used in their determination are as described previously, with any additional assumptions or information included below where required.

In each of the following subsections, the annual exposure estimates are reported for two plant configurations: single unit operational and two units operational. In general, the "two-unit dose" is twice the "single-unit dose"; however, the exposures associated with certain job functions are assumed to be independent of the number of units in operation since the functions will be performed regardless of whether one or two units are operational. These specific job functions are:

Main Control Room operations
Radwaste Control Room operations
Radwaste building routine surveillances
Radwaste building periodic testing
Radwaste building routine maintenance

For these estimates, the single-unit doses are conservatively assumed to be the same as the two-unit dose.

A summary of the direct radiation dose estimates is given in Subsection 12.4.1.3.8 and in Table 12.4-9.

12.4.1.3.1 Routine Operations Dose Estimate

During normal operations, routine patrols and surveillances are performed by plant operators. The majority of items checked are rotating equipment (pumps, fans, etc), and each is viewed to verify the absence of leaks, excessive vibrations, or other abnormal conditions. For the man-rem exposure estimation, the following assumptions were made:

a) Dose rates were estimated as outlined in Subsection 12.4.1.2. Additionally, because of the high potential dose rates associated with certain equipment, routine surveillances of such equipment will be performed from a remote location (such as the cell doorway) and credit was taken for the lower ambient radiation level at that point.

b) Exposure received during walking of patrol areas is based upon a walking speed of 200 ft per minute.

c) Patrol frequency for Zone II areas will be twice per shift, three shifts per day.

d) Patrol frequency for Zone III areas will be once per shift, three shifts per day.

e) Surveillance of equipment in Zones IV and V will not be performed regularly but only as required. A patrol frequency of once per month was used for the estimate.

f) Each patrol consists of only one man.

The results of the routine patrol exposure estimate are contained in Tables 12.4-3 through 12.4-5.

Similarly, the details and results of the exposure estimate for the periodic testing subcategory are also contained in

Tables 12.4-3 through 12.4-5. The estimated dose rates used are generally the same as for the routine patrol estimate. However, since periodic testing is assumed to occur during equipment shutdown, the estimated shutdown dose rate is used if it is different from the operating dose rate.

The remaining subcategory is control room operations exposures. This has been estimated from the estimated control room radiation levels and the staffing requirements for the main and radwaste control rooms. It is assumed that the staffing levels of both control rooms will be identical for either one or two units operational. Table 12.4-6 contains the details of the control room operations exposure estimate.

The total annual exposure estimate for the routine operations category is then the sum of the three subcategory annual exposures, see Table 12.4-7.

Annual Exposure Estimate: Routine Operations

113.1 man-rem (single unit operational)

163.8 man-rem (two units operational)

12.4.1.3.2 Routine Maintenance Dose Estimate

The estimated exposure to be received in this category was determined from a compilation of the estimated annual man-hours required for component maintenance and the estimated dose rate to which the maintenance personnel will be subjected. As with periodic testing, the estimated shutdown dose rate was used if applicable.

The first step in this estimate consisted of a detailed review of plant radiation areas to produce a listing of the types and quantities of selected equipment present in each area. Next, total annual maintenance manhours were estimated for each equipment type identified based on a combination of operating experience and engineering judgement. These total estimated manhours are shown in Table 12.4-8 and are intended to include all expected routine activities for each equipment type such as valve repacking, valve relapping, pump seal replacement, fan overhaul, etc.

In any area, the total annual manhours for routine maintenance was then the summation of the quantity-manhour products for all equipment types found in the area. Multiplying the area annual maintenance manhours by the anticipated area dose rate produced the estimated man-rem by area. These were then summed to yield

the routine maintenance man-rem by building and for the plant. Tables 12.4-3 through 12.4-5 contain the details and results of the routine maintenance exposure estimate.

A "total annual" maintenance approach was used for each component since currently available data generally does not contain sufficient information to provide a basis for manhour breakdowns by maintenance activity. In addition, the area-by-area methodology employed makes estimate compilations by system unnecessary since locations were high man-rem expenditures are expected are clearly indicated.

Annual Exposure Estimate: Routine Maintenance

237.7 man-rem (single unit operational)

395.0 man-rem (two units operational)

12.4.1.3.3 In-service Inspection Dose Estimate

The annual exposure estimate for in-service inspection is based upon the data from operating BWRs given in Table 12.4-2.

Annual Exposure Estimate: In-Service Inspection

27.5 man-rem (single unit operational)

55.0 man-rem (two units operational)

12.4.1.3.4 Special Maintenance Dose Estimate

The annual exposure estimate for special maintenance is based upon the data from operating BWRs given in Table 12.4-2.

Annual Exposure Estimate: Special Maintenance

273.1 man-rem (single unit operational)

546.2 man-rem (two units operational)

12.4.1.3.5 Waste Processing Dose Estimate

Most of the operations in the plant associated with the waste processing category are performed remotely and are therefore not suitable for evaluation by the analytical estimation technique.

Consequently, the annual man-rem estimate for waste processing is more properly taken from the historical BWR operating data of Table 12.4-2 since this will provide a conservative estimate of the anticipated exposure.

Annual Exposure Estimate: Waste Processing

37.0 man-rem (single unit operational)

74.0 man-rem (two units operational)

12.4.1.3.6 Refueling Dose Estimate

The annual exposure estimate for refueling is based upon the data from operating BWRs given in Table 12.4-2.

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Annual Exposure Estimate: Refueling

19.2 man-rem (single unit operational)

38.4 man-rem (two units operational)

12.4.1.3.7 Health Physics Dose Estimate

The annual exposure estimate for health physics monitoring is based upon the data from operating BWRs given in Table 12.4-2.

Annual Exposure Estimate: Health Physics

29.3 man-rem (single unit operational)

58.6 man-rem (two units operational)

12.4.1.3.8 Summary of Direct Radiation Dose Estimates

The annual dose estimates in the preceding seven subsections are summarized and totaled in Table 12.4-9. As shown in this table, the estimate of total annual in-plant exposure from direct radiation is:

Annual Exposure Estimate: Total

736.9 man-rem (single unit operational)

1331.0 man-rem (two units operational)

12.4.1.3.9 Methods for Estimating Doses

The contribution to the estimated cumulative station exposure from Routine Operations (RO) and Routine Maintenance (RM) in areas where radiation zone maximum design dose rates were used in the estimate are summarized below:

Calculated Manrem

	One Unit Operation		Two Unit Operation	
Bldg	RM	RO	RM	RO
Turbine	3.8	1.7	7.6	1.4
Reactor	7.2	1.9	14.4	3.8
Radwaste	11.2	1.7	11.2	3.7
Total	22.2	4.3	33.2	6.9

It can be seen that the calculated manrem in those areas where radiation zone maximum design dose rates were used in the estimate comprise 3.6 percent and 3.0 percent of the total estimated manrem for one unit and two unit operations, respectively. Since the expected radiation levels would be less than the maximum design dose rates, the impact of using expected radiation levels on the total esimated station manrem would not be significant due to the low contribution to the total from the exposure categories discussed.

Dose estimates for Inservice Inspection, Waste Processing, Special Maintenance, and Refueling were based on historical data from operating facilities. Any further breakdown of the dose estimate by individual task (such as was made for Routine Operations and Routine Maintenance) would rely primarily on historical information available. The resultant dose estimate would not be any more precise than would be an estimate based solely on reported radiation exposures. In all four areas where historical data was used in the dose estimate, the SSES design includes design features which will reduce actual exposures received by plant personnel. Due to the lack of sufficiently detailed information to allow the precise quantification of the dose reduction the calculation of the reduction cannot be performed. For example, the followig design features have been incorporated to facilitate Inservice Inspection:

a) Quick removal insulation around the reactor vessel nozzles.

b) Access panels in the shield wall to the bottom head welds.

c) Side access panels in the shield wall to the core region of the reactor vessel.

d) The use of a remote, trackless vehicle for vessel weld inspection.

e) The use of remote automatic weld inspection of the vessel nozzle welds.

f) During the pre-service inspection access will be thoroughly evaluated.

In view of the attendant uncertainties in the available data, precise quantification of the dose reduction benefit of these design features is not possible. Therefore the methodology employed in Section 12.4 gives a reasonable and conservative estimate of exposures from all activities.

12.4.2 AIRBORNE RADIOACTIVITY DOSE ESTIMATES FOR EXPOSURES WITHIN THE PLANT

The estimated exposures to plant personnel from airborne radioactivity are based upon the source distributions and radionuclide concentrations presented in Subsection 12.2.2 and Tables 12.2-30 through 12.2-37. Because of the limited geometry afforded by the finite room sizes within the plant, personnel exposures due to noble gas immersion are expected to be insignificant when compared to inhalation exposures and have therefore not been estimated.

In order to determine whether exposure contributions from airborne radioactive particulates are significant, an evaluation was made in each area of the ratio of total particulate MPC fractions to total radioiodine MPC fractions (which is equivalent to the ratio of particulate MPC-HOURS to iodine MPC-HOURS). For the turbine building areas and the reactor building areas, the particulate-to-iodine ratios were approximately 0.02 and 0.05, respectively, indicating that the particulate inhalation exposures are not significant in those areas. In the radwaste building areas, however, the particulate-to-iodine ratio was approximately 1.11. Since over 75 percent of the total particulate MPC fraction was attributable to Cobalt-60, both the thyroid inhalation dose due to radioiodines and the lung inhalation dose due to Cobalt-60 were estimated for the radwaste building (the thyroid and the lung are the critical organs for iodines and Cobalt-60, respectively).

Tables 12.4-10 through 12.4-12 are the compilations of the estimated annual occupancy times and the estimated annual exposures for each of the areas identified in Subsection 12.2.2 as being potential sources of airborne radioactivity. The occupancy times are based upon detailed reviews of each area and the determination of the operations which might occur in those areas. The exposures are based upon the estimated concentrations in Tables 12.2-35 through 12.2-37, dose factors from Table C-1 of

USNRC Regulatory Guide 1.109, and an assumed breathing rate of
1| 3.47×10^{-4} cubic meters per second.

12.4.3 EXPOSURES AT LOCATIONS OUTSIDE PLANT STRUCTURES

The radiation exposures at locations outside the plant structures were estimated for two areas: the site boundary and the visitor's center. Subsection 12.4.3.1 discusses direct radiation exposure at these locations while Subsection 12.4.3.2 deals with airborne exposures.

12.4.3.1 Direct Radiation Dose Estimates Outside Structures

At locations outside plant structures, the direct radiation exposure has two principal components:

a) Sources of activity stored outside the structures, specifically, the refueling water storage tanks (RWST) and the condensate storage tank (CST).

b) Turbine shine due to the N-16 present in the reactor steam.

Based on the calculated surface dose rates for the RWST and CST given in Subsection 12.2.1.8, the dose contribution at locations outside the plant structures due to these tanks is considered negligible.

The N-16 present in the reactor steam in the primary steam lines, turbines, and moisture separators provides a dose contribution to locations outside the plant structure as a result of the high energy gamma rays which it emits as it decays. To reduce the turbine shine doses, radiation shielding was provided around each turbine train and a roof slab was constructed over each moisture separator.

The resultant annual exposure due to turbine shine was calculated with the SKYSHINE (Section 12.3, Ref 12.4-1) computer program. Point sources were used to represent the components on the turbine deck and the source strengths are given in Table 12.2-28.

With an assumed 100 percent occupancy factor and an 80 percent
15| capacity factor, the maximum calculated dose rate occurs at the south site boundary (see Figure 12.4-1) and is 5.6 mRem/year.

The dose rate in the visitor's center was calculated by the SKYSHINE program to be 3.50×10^{-6} mRem/hr. Assuming a visitor will visit the plant one day a year for eight hours, the estimated dose for the visitor is 2.80×10^{-5} mRem/year.

12.4.3.2 Airborne Radioactivity Dose Estimates Outside Structures

Doses at the site boundary due to released activity are given in Subsection 11.3.3.

At the visitor's center, the total body gamma and beta skin doses for an assumed annual occupancy of 8 hours are also given in Subsection 11.3.3.

12.4.4 EXPOSURES TO CONSTRUCTION WORKERS

12.4.4.1 Direct Radiation and Dose Estimates

The estimated dose rates from direct radiation and turbine shine received by construction workers on Unit 2 due to the operation of Unit 1 are well within the limits of 10CFR20 for exposure to individuals in unrestricted areas.

The estimated dose rates are the sum of the direct radiation from the Unit 1 reactor building, turbine building, and radwaste building and the turbine shine doses resulting from the decay of N-16 in the steam lines and turbine equipment of Unit 1. As discussed in Subsection 12.4.3.1, dose contributions from outside storage tanks are considered negligible and were not included in the exposure estimate.

The annual dose to the construction workers employed in Unit 2 while Unit 1 is in operation has been estimated for various points in the Unit 2 construction area. The results of this estimate and their corresponding points are shown on Figure 12.4-1.

The doses from turbine shine were calculated with the SKYSHINE computer program in the manner described in Section 12.4.3.1. The resultant dose includes the direct as well as air scattered contribution. No credit was taken for the shielding which will be afforded by the partially erected Unit 2 structures. The radioactive wastes will be processed and stored in the radwaste building where shielding is provided to ensure that the dose outside the building will be minimized. With an allowance for

distance between the radwaste building and the Unit 2 construction area, the estimated direct shine dose will be less than 0.01 mRem/hr under normal conditions.

The exposure for Unit 2 construction workers has been estimated based on the following assumptions:

a) The current schedule will be met.

b) Doses to personnel in the Unit 2 structures are negligible once the exterior walls and slabs have been fully erected.

c) Manual laborers spend 80 percent of their time in the Unit 2 structures and 20 percent in the yard. Non-manual workers spend 10 percent of their time in the Unit 2 reactor building, 10 percent in the Unit 2 turbine building, and 80 percent in the field office.

Laborers assigned to the control structure or Unit 2 turbine building are assumed to work in the turbine building only. Laborers assigned to the Unit 2 reactor building or drywell are assumed to work in the reactor building only.

10 percent of the time spent in the Unit 2 turbine building will be on or above the turbine operating deck. 10 percent of the time spent in the Unit 2 reactor building will be on the refueling floor.

d) The average dose rate in the yard areas is the average of the dose rates at points 1 through 7 of Figure 12.4-1, 0.025 mRem/hr. The average dose rate in the field
15 office is 0.020 mRem/hr. Each of these dose rates include a direct shine contribution of 0.01 mRem/hr.

e) The availability factor for Unit 1 is 80 percent.

f) 40 hours per week per person at the work site, 50 weeks per year.

Exposure to personnel in various categories and locations is summarized in Table 12.4-13, which gives the total estimated exposure to Unit 2 construction workers as 30.7 man-rem.

Section 20.202 of 10CFR20 specifies that personnel monitoring
1 equipment would be required if the maximum expected whole body dose per calendar quarter for workers in an area would exceed 300 mRem. It was determined that, even in areas with the highest radiation levels (the turbine deck), no construction worker would receive a dose greater than this, so personnel monitoring

equipment will not be necessary. However, periodic radiation surveys will be made by the health physics staff. Personnel dosimetry devices will be located in areas where construction personnel are working to verify that no person will receive a dose greater than 500 mRem/yr. 1

12.4.4.2 Exposures Due to Airborne Radioactivity

Doses to Adult Workers resulting from atmospheric releases of gaseous and particulate effluents were calculated based on an occupancy factor of 2000 hours per year (40 hours per week, 50 weeks per year). At the critical location 0.06 miles from the vents in the ESE Direction doses of 16.6, 31.3 and 3.77 mRem/yr were calculated for the total body and skin due to submersion and the thyroid due to inhalation, respectively. These doses were calculated using the appropriate equations from Regulatory Guide 1.109 with slight modifications. In all cases the shielding factor for residential structures was removed from the equations and all results were multiplied by 2000/8766 to correct for the lower occupancy factor. 15

12.4.5 REFERENCES

12.4-1 M. G. Wells, D. G. collins, R. B. Small and J. M. Newell, SKYSHINE, a computer procedure for evaluations effect of the Structure Design on N-16 Gamma Ray DOse Rates, RRA-T7209, (November 1, 1972).

TABLE 12.4-3 (Continued)

Room or Area No.	Estimated Dose Rate (mRem/hr)	Routine Maintenance[1] Estimated Annual Man-hours[5]	Routine Maintenance[1] Estimated Annual Man-rem	Routine Surveillances[1] Estimated Annual Man-hours	Routine Surveillances[1] Estimated Annual Man-rem	Periodic Testing[1] Estimated Annual Man-hours	Periodic Testing[1] Estimated Annual Man-rem
El 806' 0"							
C-900, C-912	2.5	5[2]	0.013	-	-	3[2]	0.008
C-900A, C-912A	2.5	25[2]	0.063	9[2]	0.023	14[2]	0.035
C-901 to C-911	0.25	360[2]	0.090	15[2]	0.004	202[2]	0.051
Transit[3]	0.56	-	-	33[2]	0.018	-	-
	Totals	33,919	48.560	1,267	0.940	3,865	9.569

(1) All values are on a per-unit basis.

(2) Entries referencing this note are for common facilities or equipment and the man-hours are shown as on-half the estimated quantity for the room or area to reflect the per-unit basis of the table.

(3) The "transit" entries account for the estimated time spent and dose received while walking the patrol areas during routine surveillances. For elevations which entail exposures to multiple radiation levels, the estimated dose rate is the distance weighted average of the dose rates encountered. All transit times are based on an assumed walking speed of 200 feet per minute.

(4) From surveillances performed once per month.

(5) The estimates exposures in this table assume all man-hours are expended in the area in which they appear. Portions of these man-hours may actually be spent at lower radiation levels within the area. Components may also be removed to a lower background area for maintenance.

TABLE 12.4-4 (Continued)

Room or Area No.	Estimated Dose Rate (mRem/hr)	Routine Maintenance[1] Estimated Annual Man-hours[8]	Routine Maintenance[1] Estimated Annual Man-rem	Routine Surveillances[1] Estimated Annual Man-hours	Routine Surveillances[1] Estimated Annual Man-rem	Periodic Testing[1] Estimated Annual Man-hours	Periodic Testing[1] Estimated Annual Man-rem
El 719' 1"							
412	0.25	84	0.021	-	-	2	0.001
401	0.25	3,648	0.912	18	0.005	74	0.019
401A	2.5	600	1.500	37	0.093	327	0.818
472	2.5	95	0.238	18	0.045	1	0.003
402	0.25	70	0.018	-	-	2	0.001
413	0.25	10	0.003	-	-	1	0.001
406	0.25	360	0.090	-	-	62	0.016
471	15	100	1.500	5	0.075	10	0.150
407	0.25	420	0.105	-	-	62	0.016
410	0.25	120	0.030	9	0.002	24	0.006
430	13	-	-	-	-	222[7]	2.886
Transit[3]	1.64	-	-	139	0.228	-	-
El 749' 1"							
507, 510	0.25	720	0.180	-	-	124	0.031
514	16	576	9.216	-	-	57	0.912
514	2.5[6]	-	-	0.03[5]	0.001	-	-
516	0.25	45	0.011	-	-	33	0.008
513	0.25	731	0.183	210	0.053	87	0.022
512	0.25	524	0.131	26	0.007	24	0.006
517	0.25	166	0.042	9	0.002	125	0.031
515	2.5	390	0.975	-	-	30	0.075
519	200	68	13.600	-	-	59	11.800
500	0.25	404	0.101	-	-	142	0.036
502, 503	28	264	7.392	-	-	8	0.224
502, 503	28[6]	-	-	0.10[5]	0.003	-	-
501	28	91	2.548	-	-	40	1.120
504	28	58	1.624	-	-	18	0.504

TABLE 12.4-5

EXPOSURE ESTIMATES FOR THE RADWASTE BUILDING

Room or Area No.	Estimated Dose Rate (mRem/hr)	Routine Maintenance(1) Estimated Annual Man-hours(5)	Routine Maintenance(1) Estimated Annual Man-rem	Routine Surveillances(1) Estimated Annual Man-hours	Routine Surveillances(1) Estimated Annual Man-rem	Periodic Testing(1) Estimated Annual Man-hours	Periodic Testing(1) Estimated Annual Man-rem	
El 646' 0"								
R-38	15	176	2.640	-	-	-	-	|15
R-2	2.5	450	1.125	11	0.028	70	0.175	
R-36	2.5	12	0.030	-	-	-	-	
R-3	13	22	0.286	-	-	4	0.052	
R-4	2.5	376	0.940	3	0.008	22	0.055	
R-5	20	124	2.480	-	-	-	-	
R-9	2.5	674	1.685	8	0.020	360	0.900	| 1
R-6	2.5	64	0.160	3	0.008	64	0.160	
R-7	2.5	186	0.465	-	-	22	0.055	
R-8	20	12	0.240	-	-	22	0.440	
R-50	20	18	0.360	-	-	-	-	
R-50	2.5(4)	-	-	0.03(3)	0.001	-	-	|1 |15
R-10, R-21	0.25	746	0.187	18	0.005	6	0.002	
R-34	7	52	0.364	3	0.021	-	-	
R-20	3	188	0.564	5	0.015	96	0.288	
R-17, R-18, R-19	42	56	2.352	-	-	18	0.756	
R-14	3	104	0.312	5	0.015	30	0.090	
R-13	13	300	3.900	-	-	-	-	
R-15, R-16	42	28	1.176	-	-	16	0.672	
R-12	2.5	272	0.680	3	0.008	82	0.205	
R-11	4	92	0.368	-	-	42	0.168	
R-22	0.25	1,584	0.396	-	-	90	0.023	
R-31, R-32	300	79	23.700	-	-	60	18.000	
R-30	100	27	2.700	-	-	42	4.200	
R-29	10	170	1.700	-	-	34	0.340	
R-29	2.5(4)			0.12(3)	0.001			

TABLE 12.4-5 (Continued)

Room or Area No.	Estimated Dose Rate (mRem/hr)	Routine Maintenance[1] Estimated Annual Man-hours[5]	Routine Maintenance[1] Estimated Annual Man-rem	Routine Surveillances[1] Estimated Annual Man-hours	Routine Surveillances[1] Estimated Annual Man-rem	Periodic Testing[1] Estimated Annual Man-hours	Periodic Testing[1] Estimated Annual Man-rem
R-60	2.5	510	1.275	-	-	-	-
R-28	500	40	20.000	-	-	36	18.000
R-35	2.5	12	0.030	-	-	-	-
R-27	2.5	12	0.030	-	-	-	-
R-26	2.5	90	0.225	-	-	-	-
R-25	2.5	34	0.085	-	-	-	-
R-24	2.5	2	0.005	-	-	-	-
R-37	2.5	70	0.175	-	-	-	-
Transit[1]	0.83	-	-	128	0.106	-	-
El 660' 0"							
R-105	0.25	52	0.013	11	0.003	40	0.010
R-106, R-107	30	102	3.060	-	-	18	0.540
R-106, R-107	2.5[4]	-	-	0.09[3]	0.001	-	-
R-101	50	32	1.600	-	-	4	0.200
R-110	15	42	0.630	-	-	-	-
R-103	0.25	36	0.009	-	-	78	0.020
Transit[1]	0.25	-	-	22	0.006	-	-
El 676' 0"							
R-201, R-229	2.5	276	0.690	-	-	24	0.060
R-207	2.5	142	0.355	-	-	-	-
R-206	0.25	1,650	0.413	-	-	11,680	2.920
R-226	0.25	420	0.105	18	0.005	16	0.004
R-220	0.25	972	0.243	5	0.001	48	0.012
R-225, R-227	4	114	0.456	-	-	-	-

TABLE 12.4-5 (Continued)

Room or Area No.	Estimated Dose Rate (mRem/hr)	Routine Maintenance[1] Estimated Annual Man-hours[5]	Routine Maintenance[1] Estimated Annual Man-rem	Routine Surveillances[1] Estimated Annual Man-hours	Routine Surveillances[1] Estimated Annual Man-rem	Periodic Testing[1] Estimated Annual Man-hours	Periodic Testing[1] Estimated Annual Man-rem
R-250	4	280	1.120	-	-	104	0.416
Off-gas Treatment	(later)	(later)	(later)	(later)	(later)	(later)	(later)
Transit[1]	0.25	-	-	53	0.013	-	-
Walk-up[2]	0.25	-	-	137	0.034	-	-
El 691' 6"							
R-310, R-313	0.25	1,856	0.464	33	0.008	270	0.068
R-311, R-312	7	68	0.476	-	-	-	-
R-301	0.25	12	0.003	-	-	-	-
R-305	0.25	80	0.020	-	-	-	-
R-308	0.25	40	0.010	-	-	-	-
R-309	0.25	486	0.122	-	-	-	-
Transit[1]	1.77	-	-	57	0.101	-	-
	Totals	13,242	80.424	523	0.408	13,398	48.831

(1) The "transit" entries account for the estimated time spent and dose received while walking the patrol areas during routine surveillances. For elevations which entail exposures to multiple radiation levels, the estimated dose rate is the distance weighted average of the dose rates encountered. All transit times are based on an assumed walking speed of 200 feet per minute.

(2) This entry accounts for the transit between access control and the radwaste building along turbine building EL 676' 0" before and after each surveillance patrol.

TABLE 12.4-6

ESTIMATED EXPOSURE FOR OPERATORS IN RESIDENCE IN CONTROL ROOMS

Operator Designation	Assumed(1) Location	Number(2) Per Shift	Manhours(3) Per Year	Estimated Radiation Level (mRem/hr)	Estimated(2) Annual Man-rem
Shift Supervisor	MCR	1	8,760	0.25	2.2
Asst. Shift Supv.	MCR	1	8,760	0.25	2.2
Plant Control	MCR	3	26,280	0.25	6.6
Nuclear Auxiliary	RWCR	1	8,760	0.25	2.2
	TOTALS:	6	52,560		13.2

(1) MCR = Main Control Room
RWCR = Radwaste Control Room

(2) Assumed to be independent of the number of operating units

(3) Based on 8 man-hours per shift, 1095 shifts per year

TABLE 12.4-7

SUMMARY OF ROUTINE OPERATIONS EXPOSURE ESTIMATE

	Annual Estimated Man-rem Single Unit	Two Units
Routine Surveillances:		
Turbine bldg/control structure	0.9	1.8
Reactor building	1.1	2.2
Radwaste building	0.4	0.4
	2.4	4.4
Periodic Tests:		
Turbine bldg/control structure	9.6	19.2
Reactor building	39.1	78.2
Radwaste building	48.8	48.8
	97.5	146.2
Control Room Operations:		
Main Control room	11.0	11.0
Radwaste control room	2.2	2.2
	13.2	13.2
Total	113.1	163.8

TABLE 12.4-9

SUMMARY OF IN-PLANT DIRECT RADIATION EXPOSURE ESTIMATES

Category	Annual Estimated Man-rem Single Unit	Two Units
Routine Operations	113.1	163.8
Routine Maintenance	237.7	395.0
In-Service Inspection	27.5	55.0
Special Maintenance	273.1	546.2
Waste Processing	37.0	74.0
Refueling	19.2	38.4
Health Physics	29.3	58.6
TOTAL	736.9	1,331.0

TABLE 12.4-10 (Continued)

Other Equipment Areas	Estimated Annual Man-Hours: Maintenance	Surveillance	Testing	Total	Estimated Thyroid Dose (Man-rem/yr)	Estimated Tritium Dose (Man-rem/yr)
33	164	10	38	212	6.8-3	6.6-5
42,42A	860	1	144	1,005	3.2-2	3.2-4
43,151,152	66	1	2	69	2.2-3	2.2-5
110	510	0	0	510	1.6-2	1.6-4
114,115,116	1,518	0	144	1,662	5.3-2	5.2-4
117-121,123,124	812	0	38	850	2.7-2	2.6-4
125	126	0	15	141	4.5-3	4.4-5
212,214,215	1,166	0	480	1,646	5.3-2	5.2-4
300	28	0	0	28	9.0-4	8.8-6
530	2,172	155	72	2,399	7.7-2	7.4-4
531,532	10	0	0	10	3.2-4	3.2-6
C-10	639	52	95	786	2.5-2	2.4-4
C-11	33	4	8	45	1.4-3	1.4-5
C-130	10	0	0	10	3.2-4	3.2-6
C-900,C-912	5	0	3	8	2.6-4	2.4-6
C-900A,C-912A	25	20	14	59	1.9-3	1.8-5
TURBINE BUILDING TOTALS	18,998	471	2,766	22,235	2.7+0	2.6-2

(1) All values in the table are on a per-unit basis. Occupancy times for common areas within the building are shown as one-half the estimated value to agree with the per-unit basis of the table.

(2) Surveillance man-hours include transit time spent by operators walking in the area.

(3) 5.8-1 = 5.8×10^{-1}

(4) Thyroid dose attributable only to inhalation of radioiodines.

(5) Uniform dose to the total body from uptake of tritium.

TABLE 12.4-11 (Continued)

	Estimated Annual Man-Hours				Estimated[7] Thyroid Dose	Estimated[8] Tritium Dose
	Maintenance	Surveillance[2]	Testing	Total	(Man-rem/yr)	(Man-rem/yr)
Drywell Areas						
206,206A	1,045	0	470	1,515	4.5+1	--
207	422	0	0	422	1.3+1	--
400	327	0	306	633	1.9+1	--
490	447	0	164	611	1.8+1	--
516	159	0	47	206	6.2+0	--
607	67	0	67	134	4.0+0	--
Other Equipment Areas						
15	415	29	12	456	2.9-1	--
15A	58	0	0	58	3.7-2	--
401	3,648	80	74	3,802	2.4+0	--
401A	600	92	327	1,019	6.4-1	--
403,471	100	10	10	120	7.6-2	--
411, 515, 710	390	0	30	420	2.6-1	--
480	0	0	222	222	1.4-1	--
506	45	0	33	78	4.9-2	--
511	18	0	0	18	1.1-2	--
514	576	1	57	634	4.0-1	--
620	461	12	6	479	3.0-1	--
621	210	7	34	251	1.6-1	--
700	26	0	0	26	1.6-2	--
702	26	0	0	26	1.6-2	--
703	50	9	0	59	3.7-2	--
704, 712	818	130	12	960	6.0-1	--
802, C-801	0[4]	0	0	0	0.0+0	--
REACTOR BUILDING TOTALS	17,530	547	3,144	21,221	1.2+2[6]	2.8-2

TABLE 12.4-12 (Continued)

(1) All values are on a per-plant basis.

(2) Surveillance man-hours include transit time spent by operators walking in the area.

(3) 4.6-4 = 4.6×10^4

(4) Solid radwaste processing, container handling, and radwaste shipping

TABLE 12.4-13

ESTIMATED EXPOSURE TO UNIT 2 CONSTRUCTION WORKERS

Personnel	Assumed Location	Estimated Occupancy[1] (Man-hr)	Average Dose Rate (mRem/hr)	Estimated Exposure[2] (Man-rem)
Manual	Yard Area	154,000	0.025	3.08
Manual	Unit 2 Reactor Bldg Below Refueling Floor	373,000	0.010	2.98
Manual	On Refueling Floor	41,400	0.24	7.95
Manual	Unit 2 Turbine Bldg Below Turbine Deck	182,000	0.010	1.45
Manual	On and Above Turbine Deck	20,000	0.32	5.17
Non-Man.	Field Office	486,000	0.020	7.77
Non-Man.	Unit 2 Reactor Bldg Below Refueling Floor	54,700	0.010	0.44
Non-Man.	On Refueling Floor	6,080	0.24	1.17
Non-Man.	Unit 2 Turbine Bldg Below Turbine Deck	54,700	0.010	0.44
Non-Man.	On Turbine Deck	6,080	0.058	0.28
	TOTAL	1,378,160		30.73

(1) For remainder of Unit 2 construction.
(2) Based on an assumed availability of 80 percent for Unit 1.

12.5 HEALTH PHYSICS PROGRAM

12.5.1 ORGANIZATION

12.5.1.1 Introduction

The Health Physics program at Susquehanna SES is developed and implemented to evaluate and document plant radiological conditions and assure that every reasonable effort is expended to maintain personnel exposure as low as reasonably achievable (ALARA). The Health Physics Organization is displayed on Figure 12.5-1.

12.5.1.2 Responsibilities

The Health Physics Supervisor is responsible to the Superintendent of Plant. The Health Physics Supervisor is charged with the responsibility of providing the Superintendent of Plant with the information necessary to establish compliance with regulations pertaining to radiation safety, uniform enforcement of Station Health Physics requirements, and that every reasonable effort to minimize personnel exposures has been made. In addition, the Health Physics Supervisor is responsible for assuring the staff who implement the Health Physics program is trained and retrained in operational Health Physics principles applicable to Susquehanna SES.

The Health Physics Engineer is removed from the line function of day to day Health Physics activities to provide the latitude and time to develop and implement a station ALARA program that is responsive to plant status. The Health Physics Engineer's major responsibility is to provide the Health Physics Supervisor the information necessary to establish that every reasonable effort has been made to minimize personnel exposures.

The Health Physics Specialists assure implementation of the Station Health Physics program by supervision of routine and special survey and evaluation programs required by applicable regulations and procedures. The Health Physics Specialists' major responsibility is to provide the Health Physics Supervisor the information necessary to establish that survey and record keeping requirements are properly met and that plant activities receive appropriate Health Physics attention.

The Health Physics Monitors implement the Health Physics Program by performing routine and special surveys and providing Health Physics surveillance in accordance with Station Health Physics Procedures.

12.5.1.3 Authority

The Superintendent of Plant, ultimately responsible for all station activities including radiation safety, receives direct reports from the Health Physics Supervisor concerning the status of the Health Physics program. To assure uniform enforcement of Health Physics requirements, the Superintendent of Plant delegates his authority with respect to radiation safety to the Health Physics Supervisor. The Health Physics Supervisor has the authority to cease any work activity when, in his professional judgment, worker safety is jeopardized, or unnecessary personnel exposures are occurring.

The Health Physics Engineer has the independence and authority to assure that jobs are accomplished with minimal exposures. Independence from routine Health Physics activities allows the objectivity necessary for selective review and recommendation of work planning packages such as Radiation Work Permits (RWP), work requests, and special maintenance procedures, in accordance with station procedures. The Health Physics Supervisor delegates authority to the Health Physics Engineer to cease any work activity which is not being performed in accordance with As Low As Reasonably Achievable (ALARA) procedures. The Health Physics Engineer has the authority to conduct informal training and/or discussions with workers and supervisors regarding observed practices and ALARA recommendations.

The Health Physics Specialist has the authority to assure that jobs are conducted in accordance with Health Physics procedures and RWP requirements. The Health Physics Supervisor delegates the authority to the Health Physics Specialist to cease any work activity which is not being performed in accordance with RWP requirements.

In the absence of Health Physics Supervision, the authorities of the above positions may be delegated in accordance with Station Health Physics procedures to shift supervisors or assistant shift supervisors who have successfully completed Level IV training as described in Subsection 12.5.3.7. A member of Health Physics Supervision, or an individual meeting the minimum experience and qualification requirements of one or more of these positions, will be available for consultation regarding Health Physics and ALARA concerns.

The Health Physics Monitors implement Health Physics and RHP requirements under the direction of qualified supervision.

12.5.1.4 Experience and Qualification

The Health Physics staff, responsible for the Health Physics program at Susquehanna, will meet minimum experience and qualification requirements.

The Health Physics Supervisor will be an experienced professional in applied radiation protection at nuclear power plants or nuclear facilities dealing with radiation protection problems similar to those at nuclear power stations; familiar with the design features of nuclear power stations that affect the potential for exposures of persons to radiation; in possession of technical competence to establish radiation protection programs and supervisory capability to direct the work of professionals, technicians and journeyman required to implement such programs.

The Health Physics Supervisor will have a minimum of nine years of experience in applied radiation protection which is to include five years of professional experience. Four years of the experience requirement may be fulfilled by a bachelor's degree in a science or engineering subject. Three years of the professional experience will be in a nuclear power plant or nuclear facility dealing with radiological problems similar to those encountered in nuclear power stations. One year of professional experience may be fulfilled by a master's degree and two years may be fulfilled by a doctor's degree where course work related to radiation protection is involved.

The Health Physics Engineer will have a minimum of five years of experience in applied radiation protection in a nuclear power plant or a nuclear facility dealing with radiological problems similar to those encountered in nuclear power stations. Up to four years of the experience requirement may be fulfilled by related technical training or academic training in a science or engineering subject.

The Health Physics Specialist will have a minimum of four years of experience in applied radiation protection to include two

years of experience in a nuclear power plant or a nuclear facility dealing with radiological problems similar to those encountered in nuclear power stations. A maximum of two years of the experience requirement may be fulfilled by related technical training or academic training in a science or engineering subject.

To at all times assure adequate manpower for Health Physics supervisory functions, the experience and qualification requirements of the Health Physics Engineer and Health Physics Specialist positions may be reduced on a temporary basis. The Superintendent of Plant will approve or disapprove such action following review of the Health Physics Supervisor's recommendations and justification.

The Health Physics Monitor will meet the qualification requirements of Subsection 12.5.3.7.

12.5.2 FACILITIES, EQUIPMENT & INSTRUMENTATION

12.5.2.1 Control Structure Facilities

The facilities, shown in Figure 12.5-2, are located at the central access to the Controlled Zone, elevation 676', for efficiency of operation. Self-survey personnel monitoring equipment, such as hand and foot, portal, or Geiger-Mueller (G-M) type friskers, will be located at the exit from the central access control area. Self-survey requirements will be administratively imposed prior to exiting the Controlled Zone.

12.5.2.1.1 Health Physics Facilities

The Health Physics office and workroom are located along the Central Corridor. Job planning and Radiation Work Permit coordination may be conducted through the pass-thru window of the workroom. Portable radiation survey instrumentation as well as air monitoring and sampling equipment, self-reading dosimeters, and miscellaneous Health Physics supplies will be stored in the Health Physics Office and Workroom area. Health Physics equipment used for routine counting of smears and air samples such as end window G-M counters, alpha and beta scintillation detectors, and/or gas flow proportional counters will be located in the Health Physics Office to prevent cross contamination of chemistry samples and minimize counting room background variations. Health Physics samples requiring gamma isotopic

analysis and/or low level counting may be analyzed in the Counting Room. Health Physics use of the Counting Room will be coordinated with the Chemistry Group. The Health Physics office will be equipped with filing cabinets and bench tops for survey record keeping and RWP preparation.

Decontamination facilities at the central access control area consist of a main personnel decontamination area and auxiliary decontamination area. Auxiliary toilets and locker room are also provided. The personnel decontamination areas contain showers, sinks, and decontamination agents. Decontamination area ventilation is filtered through prefilter, High Efficiency Particulate Air (H.E.P.A.), and charcoal filters prior to exhaust through the Turbine Building vent. Sinks and showers drain to the chemical drain tanks for processing through the Liquid Radioactive Waste System. G-M type friskers will be located at these areas for personnel contamination monitoring.

Portable radiation survey instruments and self-reading dosimeters will normally be calibrated in the instrument calibration room using a calibration apparatus or appropriate neutron, beta and gamma sources, traceable to the National Bureau of Standards (N.B.S.). Sources will be stored in locked source containers and storage areas will be locked when not in use. Portable sources used to calibrate the area, process, and effluent radiation monitoring system as well as small solid and liquid sources used to calibrate the counting room instruments may be stored here. The calibration apparatus will utilize sources of varying strength and energy and/or varying thicknesses of shielding to provide a radiation field of known strength for use in calibrating portable radiation survey instruments. Provisions will be available for calibrating instruments in reproducible geometries. An N.B.S. calibrated condenser R-meter will be used to accurately measure radiation levels to determine source to detector distances for desired instrument calibration radiation levels. Record of calibration and repair for portable radiation detection instruments will be maintained on file. Instrument calibration may be performed by a qualified vendor. Records of such calibrations will be maintained.

The laundry room will be equipped with a washer and exhausted dryer to launder contaminated protective clothing/equipment. Facilities include a transfer table and stainless steel sorting table with exhaust hood. Protective clothing, laundered on site, will be selectively monitored for contamination, sorted and stored in the Protective Clothing area or Laundry Storage Room. Laundry detergents for protective clothing laundering, and other appropriate supplies will be stored in this area. The laundry table and dryer exhaust is discharged to the Turbine Building vent. Liquid laundry effluent will be collected in the Laundry

Drain Tanks for sampling and analysis prior to processing through the Liquid Radioactive Waste System.

The first aid room will contain a medical service center and toilet. Adequate supplies will be maintained to administer first aid for injuries requiring immediate attention. An inventory of first aid supplies will be performed at a frequency specified in station procedures to assure an adequate stock is maintained. Individuals requiring first aid will be checked by Health Physics personnel for wound contamination prior to administering first aid, when applicable.

The locker room contains approximately 100 lockers. Controlled zone workers may change from street clothing into plant clothing in the locker room. Personnel scheduled to work on Radiation Work Permit jobs may also change into clean protective clothing in the locker room. Adjacent to the locker room is a toilet and washroom, shower room and drying room.

Frequently occupied contaminated areas will have local change facilities with appropriate protective clothing supplies to minimize the spread of contamination from work areas.

Storage facilities will be located in the Central Access Control Area for storage of anti-contamination equipment, respiratory protective equipment, and miscellaneous Health Physics supplies.

The Emergency Equipment and Laundry Storage Room will be used for storage of protective clothing and emergency equipment.

12.5.2.1.2 Radiochemistry Facilities

Radiochemistry facilities consist of a sample room, radiochemistry laboratory, and counting room.

The sample room is shielded with 1'6" concrete walls and contains cabinets with worktops, sink, wall mounted storage cabinets and a fume hood assembly exhausted through prefilter, H.E.P.A. and charcoal filters to the Turbine Building vent.

The radiochemistry laboratory will be utilized for sample preparation and contains filtered fume hoods with service air connection, refrigerator, utility tables, sinks, cabinets, and drawers. The concrete walls range in thickness from 1' to 3'2". Fume hoods are exhausted through prefilter, H.E.P.A., and charcoal filters to the Turbine Building vent and the sinks drain to the Chemical Drain Tanks for processing through the Liquid

Radioactive Waste System. An emergency shower is accessible from both the radiochemistry and chemistry laboratories.

The Counting Room is constructed with 1'6" concrete walls to provide a low background environment for analysis of radiochemistry samples of station effluents and process streams. Instrumentation, such as a gas flow proportional counter, liquid scintillation counter, alpha and beta scintillators or crystals, end window G-M, and Germanium, Lithium drifted, Ge(Li) and/or Sodium Iodide (NaI), systems will be utilized for counting and/or analysis of radiochemistry samples.

12.5.2.1.3 Chemistry Laboratory

The chemistry laboratory contains an exhaust hood assembly with service air connection, drawers, worktops, sinks and laboratory equipment necessary for performing chemical analyses on non-radioactive plant materials. Station chemistry procedures will provide administrative control to assure that, under normal conditions, only non-radioactive materials are analyzed in the Chemistry Laboratory.

The laboratory exhaust hood discharges to the Turbine Building vent and the sinks drain to the neutralization tank for processing through the Liquid Radioactive Waste System.

12.5.2.2 Radwaste Building Facilities

The Radwaste Building elevation 646', 676' and 691'6" facilities are located as shown on Figures 12.5-3, 12.5.-4, and 12.5-5, respectively. Ventilation is filtered through prefilter and H.E.P.A. filters prior to exhaust to the Turbine Building vent. Drains discharge to the Chemical Drain Tank and Laundry Drain Tank for processing through the Liquid Radioactive Waste System.

12.5.2.2.1 Radwaste Building Elevation 646'0"

The facilities consist of a solid waste packaging, decontamination, and monitoring area, personnel decontamination facility and personnel decontamination facility adjacent to the laundry drain sample tank.

The solid waste packaging area contains an apparatus for remote capping operations, water spray nozzles for container

decontamination, and a remote smearing device for monitoring package surface contamination.

The two (2) personnel decontamination facilities contain showers, sinks and appropriate decontamination agents.

12.5.2.2.2 Radwaste Building Elevation 676'0"

The facilities consist of an instrument repair shop, sample room, repair area including a welding area, personnel decontamination room, controlled zone shop including a washdown area, monitoring and final decontamination area, and storage area.

The Instrument Repair Shop will be equipped with an assortment of tools and equipment necessary for work on contaminated instruments.

The sample room provides a central location for sampling various Radwaste Systems. Samples will be analyzed by the Chemistry group to determine the final disposition of the effluents being processed.

The Repair Area will be used for maintenance and welding of contaminated equipment. Appropriate tools and equipment, including welding equipment, will be stored in this area.

The personnel decontamination room will be equipped as those described at the central access control area. It is conveniently located to facilitate the rapid removal of contamination from personnel working in the instrument repair shops, sample room, repair area, or controlled zone shop.

The controlled zone shop will be equipped similar to the station machine shop. Repair of contaminated components will be performed in this area. The adjoining washdown area will be used for decontamination of components and equipment to be worked on in the controlled zone shop and is constructed with a 6" curbing.

The monitoring and final decontamination area will be used for surveying and decontamination, if necessary, of radwaste containers prior to storage.

A storage area is available on the 676'0" elevation for storage of anti-contamination equipment, respiratory protective equipment, and miscellaneous Health Physics supplies.

12.5.2.2.3 Radwaste Building Elevation 691'6"

The Health Physics facilities consist of a contaminated laundry and storage area, clean laundry and storage area, personnel decontamination area, Health Physics area, and janitor's closet.

The contaminated laundry area contains two (2) washers and two (2) exhausted dryers, an exhaust hood and sink, miscellaneous tables and carts, and a storage area for laundry detergents used in protective clothing laundering, disinfecting agents for cleaning of respiratory protective equipment, and other supplies. One washer will be labeled and administratively controlled by station procedure for use in cleaning of respiratory protective equipment only. Contamination limits will be specified by station procedure for the respiratory protective equipment washer. A separate area within the laundry facilities will be used for maintenance and repair of personnel respiratory protective equipment. Equipment will be cleaned in the designated washer, dried, inspected and disinfected, wrapped in plastic or paper bags, and stored in the Emergency Equipment and Laundry Storage Room, Radwaste Building Health Physics Area, or other designated area. The laundry effluent will be discharged to the laundry waste storage tanks for sampling prior to processing through the liquid radioactive waste system.

The clean laundry area contains two (2) washers and two (2) exhausted dryers, a sink, and miscellaneous tables and carts. The facility will normally be used for laundering of station clothing not used as anti-contamination clothing. Laundry drainage will be collected in the laundry drain tanks for sampling prior to processing through the Liquid Radioactive Waste System.

The personnel decontamination area will be equipped as those at the central access control area.

The Health Physics area will serve as an office for Health Physics personnel and storage area for Health Physics supplies and equipment in support of Radwaste activities. Equipment and instrumentation will include portable survey instruments, air samplers, counting equipment, respiratory protection equipment, contamination control supplies and other related Health Physics supplies.

12.5.2.3 Reactor Building Facilities

The Reactor Building elevation 719'1" facilities are located as shown in Figure 12.5-6.

Each unit has two (2) emergency personnel decontamination stations and a washdown area.

The two (2) emergency personnel decontamination stations contain showers and sinks, a monitoring station with frisker and protective clothing, and appropriate decontamination agents.

The washdown area will be used for equipment decontamination prior to maintenance and is constructed with a six (6) inch curb.

Ventilation from these areas is filtered through prefilter, H.E.P.A., and charcoal filters prior to exhaust through the Reactor Building vent. Drains discharge to the Reactor Building Sump, Chemical Drain Tank, and Laundry Drain Tank for processing through the Liquid Radioactive Waste System.

12.5.2.4 Turbine Building Facility

The 729' elevation of the Turbine Building contains a washdown area, with 6" curbing for turbine blading and component decontamination prior to maintenance. Ventilation from this area is filtered through prefilter, upstream H.E.P.A., charcoal, and downstream H.E.P.A. filters prior to exhaust to the Turbine Building vent. Drains discharge to the Turbine Building Chemical Radwaste Sump for processing through the Liquid Radioactive Waste System.

12.5.2.5 Guard House Building

The Guard House building serves primarily as the access control to the restricted area of the plant. Personnel dosimetry will normally be issued and stored at this area. A portal monitor and/or G-M type frisker will normally be maintained at this location as the final monitoring area prior to leaving SSES.

12.5.2.6 Health Physics Equipment

12.5.2.6.1 Protective Clothing

Protective clothing will be worn in contaminated areas to prevent personnel contamination and aid in controlling the spread of surface contamination. Protective clothing available at Susquehanna SES will include: reusable coveralls and lab coats, disposable coveralls and lab coats, plastic suits, surgeons caps, cloth hoods, plastic hoods, splash shields, cotton glove liners, cloth gloves, rubber gloves, disposable gloves, gauntlet gloves, rubber shoe covers, rubber boots, and disposable shoe covers.

Protective clothing will be stored at the Protective Clothing Area, Emergency Equipment and Laundry Storage Room (Figure 12.5-2), plant laundries (Figure 12.5-6), and selected local change areas. After use, protective clothing will be laundered and monitored, or surveyed, packaged and shipped to an off-site vendor for laundering, or discarded as radwaste.

12.5.2.6.2 Respiratory Protective Equipment

Respiratory protective equipment will be used to minimize the intake of radioactive material when engineering controls are not practicable. The Respiratory Protection Program is described in Subsection 12.5.3.5.

Respiratory Protective Equipment utilized at Susquehanna SES will consist of National Institute of Occupational Safety and Health/Mine Equipment Safety Administration, (N.I.O.S.H./M.E.S.A.) approved air purifying respirators, self-contained breathing apparatus (pressure demand), pressure demand air line respirators, constant flow air line respirators, and constant flow air line hoods, welding masks and plastic suits. A variety of respiratory devices will be available to assure proper fit of the differing facial contours of personnel requiring respiratory protection. Sufficient quantities of respiratory protective equipment will be available to allow for the use, decontamination, maintenance, and repair of equipment.

Respiratory Protective Equipment will be available at the Emergency Equipment and Laundry Storage Room (Figure 12.5-2), and Radwaste Building Health Physics Area (Figure 12.5-3, 12.5-4 and 12.5-5). Respiratory Protective Equipment will be available for emergency use at the Emergency Control Center and Control Room. N.I.O.S.H./M.E.S.A. approved emergency escape devices will be

placed at locations where the potential exists for an unexpected increase in radioactive or chemical airborne concentrations (such as the water treatment building and radwaste system). Fifteen (15) escape devices will also be located in the control room. If applicable, respiratory protective face pieces will be wrapped in plastic bags and stored individually to prohibit plastic deformation.

12.5.2.6.3 Air Sampling Equipment

Air sampling equipment will be available at the Health Physics office (Central Access Control Area, Figure 12.5-2) and the Health Physics Station (Radwaste Building, Figure 12.5-3).

Airborne activity levels will be determined by the use of continuous airborne monitors (CAMS), high and low volume portable air samplers, and breathing zone air samplers. Five (5) CAMs, five (5) high volume air samplers, five (5) low volume air samplers, and two (2) impactor attachments will be available for use at Susquehanna SES.

The CAM(s) can be used to measure particulate and gaseous activity. The air samplers can be used to measure particulate and iodine activity using the appropriate filtering medium. Particulate activity and particle size distribution can be determined using an impactor attachment. Volumes necessary for representative samples will be specified in Station Health Physics procedures. Filter media such as H.E.P.A. filters and charcoal cartridges will be stored at the Health Physics office and Workroom Area.

12.5.2.6.3.1 Continuous Air Monitors

CAMS will normally be used to sample selected areas of potential airborne concentrations. CAM sampling rates will be checked against calibrated rotometers or wet test meters on a quarterly basis and after pump replacement or repair. If CAM's are equipped with strip chart recorders or local readout, a base line sampling program will be completed prior to Unit 1 fuel load to allow estimation of naturally occurring isotopes' contribution to airborne background. CAM detector response to an appropriate check source will be performed on a quarterly basis. Manufacturer's recommended calibration or voltage plateau procedures will be performed on a quarterly basis. If applicable, operation of local alarms will be verified on a

quarterly basis.

12.5.2.6.3.2 Portable Air Samplers

When possible, each portable air sampler will be monitored for flow rate as above. Devices utilizing flow meters will be checked against calibrated rotometers or wet test meters when practicable. Manufacturers' certification of flow rate will be utilized when physical flow measurements are not possible due to equipment design.

12.5.2.6.3.3 Breathing Zone Samplers

Ten (10) battery powered breathing zone samplers will be available for use in evaluating air concentrations that radiation workers may encounter. Personnel breathing zone samplers will be checked for flow rates as above if practicable. If design prevents physical flow measurement, manufacturer's certification of rated flow or accuracy of flow meter will be utilized.

12.5.2.6.3.4 Sampling Media

Particulate air concentrations will be sampled with H.E.P.A. sampling media or impactor attachments. Manufacturer's certification of collection efficiency will be utilized in calculations of airborne concentrations.

Surveys for radioiodine concentrations will normally utilize charcoal in a reproducible geometry such as a cartridge. If studies to determine various forms of radioiodine are required, reproducible geometries of materials such as cadmium iodide, 4-iodophenol, and silver zeolite may be used with charcoal in various configurations. If charcoal impregnated filter paper is utilized in equipment such as breathing zone monitors, manufacturers recommended sampling rates and times will be followed whenever practicable.

12.5.2.6.3.5 Special Air Sampling

Water bubblers, dessicant columns, or cold traps will be available for tritium air sampling, and gas sample containers (such as Marinelli containers) will be available for special gaseous air sampling.

12.5.2.6.4 Personnel Dosimetry

The Personnel Dosimetry program is described in Subsection 12.5.3.6. Self-reading dosimeters of five different ranges for use at Susquehanna SES are as follows:

Range (mR)	Normal Use	Number Available
0-200	Low Dose Accumulating Work	400
0-500	Intermediate Dose Accumulating Work	75
0-1000	High Dose Accumulating Work	50
0-5,000	Radiation Emergency Plan Use	25
0-100,000	Radiation Emergency Plan Use	25

A total of five (5) dosimeter chargers will be available at the Central Access Control Area Health Physics Office and the Radwaste Building Health Physics Station. Self-Reading Dosimeters will also be available at these locations. Dosimeters will be tested for calibration response charging drift, and leakage prior to initial use, and on a six month frequency thereafter.

If vendor service is not utilized, approximately 1500 thermoluminescent dosimeters (TLD) will be available for use as the dosimetry of record. TLD(s) will be used for neutron, beta, and gamma exposure and will normally be evaluated on site. Approximately one hundred (100) extremity TLD devices will be available for issue when authorized by Health Physics personnel.

If applicable, a TLD reader will be installed and calibrated in accordance with vendor's instructions. Operation will be conducted by qualified individuals in accordance with approved station procedures. A performance testing program will be implemented to assure the TLD reader is properly calibrated and exposure information is accurate.

Internally deposited radioactive material will be evaluated with a whole body counter sufficiently sensitive to detect in the thyroid, lungs, or whole body a fraction of the permissible body/organ burden for gamma emitting radionuclides of interest. The whole body counter will be calibrated on a quarterly basis using phantoms and standard solutions of various radionuclides such as Co-60, Ce-137, and Ba-133. The detectors will be used in conjunction with a multi-channel analyzer and associated readout to obtain a permanent record. A vendor whole body counting system on or off site may be used as an alternative or supplement to a PP&L whole body counter.

Ten (10) battery powered personnel alarm dosimeters will be available for use when an audio alarm at a preset accumulated exposure or exposure rate may be advantageous. Personnel alarm dosimeters will be checked for accuracy on a quarterly basis and following any repair affecting calibration.

12.5.2.6.5 Miscellaneous Equipment

The following miscellaneous Health Physics equipment will be stored at various locations in the plant:

Contamination control supplies such as glove bags, contaminent tents, absorbent wipers, absorbent paper, rags, step-off pads, rope, plastic sheets, plastic bags, tape, contamination area signs, and protective clothing. Appropriate supplies may be assembled into kits and located throughout the plant to aid in the control of a contaminated spill.

Temporary shielding, such as lead bricks, lead sheets, and lead wool blankets, will be available to reduce radiation levels.

A trash compactor located on the 676' elevation of the Radwaste Building as shown in Figure 12.5-4. This location will provide adequate storage and access for loading at the rear truck access door of the Radwaste Building. The compactor and room will be vented through prefilter, H.E.P.A., and charcoal filters prior to exhaust to the Turbine Building vent.

A fitting apparatus for quantitative test fitting of individual involved in the Respiratory Protection Program. The apparatus will be a sodium chloride (NaCl) aerosol generator with flame photometer, or equivalent system, to measure airborne concentrations. Irritant smoke and/or isoamyl acetate will also be available to qualitatively test respirator fit.

12.5.2.7 Health Physics Instrumentation

Instruments for detecting and measuring alpha, beta, gamma and neutron radiation will consist of counting room, and portable radiation survey/monitoring instruments. All instruments will be subjected to operational checks and calibration to assure the accuracy of measurements of radioactivity and radiation levels. Primary and reference standards (utilizing, or prepared from, standards of Sr-90, Am-241, Cs-137, Co-60, H-3, and others, traceable to the National Bureau of Standards) will be used to maintain required accuracies of measurement. Background and efficiency checks of routinely used Health Physics counting equipment will be performed daily and these instruments will be recalibrated whenever their operation appears statistically to be out of limits specified in Station procedures. Routine calibrations will be performed on counting room instrumentation and radiation survey/monitoring instruments on a quarterly basis and after repairs affecting calibration. Efficiency curves for multi-channel analyzer systems will be determined on a semiannual basis using N.B.S. traceable sources for various reproducible geometries. Sufficient quantities of instrumentation will be available to allow for use, calibration, maintenance, and repair.

The instrumentation described in these Subsections may be replaced by equipment providing similar or improved capabilities.

12.5.2.7.1 Counting Room Instrumentation

Counting Room instruments for radioactivity measurements will include the following:

A 4096 channel analyzer, using a 3" x 3", 7% resolution Na I crystal, and a 5 Kev resolution (at 1 Mev energy full width half maximum peak) GE(Li) detector, for identification and measurement of gamma emitting radionuclides in samples of reactor primary coolant, process streams, liquid and gaseous effluents, airborne and surface contaminants.

One computer which can be interfaced with a pulse height analyzer; equipped with a teletype machine for entering instructions and printing results, a tape deck for entering programs and storing data, and an X-Y plotter for making graphs.

A low background gas flow proportional counter used for gross alpha and gross beta measurements of prepared samples.

A liquid scintillation beta counter used for measurement of tritium in reactor primary coolant, liquid and gaseous wastes, and gross beta activity other than tritium.

A NaI well crystal with counter-scaler or pulse height analyzer used for gamma analysis of various radionuclides in samples of reactor primary coolant, liquid and gaseous wastes, or prepared samples.

One beta-gamma counter-scaler, thin end window (2 mg/sq.cm, 2-inch diameter G-M) used for gross beta-gamma measurements of reactor primary coolant or prepared samples.

One alpha scintillator or semiconductor crystal used for gross alpha measurements of reactor primary coolant and prepared samples.

One beta scintillation counter-scaler used for gross beta measurements of reactor primary coolant and prepared samples.

12.5.2.7.2 Health Physics Office and Workroom Instrumentation

Health Physics instrumentation normally located in the Health Physics Office and Workroom will include the following instruments, or equivalent:

One (1) automatic and one (1) manual beta-gamma counter-scaler, thin end window (2 mg/sq.cm), 2 inch diameter G-M, used for gross beta-gamma measurements of removable contamination, air samples and nasal swabs.

An alpha scintillation or semiconductor counter-scaler used for evaluation of removable contamination, air samples and nasal swabs.

A low background gas flow proportional counter used for gross alpha and/or beta measurements of removable contamination, air samples and nasal swabs.

Ten (10) G-M beta-gamma survey meters (most sensitive range 0-.2 mR/hr., maximum range 0-2 R/hr., with internal probe) used for detection of radioactive contamination on surfaces and for low level exposure rate measurements.

Ten (10) ionization chamber beta-gamma survey meters 0-5 rem/hr. (0-5 mrem/hr. most sensitive range) used to cover the general range of dose rate measurements necessary for radiation protection evaluations.

Five (5) wide range ionization chamber beta-gamma survey meters (0-5 mR/hr. most sensitive range, maximum range 0-50 R/hr.) used for exposure rate measurements.

Four (4) remote monitoring (telescoping probe) G-M tube beta, gamma survey meters, 0-1000 R/hr, 0-2 mR/hr most sensitive range used for exposure rate measurements.

One (1) cadmium loaded polyethylene sphere, BF_3 tube, neutron Rem Counters 0-5 rem/hr (0-5 mrem/hr most sensitive range). The instrument is used to measure the dose equivalent rate due to thermal, intermediate, and fast neutron fluxes.

Two (2) alpha scintillation survey meters, 0-2M cpm (0-2K cpm most sensitive range) 30% efficiency used for measurement of alpha surface contamination.

One (1) thermal and fast BF_3 tube, paraffin moderated, neutron detectors, ("Lin-Log" decades of 0-500, 500-5,000, 0-50,000 and 50,000-500,000 cpm which is equivalent to 0-10,000 n/sq.cm/sec. for 1 Mev neutrons). Designed to detect thermal neutrons with detector removed from moderator and fast neutrons with detector inserted in moderator. An indication on the meter can be correlated with a known neutron flux and a known energy to obtain cpm/n/sq.cm - sec (flux) which in turn can be converted to mrem/hr.

12.5.2.7.3 Health Physics Radwaste Building Instrumentation

Health Physics Instrumentation normally located at the Health Physics Station in the Radwaste Building will include the following:

One (1) thin window (2 mg/sq.cm) G-M detector with counter-scaler for gross beta-gamma measurements on smears and prepared samples.

Five (5) Ionization chamber beta-gamma survey meters 0-5 rem/hr., (0-5 mrem/hr. most sensitive range) used for general survey work.

Two (2) G-M beta-gamma survey meters (most sensitive range 0-.2 mR/hr., maximum range 0-2 R/hr., with internal probe) used for detection of radioactive contamination on surfaces and for low level exposure rate measurements.

One (1) remote monitoring (telescoping Prope) G-M tube Beta, Gamma survey meter 0-1000 R/hr., 0-2 mR/hr. most sensitive range used for exposure rate measurements.

12.5.2.7.4 Personnel Contamination Monitoring Instrumentation

Personnel monitoring instruments consisting of friskers, portal monitors and a hand and foot monitor described below, will be used at the locations specified in Subsection 12.5.2.1:

Twenty-five (25) beta-gamma geiger count rate meters, 2mg/sq.cm window, 0-50,000 cpm range, adjustable audio and/or visual alarms. Gamma sensitivity for Co-60 is 3,500 cpm/mR/hr. Beta sensitivity (1" diameter source, 2 pi):

Sr-90/Yr-90 (E max. 0.54-2.2 Mev) = 45%

C-14 (E max. 0.15 Mev) = 10%

Used to detect contamination on personnel, materials, protective clothing, and equipment.

Two (2) portal monitors consisting of eight audio and/or visual alarmed G-M detectors to provide head to foot beta-gamma detection capability. Count rate alarm adjustable from 160-7000 cpm; counting time adjustable from 1 to 10 seconds.

One (1) audio and/or visual alarmed hand and foot monitor with ports monitored by G-M detectors for the hands and feet and an external probe for frisking the body.

Personnel contamination monitoring instrumentation will be calibrated on a quarterly basis or following repair, in addition to monthly source checks, to determine proper response and alarm operability.

12.5.2.7.5 Miscellaneous Health Physics Instrumentation

One (1) Condensor R-meter used to accurately measure radiation levels consisting of one (1) low energy chamber (0.025R) and three (3) high energy chambers (0.25R, 2.5R and 25R), which have been N.B.S. calibrated.

Other equipment used for Health Physics related functions will be maintained and controlled in accordance with station procedures. Such ewuipment may include:

One (1) pulse generator for calibraing pulse counting instruments. One (1) wet test meter, one (1) calibrated flow meter, one (1) velometer, and one (1) magnehelic pressure differential gauge.

The location of the area, process, and effluent radiation monitoring systems are described in Sections 11.5 and 12.3.

12.5.3 PROCEDURES

The Health Physics Procedure Program, as described in this section, will be implemented by Susquehanna SES Health Physics Technical, Operating, and As Low As Reasonably Achievable (ALARA) procedures in accordance with Section 13.5.

12.5.3.1 Control of Access and Stay Time in Radiation Areas

Physical and administrative controls will be instituted to assure the philosophy of maintaining personnel exposures as low as reasonably achievable (ALARA), as specified in Section 12.1, is implemented.

12.5.3.1.1 Physical Controls

12.5.3.1.1.1 Security Check Point

The security check point at the fence line perimeter will be a continuously manned physical control. Assigned personnel dosimetry devices and identification badges will be stored at this location when not in use. The security force will assure that all personnel who enter the station are issued appropriate badges and dosimetry in accordance with station procedures. A restricted area access list will be maintained at the security entrance. Any individual not on the access list must be accompanied by a person who is authorized unescorted restricted area access. The training, retraining and testing requirements for unescorted access are described in the Susquehanna SES Security Plan.

12.5.3.1.1.2 Security Doors

Although not primarily intended to control access to radiation areas, the security interlocked door system will assure only specifically trained and authorized individuals are able to open security entrances to the reactor, turbine, radwaste and diesel generator buildings. Security entrances will be locked or

provided with continual surveillance. Details of security access control are contained in the Susquehanna SES Security Plan.

12.5.3.1.1.3 Posting and Locking

A third physical control will be the posting and locking, as appropriate, of radiation and high radiation areas. Radiation areas, as defined in 10CFR20.202(b), will be posted in accordance with 10CFR20.203(b). Plant areas that are routinely accessible will be surveyed in accordance with station procedures to determine radiation levels. In addition to recording the results of these surveys in accordance with 10CFR20.401(b), the radiation area signs will be updated by surveyors to reflect current conditions. Every reasonable effort will be expended to erect rope or other physical barriers to minimize inadvertent entry in radiation areas.

High radiation areas, as defined in 10CFR20.202(b), will be posted in accordance with 10CFR20.203(c). These signs will be routinely updated to reflect current conditions. Surveys of high radiation areas will be performed and results recorded as above. Each entrance to a high radiation area will be equipped with audible and/or visible alarms in accordance with 10CFR20.203(c)(2)(ii) or controlled in accordance with 10CFR20.203(c)(2)(i) or (iii).

In lieu of the above controls, high radiation areas in which the intensity of radiation is greater than 100 mrem/hr. but less than 1000 mrem/hr. may be barricaded and conspicuously posted as high radiation areas and entries controlled by issuance of a Radiation Work Permit. In addition, areas in which the intensity of radiation is greater than 1000 mrem/hr. will be provided with locked doors under the administrative control of the Shift Supervisor. Controls utilized at entrances will at all times permit egress from high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

A radiation monitoring device which continuously indicates the radiation dose rate in the area.

A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a present integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control

over the activities within the area and shall perform periodic radiological surveillance at the frequency specified by Health Physics Supervision on the Radiation work permit.

Entrances to radiation areas and high radiation areas will be posted to reflect the requirement of a Radiation Work Permit (RWP) in accordance with limits specified in station procedures.

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12.5.3.1.1.4 Surveillance

When appropriate, surveillance of work activities will be provided to assure a positive control of access and stay time in radiation areas. Surveillance will be utilized when it is necessary to assure accurate record of working time as an assistance to the work group. In addition, it may be utilized for tasks involving large numbers of workers to assure control at the staging or entry point. Surveillance may also be provided for tasks in areas where conditions are unstable to assure that timely instructions to workers are issued.

12.5.3.1.2 Administrative Controls

12.5.3.1.2.1 Training

As specified in Subsection 12.5.3.7 personnel allowed unescorted restricted area access will receive Health Physics and related training in accordance with 10CFR19.12. During this training, the individual responsibility of utilizing proper Health Physics procedures in radiation areas will be emphasized. The methods utilized at Susquehanna SES to control access physically and administratively will be reviewed. Supervisory or other personnel responsible for the direction of workers may receive additional Health Physics training that will include guidance on work planning, controlling access, utilizing shielding and distance, and minimizing stay time in radiation areas.

12.5.3.1.2.2 Radiation Work Permit

The Radiation Work Permit (RWP) system described in Subsection 12.5.3.2 will be implemented to administratively control access and stay time in radiation areas. Work in radiation contamination or airborne levels greater than limits specified by station procedure will require the completion and approval of a RWP. For personnel or groups who must routinely enter specific areas as a necessary part of work duties, a Standing Radiation Work Permit (SRWP) may be issued in accordance with station procedures. Application for the SRWP will specify the need for routine entry, and expected occupancy per day, week or month. The SRWP application will receive a survey, review and approval process similar to that described in Subsection 12.5.3.2. Approved SRWP's will specify access and record keeping requirements as well as special instructions and maximum stay

time. An approved SRWP will be considered in effect until conditions warrant a change and will be subject to immediate cancellation by the Health Physics Supervisor or designated alternate. Each SRWP will be reviewed on a monthly basis by a Health Physics representative.

12.5.3.1.2.3 Reporting Requirement

The individual responsibility to report, through proper chain of command, any violation of Federal Regulations or Station procedures will be emphasized during training sessions. Violation involving potential exposure of personnel to radiation or radioactive material will be reported through appropriate channels to the Superintendent of Plant or designated alternate. Appropriate action will be taken to prevent recurrence. Any individual who violates station procedures will be subject to disciplinary action.

12.5.3.1.2.4 Independent Review

A member of Health Physics Supervision will periodically observe activities in RWP areas to review the effectiveness of specified precautions. In addition, a member of Health Physics supervision may perform independent measurements of radiation levels to assure that areas are properly posted to indicate accurate readings. During these surveys, the reviewer will assure that every reasonable effort has been expended to minimize inadvertent entry in radiation areas.

12.5.3.1.2.5 Procedure Review

Health Physics procedures related to control of access and stay time in radiation areas will at all times be subject to review to assure every reasonable administrative effort has been expended to minimize personnel exposure. Recommended changes will be evaluated and, if necessary, a proposed change will be forwarded through appropriate review and approval channels. Approved changes requiring retraining will be forwarded to the Training Supervisor for scheduling and implementation. Health Physics procedures will be reviewed annually.

12.5.3.2 Assuring that Occupational Radiation Exposure (ORE) Will Be As Low As Reasonably Achievable (ALARA)

To effectively implement the corporate ALARA commitment as stated in Section 12.1, a station ALARA program will be utilized to assure that activities are performed with the lowest practicable personnel exposure. PP&L considers it necessary to apply the basic concepts of ALARA to internal and external exposure to assure proper emphasis on both modes of potential exposure. Procedures employed to implement the program described in this section will be subject to review and revision to assure the ALARA program is responsive to plant conditions.

12.5.3.2.1 ALARA Procedures Common to External and Internal Exposure

12.5.3.2.1.1 Training

Individuals allowed unescorted restricted area access will receive Health Physics training as described in Subsection 12.5.3.7. The individual responsibility of assuring that unnecessary exposure is to be avoided will be emphasized during Health Physics Training sessions.

As appropriate, individuals involved in potentially high dose accumulating jobs will receive pre-job training in exposure reduction techniques and controls applicable to the specific job.

12.5.3.2.1.2 Radiation Work Permit

Where radiation dose rates, anticipated accumulated exposures, airborne concentrations, or contamination levels exceed limits specified by station procedures, a Radiation Work Permit (RWP) will be initiated, completed and approved prior to commencement of scheduled work. As a minimum, station procedures will specify that scheduled work in Zone IV or higher (greater than 15 mRem/hr.) will require completion of a RWP.

Health Physics will evaluate the radiological conditions associated with the work to be performed. Based upon evaluation of proposed work and surveys, Health Physics will specify the appropriate protective clothing/devices, respiratory protective

equipment, dosimetry, special samples, surveys, procedures, precautions to be taken, and expiration date

The RWP will be evaluated to assure the work will be performed from an ALARA approach. As appropriate, the evaluation will include review of proposed special tools, remote handling devices, access and communications needs, minimum manpower requirements, and work which may be performed outside of the RWP area to increase job efficiency and reduce personnel exposures. Potential incidents such as fires, spills, and equipment failure will be evaluated and proper response action discussed with radiation workers, when applicable. For high dose accumulating work, job preplanning will include Man-Rem estimates, comparison with similar jobs, establishing exposure goals, and simulated dry run, as appropriate, to increase job efficiency.

Radiological engineering controls will be used, when applicable, to minimize personnel exposures and prevent the spread of contamination and/or inhalation/injestion of radioactive material. Controls such as flushing of tanks and lines, use of temporary shielding, use of proper ventilation and purging, and properly filtered temporary exhaust will be considered. In addition, other effective methods of reducing man-rem exposures and potential intake of radioactive material will be considered. When airborne concentrations cannot be reduced below station limits, the use of respiratory protective devices will be considered.

The RWP will be approved and signed by the Health Physics Supervisor or designated alternate prior to commencement of work. RWP implementing process will be detailed in Station procedures.

A member of Health Physics supervision will selectively review completed and returned RWP's. Selection, on a variety of bases, of those RWP's which should receive a post operation evaluation will be made. Arrangements will be made, when necessary, to hold a de-briefing session with the responsible supervisor and/or workers. De-briefing and RWP review will be conducted when unexpected airborne concentrations, high man-rem exposures or high individual exposures are encountered. De-briefing will emphasize and analyze problems or difficulties encountered during performance of work. Alternative work methods will be discussed and if improvements are practicable, the responsible supervisor will initiate the review, approval and implementation process.

12.5.3.2.1.3 Work Scheduling

Use of the Radiation Work Permit system described in the previous section will establish a data base from which supervisory staff will be able to efficiently schedule workers. Health Physics will provide reports to supervisors that will indicate current individual exposure status to assist in work scheduling and assure individual exposures are minimized.

12.5.3.2.1.4 Reporting Requirements

Activities performed under an approved Radiation Work Permit must be carried out in accordance with the provisions of the RWP. Violation of RWP requirements will be reported to Health Physics and appropriate supervision. RWP violations that cause or threaten to cause unnecessary personnel exposure may result in disciplinary measures.

12.5.3.2.1.5 Internal Program Reviews

In an effort to provide more efficient methods of control, evaluation, and reporting, a member of Health Physics supervision will conduct reviews of the RWP program and procedures utilized to minimize personnel exposure. Results of internal reviews will be reported to appropriate levels of station management and the ALARA Review Committee. In addition, the Health Physics group will perform special reviews or studies requested by corporate committees to assist management in assuring that all aspects of the ALARA program are implemented.

12.5.3.2.1.6 Exposure Goals

On major dose accumulating job functions, total man-rem and/or man-MPC-hours (man-hours x ratio of measured airborne concentration to Maximum Permissible Concentration), exposure goals may be established prior to commencement of scheduled work. A general goal will be based on the lowest dose commitment recorded on jobs of similar nature. A general goal of equaling or bettering the lowest total worktime expended on jobs of similar nature may be utilized when airborne concentrations or dose rates are unpredictable or subject to variations. These general goals may be modified if work tasks are not identical or estimated if there is no available historical data. Significant

areas. The data from analyses of these air samples will be used to assist in future job planning and demonstrate that exposures to airborne material are as low as reasonably achievable. When portable BZ samplers are not practicable, temporary air samplers located close to the breathing zone of workers may be utilized.

12.5.3.2.3.5 Routine Air Sampling

Continuous air monitors will be placed in representative areas to sample those locations where airborne concentrations may be generated. These samplers will be periodically checked to verify proper function and assure that unexpected airborne concentrations are detected at the earliest possible time. The air sampling program is described in Subsection 12.5.3.5.

12.5.3.2.3.6 Control of Absorption and Ingestion

When work is scheduled on equipment or systems that contained or may contain radioactive liquids, every reasonable effort to prevent skin contact with radioactive solutions will be expended. Items such as plastic suits, rubber gloves and/or gauntlets, high rubber boots, face shields and hoods may be utilized as appropriate to the task to be completed.

Ingestion of radioactive materials will be minimized by assuring that adequate protective equipment is properly worn, removed, stored, laundered and surveyed. These physical controls in conjunction with administrative requirements and training in the areas of self-survey, prohibition of eating and smoking in contaminated areas and, decontamination techniques will assure that potential ingestion of radioactive material is minimized.

12.5.3.2.3.7 Control of Area and Equipment Contamination Levels

Contaminated areas and equipment will be decontaminated to as low a level, as practicable. Special emphasis will be placed on items that may be inadvertently touched by personnel and areas sufficiently contaminated so as to pose the potential for an airborne concentration. Supervisory staff will be responsible for assuring that work areas are maintained in a neat and orderly manner. The housekeeping practices employed will facilitate clean-up and decontamination efforts and thus minimize personnel stay time in radiation/contamination areas.

12.5.3.2.3.8 Airborne Exposure Evaluation

Exposures to airborne radioactive material will be tabulated to aid in work planning and demonstrate the effectiveness of the internal ALARA program. Air sample results in terms of a fraction or multiple of the maximum permissible concentration (MPC) for identified or unidentified isotopes multiplied by the work times will permit a running tabulation of individual and group MPC-hour exposures. When respiratory protection is employed, appropriate reductions of intake will be based on recommended protection factors. Subsection 12.5.3.5 describes the respiratory protection program.

12.5.3.2.3.9 Administrative Limits

To minimize potential intake of radioactive material in excess of Federal limits, station limits will be established. Airborne exposure or intake in excess of these limits may require work restriction, use of respiratory protection, or special in-vivo or bioassay studies.

12.5.3.3 Radiation Surveys

The Health Physics program will utilize a comprehensive system of radiation surveys to document plant radiological conditions and identify sources of radiation that contribute to occupational radiation exposure. The radiation survey program will be subject to evaluation by Health Physics supervision to assure that necessary data is collected while exposures to surveyors are as low as reasonably achievable.

12.5.3.3.1 Radiation Survey Program Controls

12.5.3.3.1.1 Record Review

A member of Health Physics supervision will review radiation survey records to assure that adequate readings are taken and properly recorded. If a need for additional data is noted, supervision will assure that such readings or supplemental surveys are taken and recorded. In addition, supervision will review data to assure that unwarranted readings that contribute to time spent in radiation areas are not taken. If appropriate,

deviations above established goals will be investigated by Health Physics and/or the responsible supervisor. Methods to improve performance on future jobs will be investigated and implemented, if appropriate.

12.5.3.2.1.7 Job Pre-planning

When applicable, tasks to be performed under the provisions of a Radiation Work Permit will be pre-planned. The responsible supervisor will assure that individuals selected to perform the task are familiar with the appropriate procedures to be employed. Supervision will also assure that, when applicable, a tool list to include special tools that will reduce exposures is completed and reviewed. When practicable, the responsible supervisor will observe dry-run procedure performance. This training may be observed by a Health Physics representative to make time study records as an aid in estimation of exposure or worktime goals. Special emphasis will be placed on job pre-planning for work in high radiation areas to maximize the use of temporary shielding and distance and minimize the work time.

12.5.3.2.1.8 Worker's Recommendations

An informal mechanism of soliciting worker's recommendations for improvement of job efficiency will be utilized to evaluate alternative work methods. Supervisors will encourage workers to present alternatives that will reduce work time in radiation areas and airborne concentrations. Responsible supervisors may consult with Health Physics during or following evaluation of a recommended change to assure that individual and group exposures will not be adversely affected. Changes in methods or equipment that are anticipated to improve efficiency and reduce exposure will be reviewed, approved and implemented in accordance with station procedures.

12.5.3.2.2 External ALARA

12.5.3.2.2.1 Dosimeter Evaluations

Each RWP issued to permit work or entry in a radiation field will require each worker to wear at least one pocket dosimeter. Time and exposure record log sheets will be posted with the Radiation Work Permit near the general work location. The dosimeter log

sheets will be reviewed and running totals will be updated. The responsible individual will assure required data is properly recorded and will forward the final dosimeter log sheets and the completed RWP to Health Physics following work completion.

12.5.3.2.2.2 Categorization of Exposures

Exposures incurred on RWP tasks will be categorized by type of worker(s), work group, and job function. To facilitate collation of data, scheduled work functions will be coded and entered on the Radiation Work Permit, when applicable. In addition, plant system codes will be developed for RWP use. Whenever applicable, the equipment component number will also be recorded on the RWP. This system will allow an exposure history data base to be collected by equipment, system, and work function, and thus permit supervisors and Health Physics personnel access to definitive records when planning RWP tasks.

12.5.3.2.2.3 Work Time Evaluation

Recording entry and exit times will allow total man hours spent on particular tasks to be tabulated. Under favorable conditions, a comparison of exposure rate multiplied by man hours expended and measured dosimeter individual or group totals may be made to assure proper data entry and verify that no significant exposure rate changes occurred. The man hours expended will also be used as a data base to assist supervisory staff in planning work of similar nature.

12.5.3.2.2.4 Special Alarms and Instruments

The use of special alarms and instruments will be evaluated. Alarmed timers may be used to warn workers they are approaching the maximum allowable work time. Remote radiation monitors may be installed in the general work area to allow readouts in lower radiation areas. Portable survey instruments may be placed in the work area to allow workers to monitor changes in exposure rate. Radiation rate meters and integrating devices with audible pre-set alarms may be used to warn workers of unexpected radiation levels or dose accumulation.

12.5.3.2.2.5 Temporary Shielding

During the planning phase of RWP work, supervision will evaluate the use of temporary shielding. Care will be taken to assure that installation and removal of shielding will not cause larger man-rem total exposures than expected without its use. Every reasonable effort will be made to utilize temporary shielding, such as lead blankets, that can be quickly installed on initial entry and easily removed upon exit.

12.5.3.2.2.6 Special Tools and Apparatus

Every reasonable effort will be expended to assure special or modified tools are available for specific tasks. Available tools that will significantly reduce stay time in radiation areas and maximize distance from radioactive sources will be included on job procedure tool lists. Appropriate supervisors will review tasks to identify procedures that may be improved by modifications or replacement of tools and/or apparatus.

12.5.3.2.2.7 Non-RWP Work Review

Health Physics personnel will review radiation surveys to identify areas not normally meeting RWP criteria. These areas will be studied to locate those of the highest occupancy frequency and/or duration of stay time. Health Physics may make recommendations pertaining to shielding or occupancy limits. These recommendations will be implemented whenever practicable to assure that the exposures incurred in low dose rate areas are as low as reasonably achievable.

12.5.3.2.2.8 Administrative Limits

Administrative limits will be implemented by station procedures to maintain personnel exposures ALARA with respect to Federal Limits. Station exposure limits may be exceeded only after approval of the Health Physics Supervisor, or designated alternate. Unapproved exposure exceeding station limits will be investigated by Health Physics to identify causes and establish methods to prevent recurrence.

12.5.3.2.3 Internal ALARA

12.5.3.2.3.1 Engineering Controls

Minimizing airborne concentrations by utilizing practicable engineering or physical controls will assure that occupational exposures are as low as reasonably achievable. Airborne concentrations will be minimized by appropriate use of containment techniques, temporary exhaust mechanisms, and review of air flow patterns and velocities. Control and evaluation of airborne radioactivity is described in Subsection 12.5.3.5.

12.5.3.2.3.2 Respiratory Protection

When engineering controls are not practicable, the use of respiratory protection will be evaluated. Respiratory protection may be utilized to minimize the intake of radioactive material. The respiratory protection fitting and training program is described in Subsection 12.5.3.5.

12.5.3.2.3.3 Pre Work Air Surveys

When RWP requests indicate that work is required in airborne radioactive material concentrations, appropriate air samples will be taken. These samples will normally be of short-term, high volume nature in order to obtain representative data in the shortest period of time. Any area that is posted as an Airborne Radioactivity Area will be sampled and analyzed prior to commencement of scheduled work. Whenever practicable, surveyors will utilize respiratory protection and/or remote air samplers to minimize their exposures. When existing airborne radioactive materials are not specifically identified, the MPC (Maximum Permissible Concentration) for unidentified alpha and/or beta-gamma materials will be used for scheduling, criteria for respiratory protection, and calculations of anticipated MPC-hours of exposure.

12.5.3.2.3.4 Special Air Sampling

When applicable, air samples will be taken with portable breathing zone (BZ) air samplers equipped with appropriate filter media during work in actual or potential airborne radioactivity

Health Physics supervision will assure that proper corrective measures are taken.

12.5.3.3.1.2 Independent Reviews

To assure proper performance of job duties by surveyors, a member of Health Physics supervision will perform independent reviews which may include physical measurement of radiation levels in areas previously surveyed. Review data will be compared with survey records and posting of warning signs. The reviewer may accompany surveyors to observe and verify proper survey techniques. Deviations from approved Health Physics procedures or discrepancies in radiation measurements will be investigated and results reported to the Health Physics Supervisor or designated alternate. Appropriate corrective measures will be taken to prevent recurrence.

12.5.3.3.1.3 Surveyor Dose Evaluation

Every reasonable effort will be expended to assure that occupational radiation exposure to surveyors is maintained as low as reasonably achievable consistent with providing sufficient survey data required for minimizing total plant exposures. Surveyors' radiation exposure will be tabulated in accordance with the ALARA program described in Subsection 12.5.3.2. Health Physics personnel will be issued appropriate dosimetry to be worn during radiation survey work. Beginning and ending dosimeter readings will be recorded. Individual exposures incurred during the survey may be reviewed and compared with previous surveyor exposures. This dosimeter data will be updated to reflect group man-rem exposures incurred during radiation survey work. Analyses of exposures incurred during survey work will allow investigation and implementation of methods to control and minimize radiation exposure of surveyor personnel.

12.5.3.3.1.4 Surveyor Work Rotation

Every reasonable effort will be made to assure that surveyor exposure is evenly distributed by work assignment scheduling and rotation of Health Physics personnel. This rotation will allow comparison of surveyor performance, minimize individual exposures, and assure maintenance of familiarity with all areas of the plant.

12.5.3.3.1.5 Training

Training of radiation workers will aid in the reduction of man-hours expended in radiation fields. All station personnel requiring Level II Health Physics training as described in Subsection 12.5.3.7 will receive training in the types of radiation and methods of detection, self-survey and radiation rate survey. This training will include high radiation area survey techniques, data evaluation and special instrument operation. Retraining of station personnel requiring Level II Health Physics training will include the areas of radiation survey techniques and procedures.

Health Physics personnel will receive formal and on-the-job training in survey techniques prior to fuel load at Susquehanna SES. Special emphasis will be directed toward assuring that efficient high radiation area survey techniques are exercised by Health Physics personnel. Impromptu training sessions will be held as needed to assure state-of-the-art understanding or improved performance in areas where reviews have indicated the need for additional training.

Training sessions will emphasize the importance of collecting necessary data while exercising the factors of time, distance and shielding to minimize occupational exposures.

12.5.3.3.2 Radiation Survey Program

12.5.3.3.2.1 Instrument Selection

Health Physics procedures will describe the instrument type(s) to be utilized during radiation survey work. The surveyor will be required to enter instrument description(s) and identification number(s) on survey forms. Prior to performing a radiation survey, the surveyor will check the calibration status of the portable instrument(s) selected for use to assure not more than three months have elapsed since the last calibration. The instrument selected will be checked for battery strength, if applicable, and, in a reproducible geometry, at least one scale's response to known check source(s) will be verified. Instruments overdue for calibration will not be used for radiation survey work. Personnel will be instructed to report instrumentation suspected to be malfunctioning. A properly checked replacement or equivalent survey instrument will be utilized.

12.5.3.3.2.2 Routine Radiation Area Surveys

Each area on site found to produce a radiation dose rate such that an individual could receive 5 mrem in any one hour or 100 mrem in any five consecutive days will be conspicuously posted as a Radiation Area in accordance with 10CFR20.203. Every reasonable effort will be made to minimize inadvertent entries in such areas. The "Caution Radiation Area" signs posted at the boundaries will be updated to reflect the date of the latest survey. Whenever practicable, the signs will also reflect the general and maximum radiation levels within the area and any special conditions required for entry. Routine surveys of radiation areas will normally be taken to assure that each area is surveyed once per week. Areas subject to variations in radiation levels or increased time of occupancy may be surveyed on a more frequent basis, as appropriate. When reactor conditions are operationally stable, survey frequency in radiation areas may be reduced to spot checks at the boundaries to minimize Health Physics personnel exposures.

12.5.3.3.2.3 High Radiation Area Surveys

Each area on site found to produce a radiation dose rate equal to or greater than 100 mrem/hr. will be posted as a High Radiation Area and access will be controlled in accordance with Subsection 12.5.3.1. Routine surveys within such areas will not normally be performed with conventional portable survey instruments. Every reasonable effort will be made to utilize readings from the Area Radiation Monitoring (ARM) System to identify changes of radiation levels. Analyses of maximum and general radiation levels within high radiation areas will normally be performed with remote probe survey instruments, long reach survey instruments, retrievable TLD's or dosimeters. When practicable, findings from these surveys will be correlated to the appropriate ARM readings and reactor operating conditions. Correlation readings and/or perimeter readings will be taken to assure each high radiation area is surveyed once per week. In addition, radiation surveys will be taken at the entrances to high radiation areas on a frequency dependent upon occupancy in the vicinity and variation in radiation levels. Signs will be updated to reflect the latest readings. If surveys at entrances or ARM readings show significant change, additional surveys may be performed to update the readings within the area. In order to minimize occupational exposure of surveyors, high radiation area survey frequency may be reduced when operating conditions are stable.

12.5.3.3.2.4 Non-Radiation Area Surveys

Areas in and around the Controlled Zone not considered potential radiation areas will be selectively surveyed to establish that every reasonable effort has been made to keep measurable radiation levels as low as reasonably achievable. Portable instrument surveys will be performed so as to assure a representative number of non-radiation areas are surveyed once per month. Areas subject to significant change or variation will be surveyed on a more frequent basis as appropriate. Any area, not previously noted, that is found to be a radiation area will be promptly posted with a "Caution Radiation Area" sign and reported to Health Physics supervision. If the radiation dose rate cannot be eliminated, every reasonable effort will be made to minimize the dose rate and inadvertent entry. The area will be placed on the radiation area survey routine.

Areas within Susquehanna SES security fence not covered by portable instrument survey programs will be selectively monitored by area TLD's to document integrated exposures. Area TLD's will normally be changed and evaluated on a monthly basis.

12.5.3.3.2.5 Radiation Work Permit Surveys

RWP surveyors will wear self-reading dosimeters. The surveyor will enter the exposure incurred on the RWP request to assure this exposure category is included in the RWP job function as well as the system and/or equipment exposure totals.

A member of Health Physics supervision will screen incoming RWP requests to assure inclusion of special measurements or considerations. Special instructions may be developed or impromptu training performed to assure that necessary data is collected in the minimum of time.

12.5.3.3.2.6 Special Radiation Surveys

Special radiation surveys will be performed as requested by operating groups, regulatory agencies, or corporate committees. These survey requests will be coordinated by Health Physics supervision to assure the need for the survey justifies occupational exposure of surveyors. A member of Health Physics supervision may draft special instructions for performance of the survey and/or perform impromptu training sessions with surveyors. Emphasis will be placed on assuring that necessary data is

collected in the minimum of time. Individual and man-rem exposure incurred during special surveys will be logged by job function, equipment and/or system.

12.5.3.3.2.7 Unit 2 Construction Surveys

During the start-up/operation phase of Unit 1 and the construction phase of Unit 2, routinely occupied areas in the proximity of Unit 1 will be surveyed on a weekly basis with portable instrumentation. Any area found to contain a dose rate such that if an individual were continuously present he would receive a dose in excess of 100 mrem in any seven consecutive days due to the operation of Unit 1 will be reported to Health Physics supervision. Special shielding, barricading or access control may be employed to eliminate or minimize the potential for personnel exposure. If such areas are identified, portable instrument survey frequency may be increased depending on potential for occupancy and degree of access control exercised.

In addition to portable instrument surveys a program of area TLD monitors will be used to supplement and verify instrument findings. These TLD's will be placed in representative locations of routinely occupied areas near Unit 1 and will normally be changed on a weekly basis. An investigation will be performed if, after natural background subtraction, administrative limits have been exceeded. Health Physics supervision will assure that areas monitored are representative of construction activities in progress.

12.5.3.3.2.8 Radiation Survey Records

Radiation surveys performed at Susquehanna SES will be documented in accordance with approved station procedures. A member of Health Physics supervision will review the record(s) completed by surveyors to assure proper data entry. The reviewer will initial and date the record and forward it for permanent filing.

12.5.3.4 Contamination Survey Procedures

A system of contamination evaluation will be utilized to minimize the spread of radioactive material. Evaluation of personnel, equipment and surface contamination will also be made to demonstrate the efficiency of engineering and procedural controls. In addition, the contamination survey programs will be

evaluated to assure that surveyor exposures are as low as reasonably achievable.

12.5.3.4.1 Personnel Contamination Surveys

Evaluation of exposures due to personnel contamination will be conducted in accordance with Subsection 12.5.3.6.

12.5.3.4.1.1 Frisker Survey

G-M personnel friskers will be placed in strategic locations within the controlled zone. Every effort will be made to locate these instruments in as low a radiation background area as possible in order to maximize sensitivity. Personnel will be trained in the use of the instrument(s) and interpretation of the readings.

In the event of frisker malfunction, personnel will be required to notify Health Physics. Audible or visible alarms will be preset at a suitable point above background to minimize spurious alarms and maximize sensitivity. Limits will be conspicuously posted for instruments without automatic alarms.

Personnel contamination causing frisker alarm will require notification of Health Physics. Health Physics will take appropriate actions to minimize further spread of contamination, and direct appropriate decontamination of affected areas and personnel.

When personnel contamination is noted, a Health Physics investigation appropriate to the incident will be performed. A contamination incident found to have caused an intake of radioactive material will be promptly reported to appropriate supervision. When applicable, recommended methods to prevent recurrence will be forwarded to the Superintendent of Plant for concurrence and implementation by his directive.

12.5.3.4.1.2 Nasal Swab

Nasal swabbing procedures will be implemented as requested by Health Physics or when contamination exceeding station limits is detected on facial areas to qualitatively determine if inhalation of radioactive material occurred. Health Physics personnel will evaluate the swab as soon as practicable. Findings in excess of

station limits will require nasal clearance, shower and scrub-down, a whole body count and/or bioassay, and a documented investigation and evaluation.

12.5.3.4.1.3 Ingestion Procedures

If contamination is detected in or on the mouth, a shower and scrubdown and a whole body count will be performed. Fecal and/or urine collection may be initiated to more accurately determine ingested amounts. All cases of ingestion will be investigated, evaluated, documented and reported to appropriate supervision and the Superintendent of Plant, and appropriate corrective measures will be taken.

12.5.3.4.1.4 Wound, Cut, Abrasion Surveys

To control inadvertent entry of radioactive material in wounds, cuts or abrasions, individuals will be responsible for bringing such matters to the attention of supervisors and/or Health Physics prior to work commencement. Supervisory personnel will assure that reported skin breaks are brought to the attention of the Health Physics group during job planning or RWP request. Health Physics will be responsible for assuring that skin breaks are properly protected prior to work commencement. Open wounds that cannot be adequately sealed will be sufficient grounds to restrict the worker from contamination work.

Any injury that may have caused contamination of a wound will require the worker to immediately exit the work area and report the incident to Health Physics and appropriate supervision. The wound will be flushed and surveyed with portable instrumentation. If contamination is detected in the wound, the Shift Supervisor may initiate the Susquehanna SES Emergency Plan in accordance with written Emergency Plan Implementing Procedures. If injury is sufficient to prevent the worker from moving or exiting the area, the Shift Supervisor will be immediately notified and the Emergency Plan will be initiated, if appropriate. Appropriate whole body counts and/or bioassays will be taken following any needed medical treatment.

12.5.3.4.2 Equipment Contamination Surveys

12.5.3.4.2.1 Contamination Zone Equipment Surveys

Movement of equipment from a contamination zone will require notification of Health Physics personnel. Fixed and removable contamination levels will be evaluated as appropriate and a clearance for removal will be issued in accordance with station procedures.

Routinely used tools may be permanently marked to indicate they are contaminated and will normally be stored inside well marked contamination areas. Repair or use outside contamination zones will require Health Physics approval. Permanently marked tools will be surveyed by Health Physics personnel as necessary and at the request of the appropriate supervisor. Contaminated items that cannot practicably be decontaminated will be covered with plastic or other material and appropriately posted.

12.5.3.4.2.2 Personal Item Surveys

Change-out procedures will require that individuals leaving a contamination zone perform surveys of personal items that may have become contaminated during work. Items such as dosimeters, TLD or badge holders, pens and pencils, will be scanned with a G-M frisker. Contamination noted on such items will be reported to Health Physics personnel. Additional surveys will be performed and the items decontaminated or discarded as radioactive waste as appropriate.

12.5.3.4.2.3 Protective Clothing Surveys

Reusable protective clothing and shoe covers used in contamination zones will be collected in recepticles at step-off areas and sent for laundering/decontamination. If clothing is cleaned at Station laundry facilities it will be removed from containers, sorted in an exhausted area of the laundry and scanned with a G-M detector to locate highly contaminated items that may require separate decontamination or disposal. Following washing and drying, clothing will be re-surveyed to assure that items are within station limits. Records of the range of survey results before and after laundering will be maintained. Every reasonable effort will be expended to assure that clothing is maintained at as low a contamination level as practicable.

Protective clothing that is shipped off site for laundering will be prepared for shipment and labeled in accordance with applicable U.S. Department of Transportation (USDOT) regulations. Items returned from vendors will be spot checked with survey instruments to assure that residual contamination levels are less than applicable station limits. Records of survey results will be maintained for each shipment.

12.5.3.4.2.4 Respiratory Protection Device Surveys

Respiratory protective masks will be checked for contamination prior to cleaning and disinfection. Following decontamination and cleaning, masks will be checked for removable and fixed contamination levels prior to disinfection, storage and/or reissue. Survey results will be recorded.

Exterior surfaces of other protective devices, such as supplied air hoods and suits, self contained breathing apparatus and hoses, will be checked for contamination levels following job completion. Items other than face pieces that will routinely be reused in contamination zones may be bagged and labeled to reflect the latest survey findings.

12.5.3.4.2.5 Fixed Equipment Surveys

Routinely accessible plant equipment that may become inadvertently contaminated will be spot checked to assure items are less than appropriate station limits. Fixed equipment of this category found to exceed removable contamination limits will be wiped down and resurveyed. If decontamination efforts are not successful or if the item is prone to recurrent contamination it will be posted as "Contaminated". An equipment contamination list will be tabulated to assure items in this category are resurveyed. If an item is found not to be a recurrent contamination problem it will be removed from the survey list.

A representative number of smears will be taken on items such as door knobs and stair railings, to assure that other controls exercised are minimizing the spread of contamination.

12.5.3.4.2.6 Surveys Involving Receipt/Shipment of Radioactive Material

The security staff will be instructed to notify the Health Physics Supervisor or designated alternate upon arrival of shipments in excess of "Type A" quantities at the site. Shipping containers will be monitored for radiation and/or contamination in accordance with 10CFR20.205. Whenever practicable, the container will be monitored prior to removal from the vehicle. If removable contamination or radiation levels are found to exceed the limits of 10CFR20.205, the Superintendent of Plant or designated alternate will notify the final delivering carrier and the Nuclear Regulatory Commission (NRC) Inspection and Enforcement Regional Office.

When applicable, Health Physics Supervision will assure that, prior to leaving the site, exclusive use transport vehicle surface contamination and radiation levels are within limits specified in 49CFR173.

Station procedures will specify special procedures and precautions to be taken when opening packages containing licensed material, including instructions pertaining to specific types of shipments normally received at Susquehanna SES.

Radioactive material will be shipped in accordance with USDOT and NRC regulations. Station procedures will implement the applicable regulations with regard to proper packaging and labeling requirements. Appropriate removable contamination and dose rate surveys will be taken, records completed, and shipments labeled accordingly.

12.5.3.4.3 Surface Contamination Surveys

12.5.3.4.3.1 Controlled Access Areas

A smear survey program will be developed and implemented to assure that a representative number of routinely accessible surface areas within the controlled zone are checked for removable contamination. Special emphasis will be placed on survey of the clean side of established contamination zone step-off areas. Smears will be analyzed on appropriate counting equipment and records of results will be maintained in disintegrations per minute (dpm) per 100 sq.cm. If results indicate removable contamination exceeds station limits, the area will be posted as a contamination zone. The area will be decontaminated and resurveyed as soon as practicable. Area signs and barriers will be removed when surveys indicate that removable contamination is below station limits.

In representative areas where gamma background permits, surveys will be performed with portable detectors to establish the level of fixed contamination on normally occupied controlled zone surfaces. A fixed contamination survey will be performed prior to any sanding, chipping, welding, grinding and sawing, of potentially contaminated Controlled Zone surfaces.

12.5.3.4.3.2 Non-Controlled Zone Areas

Occupied plant areas outside the controlled zone will be surveyed to assure that a representative number of floor surfaces are checked for removable contamination. The exit areas from the controlled zone will receive special emphasis to minimize the spread of contamination. Smear survey, analyses and record keeping techniques will be as described above. Non-controlled zone areas found to have removable contamination levels exceeding station limits will be decontaminated and resurveyed.

12.5.3.4.3.3 Special Area Surveys

Lunch room facilities and vending machine areas frequented by controlled zone workers will be checked for removable contamination. Stoves, benches, table tops, and floor surfaces will be representatively smeared to assure minimal contamination in eating areas. Removable contamination in excess of non-controlled zone limits will be reported to Health Physics or

Shift Supervision and the area will be restricted from further use until decontaminated. Special emphasis will be placed on eating or cooking surfaces to assure that these items are as far below non-controlled zone limits as reasonably achievable.

Other specific areas will be checked for removable contamination to demonstrate the effectiveness of the contamination controls exercised within controlled zone areas. These areas include:

(1) Entrances to the control room and the Control Structure.

(2) The Guard House at the site perimeter.

(3) General floor areas of shower and locker room facilities.

Floor surfaces in areas that offer a repeated potential for contamination may be maintained as contamination zones to assure positive contamination control. In addition to the routine check outside step-off areas, a general survey of contamination levels inside the areas will be performed whenever practicable. Dose rates within the areas, frequency of occupancy, past survey results, and actual need for such surveys will be evaluated by Health Physics supervision when selecting established contamination zones to be surveyed. When area dose rates permit, every reasonable effort will be expended to minimize contamination levels.

12.5.3.4.3.4 Implementation, Review, and Reporting Practices

Contamination limits, general survey locations and survey frequencies will be specified in station Health Physics Procedures. Procedures will be subject to review by Health Physics Supervision to assure contamination survey implementation is responsive to plant status.

A member of Health Physics supervision will review records of contamination survey results to assure proper completion and adequate survey. In the event of contamination in excess of station limits, a member of Health Physics Supervision will be responsible for assuring that corrective measures are implemented and that further reports through appropriate channels are initiated if required.

12.5.3.5 Airborne Radioactive Material

Every reasonable effort will be expended to assure that material released as airborne concentrations within the plant is minimized. A sampling and analysis program will be utilized to determine airborne concentrations in representative numbers of routinely occupied areas. These routine measurements as well as special surveys, respiratory protection procedures and administrative procedures will be implemented to minimize airborne contamination and the potential intake of radioactive material.

12.5.3.5.1 Physical Controls

12.5.3.5.1.1 Air Flow Patterns

A survey program for determining air flow patterns within the controlled zone will be implemented prior to Unit 1 fuel load. After Unit 1 fuel load these surveys will be periodically performed to demonstrate that air flow patterns are toward areas of higher actual, or expected, airborne concentrations. Affected areas will be re-surveyed following ventilation modifications to assure proper air movement. Appropriate measures will be taken if flow patterns are found to be unacceptable.

12.5.3.5.1.2 Contamination Confinement

Contaminated items will be properly confined to prevent inadvertent airborne contamination. Such items will be sealed in appropriate material or stored in ventilated areas whenever practicable. When necessary, alternatives such as temporary tents or enclosures, storage in rooms or areas where air movement is away from occupied areas, or wetting of the item may be utilized to minimize airborne concentrations. Contaminated trash will be sealed in plastic prior to disposal whenever practicable. Every reasonable effort will be made to assure that contaminated trash recepticals are closed when not in use.

12.5.3.5.1.3 Air Exhaust

Exhaust of areas or items where airborne concentrations may be generated will be employed whenever practicable. Contaminated laundry sorting areas, trash compactors, fume hoods, and sampling stations are typical locations where air exhaust will be utilized. Exhaust flow rates or face velocities on such equipment will be verified periodically and after ventilation modifications to assure proper function. Items that may contain highly contaminated materials such as trash compactors or high level fume hoods will be equipped with a visual indicator or alarm to warn individuals upon loss of exhaust flow. Portable exhaust fans will be directly discharged to building exhaust whenever practicable. When discharge to building exhaust is not practicable the portable exhaust fan will be filtered to minimize airborne concentrations.

12.5.3.5.1.4 Posting and Locking

Accessible areas containing airborne concentrations exceeding the limits specified in 10CFR20.203 will be posted with a "Caution - Airborne Radioactivity Area" sign. Whenever practicable, access points to such areas will be locked or barricaded to reduce the risk of inadvertent entry.

12.5.3.5.2 Administrative Controls

12.5.3.5.2.1 Health Physics Review

All posted airborne radioactivity areas will be reviewed by a member Health Physics supervision on a quarterly basis. Methods to reduce existing airborne concentrations will be forwarded through appropriate channels for review, approval, and implementation. During the review, Health Physics Supervision will assure that every reasonable effort has been expended to reduce the risk of inadvertent entry in airborne radioactivity areas.

12.5.3.5.2.2 Health Physics Investigation

When an occurrence produces unusually high airborne concentrations in occupied areas, Health Physics Supervision will assure that an investigation appropriate to the incident is

completed. The first priority will be evaluation and follow-up of personnel intake of radioactive material if applicable. The second portion of investigation will emphasize determination of the events leading to the release. Recommendations to prevent recurrence will be forwarded through appropriate channels for implementation.

12.5.3.5.2.3 RWP Procedures

Radiation Work Permit procedures, as described in Subsection 12.5.3.2, will be a primary administrative control of exposure to airborne radioactive material. Health Physics review prior to approval will assure that every reasonable effort is expended to minimize the production of, or reduce existing, airborne concentrations before work commencement.

12.5.3.5.3 Air Sampling Equipment

A description of the use, calibration methods and frequencies of specific air sampling equipment utilized at Susquehanna SES is contained in Subsection 12.5.2.

12.5.3.5.4 Airborne Concentration Sampling

12.5.3.5.4.1 Routine Sampling

Routine sampling in selected areas of potential airborne concentrations will be accomplished with continuous air monitors (CAM) or portable air monitors. CAM sampling media and detector will be selected as appropriate to the intended use of the device. CAM's will be routinely checked for proper operation. Abnormal readings or equipment malfunction will be reported through appropriate channels for investigation and/or repair. Alarms, if applicable, will be checked for operability during source check and calibration procedures. Fixed filter devices will be changed on a frequency specified by Health Physics procedures to assure optimum sampling time, meaningful results, and proper equipment operation.

12.5.3.5.4.2 Special Air Sampling

Records will be maintained to reflect the reason for the special surveys, device(s) and sampling media used and final results. The majority of special air samples will be taken as result of Radiation Work Permit requests and pertinent results will be recorded thereon.

12.5.3.5.5 Air Sample Evaluation

12.5.3.5.5.1 Particulate Initial Evaluation

A data sheet will be completed to reflect sample location, date, starting flow rate, starting time, sampler and collection media used, and collection efficiency. At completion of sampling, the date, time, and ending flow rate will be recorded. Air sample filters will be counted as soon as practicable following collection. Results will be recorded on an analysis form to reflect counter used, efficiency, counting time, background count rate, gross sample count rate, net sample count rate, and sample disintegrations per minute beta, and/or beta-gamma, and/or alpha. Sample disintegrations per minute divided by collection efficiency of the media, the number of disintegrations per minute per microcurie and the total volume of air sampled will yield the initial estimate of airborne concentration.

Prior to Unit 1 fuel load an air sampling program will be implemented to obtain a base line of information concerning naturally occurring radioactive concentrations. This data will enable development of an average beta to alpha ratio of naturally occurring airborne emitters. This "First Count Factor" may be utilized as an initial evaluation technique for low level particulate air samples.

12.5.3.5.5.2 Subsequent Particulate Evaluations

Every effort will be made to initially evaluate air samples as soon as practicable following collection. In instances where time delay before analysis in conjunction with suspected short lived isotopes is significant, repeated counts may be performed to obtain a decay curve. Extrapolation and subtraction techniques may be used to determine initial amounts and half lives of component isotopes.

When statistically possible, fixed filter samples may be gamma scanned with a NaI or Ge(Li) detector to identify gamma emitting isotopes. When this or other specific analyses are not practicable, the MPC for unidentified beta-gamma emitters will be used for exposure evaluation, and procedural controls.

Other evaluations that may be utilized are beta absorption counting, radiochemical separations and analysis, and liquid scintillation counting.

12.5.3.5.5.3 Gaseous Evaluations

Airborne radioiodine samples will normally be collected on charcoal canister or cartridges, and analyzed on a NaI or Ge(Li) detector. Appropriate standard sources in reproducible geometries will be used to obtain efficiency curves for analysis equipment. Photopeak areas, counting efficiency and branching ratios for the identified isotope will be utilized to calculate the amount of deposit. Collection efficiency and total volume of sampled air will be incorporated to calculate airborne concentrations.

Airborne tritium samples will normally be collected in water bubblers or dessicant columns. Collection and counting efficiencies and total air volume will be verified and used to calculate airborne concentrations.

If analyses of restricted area air for noble gases are required, sample chambers may be analyzed with NaI or Ge(Li) detectors to identify isotopes.

12.5.3.5.6 Respiratory Protection

The respiratory protection program will assure that personnel intake of radioactive material is minimized. The respiratory protection program will not be used in place of practicable engineering controls and prudent radiation safety practices. Every reasonable effort will be expended to prevent potential, and minimize existing, airborne concentrations. When controls are not practicable, or conditions unpredictable, respiratory protective devices may be utilized to minimize potential intake of airborne radioactive material.

The Susquehanna SES Respiratory Protection Program will ensure that the following minimum criteria are met: written standard operating procedures; proper selection of equipment, based on the

hazard; proper training and instruction of users; proper fitting, use, cleaning, storage, inspection, quality assurance, and maintenance of equipment; appropriate surveillance of work area conditions, consideration of the degree of employee exposure to stress; regular inspection and evaluation to determine the continued program effectiveness; program responsibility vested in one qualified individual and an adequate medical surveillance program for respirator users.

12.5.3.5.6.1 Training and Fitting

The training and fitting program is described in Subsection 12.5.3.7.

12.5.3.5.6.2 Written Procedures

The Respiratory Protection Program and program responsibility will be implemented by Health Physics procedures. Applicable Health Physics Procedures will include as a minimum: description of equipment; information regarding issuance, maintenance, selection, use, and return of equipment; and training techniques. Information regarding air sampling and bioassay programs will be referenced.

12.5.3.5.6.3 Selection of Equipment

The need for respiratory protection will be determined by Health Physics personnel after evaluation of appropriate engineering controls. Airborne concentrations will be determined by air sampling methods described in this section. The hazard will be evaluated and applicable respiratory protection prescribed in accordance with the RWP evaluation, review, approval and implementation process as described in Subsection 12.5.3.2.

12.5.3.5.6.4 Issue and Use

For normal work situations, respirators will be issued after approval of a Radiation Work Permit. Individuals' I.D. cards or qualification list will be utilized to assure only the specific models approved for the worker are issued. After issuance, the worker will be responsible for proper use and storage of the device. Approved Health Physics procedures for use, storage and

return of respirators will be reviewed during qualification training sessions.

12.5.3.5.6.5 Contamination Surveys

Whenever practicable, respirators will be scanned with a G-M detector during final change-out procedures upon completion of assigned work. Detectable radiation levels on inside surfaces of the device will require notification of Health Physics. The inside surfaces will then be monitored for removable contamination and/or a nasal swab will be taken. Based upon findings and suspected isotopes, further evaluations may be required in accordance with Subsection 12.5.3.6.

12.5.3.5.6.6 Cleaning, Decontamination, Inspection, Maintenance, Disinfection and Storage

Station procedures will specify cleaning, decontamination, survey, inspection, maintenance, disinfection and storage requirements. Respirators will normally be used no more than one day (shift) prior to return for cleaning survey, inspection, maintenance if needed, and disinfection. In no case will a respirator be issued to another individual prior to cleaning survey, inspection and disinfection. Respiratory face pieces will be washed, dried, surveyed for removable and fixed contamination levels, inspected, disinfected and stored in accordance with approved Health Physics procedures. Inspection of masks will emphasize defects at critical points, proper function of attached fittings and valves, and proper shape of face-piece. Simple maintenance and repair will be performed as necessary. Maintenance and repair of regulators will be performed only by specially trained and qualified individuals. Masks ready for reissue will be stored in plastic or paper bags in cabinets or containers. Every effort will be made to assure proper storage of masks to prevent deformation of face piece parts.

12.5.3.5.6.7 Quality Controls

Inspection and testing of new equipment will be implemented by written station procedures to detect instances of human error or defective materials in the manufacture and assembly of the devices. Procedures will specify the components of each device to be inspected and the acceptance criteria when applicable.

Respiratory protection devices will be routinely inspected and tested after cleaning and maintenance. The inspection will be performed to detect any damage or defects caused by cleaning or wear. Testing will normally consist of a positive or negative pressure leak detection test or exposure to a challenge atmosphere.

In accordance with station procedures periodic checks of items in storage will be performed to ensure that the facepiece rubber is not taking a set, rubber parts are not hardening or deteriorating, sorbent canisters have not exceeded their shelf life, and breathing air or oxygen cylinders contain sufficient pressure.

12.5.3.5.6.8 Surveillance of Work Area Conditions

For work conditions involving respiratory protection, air sampling surveillance will provide an estimate of the potential intake of airborne radioactive materials and resulting exposure of the individual worker, indicate the continuing effectiveness of existing controls, and warn of the deterioration of control equipment or operating procedures.

The periods of time respirators are worn continuously and the overall durations of use will be kept to a minimum by procedural controls and work surveillance. Workers will be instructed of provisions to leave areas where respirator use is required for relief in case of equipment malfunction, undue physical or psychological distress, procedural or communication failure, significant deterioration of operational conditions, or any other condition that might require such relief.

12.5.3.5.6.9 Evaluation of Program Effectiveness

Respirator failures, evidence of respirator leakage, and equipment problems encountered will be investigated by Health Physics. Problems will be solicited from respirator users during activities such as plant safety meetings and training sessions. Proposed changes to prevent recurrence or improve efficiency of the program will be forwarded through appropriate channels for review, approval and implementation.

Respiratory protection will be evaluated by bioassay results correlated with air sampling results as described in Subsection 12.5.3.6. Evidence of a rise in exposure levels attributable to inhalation will be investigated.

12.5.3.5.6.10 Medical Surveillance

Prior to participation in the Susquehanna SES Respiratory Protection Program, individuals will be evaluated by competent medical personnel to ensure they are physically and mentally able to wear respirators under anticipated working conditions.

Individuals involved in the respiratory protection program will also be re-evaluated as part of their routine company physical with respect to physiological and psychological factors affecting respirator use.

Details of the medical surveillance program will be specified in Station Health Physics Procedures.

12.5.3.5.7 Handling of Radioactive Material

12.5.3.5.7.1 Unsealed Material

Radioactive material in liquid form will be stored in sealed or vented/ exhausted containers whenever practicable. When containers are opened to atmosphere and generation of airborne concentrations is possible, they will be opened in fume hoods, exhausted areas, or in locations where air movement is away from workers' breathing zones. Whenever practicable, liquid radioactive material will be transported in unbreakable containers or in a secondary container to collect material in case of breakage.

Gaseous radioactive material will be similarly stored and opened. Transport of gaseous samples will be done in sealed, gas tight containers.

Solid articles that are sufficiently contaminated with particulate and/or volatile material so as to pose a potential airborne hazard will be handled and stored as described in Subsection 12.5.3.5.1.2.

Protective clothing, respiratory protection, and special precautions will be specified by Health Physics procedures and/or Radiation Work Permit for handling unsealed material.

12.5.3.5.7.2 Sealed Materials

Sources will be stored in appropriate shielded containers when not in use. Containers and storage locations will be posted to reflect contents and radiation levels. Sources will be locked inside containers or containers will be locked in a storage location when not in use. When sources produce a whole body or contact radiation dose rate greater than limits established by station procedure, a Radiation Work Permit will be completed and approved prior to use. Remote devices such as forceps, tongs or manipulators will be used whenever practicable or required by Radiation Work Permit.

Licensed sealed sources will be monitored for leakage to assure that storage or use is not causing the spread of contamination or airborne radioactive material. When monitoring of the source capsule is not practicable, removable contamination surveys will be performed at places on the container or source holder where contamination might be expected to accumulate if the source were leaking. Samples will be analyzed on counting equipment appropriate to the source material and records of results maintained. Frequency, materials to be tested and record keeping requirements of NRC license or Technical Specifications will be implemented by Station Health Physics Procedures. Sealed sources found to be leaking will be sealed from atmosphere whenever practicable and/or stored in ventilated areas until disposal or repair.

12.5.3.6 Personnel Monitoring

12.5.3.6.1 External Personnel Monitoring

Personnel monitoring devices will be used at Susquehanna SES to evaluate external occupational exposure to radiation sources. Exposure information will be used for work function exposure evaluation, job planning, reporting requirements, incident analysis, and an indication of the effectiveness of ALARA practices.

12.5.3.6.1.1 Personnel Dosimetry Evaluation

Routinely used personnel dosimetry will include self-reading dosimeters, thermoluminescent dosimeters (TLD), and/or film badges. Individuals requiring personnel dosimetry will be

instructed in the purpose and use of the devices, station administrative exposure limits, and interpretation of self-reading dosimeter readings. Appropriate dosimetry devices will be issued in accordance with station procedures implementing 10CFR20.202.

Dosimetry will normally be worn on the front of the body between the neck and the waist in a clearly visible location. When appropriate, dosimetry will be issued and worn on the extremities. Dosimetry may be wrapped in plastic to prevent the contamination of personnel monitoring devices when entering contaminated areas.

As described in Subsection 12.5.3.2, self-reading dosimeter results will be used for specific ALARA job exposure evaluation as well as to indicate current individual exposure status. Dosimeters of appropriate ranges will be available for use during work in radiation and high radiation areas. Radiation workers will be responsible for checking their dosimeter readings when working in RWP areas. The frequency of dosimeter checking will depend upon the nature of the job and whole body dose rates, and will be discussed with the radiation workers during RWP pre-job planning. Off-scale or malfunctioning dosimeters will be reported to Health Physics. Health Physics personnel will evaluate the occurrence, issue a replacement dosimeter and test the suspect dosimeter for response and leakage. Dosimeters will be removed from service if the calibration response, 24 hour leakage, or changing drift test results exceed acceptance criteria specified in the Station Health Physics procedures.

Self-reading dosimeters will normally be used to monitor gamma exposure only. They may be used to determine neutron dose equivalent in a mixed radiation field provided the neutron dose equivalent rate and gamma exposure rate at the point of personnel exposure are known from separately made determinations; the neutron-to-gamma ratio is essentially constant during the period of personnel exposure; and the degree of response of the dosimeter to the neutron flux density is known. Methods of evaluation of dosimeter readings to determine neutron dose equivalent will be specified in Station Health Physics procedures. When neutron dose equivalent is determined from self-reading dosimeters, it will be added to the whole body gamma dose equivalent.

TLD devices will normally be used as the dosimetry of record. Personnel TLD(s) will normally be evaluated on a monthly basis or more frequently as determined by Health Physics Supervision. The data obtained from TLD's will be evaluated to determine dose equivalents. Gamma TLD chip readings indicate the dose equivalent to be attributed to whole body. Appropriate

correction and quality factors will be applied to neutron chip readings to determine the neutron dose equivalent. Neutron and gamma doses will normally be added together to yield the whole body dose equivalent. Appropriate correction factors will be applied to the Beta TLD chip readings to determine the beta dose. The beta dose will normally be added to the whole body dose equivalent to determine the skin dose equivalent. When appropriate, the skin of whole body dose equivalent will be added to the gamma dose equivalent, determined by issued extremity monitoring devices, to determine total extremities' dose equivalent. For individuals who do not utilize extremities devices during a calendar quarter, the skin of whole body dose equivalent will be assigned as extremities' dose equivalent.

If film badges are used as the dosimetry of record, the service will be purchased from an outside vendor and evaluated by the vendor on a monthly basis or as specified by Health Physics Supervision. A program will be implemented to verify film badge accuracy. Film badge results will be evaluated and categorized according to whole body, skin of the whole body, and extremity dose equivalent. Film badges may be used to determine neutron dose equivalent when the effects of image fading, low sensitivity, and masking in high gamma fields are not critical.

Personnel exposures will be accumulated and evaluated against applicable station and federal limits by Health Physics personnel.

12.5.3.6.1.2 Administrative Exposure Control

Administrative exposure limits will be established and implemented by Health Physics procedures to assure the limits of 10CFR20.101 are not exceeded and personnel occupational exposures are maintained ALARA.

12.5.3.6.1.3 Methods of Recording and Reporting

Designated supervisors will receive reports of their employees' accumulated exposures for use in RHP job planning and scheduling. Updates of exposure totals will be compiled from self-reading dosimeter readings. Unapproved exposures exceeding station limits will be reported to the Superintendent of Plant and appropriate supervision, and investigated by Health Physics to identify causes and establish methods to prevent recurrence.

Occupational radiation exposure received during previous employment will be used in preparation of individuals' Forms NRC-4, or equivalent. When an individual's occupational exposure history cannot be obtained, the values specified in 10CFR20.102(c)(1) will be used. Records used in preparing Form NRC-4, or equivalent, will be retained and preserved until the NRC authorizes disposition.

Records of the radiation exposure of all individuals issued personnel dosimetry in accordance with 10CFR20.202 will be maintained on Form NRC-5, or equivalent. Exposures will be tabulated for periods not exceeding one calendar quarter. A separate record will be completed when it is necessary to enter information for exposure to the extremities or skin of the whole body. Records of radiation exposure received during employment at Susquehanna SES will be maintained indefinitely or until NRC authorizes disposal.

Reports of exposure to radiation or radioactive materials will be made to individuals as specified in 10CFR19.13. When reports of individual exposure to radiation or radioactive material are made to the NRC, the individual(s) concerned will also be notified. This notice will be forwarded to the individual(s) at a time no later than the transmittal to the Commission and will comply with 10CFR19.13.

A report of the individual's exposure to radiation or radioactive material incurred while employed or working at Susquehanna SES will be furnished to the NRC in accordance with 10CFR20.408 and to the individual upon termination of employment or work assignment at Susquehanna SES.

A personnel monitoring information report will be submitted, in accordance with 10CFR20.407, within the first quarter of each calendar year. As part of a routine annual operating report, personnel exposure information will be submitted within the first quarter of each calendar year. It will include a tabulation of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr. and associated man-rem exposure according to work and job functions. It will also include for each outage or forced reduction in power of over 20 percent of design power level, where the reduction extends for greater than four hours, a report of radiation exposure associated with the outage which accounts for more than 10 percent of the allowable annual values.

In the event of an exposure in excess of 10CFR20.101 limits, Health Physics Supervision will investigate the event and document the description of the occurrence; conditions under which the exposure occurred; names of personnel involved and

amount of exposure received; action taken at time of occurrence; recommendations for corrective measures and means of implementation to prevent a similar occurrence.

In the event of an unauthorized exposure in excess of station administrative limits, Health Physics Supervision will investigate the event to determine the cause(s). Recommendations for corrective measures will be forwarded for review, approval, and implementation in accordance with station procedures.

Reports of overexposures at Susquehanna SES will be submitted to the NRC and the individual(s) involved in accordance with 10CFR19.13 and 10CFR20.405. Reports will also be forwarded to appropriate committees for review and recommendation for follow-up action.

12.5.3.6.2 Internal Radiation Exposure Assessment

When engineering controls are impracticable and airborne concentrations exceed station limits, trained individuals will be equipped with properly fitted respirators. Internal exposure evaluation will be utilized to determine the effectiveness of the Respiratory Protection Program and evaluate suspected intake of radioactive material. The Respiratory Protection Program is described in Subsection 12.5.3.5. Whole body counting and/or bioassay techniques will be used to compare the quantity of radioactive material present in the body to that quantity which would result from inhalation for 40 hours per week for 13 weeks at uniform airborne concentrations specified in Appendix B, Table 1, Column 1, 10CFR20.

12.5.3.6.2.1 Bioassay Methods

Whole body counting will be used to qualitatively and quantitatively identify radionuclides deposited in the body which emit penetrating radiations. Depending upon the physical construction and geometry of the whole body counter, sensitivity of the detector(s), and biological factors, concentrations of radionuclides may be detected in the whole body, thyroid, lung, or wounds. The whole body counter will be set up and calibrated and/or utilized in accordance with Subsection 12.5.2.

Urine analysis may be conducted to identify the presence of pure alpha or beta emitters in extracellular body fluids. Under favorable circumstances, with a full 24-hour sample and further

analyses, the amount of radionuclides may be qualitatively and quantitatively determined. Results may be utilized to substantiate in vivo analyses findings.

Fecal analysis will normally be used to evaluate intake of non-transportable (i.e. insoluble) material and provide evidence of the clearance of such material from the lungs. When it is suspected that a nontransportable radionuclide has been inhaled, the total amount excreted in feces during the succeeding few days may be used to estimate the amount initially deposited in the lungs. Standard lung models recommended by International Commission on Radiological Protection (ICRP) may then be used to evaluate the amount inhaled.

Dose commitment for internal deposits may be estimated by calculating the amount of airborne radioactive material inhaled, based on airborne radioactive material measurements, exposure times, standard lung models and breathing rates.

12.5.3.6.2.2 Administrative Controls

Records, approved station procedures, program reviews, and investigation will assure proper administrative control over the internal personnel monitoring program. Reviews of the internal personnel monitoring program and investigations of individual cases of suspected or known intakes will be performed and documented by Health Physics Supervision and reported to appropriate committees.

12.5.3.6.2.3 Criteria for Participation or Selection

Selection of personnel and frequency of routine whole body counting and bioassay analyses will be implemented by Health Physics Procedures.

The following is a guideline for participation in special whole body counting and/or bioassay analyses:

(1) Personnel evaluated by means of a nasal swab as having contamination in the nasal passages in excess of limits specified in Health Physics Procedures.

(2) Personnel suspected to have ingested a detectable level of radioactive material, or absorbed a detectable level of radioactive material through a wound or break in the skin.

(3) Personnel physically present without respiratory protection, or those experiencing respirator failure, in a concentration resulting in greater than 40 MPC-Hours exposure in any seven consecutive days may be counted. An evaluation will be performed in accordance with 10CFR20.103 and whole body, lung, or thyroid counting will be performed if calculations show potential deposit of greater than the Minimum Detectable Activity (M.D.A.) of the counter for long lived isotopes.

The following is a guideline for selection of personnel for special, non-routine urine analysis:

(1) When there is suspicion of an intake of a beta or alpha emitter only.

(2) In conjunction with non-routine fecal analysis.

In addition to the above criteria, personnel may be required to submit urine samples to evaluate clearance rates of radioactive material identified by special or routine whole body counts, or as directed by Health Physics Supervision.

Fecal sampling and analysis will normally be done on a non-routine basis as designated by Health Physics Supervision. Fecal analysis may be done as a follow up on whole body or lung counts.

12.5.3.6.2.4 Evaluation and Reporting

Identifiable deposits will be evaluated against the criteria of 10CFR20.103 assuming conservative conditions and time frames with respect to the time of intake. Reports will be generated when internal deposits indicate greater than 40 MPC-Hours exposure in any seven consecutive days. The reports will be reviewed by appropriate supervision and maintained on file subject to NRC inspection. Reports of overexposure will be completed and submitted to the NRC when it is determined a quantity greater than specified in 10CFR20.103 has been inhaled.

Whole body dose commitment resulting from internal deposits exceeding station limits will be calculated and included on the individual's Form NRC-5 or equivalent.

Specific organ counting may be performed if appropriate. Organ content may be assigned using whole body measurements and ICRP-2 recommended fractions and clearance times, when organ counting is not possible. Dose commitment to blood forming organs, gonads,

whole body or eyes resulting from deposits in other organs may be calculated using Medical Internal Radiation Dose Committee (MIRD) equations. Whole body dose commitment resulting from internal deposits exceeding station limits will be calculated and included on the individual's Form NRC-5.

12.5.3.7 Health Physics Training Programs

Health Physics Training Programs will assure that personnel, who have unescorted access to the restricted area, possess an adequate understanding of radiation protection to maintain occupational radiation exposures as low as reasonably achievable. Special training/retraining will be administered upon recommendation of the Superintendent of Plant or Health Physics Supervisor. Record keeping and training scheduling will be performed by the Training Supervisor or designated alternate.

12.5.3.7.1 Program Controls

12.5.3.7.1.1 Management Review

Management will formally review Health Physics Training Programs once every three (3) years. Consideration will be given to workers' suggestions and instructors' comments. Management will evaluate the program's influence on maintaining radiation exposures as low as reasonably achievable. The review will be documented and comments/changes will be recorded and incorporated into the training program when applicable.

12.5.3.7.1.2 Health Physics Training Program Review

Health Physics Training Programs will be reviewed by Health Physics Supervision and pertinent committees to assure implementation of ALARA philosophy. Recommendations for improvements to training programs will be forwarded through appropriate channels for review, approval, and implementation.

12.5.3.7.1.3 Access Control

An access control list will be compiled and maintained. The list will specify personnel qualified for unescorted access to the

Restricted Area by having met the requirements of Level I Health Physics Training and appropriate plans and procedures. A listing specifying individuals' retraining dates will be maintained. A copy of the access list will be maintained at the security guard house.

During appropriate training sessions, individuals whose job duties do not require entry in radiation, contamination, or RWP areas will be informed of the reasons they are denied access to such areas.

12.5.3.7.1.4 Retraining/Replacement Training

To assure individual proficiency in Radiation Protection practices, retesting will be performed on a yearly basis. Retraining will be performed every two (2) years or as recommended by the Superintendent of Plant. Scheduling, records, and test results will be maintained by the Training Supervisor or designated alternate. Individuals changing job classification will receive training of the level required by their new job classification.

Training/retraining will be administered, under the direction of the Training Supervisor or designated alternate, to candidates for Nuclear Regulatory Commission (NRC) operating licenses and those holding NRC licenses. The Training Supervisor or designated alternate may request the Health Physics Supervisor to provide instruction on selected Health Physics topics.

12.5.3.7.2 Training Programs

12.5.3.7.2.1 Level I Training

All persons allowed unescorted access to the restricted area will, as a minimum, receive Level I Health Physics training. To be qualified in Level I Health Physics an individual will demonstrate proficiency in the following areas as evidenced by passing a written examination:

Requirements of 10CFR 19.12

Radiation/Contamination (examples and control)

ALARA (Corporate commitments, meaning and individual responsibility)

Personnel Monitoring and Self-Survey Requirements

Radiological Control Signs and Posting Requirements

Radiation Exposure Control and Limits

Radiation Emergency Plan and Applicable Procedures

Prenatal Radiation Exposure

12.5.3.7.2.2 Level II Training

Level II Health Physics Training will normally be administered to individuals who have successfully completed Level I and require access to Radiation Work Permit Areas. The need for such training will be evaluated and scheduled by the Training Supervisor, or designated alternate. Level II training will be administered to provide radiation workers with an adequate knowledge to effectively cope with job situations while maintaining radiation exposures as low as reasonably achievable. The individual will demonstrate proficiency in the following areas as evidenced by passing a written examination:

ALARA (applicable procedures)

Contamination Control and Self-Survey Requirements

Fundamentals of Radioactivity

Radiation Dose Units and Biological Effects

Radiation and High Radiation Area Survey Techniques

Principles of Radiation Safety (Time, Distance and Shielding)

Radiation Work Permits (RWP)

Use of protective clothing/devices

12.5.3.7.2.3 Level III Training

Level III training will emphasize special applications of ALARA practices and will normally be directed at supervisors of radiation workers. ALARA training in the planning of radiation work permit jobs will include man-rem reviewing techniques, methods for reducing personnel exposures, and other areas

recommended by the ALARA Review Committee and Health Physics supervision. In addition, effective methods of improving work efficiency, such as mock-up situations, dry-runs, and maintenance oriented photographs for job planning will be discussed.

12.5.3.7.2.4 Level IV Training

Level IV training will emphasize ALARA and Health Physics aspects of the RWP review process discussed in Subsection 12.5.3.2. The training will be directed at qualifying members of Shift Supervision for RWP review and approval authority in the absence of Health Physics Supervision.

12.5.3.7.2.5 Respiratory Protection Training Program

Individuals and their supervisors requiring access to areas where respiratory protection will be utilized will complete the Respiratory Protection Training Program. The instructor will be a qualified individual with a thorough knowledge and considerable experience regarding the application and use of respiratory protective equipment and the hazards associated with radioactive airborne contaminants.

Training will include lectures, demonstrations, discussions of pertinent station procedures, and actual wearing of respirators to become familiar with the various devices utilized at Susquehanna SES. The program will include as a minimum: discussion of the airborne contaminants against which the wearer is to be protected, including their physical properties, MPC's, physiological action, toxicity, and means of detection; discussion of the construction, operating principles, and limitations of the respirator and the reasons the respirator is the proper type for the particular purpose; discussion of the reasons for using the respirators and an explanation of why more positive control is not immediately feasible, including recognition that every reasonable effort is being made to reduce or eliminate the need for respirators; instruction in procedures for ensuring that the respirator is in proper working condition; instruction in fitting the respirator properly and checking adequacy of fit; instruction in the proper use and maintenance of the respirator; discussion of the application of various cartridges and canisters available for air-purifying respirators; instruction in emergency action to be taken in the event of malfunction of the respiratory protective devices; review of radiation and contamination hazards, including the use of other protective equipment that may be used with respirators; classroom

and field training to recognize and cope with emergency situations; and other special training as needed for special use.

Individuals will be required to don the device(s) that may be used, perform appropriate pressure tests for leak detection, and be exposed to a challenge atmosphere. If a quantitative test device is available, it will be utilized to quantitatively measure and record leakage. If leakage exceeds the devices rated protection factor and retests confirm this, the individual will not be approved to use the device. If quantitative testing is not practicable or unavailable, qualitative tests such as irritant smoke or isoamyl acetate may be used as a challenge atmosphere. Detection of odor will be considered a fitting failure. After successful completion of training and fitting programs, appropriate records will be maintained to assure individuals are issued only the approved type and model of protective device(s). These records will reflect expiration dates. Individuals will receive retraining and reconfirmation of respirator fit on an annual basis. Related records will be maintained by the Training Supervisor or designated alternate.

12.5.3.7.2.6 Health Physics Monitor Initial Training Program

A Health Physics Training Program will be administered to applicants for the position of Health Physics Monitor under the direction of the Health Physics Supervisor or designated alternate. The content of instruction will depend upon the experience and qualifications of the applicant with course content outlined in approved station procedures. Applicants with Health Physics experience may be waived from participation in part or all of the initial monitor training program. All applicants must demonstrate their proficiency by successfully completing the Monitor Qualification Examination.

The initial training program will cover a period of approximately one (1) year for the applicant lacking Health Physics experience. The formal training may include instruction by outside consultants, and participation at operating reactor facilities in addition to on the job training, in-house instruction and examinations. The following is an outline of the Initial Monitor Training Program:

Introduction to Health Physics, (General topics: Mathematical computations, Basic Atomic and Nuclear Physics, Radiation and Radioactive Decay, Isotope production and disposal, Reactor Fundamentals).

Health Physics Course (General topics: Radiation and Contamination Surveys and Control, Posting Requirements, ALARA Applications, Respiratory Protection, Protective Clothing, Health Physics Procedures, Decontamination of Personnel and Equipment, Air Monitor Operation and Results Interpretation, Health Physics Record Keeping, Appropriate Station Plans and Procedures, Applicable Regulations and Limits, Radiological Emergency Monitoring Program, Radiation Work Permits (RWP), Health Physics Job Coverage, Personnel Monitoring)

BWR Health Physics (General Topics: BWR Systems, BWR Outage/Refueling, BWR Operational Health Physics)

Review and Monitor Qualifying Examination

Health Physics Supervision will review the applicant's proficiency as displayed during the training programs, examinations and the monitor qualification examination. The successful candidate will be assigned the responsibilities of Health Physics Monitor.

12.5.3.7.3.7 Health Physics Monitor Retraining Program

All Health Physics Monitors will receive a retraining review on an annual basis. The purpose of the review will be to strengthen the monitor's understanding of Health Physics applications and state of the art Health Physics technology. Review will consist of formal and/or informal training sessions that will include topics similar to those described in the Health Physics course above. One method of evaluating the monitor's competence in several areas may be the presentation of a hypothetical work situation problem requiring demonstration of Health Physics knowledge in a logical progression.

Areas not covered by the problem solving process will be evaluated by means of written and/or oral examinations. Records of training sessions and examinations will be forwarded to the Training Supervisor. An evaluation will be performed to identify areas where supplementary retraining may be necessary. Informal sessions will be held with the monitor by a member of Health Physics Supervision to discuss areas of individual concern and additional retraining needs.

Health Physics Monitors will be subject to all or any portion of the retraining process when deemed necessary by the Health Physics Supervisor or designated alternate based on job performance. Monitors may also request additional training in

areas of individual interest. A member of Health Physics Supervision will evaluate such requests and, if appropriate, administer specialized informal training to suit individual needs. In this case, the monitor's performance will not be subject to formal, documented evaluation.

15.0 ACCIDENT ANALYSES

In this chapter the effects of anticipated process disturbances and postulated component failures are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and events.

The scope of the situations analyzed includes anticipated (expected) operational occurrences (e.g., loss of electrical load), abnormal (unexpected) transients that induce system operations condition disturbances, postulated accidents of low probability (e.g., the sudden loss of integrity of a major component), and finally hypothetical events of extremely low probability (e.g., an anticipated transient without the operation of the entire control rod drive system).

15.0.1 ANALYTICAL OBJECTIVE

The spectrum of postulated initiating events is divided into categories based upon the type of disturbance and the expected frequency of the initiating occurrence; the limiting events in each combination of category and frequency are quantitatively analyzed. The plant safety analysis evaluates the ability of the plant to operate within regulatory guidelines, without undue risk to the public health and safety.

15.0.2 ANALYTICAL CATEGORIES

Transient and accident events contained in this report are discussed in individual categories as required by Reference 15.0-1. The results of the events are summarized in Table 15.0-1. Each event is assigned to one of the following applicable categories:

1. Decrease in Core Coolant Temperature:

   Reactor vessel water (moderator) temperature reduction results in an increase in core reactivity. This could lead to fuel-cladding damage.

2. Increase in Reactor Pressure:

   Nuclear system pressure increases threaten to rupture the reactor coolant pressure boundary (RCPB). Increasing pressure also collapses the voids in the core-moderator thereby increasing core reactivity and power level which threaten fuel cladding due to overheating.

3. Decrease in Reactor Core Coolant Flow Rate:

   A reduction in the core coolant flow rate threatens to overheat the cladding as the coolant becomes unable to adequately remove the heat generated by the fuel.

4. Reactivity and Power Distribution Anomalies:

   Transient events included in this category are those which cause rapid increases in power which are due to increased core flow disturbance events. Increased core flow reduces the void content of the moderator increasing core reactivity and power level.

5. Increase in Reactor Coolant Inventory:

   Increasing coolant inventory could result in excessive moisture carryover to the main turbine, feedwater turbines, etc.

6. Decrease in Reactor Coolant Inventory:

   Reductions in coolant inventory could threaten the fuel as the coolant becomes less able to remove the heat generated in the core.

7. Radioactive Release from a Subsystem or Component:

   Loss of integrity of a radioactive containment component is postulated.

8. Anticipated Transients Without Scram:

   In order to determine the capability of plant design to accommodate an extremely low probability event, a multi-system maloperation plus multi-single active component failures (SACF) situation is postulated.

15.0.3 EVENT EVALUATION

15.0.3.1 Identification of Causes and Frequency Classification

Situations and causes which lead to the initiating event analyzed are described within the categories designated above. The frequency of occurrence of each event is summarized based upon currently available operating plant history for the transient event. Events for which inconclusive data exists are discussed separately within each event section.

Each initiating event within the major groups is assigned to one of the following frequency groups:

1. Incidents of moderate frequency - these are incidents that may occur during a calendar year to once per 20 years for a particular plant. This event is referred to as an "anticipated (expected) operational transient."

2. Infrequent incidents - these are incidents that may occur during the life of the particular plant (spanning once in 20 years to once in 100 years). This event is referred to as an "abnormal (unexpected) operational transient."

3. Limiting faults - these are occurrences that are not expected to occur but are postulated because their consequences may result in the release of significant amounts of radioactive material. This event is referred to as a "design basis (postulated) accident."

4. Normal operation - operations of high frequency are not discussed here but are examined along with (1), (2), and (3) in the nuclear systems operational analyses in Appendix 15A.

15.0.3.1.1 Unacceptable Results for Incidents of Moderate Frequency (Anticipated (Expected) Operational Transients)

The following are considered to be unacceptable safety results for incidents of moderate frequency (anticipated operational transients):

1. A release of radioactive material to the environs that exceeds the limits of 10CFR20.

2. Reactor operation induced fuel cladding failure.

3. Nuclear system stresses in excess of that allowed for the transient classification by applicable industry codes.

4. Containment stresses in excess of that allowed for the transient classification by applicable industry codes.

15.0.3.1.2 Unacceptable Results for Infrequent Incidents (Abnormal (Unexpected) Operational Transients)

The following are considered to be unacceptable safety results for infrequent incidents (abnormal operational transients);

1. Release of radioactivity which results in dose consequences that exceed a small fraction of 10CFR100.

2. Fuel damage that would preclude resumption of normal operation after a normal restart.

3. Generation of a condition that results in consequential loss of function of the reactor coolant system.

4. Generation of a condition that results in a consequential loss of function of a necessary containment barrier.

15.0.3.1.3 Unacceptable Results for Limiting Faults (Design Basis (Postulated) Accidents)

The following are considered to be unacceptable safety results for limiting faults (design basis accidents):

1. Radioactive material release which results in dose consequences that exceed the guideline values of 10CFR100.

2. Failure of fuel cladding which would cause changes in core geometry such that core cooling would be inhibited.

3. Nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes.

4. Containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required.

5. Radiation exposure to plant operations personnel in the main control room in excess of 5 Rem whole body, 30 Rem inhalation, and 75 Rem skin.

15.0.3.2 Sequence of Events and Systems Operations

Each transient or accident is discussed and evaluated in terms of:

1. A step-by-step sequence of events from initiation to final stabilized condition.

2. The extent to which normally operating plant instrumentation and controls are assumed to function.

3. The extent to which plant and reactor protection systems are required to function.

4. The credit taken for the functioning of normally operating plant systems.

5. The operation of engineered safety systems that is required.

6. The effect of a single failure or an operator error on the event.

15.0.3.2.1 Single Failures or Operator Errors

15.0.3.2.1.1 General

This paragraph discusses a very important concept pertaining to the application of single failures and operator errors to analyses of the postulated events. Single active component failure (SACF) criteria have been required and successfully applied on past NRC approved docket applications to design basis accident categories only. Reference 15.0-1 infers that a "single failures and Operator errors" requirement should be applied to transient events (both high, moderate, and low probability occurrences) as well as accident (very low probability) situations.

Transient evaluations have been judged against a criteria of one single equipment failure "or" one single operator error as the initiating event with no additional single failure assumptions to the protective sequences although a great majority of these protective sequences utilized safety systems which can accommodate SACF aspects. Even under these postulated events, the plant damage allowances or limits were very much the same as those for normal operation.

Reference 15.0-1 suggests that the transient and accident scenarios should now include "and" (multi-failure) event sequences. The format request follows:

For initiating occurrence	1)	an equipment failure or an operator error, and
For single equipment failure or operator error analysis	2)	another equipment failure or failures and/or another operator error or errors.

This is considered a new requirement and the impact will need to be completely evaluated. While this is under consideration GE has evaluated and presented the transients and accidents in this chapter in the above new requirement manner.

Event categorization relative to transient and accident analysis is discussed here. If the evaluation is done per the new multi-failure methods, the event frequency categories should be modified.

The original categorization of events was based on frequency of the initiating event alone and thus the allowance or limit was accordingly established based on that high frequency level. With the introduction of additional assumptions and conditions (initial event and SCF and/or SOE), the total event would not fall into a lower frequency/probability category. Thus, less restrictive limits or allowances should be applied in the analysis of transients and accidents. This needs to be considered and evaluated.

GE has evaluated and presented the transients and accidents in this chapter by the more restrictive old allowances and limits of the event categorization presently in effect.

Most events postulated for consideration are already the results of single equipment failures or single operator errors that have been postulated during any normal or planned mode of plant operations. The types of operational single failures and operators errors considered as initiating events and subsequent protective sequence challenges are identified in the following paragraphs:

15.0.3.2.1.2 Initiating Event Analysis

1. The undesired opening or closing of any single valve (a check valve is not assumed to close against normal flow)

or

2. The undesired starting or stopping of any single component

or

3. The malfunction or maloperation of any single control device

or

4. Any single electrical component failure

or

5. Any single operator error.

Operator error is defined as an active deviation from written operating procedures or nuclear plant standard operating practices. A single operator error is the set of actions that is a direct consequence of a single erroneous decision. The set of actions is limited as follows:

1. Those actions that could be performed by one person.

2. Those actions that would have constituted a correct procedure had the initial decision been correct.

3. Those actions that are subsequent to the initial operator error and have an effect on the designed operation of the plant, but are not necessarily directly related to the operator error.

Examples of single operator errors are as follows:

1. An increase in power above the established flow control power limits by control rod withdrawal in the specified sequences.

2. The selection and complete withdrawal of a single control rod out of sequence.

3. An incorrect calibration of an average power range monitor.

4. Manual isolation of the main steam lines as a result of operator misinterpretation of an alarm or indication.

15.0.3.2.1.3 Single Active Component Failure or Single Operator Failure Analysis

1. The undesired action or maloperation of a single active component

or

2. Any single operator error where operator errors are defined as in Subsection 15.0.3.2.1.2.

15.0.3.3 Core and System Performance

15.0.3.3.1 Introduction

Section 4.4 describes the various fuel failure mechanisms. Avoidance of unacceptable safety limits 1 and 2 (Subsection

4.4.1.4) for incidents of moderate frequency is verified statistically with consideration given to date, calculation, manufacturing, and operating uncertainties. An acceptable criterion was determined to be that 99.9% of the fuel rods in the core would not be expected to experience boiling transition (see Reference 15.0-2). This criterion is met by demonstrating that transients do not result in a minimum critical power ratio (MCPR) less than 1.06. The reactor steady state CPR operating limit is derived by determining the decrease in MCPR for the most limiting event. All other event result in smaller MCPR decreases and are not reviewed in depth in this chapter. The MCPR during significant abnormal events is calculated using a transient core heat transfer analysis computer program. The computer program is based on a multinode, single channel thermal hydraulic model which requires simultaneous solution of the partial differential equations for the conservation of mass, energy, and momentum in the bundle, and which accounts for axial variation in power generation. The primary inputs to the model include a physical description of the bundle, and channel inlet flow and enthalpy, pressure and power generation as functions of time.

A detailed description of the analytical model may be found in Appendix C of Reference 15.0-2. Determination of the steady-state operating limit is accomplished as follows:

1. The change in critical power ratio (ΔCPR) which would result in the safety limit CPR (1.06) being reached, is calculated for each event. These values are shown in Table 15.0-1.

2. The ΔCPR value is then added to the safety limit CPR value (1.06) to result in the event based MCPR except for events whose ΔCPR is calculated using ODYN.

3. For events whose ΔCPR is determined by ODYN (all rapid pressurzation events) the event based MCPR is determined in conjunction with correction factors, the ΔCPR and the safety limit CPR. These correction factors are explained in detail in Section 3/4.2.3 of the Technical Specifications.

These results are given in Table 15.0-5 and Figure 15.0-3 for the limiting transients.

The operating limit MCPR is the maximum value of the event MCPRs calculated from the transient analysis. The maximum calculated transient MCPR is depicted by the solid line in Figure 15.0-3. Maintaining the CPR operating limit at or above this operating limit assures that the safety limit CPR of 1.06 is never violated.

For situations in which fuel damage is sustained, the event of damage is determined by correlating fuel energy content, cladding

temperature, fuel rod internal pressure, and cladding mechanical characteristics.

These correlations are substantiated by fuel rod failure tests and are discussed in Section 4.4 and Section 6.3.

15.0.3.3.2 Input Parameters and Initial Conditions for Analyzed Events

In general the events analyzed within this section have values for input parameters and initial conditions as specified in Table 15.0-2. Analyses which assume data inputs different than these values are designated accordingly in the appropriate event discussion.

15.0.3.3.3 Initial Power/Flow Operating Constraints

The analysis basis for most of the transient safety analyses is the thermal power at rated core flow (100%) corresponding to 105% Nuclear Boiler Rated steam flow. This operating point is the apex of a bounded operating power/flow map which, in response to any classified abnormal operational transients, will yield the minimum pressure and thermal margins of any operating point within the bounded map. Referring to Figure 15.0-1, the apex of the bounded power/flow map is point A, the upper bound is the design flow control line (105%, rod line A-D'), the lower bound is the zero power line H'-J', the right bound is the rated pump speed line A-H', and the left bound is either the minimum pump speed line D-J or the natural circulation line D'-J'.

The power/flow map, A-D'-J'-H-A, represents the acceptable operational constraints for abnormal operational transient evaluations.

Any other constraint which may truncate the bounded power/flow map must be observed, such as the recirculation valve and pump cavitation regions, the licensed power limit and other restrictions based on pressure and thermal margin criteria. For instance, if the licensed power is 100% nuclear power rated (NBR), the power/flow map is truncated by the line B-C and reactor operation must be confined within the boundary B-C-D'-J'-J-L-K-B. If the maximum operating power level has to be limited, such as point F, to satisfy pressure margin criteria, the upper constraint on power/flow is correspondingly reduced to the rod line, such as line F G', which intersects the power/flow coordinate of the new operating basis. In this case, the operating bounds would be F-G'-J'-J'-J-L-K-F. Operation would not be allowed at any point along line F-M, removed from point F,

at the derated power but at reduced flow. If, however, operating limitations are imposed by GETAB derived from transient data with an operating basis at point A, the power/flow boundary for 100% NBR licensed power would be B-C-D'-J'-J-L-K-B. This power/flow boundary would be truncated by the MCPR operating limit for which there is no direct correlation to a line on the power/flow boundary and within the constraints imposed by GETAB. If operation is restricted to point F by the MCPR operating limit, operation at point M would be allowed provided the MCPR limit is not violated.

Consequently, the upper operating power/flow limit of a reactor is predicated on the operating basis of the analysis and the corresponding constant rod pattern line. This boundary may be truncated by the licensed power and the GETAB operating limit.

Certain localized events are evaluated at other than the above mentioned conditions. These conditions are discussed pertinent to the appropriate event.

15.0.3.3.4 Results

The results of analytical evaluations are provided for each event. In addition critical parameters are shown in Table 15.0-1. From the data in Table 15.0-1 an evaluation of the limiting event for that particular category and parameter can be made. In Table 15.0-1A a summary of applicable accidents is provided. This table compares the GE calculated amount of failed fuel to that used in worst case Radiological Calculations.

15.0.3.5 Barrier Performance

This section primarily evaluates the performance of the Reactor Coolant Pressure Boundary (RCPB) and the Containment System during transients and accidents.

During transients that occur with no release of coolant to the containment only RCPB performance is considered. If release to the containment occurs as in the case of limiting faults, then challenges to the containment are evaluated as well.

Containment integrity is maintained so long as internal pressures remain below the maximum allowable values. The design internal pressures are as follows:

Drywell (primary containment)	53 psig
Suppression Chamber (primary containment	53 psig

Secondary Containment 7 in. H_2O

Damage to any of the radioactive material barriers as a result of accident-initiated fluid impingement and jet forces is considered in the other portions of the FSAR where the mechanical design features of systems and components are described. Design basis accidents are used in determining the sizing and strength requirements of the essential nuclear system components. A comparison of the accidents considered in this section with those used in the mechanical design of equipment reveals either that the applicable accidents are the same or that the accident in this section results in less severe stresses than those assumed for mechanical design.

15.0.3.6 Radiological Consequences

In this chapter, the consequences of radioactiveity released during the three types of events: a) incidents of moderate frequency (anticipated operational transients), (b) infrequent incidents (abnormal operational transients), and c) limiting faults (design basis accidents) are considered. For all events whose consequences are limiting a detailed quantitative evaluation is presented. For non-limiting events a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

For limiting faults (design basis accidents) two quantitative analyses are considered:

1. The first is based on conservative assumptions considered to be acceptable to the NRC for the purposes of bounding the event and determining the adequacy of the plant design to meet 10 CFR Part 100 guidelines. This analyses is referred to as the "design basis analysis."

2. The second is based on realistic assumptions considered to reflect expected radiological consequences. This analysis is referred to as the "realistic analysis."

Results for both are shown to be within NRC guidelines.

Atmospheric Dispersion Parameters

Short-term site-specific X/Q's were calculated as described in Section 2.3. For the conservative case, the 5 percent probability level X/Q's were used in the dose calculations. The resultant offsite doses are conservative. For the realistic case, 50 percent probability level X/Q's were used. The 5 and 50

percent level X/Q's are given in Tables 15.0-3 and 15.0-4, respectively.

15.0.4 Nuclear Safety Operational Analysis (NSQA) Relationship

Appendix 15A is a comprehensive system-level, qualitative FMEA, relative to all the events considered, the protective sequences utilized to accommodate the events and their effects, and the systems involved in the protective actions.

Interdependency of analysis and cross-referral of protective actions is an integral part of this chapter and the appendix.

Contained in Appendix 15A is a summary table which classifies events by frequency only (i.e., not just within a given category such as Decrease in Core Coolant Temperature).

15.0.5 REFERENCES

15.0-1 United States Nuclear Regulatory Commission Regulation Guide 1.70 Revision 2 (Preliminary), September 1975, "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants, Light Water Reactor Edition."

15.0-2 "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application," November 1973, (NEDO-10959 and NEDE-10958).

Subsection I.D.	Figure I.D.	Description	Maximum Neutron Flux NBR	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam Line Pressure psig	Maximum Core Average Surface Heat Flux % of Initial	ΔCPR** -	Frequency Category*	Duration of Blowdown: No. of Valves 1st Blow-down	Duration of Blowdown: Duration-of Blowdown sec
15.2.5	15.2-6	Loss of Condenser Vacuum	167.5	1140	1165	1131	101.3	<0.09	a	16	20
15.2.6	15.2-7	Loss of Auxiliary Power Transformer	104.5	1145	1160	1140	100.1	~0.0(1)	a	16	16
15.2.6	15.2-8	Loss of All Grid Connections	107.2	1140	1161	1130	100.1	~0.0(1)	a	13	17
15.2.7	15.2-9	Loss of All Feedwater Flow	103.8	1094	1105	1094	100.1	~0.0(1)	a	2	5
15.2.8		Feedwater Piping Break	See 15.6.6								
15.2.9		Failure of RHR Shutdown Cooling	See Text								
15.3		DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE									
15.3.1	15.3-1	Trip of One Recirculation Pump Motor	103.6	1015	1053	998	100.0	~0.0(1)	a	0	0
15.3.1	15.3-2	Trip of Both Recirculation Pump Motors	103.5	1113	1127	1109	100.1	~0.0(1)	a	10	28
15.3.2		Recirculation Flow Control Failure Decreasing Flow	See 15.3.1								
15.3.3	15.3-3	Seizure of One Recirculation Pump	103.2	1126	1137	1120	100.4	~0.0(1)	c	13	
15.3.4		Recirc.Pump Shaft Break	See 15.3.3						c		
15.4		REACTIVITY AND POWER DISTRIBUTION ANOMALIES									
15.4.1.1		RWE - Refueling	See Text						b		

TABLE 15.0-1 (cont'd)

Subsection I.D.	Figure I.D.	Description	Maximum Neutron Flux NBR	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam Line Pressure psig	Maximum Core Average Surface Heat Flux % of Initial	ΔCPR** -	Frequency Category*	Duration of Blowdown: No. of Valves 1st Blow-down	Duration of Blowdown: Duration-of Blowdown sec
15.4.1.2		RWE - Startup	See Text						b		
15.4.2		RWE - At Power	See Text						a		
15.4.3		Control Rod Misoperation	See Subsections 15.4.1 and 15.4.2								
15.4.4	15.4-6	Startup of Idle Recirculation Loop	323.4	973	988	967	134.9	(3)	a	0	0
15.4.5	15.4-7	Recirculation Flow Control Failure - Increasing Flow	264.6	982	1008	973	130.3	(3)	a	0	0
15.4.7		Misplaced Bundle Accident	See Text						b		
15.5		INCREASED IN REACTOR COOLANT INVENTORY									
15.5.1	15.5-1	Inadvertent HPCI Pump Start	118.2	1023	1061	1004	11.4	0.11	a	0	0
15.5.3		BWR Transients	See appropriate Events in Sections 15.1 and 15.2								

* a - incidents of moderate frequency
b - infrequent incidents
c - limiting faults

** ΔCPR based on an initial CPR which yields an MCPR = 1.06
(1) estimated values
(2) ODYN results without adjustment factors
(3) These events are initiated from low power levels - MCPR > 1.06

Table 15.0-2 (Continued)

18. Core Average Rated Void** Fraction, %	40.74
19. Scram Reactivity, Δk Analysis Data	Figure 15.0-2
20. Control Rod Drive Speed, Position versus time	Figure 15.0-2
21. Jet Pump Ratio, M	1.84
22. Safety/Relief Valve Capacity, % NBR	
@ 1091 psig	99.0
Manufacturer	CROSBY
Quantity Installed	16
23. Relief Function Delay, seconds	0.4
24. Relief Function Response, seconds	0.15
25. Set Points for Safety/Relief Valves, psig	1110, 1120, 1130, 1140, 1150
26. Number of Valve Groupings Simulated	5
27. High Flux Trip, % NBR Analysis set point (120 x 1.044), % NBR	125.3
28. High Pressure Scram Set Point, psig	1071
29. Vessel Level Trips, Inches Above (+), Below (-) Separator Skirt Bottom	
Level 8 - (L8), inches	+54
Level 4 - (L4), inches	+30
Level 3 - (L3), inches	+12.5
Level 2 - (L2), inches	-38
30. APRM Thermal Trip Set Point, % NBR	125.0
31. Recirculation Pump Trip Delay, Seconds	0.175
32. Recirculation Pump Trip Inertia for Analysis, seconds*	4.5

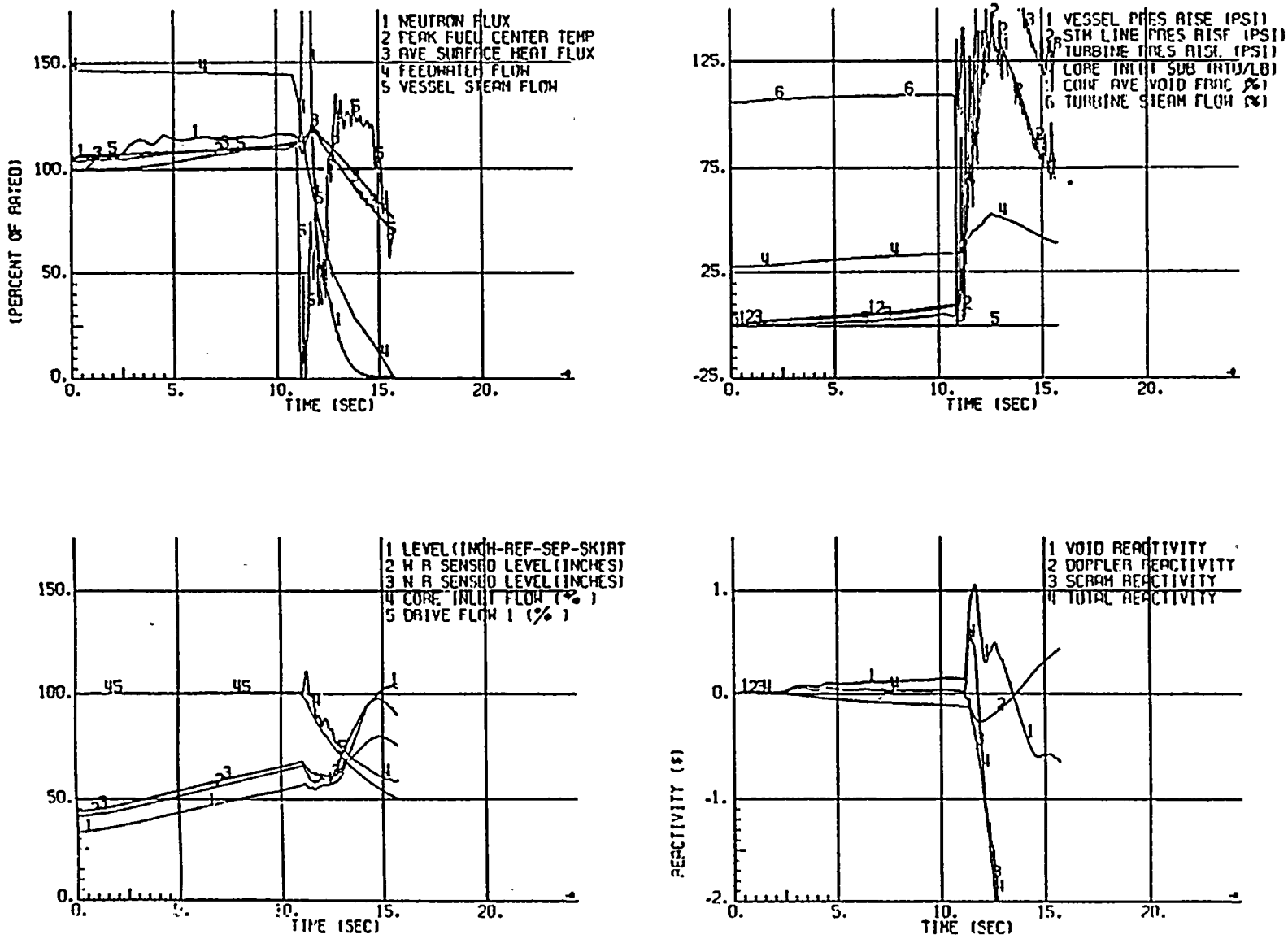
*The inertia time constant is defined by the expression:

Table 15.0-2 (Continued)

$$t = \frac{2\pi J_o n}{g T_o}$$

, where t = inertia time constant (Sec).
J_o = pump motor inertia (lb-ft²)
n = rated pump speed (rps)
g = gravitational constant (ft/sec²)
T_o = pump shaft torque (lb-ft)

** Parameters used in REDY only. ODYN values are calculated within the code for equilibrium cycle conditions.



Rev. 26, 9/81

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER FAILURE, MAXIMUM DEMAND, WITH HIGH WATER LEVEL TRIPS

FIGURE 15.1-2

15.2 INCREASE IN REACTOR PRESSURE

15.2.1 PRESSURE REGULATOR FAILURE - CLOSED

15.2.1.1 Identification of Causes and Frequency Classification

15.2.1.1.1 Identification of Causes

Two identical pressure regulators are provided to maintain primary system pressure control. They independently sense pressure just upstream of the main turbine stop valves and compare it to two separate set points to create proportional error signals that produce each regulator output. The output of both regulators feeds in a high gate value. The regulator with the highest output controls the main turbine control valves. The lowest pressure set point gives the largest pressure error and thereby largest regulator output. The backup regulator is set 5 psi higher giving a slightly smaller error and a slightly smaller effective output of the controller.

It is assumed for purposes of this transient analysis that a single failure occurs which erroneously causes the controlling regulator to close the main turbine control valves and thereby increases reactor pressure. If this occurs, the backup regulator is ready to take control.

15.2.1.1.2 Frequency Classification

This event is treated as a moderate frequency event.

15.2.1.2 Sequence of Events and System Operation

15.2.1.2.1 Sequence of Events

Postulating a failure of the primary or controlling pressure regulator in the closed mode as discussed in Subsection 15.2.1.1.1 will cause the valves to close momentarily. The pressure will increase, because the reactor is still generating the initial steam flow. The backup regulator will reopen the valves and re-establish steady-state operation above the initial pressure equal to the set point difference of 5 psi.

15.2.1.2.1.1 Identification of Operator Actions

The operator will verify that the backup regulator assumes proper control. However, this action is not required as discussed below in Subsection 15.2.1.2.3.

15.2.1.2.2 Systems Operation

Normal plant instrumentation and control is assumed to function. This event requires no protection system or safeguard systems operation.

15.2.1.2.3 The Effect of Single Failures and Operator Errors

The nature of the first assumed failure produces a slight pressure increase in the reactor until the backup regulator gains control. If we fail the backup regulator at this time (the second assumed failure), the control valves would start to close, raising the reactor pressure to the point where a flux or pressure scram trip would be initiated to shut down the reactor. This event is less severe than the turbine trip where stop valve closure occurs (Subsection 15.2.3).

15.2.1.3 Core and System Performance

The disturbance is mild, similar to a pressure set point change and no significant reductions in fuel thermal margins occur. This transient is much less severe than the generator and turbine trip transients described in Subsections 15.2.2 and 15.2.3.

15.2.1.3.1 Mathematical Model

Only qualitative evaluation is provided.

15.2.1.3.2 Input Parameters and Initial Conditions

Only qualitative evaluation is provided.

15.2.1.3.3 Results

Response of the reactor during this regulator failure is such that pressure at the turbine inlet increases quickly, less than 2 seconds or so, due to the sharp closing action of the turbine control valves which reopen when the backup regulator gains control. This pressure disturbance in the vessel is not expected to exceed flux or pressure scram trip set points.

15.2.1.3.4 Consideration of Uncertainties

All systems utilized for protection in this event were assumed to have the poorest allowable response (e.g., relief set points, scram stroke time, and work characteristics). Plant behavior is, therefore, expected to reduce the actual severity of the transient.

15.2.1.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.1.5 Radiological Consequences

Since this event does not result in any additional fuel failures or any release of primary coolant to either the secondary containment or to the environment, there are no radiological consequences associated with this event.

15.2.2 Generator Load Rejection

15.2.2.1 Identification of Causes and Frequency Classification

15.2.2.1.1 Identification of Causes

Fast closure of the turbine control valves (TCV) is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The turbine control valves are required to close as rapidly as possible to prevent excessive overspeed of the turbine-generator (T-G) rotor. Closure of the main turbine control valves will cause a sudden reduction in steam flow which results in an increase in system pressure and reactor shutdown.

15.2.2.1.2 Frequency Classification

15.2.2.1.2.1 Generator Load Rejection With or Without Bypass

This event is categorized as an incident of moderate frequency.

15.2.2.2 Sequence of Events and System Operation

15.2.2.2.1 Sequence of Events

15.2.2.2.1.1 Generator Load Rejection - Turbine Control Valve Fast Closure

A loss of generator electrical load from high power conditions produces the sequence of events listed in Table 15.2-1.

15.2.2.2.1.2 Generator Load Rejection with Failure of Bypass

A loss of generator electrical load at high power with bypass failure produces the sequence of events listed in Table 15.2-2.

15.2.2.2.1.3 Identification of Operator Actions

(1) Verify proper bypass valve performance.

(2) Observe that the feedwater/level controls have maintained the reactor water level at a satisfactory value.

(3) Observe that the pressure regulator is controlling reactor pressure at the desired value.

(4) Record peak power and pressure.

(5) Verify relief valve operation.

15.2.2.2.2 System Operation

15.2.2.2.2.1 Generator Load Rejection with Bypass

In order to properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems.

Turbine control valve (TCV) fast closure initiates a scram trip signal for power levels greater than 30% NB rated. In addition, recirculation pump trip is initiated. Both of these trip signals satisfy single failure criterion and credit is taken for these protection features.

The pressure relief system which operates the relief valves independently when system pressure exceeds relief valve instrumentation set points is assumed to function normally during the time period analyzed.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.2.2.2.2 Generator Load Rejection with Failure of Bypass

Same as Subsection 15.2.2.2.2.1 except that failure of the main turbine bypass valves is assumed for the entire transient.

15.2.2.2.3 The Effect of Single Failures and Operator Errors

Mitigation of pressure increase is accomplished by the reactor protection system functions. Turbine control valve trip scram and RPT are designed to satisfy the single failure criterion. An evaluation of the most limiting single failure (i.e., failure of the bypass system) was considered in this event. Details of single failure analysis can be found in Appendix 15A.

15.2.2.3 Core and System Performance

15.2.2.3.1 Mathematical Model

The computer model described in Subsection 15.1.2.3.1 was used to simulate this event.

15.2.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15.0-2.

The turbine electrohydraulic control system (EHC) power/load imbalance device detects load rejection before a measurable speed change takes place.

The closure characteristics of the turbine control valves are assumed such that the valves operate in the full arc (FA) mode and have a full stroke closure time, from fully open to fully closed of 0.15 seconds.

Auxiliary power would normally be independent of any turbine-generator overspeed effects and continuously supplied at rated frequency since automatic fast transfer to auxiliary power supplies normally occurs. For the purposes of worst case analysis, the recirculation pumps are assumed to remain tied to the main generator and thus increase in speed with the T-G overspeed until tripped by the Recirculation Pump Trip system (RPT).

The reactor is operating in the manual flow-control mode when load rejection occurs. Results do not significantly differ if the plant had been operating in the automatic flow-control mode.

The bypass valve opening characteristics are simulated using the specified delay together with the specified opening characteristic required for bypass system operation.

Although the closure of main steam isolation valves as caused by low water level trip (L2) is included in the simulation, the flows from initiation of RCIC and HPCI core cooling system functions are not included. If these events occur, they will follow sometime after the primary concerns of fuel margin and overpressure effects have passed and are expected to result in effects less severe than those already experienced by the reactor system.

15.2.2.3.3 Results

15.2.2.3.3.1 Generator Load Rejection with Bypass

Figure 15.2-1 shows the results of the generator trip from rated power. Peak neutron flux rises 282% of the rated value.

The average surface heat flux peaks at 110% of its initial value and MCPR does not significantly decrease below its initial value.

15.2.2.3.3.2 Generator Load Rejection with Failure of Bypass

Figure 15.2-2 shows that, for the case of Bypass failure, peak neutron flux reaches about 466% of rated, average surface heat flux reaches 118% of its initial value.

15.2.2.3.4 Consideration of Uncertainties

The full stroke closure rate of the turbine control valve of 0.15 seconds is conservative. Typically, the actual closure rate is more like 0.2 seconds. Clearly the less time it takes to close, the more severe the pressurization effect.

All systems utilized for protection in this event were assumed to have the poorest allowable response (e.g., relief set points, scram stroke time and work characteristics). Plant behavior is, therefore, expected to reduce the actual severity of the transient.

15.2.2.4 Barrier Performance

15.2.2.4.1 Generator Load Rejection

Peak pressure remains within normal safety range and no threat to the barrier exists.

15.2.2.4.2 Generator Load Rejection with Failure of Bypass

Peak pressure at the valves reaches 1189 psig. The peak nuclear system pressure reaches 1218 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1375 psig.

15.2.2.5 Radiological Consequences

While the consequence of this event does not result in fuel failures, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.3 TURBINE TRIP

15.2.3.1 Identification of Causes and Frequency Classification

15.2.3.1.1 Identification of Causes

A variety of turbine or nuclear system malfunctions will initiate a turbine trip. Some examples are moisture separator and heater drain tank high levels, large vibrations, operator lock out, loss of control fluid pressure, low condenser vacuum and reactor high water level.

15.2.3.1.2 Frequency Classification

15.2.3.1.2.1 Turbine Trip

This transient is categorized as an incident of moderate frequency. In defining the frequency of this event, turbine trips which occur as a byproduct of other transients such as loss of condenser vacuum or reactor high level trip events are not included. However, spurious low vacuum or high level trip signals which cause an unnecessary turbine trip are included in defining the frequency. In order to get an accurate event-by-event frequency breakdown, this type of division of initiating causes is required.

15.2.3.2 Sequence of Events and Systems Operation

15.2.3.2.1 Sequence of Events

15.2.3.2.1.1 Turbine Trip

Turbine trip at high power produces the sequence of events listed in Table 15.2-3.

15.2.3.2.1.2 Turbine Trip with Failure of the Bypass

Turbine trip at high power with bypass failure produces the sequence of events listed in Table 15.2-4.

15.2.3.2.1.3 Identification of Operator Actions

The operator must:

(1) Verify auto transfer of buses supplied by generator to incoming power; if automatic transfer does not occur, manual transfer must be made.

(2) Monitor and maintain reactor water level at required level.

(3) Check turbine for proper operation of all auxiliaries during coastdown.

(4) Depending on conditions, initiate normal operating procedures for cool-down, or maintain pressure for restart purposes.

(5) Put the mode switch in the startup position before the reactor pressure decays to < 850 psig.

(6) Secure the RCIC operation if auto initiation occurred due to low water level.

(7) Monitor control rod drive positions and insert both the IRMs and SRMs.

(8) Investigate the cause of the trip, make repairs as necessary, and complete the scram report.

(9) Cool down the reactor per standard procedure if a restart is not intended.

15.2.3.2.2 Systems Operation

15.2.3.2.2.1 Turbine Trip

All plant control systems maintain normal operation unless specifically designated to the contrary.

Turbine stop valve closure initiates a reactor scram trip via position signals to the protection system. Credit is taken for successful operation of the reactor protection system.

Turbine stop valve closure initiates recirculation pump trip (RPT) thereby terminating the jet pump drive flow.

The pressure relief system which operates the relief valves independently when system pressure exceeds relief valve instrumentation set points is assumed to function normally during the time period analyzed.

15.2.3.2.2.2 Turbine Trip with Failure of the Bypass

Same as Subsection 15.2.3.2.2.1 except that failure of the main turbine bypass system is assumed for the entire transient time period analyzed.

15.2.3.2.2.3 Turbine Trip at Low Power with Failure of the Bypass

Same as Subsection 15.2.3.2.2.1 except that failure of the main turbine bypass system is assumed.

It should be noted that below 30% NB rated power level, a main stop valve scram trip inhibit signal derived from the first stage pressure of the turbine is activated. This is done to eliminate the stop valve scram trip signal from scramming the reactor provided the bypass system functions properly. In other words, the bypass would be sufficient at this low power to accommodate a turbine trip without the necessity of shutting down the reactor. All other protection system functions remain functional as before and credit is taken for those protection system trips.

15.2.3.2.3 The Effect of Single Failures and Operator Errors

15.2.3.2.3.1 Turbine Trips at Power Levels Greater Than 30% NBR

Mitigation of pressure increase is accomplished by the reactor protection system functions. Main stop valve closure scram trip and RPT are designed to satisfy single failure criterion.

15.2.3.2.3.2 Turbine Trips at Power Levels Less Than 30% NBR

Same as Subsection 15.2.3.2.3.1 except RPT and stop valve closure scram trip is normally inoperative. Since protection is still provided by high flux, high pressure, etc., these will also continue to function and scram the reactor should a single failure occur.

15.2.3.3 Core and System Performance

15.2.3.3.1 Mathematical Model

The computer model described in Subsection 15.1.1.3.1 was used to simulate the turbine trip with bypass event, and one in Subsection 15.1.2.3.1 was used for the turbine trip with failure of bypass event.

15.2.3.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

Turbine stop valves full stroke closure time is 0.1 second.

A reactor scram is initiated by position switches on the stop valves when the valves are less than 90% open. This stop valve scram trip signal is automatically bypassed when the reactor is below 30% NB rated power level.

Reduction in core recirculation flow is initiated by position switches on the main stop valves, which actuate trip circuitry which trips the recirculation pumps.

15.2.3.3.3 Results

15.2.3.3.3.1 Turbine Trip

A turbine trip with the bypass system operating normally is simulated at 105% NB rated steam flow conditions in Figure 15.2-3.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited to 167% of rated by the stop valve scram and the RPT system. Peak fuel surface heat flux does not exceed 101% of its initial value.

15.2.3.3.3.2 Turbine Trip with Failure of Bypass

A turbine trip with failure of the bypass system is simulated at 105% NB rated steam flow conditions in Figure 15.2-4.

Peak neutron flux reaches 447% of its rated value, and the peak surface heat flux reaches 116% of its initial value.

15.2.3.3.3.3 Turbine Trip with Bypass Valve Failure, Low Power

This transient is less severe than a similar one at high power. Below 30% of rated power, the turbine stop valve closure and turbine control valve closure scrams are automatically bypassed.

At these lower power levels, turbine first stage pressure is used to initiate the scram logic bypass. The scram which terminates the transient is initiated by high vessel pressure. The bypass valves are assumed to fail; therefore, system pressure will increase until the pressure relief set points are reached. At this time, because of the relatively low power of this transient event, relatively few relief valves will open to limit reactor pressure. Peak pressures are not expected to greatly exceed the pressure relief valve set points and will be significantly below the RCPB transient limit of 1375 psig. Peak surface heat flux and peak fuel center temperature remain at relatively low values and MCPR is expected to remain well above the GETAB safety limit.

15.2.3.3.4 Considerations of Uncertainties

Uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

(1) Slowest allowable control rod scram motion is assumed.

(2) Scram worth shape for all-rod-out conditions is assumed.

(3) Minimum specified valve capacities are utilized for over- pressure protection.

(4) Set points of the safety/relief valves include errors (high) for all valves.

15.2.3.4 Barrier Performance

15.2.3.4.1 Turbine Trip

Peak pressure in the bottom of the vessel reaches 1167 psig, which is below the ASME code limit of 1375 psig for the reactor cooling pressure boundary. Vessel dome pressure does not exceed 1143 psig. The severity of turbine trips from lower initial power levels decreases to the point where a scram can be avoided if auxiliary power is available from an external source and the power level is within the bypass capability.

15.2.3.4.2 Turbine Trip with Failure of the Bypass

The safety/relief valves are open and close sequentially as the stored energy is dissipated and the pressure falls below the set points of the valves. Peak nuclear system pressure reaches 1215 psig at the vessel bottom, therefore, the overpressure transient is clearly below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Peak dome pressure does not exceed 1185 psig.

15.2.3.4.2.1 Turbine Trip with Failure of Bypass at Low Power

Qualitative discussion is provided in Subsection 15.2.3.3.3.3.

15.2.3.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release condition. If purging of the containment is chosen, the release will have to be in accordance with established technical specifications; therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.4 MSIV CLOSURES

15.2.4.1 Identification of Causes and Frequency Classification

15.2.4.1.1 Identification of Causes

Various steam line and nuclear system malfunctions, or operator actions, can initiate main steam isolation valve (MSIV) closure. Examples are low steamline pressure, high steamline flow, high steamline radiation, low water level or manual action.

15.2.4.1.2 Frequency Classification

15.2.4.1.2.1 Closure of All Main Steam Isolation Valves

This event is categorized as an incident of moderate frequency. To define the frequency of this event as an initiating event and not the byproduct of another transient, only the following contribute to the frequency: manual action (purposely or inadvertent); spurious signals such as low pressure, low reactor water level, low condenser vacuum, etc.; and finally, equipment malfunctions such as faulty valves or operating mechanisms. A closure of one MSIV may cause an immediate closure of all the other MSIVs depending on reactor conditions. If this occurs, it is also included in this category. During the main steam isolation valve closure, position switches on the valves provide a reactor scram if the valves in three or more main steam lines

are less than 90% open (except for interlocks which permit proper plant startup.). Protection system logic, however, permits the test closure of one valve without initiating scram from the position switches.

15.2.4.1.2.2 Closure of One Main Steam Isolation Valve

This event is categorized as an incident of moderate frequency. One MSIV may be closed at a time for testing purposes; this is done manually. Operator error or equipment malfunction may cause a single MSIV to be closed inadvertently. If reactor power is greater than about 80% when this occurs, a high flux or high steam line flow scram may result, (if all MSIVs close as a result of the single closure, the event is considered as a closure of all MSIVs).

15.2.4.2 Sequence of Events and Systems Operation

15.2.4.2.1 Sequence of Events

Table 15.2-5 lists the sequence of events for Figure 15.2-5.

15.2.4.2.1.1 Identification of Operator Actions

The following is the sequence of operator actions expected during the course of the event assuming no restart of the reactor. The operator should

(1) Observe that all rods have inserted.

(2) Observe that the relief valves have opened for reactor pressure control.

(3) Check that RCIC auto starts on the impending low reactor water level condition.

(4) Switch the feedwater controller to the manual position.

(5) Initiate operation of the RHR system in the steam condensing mode only.

(6) When the reactor vessel level has recovered to a satisfactory level, switch RCIC controller to manual and secure when level is under control.

(7) When the reactor has cooled sufficiently for RHR operation, put it into service per procedure.

(8) Before resetting the MSIV isolation, determine the cause of valve closure.

(9) Observe turbine coastdown and break vacuum before the loss of sealing steam. Check T-G auxiliaries for proper operation.

(10) Not reset and open MSIVs unless conditions warrant and be sure the pressure regulator set point is above vessel pressure.

(11) Survey maintenance requirements and complete the scram report.

15.2.4.2.2 Systems Operation

15.2.4.2.2.1 Closure of All Main Steam Isolation Valves

MSIV closures initiate a reactor scram trip via position signals to the protection system. Credit is taken for successful operation of the protection system.

The pressure relief system which initiates opening of the relief valves when system pressure exceeds relief valve instrumentation set points is assumed to function normally during the time period analyzed.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.4.2.2.2 Closure of One Main Steam Isolation Valve

A closure of a single MSIV at any given time will not initiate a reactor scram. This is because the valve position scram trip logic is designed to accommodate single valve operation and testability during normal reactor operation at limited power levels. Credit is taken for the operation of the pressure and flux signals to initiate a reactor scram.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.4.2.3 The Effect of Single Failures and Operator Errors

Mitigation of pressure increase is accomplished by initiation of the reactor scram via MSIV position switches and the protection system. Relief valves also operate to limit system pressure. All of these aspects are designed to single failure criterion and additional single failures would not alter the results of this analysis. Closure of one MSIV plus a single active component failure (the second MSIV) results in a situation no worse than the analysis of the four closed MSIVs.

Failure of a single relief valve to open is not expected to have any significant effect. Such a failure is expected to result in less than a 20 psi increase in the maximum vessel pressure rise. The peak pressure will still remain considerably below 1375 psig. The design basis and performance of the pressure relief system is discussed in Chapter 5.

15.2.4.3 Core and System Performance

15.2.4.3.1 Mathematical Model

The computer model described in Subsection 15.1.1.3.1 was used to simulate these transient events.

15.2.4.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

The main steam isolation valves close in 3 to 5 seconds. The worst case, the 3-second closure time, is assumed in this analysis.

Position switches on the valves initiate a reactor scram when the valves are less than 90% open. Closure of these valves inhibits steam flow to the feedwater turbines terminating feedwater flow.

Valve closure indirectly causes a trip of the main turbine and generator.

Because of the loss of feedwater flow, water level within the vessel decreases sufficiently to initiate trip of the recirculation pump and initiate the HPCI and RCIC systems.

15.2.4.3.3 Results

15.2.4.3.3.1 Closure of All Main Steam Isolation Valves

Figures 15.2-5 shows the changes in important nuclear system variables for the simultaneous isolation of all main steamlines while the reactor is operating at 105% of NB rated steam flow. Peak neutron flux reaches 164% of rated after approximately 2.4 seconds. At this time, the nonlinear valve closure becomes a strong effect and the conservative scram characteristic assumption has not yet allowed credit for the full shutdown of the reactor.

15.2.4.3.3.2 Closure of One Main Steam Isolation Valve

Only one isolation valve is permitted to be closed at a time for testing purposes to prevent scram. Normal test procedure requires an initial power reduction to approximately 80 to 90% of design conditions in order to avoid high flux scram, high pressure scram, or full isolation from high steam flow in the "live" lines. With a 3-second closure of one main steam isolation valve during 105% rated power conditions, the steam flow disturbance raises vessel pressure and reactor power enough to initiate a high neutron flux scram. This transient is considerably milder than the full power case. No quantitative analysis is furnished for this event. However, no significant change in thermal margins is experienced and no fuel damage occurs. Peak pressure remains below SRV set points.

Inadvertent closure of one or all of the isolation valves while the reactor is shut down (such as operating state C, as defined in Appendix 15A) will produce no significant transient. Closures during plant heatup (operating state D) will be less severe than the maximum power cases (maximum stored and decay heat) discussed in Subsection 15.2.4.3.3.1.

15.2.4.3.4 Considerations of Uncertainties

Uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For examples:

(1) Slowest allowable control rod scram motion is assumed.

(2) Scram worth shape for all-rod-out conditions is assumed.

(3) Minimum specified valve capacities are utilized for over-pressure protection.

(4) Set points of the safety/relief valves are assumed to be 1% higher than the valve's nominal set point.

15.2.4.4 Barrier Performance

15.2.4.4.1 Closure of All Main Steam Isolation Valves

The nuclear system relief valves begin to open at approximately 2.7 seconds after the start of isolation. The SRVs close sequentially as the stored heat is dissipated but continue to discharge the decay heat intermittently. Peak pressure at the vessel bottom reaches 1187 psig, clearly below the pressure limits of the reactor coolant pressure boundary. Peak pressure in the main steamline is 1146 psig.

15.2.4.4.2 Closure of One Main Steam Isolation Valve

If closure of the valve occurs at an unacceptably high operating power level, a flux or pressure scram will result; therefore, no significant effect is imposed on the RCPB. The main turbine bypass system will continue to regulate system pressure via the other three "live" steamlines.

15.2.4.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release condition. If purging of the containment is chosen, the release will have to be in accordance with established technical specifications; therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

The activity released to the suppression chamber can be contained for some period of time. It is, therefore, assumed that the activity airborne above the suppression pool will be released

under controlled conditions. The operator can choose to release the activity after decay has reduced the amount of activity to levels where the offsite dose consequence is minimal. For example, consider the case when the activity is released through the containment purge at an assumed time of 4 hours after the blowdown is complete (8 hours after the transient begins).

The containment airborne activity is discharged via the SGTS, which has a filter efficiency of 99 percent for the iodine activity. For this example, the airborne activities above the suppression pool and the activity released to the environs are listed in Tables 15.2-6 and 15.2-7 respectively.

The offsite radiological doses are presented in Table 15.2-8.

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15.2.5 LOSS OF CONDENSER VACUUM

15.2.5.1 Identification of Causes and Frequency Classification

15.2.5.1.1 Identification of Causes

Various system malfunctions which can cause a loss of condenser vacuum due to some single equipment failure are designated in Table 15.2-9.

15.2.5.1.2 Frequency Classification

This event is categorized as an incident of moderate frequency.

15.2.5.2 Sequence of Events and Systems Operation

15.2.5.2.1 Sequence of Events

Table 15.2-10 lists the sequence of events for Figure 15.2-6.

15.2.5.2.1.1 Identification of Operator Actions

The operator must:

(1) Verify auto transfer of buses supplied by generator to incoming power - if automatic transfer does not occur, manual transfer must be made.

(2) Monitor and maintain reactor water level at required level.

(3) Check turbine for proper operation of all auxiliaries during coastdown.

(4) Depending on conditions, initiate normal operating procedures for cooldown, or maintain pressure for restart purposes.

(5) Put the mode switch in the "STARTUP" position before the reactor pressure decays to < 850 psig.

(6) Secure the RCIC operation if auto initiation occurred due to low water level.

(7) Monitor control rod drive positions and insert both the IRMs and SRMs.

(8) Investigate the cause of the trip, make repairs as necessary, and complete the scram report.

(9) Cool down the reactor per standard procedure if a restart is not intended.

15.2.5.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems. Tripping functions incurred by sensing main turbine condenser vacuum pressure are designated in Table 15.2-11.

15.2.5.2.3 The Effect of Single Failures and Operator Errors

This event does not lead to a general increase in reactor power level. Mitigation of power increase is accomplished by the protection system initiation of scram. Failure of the integrity of the condenser unit itself is considered to be an accident situation and is described in Subsection 15.7.1.

Single failures will not effect the vacuum monitoring and turbine trip devices which are redundant. The protective sequences of the anticipated operational transient are shown to be single failure proof. See Appendix 15A for details.

15.2.5.3 Core and System Performance

15.2.5.3.1 Mathematical Model

The computer model described in Subsection 15.1.1.3.1 was used to simulate this transient event.

15.2.5.3.2 Input Parameters and Initial Conditions

This analysis was performed with plant conditions tabulated in Table 15.0-2 unless otherwise noted. Turbine stop valves full stroke closure time is 0.1 second.

A reactor scram is initiated by position switches on the stop valves when the valves are less than 90% open. This stop valve scram trip signal is automatically bypassed when the reactor is below 30% NB rated power level.

The analysis presented here is a hypothetical case with a conservative .8 inches Hg per second vacuum decay rate. Thus, the bypass system is available for several seconds since the bypass is signaled to close at a vacuum level of about 10 inches Hg less than the stop valve closure.

15.2.5.3.3 Results

Under this hypothetical .8 inches Hg per second vacuum decay condition, the turbine bypass valve and main steam isolation valve closure would follow main turbine and feedwater turbine trips about 12 seconds after they initiate the transient. This transient, therefore, is similar to a normal turbine trip with bypass. The effect of main steam isolation valve closure tends to be minimal since the closure of main turbine stop valves and subsequently the bypass valves have already shut off the main steam line flow. Figure 15.2-6 shows the transient expected for this event. It is assumed that the plant is initially operating at 105% of Nuclear Boiler rated steam flow conditions. Peak neutron flux reaches 168% of NB rated power while average fuel surface heat flux reaches 105% of rated value. Safety/relief valves open to limit the pressure rise, then sequentially reclose as the stored energy is dissipated.

15.2.5.3.4 Considerations of Uncertainties

The reduction or loss of vacuum in the main turbine condenser will sequentially trip the main and feedwater turbines and close the main steamline isolation valves and bypass valves. While these are the major events occurring, other resultant actions will include scram (from stop valve closure) and bypass opening with the main turbine trip. Because the protective actions are actuated at various levels of condenser vacuum, the severity of the resulting transient is directly dependent upon the rate at which the vacuum pressure is lost. Normal loss of vacuum due to loss of cooling water pumps or steam jet air ejector problem

produces a very slow rate of loss of vacuum (minutes, not seconds). See Table 15.2-9. If corrective actions by the reactor operators are not successful, then simultaneous trips of the main and feedwater turbines, and ultimately complete isolation by closing the bypass valves (opened with the main turbine trip) and the MSIVs, will occur.

A faster rate of loss of the condenser vacuum would reduce the anticipatory action of the scram and the overall effectiveness of the bypass valves since they would be closed more quickly.

Other uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

(1) Slowest allowable control rod scram motion is assumed.

(2) Scram worth shape for all-rod-out conditions is assumed.

(3) Minimum specified valve capacities are utilized for overpressure protection.

(4) Set points of the safety/relief valves are assumed to be at the upper limit of Technical Specifications for all valves.

15.2.5.4 Barrier Performance

Peak nuclear system pressure is 1165 psig at the vessel bottom. Clearly, the overpressure transient is below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Vessel dome pressure does not exceed 1140 psig. A comparison of these values to those for Turbine Trip with Bypass Failure, at high power shows the similarities between these two transients. The prime differences are the loss of feedwater and main steamline isolation, and the resulting low water level trips.

15.2.5.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled

release conditions. If purging of the containment is chosen, the release will have to be in accordance with established technical specifications; therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.6 LOSS OF AC POWER

15.2.6.1 Identification of Causes and Frequency Classification

15.2.6.1.1 Identification of Causes

15.2.6.1.1.1 Loss of Auxiliary Power Transformer

Causes for interruption or loss of the auxiliary power transformer power can arise from normal operation or malfunctioning of transformer protection circuitry. These can include high transformer oil temperature, reverse or high current operation as well as operator error which trips the transformer breakers.

15.2.6.1.1.2 Loss of All Grid Connections

Loss of all grid connections can result from major shifts in electrical loads, loss of loads, lightning, storms, wind, etc., which contribute to electrical grid instabilities. These instabilities will cause equipment damage if unchecked. Protective relay schemes automatically disconnect electrical sources and loads to mitigate damage and regain electrical grid stability.

15.2.6.1.2 Frequency Classification

15.2.6.1.2.1 Loss of Auxiliary Power Transformer

This transient disturbance is categorized as an incident of moderate frequency.

15.2.6.1.2.2 Loss of All Grid Connections

This transient disturbance is categorized as an incident of moderate frequency.

15.2.6.2 Sequence of Events and Systems Operation

15.2.6.2.1 Sequence of Events

15.2.6.2.1.1 Loss of Auxiliary Power Transformer

Table 15.2-12 lists the sequence of events for Figure 15.2-7.

15.2.6.2.1.2 Loss of All Grid Connections

Table 15.2-13 lists the sequence of events for Figure 15.2-8.

15.2.6.2.1.3 Identification of Operator Actions

The operator should maintain the reactor water level by use of the RCIC or HPCI system, control reactor pressure by use of the relief valves and steam condensing mode of the RHR and verify that the turbine d-c oil pump is operating satisfactorily to prevent turbine bearing damage. Also, he should verify proper switching and loading of the emergency diesel generators.

The following is the sequence of operator actions expected during the course of the events when no immediate restart is assumed. The operator should:

(1) Following the scram, verify all rods in.

(2) Check that diesel generators start and carry the vital loads.

(3) Check that relays on the reactor protection system (RPS) drop out.

(4) Check that both RCIC and HPCI start when reactor vessel level drops to the initiation point after the relief opens.

(5) Break vacuum before the loss of sealing steam cccurs.

(6) Check T-G auxiliaries during coastdown.

(7) When both the reactor pressure and level are under control, secure both HPCI and RCIC as necessary.

(8) Continue cooldown per the normal procedure

(9) Complete the scram report and survey the maintenance requirements.

15.2.6.2.2 Systems Operation

15.2.6.2.2.1 Loss of Auxiliary Power Transformer

This event, unless otherwise stated, assumes and takes credit for normal functioning of plant instrumentation and controls, plant protection and reactor protection systems.

The reactor is subjected to a complex sequence of events when the plant loses all auxiliary power. Estimates of the responses of the various reactor systems (assuming loss of the auxiliary transformer) provide the following simulation sequence:

(1) The recirculation pumps are tripped at a reference time, t=0, with normal coastdown times.

(2) At approximately 2 seconds, independent MSIV closure and scram are initiated due to loss of power to MSIV logic and actuator solenoids.

(3) At approximately 4 seconds, feedwater pump trips are initiated.

Operation of the HPCI and RCIC system functions are not simulated in this analysis. Their operation occurs at some time beyond the primary concerns of fuel thermal margin and overpressure effects of this analysis.

15.2.6.2.2.2 Loss of All Grid Connections

Same as Subsection 15.2.6.2.2.1 with the following additional concern.

The loss of all grid connections is another feasible, although improbable, way to lose all auxiliary power. This event would add a generator load rejection to the above sequence at time, t=0. The load rejection immediately forces the turbine control

valves closed, causes a scram and initiates recirculation pump trip (RPT) (already tripped at reference time t=0).

15.2.6.2.3 The Effect of Single Failures and Operator Errors

Loss of the auxiliary power transformer in general leads to a reduction in power level due to rapid pump coastdown with pressurization effects due to turbine trip occurring after the reactor scram has occurred. Additional failures of the other systems assumed to protect the reactor would not result in an effect different from those reported. Failures of the protection systems have been considered and satisfy single failure criteria and as such no change in analyzed consequences is expected. See Appendix 15A for details on single failure analysis.

15.2.6.3 Core and System Performance

15.2.6.3.1 Mathematical Model

The computer model described in Subsection 15.1.1.3.1 was used to simulate this event.

Operation of the RCIC or HPCI systems is not included in the simulation of this transient, since startup of these pumps does not permit flow in the time period of this simulation.

15.2.6.3.2 Input Parameters and Initial Conditions

15.2.6.3.2.1 Loss of Auxiliary Power Transformer

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2 and under the assumed systems constraints described in Subsection 15.2.6.2.2.

15.2.6.3.2.2 Loss of All Grid Connections

Same as Subsection 15.2.6.3.2.1

15.2.6.3.3 Results

15.2.6.3.3.1 Loss of Auxiliary Power Transformer

Figure 15.2-7 shows graphically the simulated transient. The initial portion of the transient is similar to the loss-of-feedwater transient. At 2 seconds MSIV's start to close and the reactor is scrammed. The feedwater turbines trip off at about 4 seconds.

The RHRS, in the shutdown cooling mode, is initiated to dissipate the heat. Sensed level drops to the RCIC and HPCI initiation set point at approximately 32 sec after loss of auxiliary power.

There is no significant increase in fuel temperature or decrease in the operating MCPR value, fuel thermal margins are not threatened and the design basis is satisfied.

15.2.6.3.3.2 Loss of All Grid Connections

Loss of all grid connections is a more general form of loss of auxiliary power. It essentially takes on the characteristic response of the standard full load rejection discussed in Subsection 15.2.2. Figure 15.2-8 shows graphically the simulated event. Peak neutron flux reaches 107% of NB rated power while fuel surface heat flux peaks at 100% of initial value. There is no significant increase in fuel temperature.

15.2.6.3.4 Consideration of Uncertainties

The most conservative characteristics of protection features are assumed. Any actual deviations in plant performance are expected to make the results of this event less severe.

Operation of the RCIC or HPCI systems is not included in the simulation of the first 50 seconds of this transient. Startup of these pumps occurs in the latter part of this time period but these systems have no significant effect on the results of this transient.

Following main steam line isolation and RHR initiation the reactor pressure is expected to increase until the safety/relief valve set point(s) (5) are reached. At this time the valves operate in a cyclic manner to discharge the decay heat to the suppression pool.

15.2.6.4 Barrier Performance

15.2.6.4.1 Loss of Auxiliary Power Transformer

The consequences of this event do not result in any significant temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.6.4.2 Loss of All Grid Connections

Safety/relief valves open in the pressure relief mode of operation as the pressure increases beyond their set points. The pressure in the dome is limited to a maximum value of 1140 psig, well below the vessel pressure limit of 1375 psig.

15.2.6.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.7 LOSS OF FEEDWATER FLOW

15.2.7.1 Identification of Causes and Frequency Classification

15.2.7.1.1 Identification of Causes

A loss of feedwater flow could occur from pump failures, feedwater controller failures, operator errors, or reactor system variables such as high vessel water level (L8) trip signal.

15.2.7.1.2 Frequency Classification

This transient disturbance is categorized as an incident of moderate frequency.

15.2.7.2 Sequence of Events and Systems Operation

15.2.7.2.1 Sequence of Events

Table 15.2-14 lists the sequence of events for Figure 15.2-9.

15.2.7.2.1.1 Identification of Operator Actions

The operator should verify MSIV closure, and ensure RCIC and HPCI actuation so that water inventory is maintained in the reactor vessel. Initiate the steam condensing mode of the RHR system to complement the RCIC system. Monitor reactor water level and pressure control, and T-G auxiliaries during shutdown.

The following is the sequence of operator actions expected during the course of the event when no immediate restart is assumed. The operator should:

(1) Verify all rods in, following the scram.

(2) Verify HPCI and RCIC initiation.

(3) Verify MSIV closure.

(4) Verify that the relief valves open on reactor high pressure.

(5) Verify that the recirculation pumps trip on reactor low-low level.

(6) Secure HPCI when reactor level and pressure are under control.

(7) Continue operation of the RCIC until decay heat diminishes to a point where the RHR system can be put into service.

(8) Monitor the turbine coastdown, break vacuum as necessary.

(9) Complete the scram report and survey the maintenance requirements.

15.2.7.2.2 Systems Operation

Loss of feedwater flow results in a proportional reduction of vessel inventory causing the vessel water level to drop. The first corrective action is the low level (L3) scram trip actuation. Reactor protection system responds within 1 second after this trip to scram the reactor. The low level (L3) scram trip function meets single failure criterion.

Containment isolation, when it occurs, would also initiate a main steam line isolation valve position scram trip signal as part of the normal isolation event. The reactor, however, is already scrammed and shut down by this time.

Credit is taken for operation of the safety relief valve (low set point) to remove decay heat since the bypass becomes ineffective due to main steam line isolation.

15.2.7.2.3 The Effect of Single Failures and Operator Errors

The nature of this event, as explained above, results in a lowering of vessel water level. Key corrective efforts to shut down the reactor are automatic and designed to satisfy single failure criterion; therefore, any additional failure in these shutdown methods would not aggravate or change the simulated transient. See Appendix 15A for details.

The potential exists for a single relief valve failing to close once it is opened. This would result in a complete depressurization of the reactor. This is discussed in Subsection 15.1.4. Either the HPCI or RCIC system is capable of maintaining

adequate core coverage and will provide long-term inventory control.

15.2.7.3 Core and System Performance

15.2.7.3.1 Mathematical Model

The computer model described in Subsection 15.1.1.3.1 was used to simulate this event.

15.2.7.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

15.2.7.3.3 Results

The results of this transient simulation are shown in Figure 15.2-9. Feedwater flow terminates at approximately 5 seconds. Subcooling decreases causing a reduction in core power level and pressure. As power level is lowered, the turbine steam flow starts to drop off because the pressure regulator is attempting to maintain pressure for the first 8 seconds or so. Water level continues to drop until the vessel level (L3) scram trip set point is reached whereupon the reactor is shut down. Main steam line isolation occurs at 13 seconds due to vessel water dropping to the L2 trip. Also at this time, the recirculation system is tripped and HPCI and RCIC operation is initiated. MCPR remains considerably above the safety limit since increases in heat flux are not experienced.

15.2.7.3.4 Considerations of Uncertainties

End-of-cycle scram characteristics are assumed.

This transient is most severe from high power conditions, because the rate of level decrease is greatest and the amount of stored and decay heat to be dissipated are highest.

Operation of the RCIC or HPCI systems is not included in the simulation of the first 50 seconds of this transient since startup of these pumps occurs in the latter part of this time period and therefore these systems have no significant effects on

the results of this transient except perhaps as discussed in Subsection 15.2.7.2.3.

15.2.7.4 Barrier Performance

Peak pressure in the bottom of the vessel reaches 1105 psig, which is below the ASME Code limit of 1375 psig for the RCPB. Vessel dome pressure does not exceed 1094 psig. The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.7.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with established technical specifications; therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.8 FEEDWATER LINE BREAK

(Refer to Subsection 15.6.6)

15.2.9 FAILURE OF RHR SHUTDOWN COOLING

Normally, in evaluating component failure considerations associated with the RHRS - Shutdown Cooling mode operation, active pumps or instrumentation (all of which are redundant for safety system portions of the RHRS aspects) would be assumed to be the likely failed equipment. For purposes of worst case analysis, the single recirculation loop suction valve to the redundant RHRS loops is assumed to fail. This failure would, of course, still leave two complete RHRS loops for LPCI, suppression pool, and containment cooling minus the normal RHRS - Shutdown

Cooling loop connection. Although the suction valve could be manually manipulated open, it is assumed failed indefinitely. If it is now assumed that the SACF criteria is applied, the plant operator has one complete RHRS loop available with the further selective worst case assumption that the other RHRS loop is lost.

Recent analytical evaluations of this event have required additional worst case assumptions. These included:

(1) loss of all offsite ac power

(2) utilization of safety shutdown equipment only

(3) operator involvement no earlier than 10 minutes after coincident assumptions.

These accident-type assumptions would change the initial incident (malfunction of RHRS suction valve) from a moderate frequency incident to a classification in the design basis accident status. However, the event is evaluated as a moderate frequency event with its subsequent limits.

15.2.9.1 Identification of Causes and Frequency Classification

15.2.9.1.1 Identification of Causes

The plant is operating at 105% NBR rated steam flow when a long-term loss of offsite power occurs, causing multiple safety-relief valve actuation (see Subsection 15.2.6) and subsequent heatup of the suppression pool. Reactor vessel depressurization is initiated to bring the reactor pressure to approximately 100 psig. Concurrent with the loss of offsite power, an additional (divisional) single failure occurs which prevents the operator from establishing the normal shutdown cooling path through the RHR shutdown cooling lines. He then establishes a shutdown cooling path for the vessel through the ADS valves.

15.2.9.1.2 Frequency Classification

This event is evaluated as a moderate frequency event. However, for the following reasons it could be considered an infrequent incident:

(1) No RHR valves have failed in the shutdown cooling mode in BWR total operating experience.

(2) The set of conditions evaluated is for multiple failure as described above and is only postulated (not expected) to occur.

15.2.9.2 Sequence of Events and System Operation

15.2.9.2.1 Sequence of Events

The sequence of events for this event is shown in Table 15.2-15.

15.2.9.2.1.1 Identification of Operator Actions

For the early part of the transient, the operator actions are identical to those described in Subsection 15.2.6 resulting in an isolation. The operator then proceeds to do the following:

(1) at approximately 10 minutes into the transient, initiate RPV shutdown depressurization at ~100°F/hr by manual actuation of the safety/relief valves;

(2) at approximately 15 minutes into the transient, initiate suppression pool cooling (again for purposes of this analysis, "worst case," it is assumed that only one RHR heat exchanger is available);

(3) when the reactor pressure vessel is depressurized to approximately 100 psig, opens the RHR shutdown cooling system isolation valves. However, it is assumed that a failure occurs and the operator cannot open one of the isolation valves on the RHR suction line and the normal RHR shutdown cooling path is not established;

(4) selectively opens safety/relief valves (ADS) to complete blowdown and floods the vessel up through the safety/relief valves thereby establishing a closed cooling path as described in the notes for Figure 15.2-11.

15.2.9.2.2 System Operation

Plant instrumentation and control is assumed to be functioning normally except as noted. In this evaluation credit is taken for the plant and reactor protection systems and/or the ESF utilized.

15.2.9.2.3 The Effect of Single Failures and Operator Errors

The worst case single failure (Loss of Division Power) has already been analyzed in this event. Therefore, no single failure or operator error can make the consequences of this event any worse. See Appendix 15A for further discussion.

15.2.9.3 Core and System Performance

15.2.9.3.1 Methods, Assumptions, and Conditions

An event that can directly cause reactor vessel water temperature increase is one in which the energy removal rate is less than the decay heat rate. The applicable event is loss of RHR shutdown cooling. This event can occur only during the low pressure portion of a normal reactor shutdown and cooldown, when the RHR system is operating in the shutdown cooling mode. During this time MCPR remains high and nucleate boiling heat transfer is not exceeded at any time. Therefore, the core thermal safety margin remains essentially unchanged. The 10-minute time period assumed for operator action is an estimate of how long it would take the operator to initiate the necessary actions; it is not a time by which he must initiate action. 1 19 19

15.2.9.3.2 Results

For most single failures that could result in loss of shutdown cooling, no unique safety actions are required. In these cases, shutdown cooling is simply re-established using other, normal shutdown cooling equipment. In cases where both of the RHRS shutdown cooling suction valves cannot be opened, alternate paths are available to accomplish the shutdown cooling function (Figure 15.2-10). An evaluation has been performed assuming the worst single failure that could disable the RHRS shutdown cooling valves.

This evaluation demonstrates the capability to safely transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. The evaluation assures that, for

onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the safety function can be accomplished, assuming a worst-case single failure.

The alternate cooldown path chosen to accomplish the shutdown cooling function utilizes the RHR and ADS or normal relief valve systems (see Reference 15.2-1 and Figure 15.2-11).

The alternate shutdown systems are capable of performing the function of transferring heat from the reactor to the environment
1| using only safety grade systems. Even if it is additionally postulated that all of the ADS or relief valve discharge piping also fails, the shutdown cooling function would eventually be accomplished as the cooling water would run directly out of the ADS or safety/relief valves, flooding into the drywell.

The systems have suitable redundancy in components such that, for onsite electrical power operation (assuming offsite power is not available) and for offsite electrical power operation (assuming onsite power is also not available), the systems' safety function
1| can be accomplished assuming an additional single failure. The systems can be fully operated from the main control room.

The design evaluation is divided into two phases: (1) full power operation to approximately 100 psig vessel pressure, and (2) approximately 100 psig vessel pressure to cold shutdown (14.7
1| psia, 200°F) conditions.

19| 15.2.9.3.2.1 Full Power to Approximately 100 psig

Independent of the event that initiated plant shutdown (whether it be a normal plant shutdown or a forced plant shutdown), the reactor is normally brought to approximately 100 psig using either the main condenser or, in the case where the main
1| condenser is unavailable, the RCIC/HPCI systems, together with the nuclear boiler pressure relief system.

For evaluation purposes, however, it is assumed that plant
1| shutdown is initiated by transient event 15.2.6, which results in relief valve actuation and subsequent suppression pool heatup. For this postulated condition, the reactor is shut down and the reactor vessel pressure and temperature are reduced to and maintained at saturated conditions at approximately 100 psig. The reactor vessel is depressurized by manually opening selected
1| safety/relief valves. Reactor vessel makeup water is automatically provided

via the RCIC/HPCI systems. While in this condition, the RHR system (suppression pool cooling mode) is used to maintain the suppression pool temperature within shutdown limits.

These systems are designed to routinely perform their functions for both normal and forced plant shutdown. Since the RCIC, HPCI and RHR systems are divisionally separated, no single failure, together with the loss of offsite power, is capable of preventing reaching the 100 psig level.

15.2.9.3.2.2 Approximately 100 psig to Cold Shutdown |19

The following assumptions are used for the analyses of the procedures for attaining cold shutdown from a pressure of approximately 100 psig:

(1) the vessel is at 100 psig and saturated conditions;

(2) a worst-case single failure is assumed to occur (i.e., loss of a division of emergency power); and

(3) there is no offsite power available.

In the event that the RHR shutdown suction line is not available because of single failure, the first action to be taken will be for personnel to gain access and effect repairs. For example, if a single electrical failure caused a suction valve to fail in the closed position, a hand wheel is provided on the valve to allow manual operation. If for some reason the normal shutdown cooling suction line cannot be repaired, the capabilities described below will satisfy the normal shutdown cooling requirements and thus fully comply with GDC 34. |1 |1

The RHR shutdown cooling line valves are in two divisions (Division 1 = the outboard valve, and Division 2 = the inboard valve) to satisfy containment isolation criteria. For evaluation purposes, the worst-case failure is assumed to be the loss of a division of emergency power, since this also prevents actuation of one shutdown cooling line valve. Engineered safety feature equipment available for accomplishing the shutdown cooling function includes (for the selected path):

ADS (DC Division 1 and DC Division 2)

RHR Loop A (Division 1)

RHR Loop B (Division 2)

HPCI (DC Division 2)

RCIC (DC Division 1)

Core Spray A (Division 1)

Core Spray B (Division 2)

For failures of Division 1 or 2, the following systems are assumed functional:

(1) Division 1 Fails, Division 2 Functional:

Failed Systems	Functional Systems
RHR Pumps A & C	HPCI
CS Loop A	ADS
RCIC	RHR Loop B
	CS Loop B
	RHR Pumps B & D

(2) Division 2 Fails, Division 1 Functional:

Failed Systems	Functional Systems
RHR Pumps B & D	CS Loop A
CS Loop B	RCIC
HPCI	RHR Loop A
	ADS
	RHR Pumps A & C

Assuming the single failure is the failure of Division 2, the safety function is accomplished by establishing one of the cooling loops described in Activity C2 of Figure 15.2-11. If the assumed single failure is Division 1, the safety function is accomplished by establishing one of the cooling loops described as Activity C1 of Figure 15.2-11.

Using the above assumptions and following the depressurization transient shown in Figure 15.2-12, the suppression pool temperature is shown in Figure 15.2-13.

15.2.9.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed. Release of coolant to the containment occurs via SRV actuation. Release of radiation to the environment is described below.

15.2.9.5 Radiological Consequences

While this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposures to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.2.10 REFERENCES

15.2-1 Letter - R.S. Boyd to I. F. Stuart; dated November 12, 1975, Subject: Requirements Delineated for RHRS - Shutdown Cooling System--Single Failure Analysis.

15.2-2 Fukushima, T. Y, "Hex 01 User Manual", NEDE-23014, July 1976.

15.2-3 Brutschy, F.G., et al, "Behavior of Iodine in Reactor Water During Plant Shutdown and Startup".

15.2-4 Nguyen, D., "Realistic Accident Analysis for General Electric Boiling Water Reactor - The RELAP Code and User's Guide," NEDO-21142, to be issued (December 1977).

TABLE 15.2-10

SEQUENCE OF EVENTS FOR FIGURE 15.2-6

Time-sec	Event
-0.0 (est)	Initiate simulated loss of condenser vacuum at 2 inches of Hg per second.
0.0 (est)	Low condenser vacuum main turbine trip actuated.
0.0 (est)	Low condenser vacuum feedwater trip actuated.
0.01 (est)	Main turbine trip initiates reactor scram.
0.01 (est)	Main turbine trip initiates recirculation pump trip (RPT)
.1 (est)	Turbine stop valve closes
1.7	Group 1 relief valves set points actuated.
1.9	Group 2 relief valves set points actuated.
2.2	Group 3 relief valves set points actuated.
2.4	Group 4 relief valves set points actuated.
2.6	Group 5 relief valves set points actuated.
12.1	Low condenser vacuum initiates main steam line isolation valve closure.
12.1	Low condenser vacuum initiates bypass valve closure.
21.5	Group 1 relief valves close.
23.5	L2 Vessel level set point isolation.
53 (est)	HPCI/RCIC system flow enters vessel (not included in simulation).
90+	Relief valves cycle as required on pressure.

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15.4.1 Rod Withdrawal Error - Low Power

15.4.1.1 Control Rod Removal Error During Refueling

15.4.1.1.1 Identification of Causes and Frequency Classification

The event considered here is inadvertent criticality due to the complete withdrawal or removal of the most reactive rod during refueling. The probability of the initial causes alone is considered low enough to warrant its being categorized as an infrequent incident, since there is no postulated set of circumstances which results in an inadvertent RWE while in the REFUEL mode.

15.4.1.1.2 Sequence of Events and Systems Operation

15.4.1.1.2.1 Initial Control Rod Removal

During refueling operations safety system interlocks provide assurance that inadvertant criticality does not occur because a control rod has been removed or is withdrawn in coincidence with another control rod.

15.4.1.1.2.2 Fuel Movement With Control Rod Removed

Fuel movement and other core alterations with control rods removed will be controlled by section 3/4.9 of the Technical Specifications. These requirements along with the associated refueling interlocks sufficiently minimize the possibility of loading fuel into a cell containing no control rod, moving the refueling platform over the core and withdrawing additional control rods when there is uncontrolled fuel in the core.

15.4.1.1.2.3 Control Rod Removal Without Fuel Removal

Finally, the design of the control rod, incorporating the velocity limiter, does not physically permit the upward removal of the control rod without the simultaneous or prior removal of the four adjacent fuel bundles. This precludes any hazardous condition.

15.4.1.1.2.4 Identification of Operator Actions

No operator actions are required to preclude this event since the plant design as discussed above prevents its occurrence.

15.4.1.1.2.5 Effect of Single Failure and Operator Errors

If any one of the operations involved in initial failure or error is followed by any other SEF or SOE, the necessary safety actions are taken (e.g., rod block or scram) automatically prior to limit violation. Refer to Appendix 15A for details.

15.4.1.1.3 Core and System Performances

Since the probability of inadvertent criticality during refueling is precluded, the core and system performances were not analyzed. However, it is well known that withdrawal of the highest worth control rod during refueling results in a positive reactivity insertion but not enough to cause criticality. This is verified experimentally by performing shutdown margin checks. (See Subsection 4.3.2 for a description of the methods and results of the shutdown margin analysis.) Additional reactivity insertion is precluded by interlocks. (See Subsection 7.6.1a.1) As a result, no radioactive material is ever released from the fuel, making it unnecessary to assess any radiological consequences.

No mathematical models are involved in this event. The need for input parameters or initial conditions is not required as there are no results to report. Consideration of uncertainties is not appropriate.

15.4.1.1.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since it is a highly localized event and does not result in any change in the core pressure or temperature.

15.4.1.1.5 Radiological Consequences

An evaluation of the radiological consequences was not made for this event since no radioactive material is released from the fuel.

15.4.1.2 Continuous Rod Withdrawal During Reactor Startup

15.4.1.2.1 Identification of Causes and Frequency Classification

The probability of initial causes or errors of this event alone is considered low enough to warrant its being categorized as an infrequent incident. The probability of further development of this event is extremely low because it is contingent upon the simultaneous failure of two redundant systems, the RSCS and the RWM systems, concurrent with a high worth rod, out-of-sequence rod selection contrary to procedures, plus operator ignorance of continuous alarm annunciations prior to safety system actuation.

15.4.1.2.2 Sequence of Events and Systems Operation

15.4.1.2.2.1 Sequence of Events

Control rod withdrawal errors are not considered credible in the startup and low power ranges. The RSCS and RWM prevent the operator from selecting and withdrawing an out-of-sequence control rod.

Continuous control rod withdrawal errors during reactor startup are precluded by the RSCS. The RSCS prevents the withdrawal of

an out-of-sequence control rod in the 100% to 75% control rod density range and limits rod movement to the banked position mode of rod withdrawal from the 75% rod density to the preset power level. Since only in-sequence control rods can be withdrawn in the 100% to 75% control rod density and control rods are withdrawn in the banked position mode from the 75% control rod density point to the preset power level, there is no basis for the continuous control rod withdrawal error in the startup and low power range. The low power range is defined as zero power to the RSCS low power setpoint, i.e., 20% of rated core power. For RWE above low power setpoint see Subsection 15.4.2. The banked position mode of the RSCS is described in Reference 15.4-2.

15.4.1.2.2.2 Identification of Operator Actions

No operator actions are required to preclude this event since the plant design as discussed above prevents its occurrence.

15.4.1.2.2.3 Effects of Single Failure and Operator Errors

If any one of the operations involved the initial failure or error and is followed by another SEF or SOE, the necessary safety actions are taken (e.g., rod blocks) prior to any limit violation. Refer to Appendix 15A for details.

15.4.1.2.3 Core and System Performance

The performance of the RSCS and RWM prevent erroneous selection and withdrawal of an out-of-sequence control rod. The core and system performance is not affected by such an operator error.

No mathematical models are involved in this event. The need for input parameters or initial conditions is not required as there are no results to report. Consideration of uncertainties is not appropriate.

15.4.1.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since there is no postulated set of circumstances for which this error could occur.

15.4.1.2.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

15.4.2 Rod Withdrawal Error - at Power

15.4.2.1 Identification of Causes and Frequency Classifications

15.4.2.1.1 Identification of Causes

While operating in the power range in a normal mode of operation the reactor operator makes a procedural error and withdraws the maximum worth control rod until the Rod Block Monitor (RBM) System inhibits further withdrawal.

15.4.2.1.2 Frequency Classification

The probability of this event is considered low enough to warrant its being categorized as an infrequent incident. However, because of the lack of sufficient frequency data base, this transient disturbance is analyzed as an incident of moderate frequency until the frequency of this event can be further evaluated and justified.

15.4.2.2 Sequence of Events and Systems Operation

15.4.2.2.1 Sequence of Events

The sequence of events for this transient, as calculated with conservative assumptions, is presented in Table 15.4-1. No operator actions are required during this event. However, operator actions expected to occur are shown in the above referenced table.

15.4.2.2.2 System Operations

The focal point of this event is localized to a small portion of the core. Therefore, although reactor control and instrumentation is assumed to function normally, credit is taken only for RBM system. A discussion of the event follows below.

While operating in the power range in a normal mode (except as noted in Subsection 15.4.2.3.2) of operation, the reactor operator makes a procedural error and withdraws the maximum worth control rod until the RBM system inhibits further withdrawal.

Under most normal operating conditions no operator action is required since the transient which would occur would be very mild. Should the peak linear power design limits be exceeded, the nearest local power range monitor (LPRM) would detect this phenomenon and sound an alarm. The operator must acknowledge this alarm and take appropriate action to rectify the situation.

If the rod withdrawal error is severe enough, the rod block monitor (RBM) system would sound alarms, at which time the operator would acknowledge the alarm and take corrective action. Even for extremely severe conditions (i.e., for highly abnormal control rod patterns, operating conditions, and assuming that the operator ignores all alarms and warnings and continues to withdraw the control rod), the RBM system will block further withdrawal of the control rod before the fuel reaches the point of boiling transition or the 1% plastic strain limit imposed on the clad.

15.4.2.2.3 Effect of Single Failure and Operator Errors

The effect of operator errors has been discussed above. It was shown that operator errors (which initiated this transient) cannot impact the consequences of this event due to the RBM system. The RBM system is designed to be single failure proof, therefore termination of this transient is assured. See Appendix 15A for details.

15.4.2.3 Core and System Performance

15.4.2.3.1 Mathematical Model

For this transient the reactivity insertion rate is very slow; therefore, it is adequate to assume that the core has sufficient time to equilibriate (i.e., that both the neutron flux and heat flux are in phase). Making use of the above assumption, this transient is calculated using a steadystate three-dimensional coupled nuclear-thermal-hydraulics computer program. The program is described in detail in Reference 4.3-2 of Section 4.3. All spatial effects are included in the calculation.

The primary output from this code, in addition to the basic nuclear parameters, is: the variation of the linear heat generator rate (LHGR); the variation of the minimum critical power ratio (MCPR); the total reactor power; and the variation of the in-core instruments during the transient. A detector response code used the instrument responses to predict the Rod Block Monitor action under the specified condition for the Rod Withdrawal Error.

The analytical methods and assumptions which are used in evaluating the consequences of this accident are considered to provide a realistic, yet conservative assessment of the consequences.

15.4.2.3.2 Input Parameters and Initial Conditions

The number of possible RWE transients is extremely large due to the number of control rods and the wide range of exposures and power levels. In order to encompass all of the possible RWEs which could conceivably occur, a limiting analysis is defined such that a conservative assessment of the consequences is provided.

The conservative assumptions are:

(1) The assumed error is a continuous withdrawal of the maximum worth rod at its maximum drive speed.

(2) The core is assumed to be operating at rated conditions.

(3) The reactor is presumed to be in its most reactive state and devoid of all xenon. This insures that the amount of excess reactivity which must be controlled by the movable control rods is maximum.

(4) Furthermore, it is assumed that the operator has fully inserted the maximum worth rod prior to its removal and selected the remaining control rod pattern in such a way as to approach thermal limits in the fuel bundles in the vicinity of the rod to be withdrawn. (See Figure 15.4-1). It should be emphasized that this control rod configuration would be highly abnormal and could only be achieved by deliberate operator action or by numerous operator errors.

(5) The operator is assumed to ignore all warnings during the transient.

(6) Of the four LPRM strings nearest to the control rod being withdrawn, the two highest reading LPRMs during the transient are assumed to have failed.

(7) One of the two instrument channels is assumed to be bypassed and out-of-service. The A and C LPRM chambers input to one channel while the B and D chambers input to the other. The channel with the greatest response is assumed to be bypassed.

The conservative assumptions indicated above provides a high degree of assurance that the transient as analyzed bounds all RWE which could possibly occur. Table 15.4-2 presents the other parameters used in the analysis of this event.

15.4.2.3.2.1 RBM System Operation

The RBM system minimizes the consequences of a RWE by blocking the motion of the control rod before the safety limits are exceeded.

The RBM has three trip levels (rod withdrawal permissive removed). The trip levels may be adjusted and are nominally 8% of reactor power apart. The highest trip level is set so that the safety limit is not exceeded. The lower two trip levels are intended to provide a warning to the operator. Settings are 107%, 99% and 91% of initial, steady-state, operating power at 100% flow. The trip levels are automatically varied with reactor coolant flow to protect against fuel damage at lower flows. The variation is set to assure that no fuel damage will occur at any indicated coolant flow. The operator may encounter any number (up to three) of trip points depending on the starting power of a given control rod withdrawal. The lower two points may be passed up (reset) by manual operation of a push button. The reset permissive is actuated (and indicated by a light) when the RMB reaches 2% power less than the trip point. The operator should

then assess his local power and either reset or select a new rod. The highest (power) trip point may not be reset.

15.4.2.3.3 Results

The consequences of this transient are relatively mild, and neither localized nor gross occurrence of boiling transition or violation of 1% plastic strain limit on the cladding occur. The variation in the MCPR and MLHGR, as a function of withdrawal of the highest worth rod, is presented in Figures 15.4-2 and 15.4-3, respectively. The bundles presented in Figures 15.4-2 and 15.4-3 represent the envelope of the MCPR and the MLHGR for each two-foot interval during the transient. Variation in the total reactor power is also shown in these figures. Although these figures show the change in thermal limits from the fully inserted to the fully withdrawn position, the control rod is automatically blocked at 5.0 feet, even under the worst set of assumptions. The variation in the signal response of the two independent channels is shown in Figures 15.4-4 and 15.4-5. With a setpoint of 106% the rod is shown to block at 5.0 feet resulting in a Δ MCPR of 0.183 and MLHGR of 14.84 kw/ft.

15.4.2.3.4 Considerations of Uncertainties

The conservative assumptions which assure that this event has been conservatively analyzed have been previously discussed in Subsection 15.4.2.3.2.

15.4.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since this is a localized event with very little change in the gross core characteristics. Typically, an increase in total core power is less than 5% and the changes in pressure are negligible.

15.4.2.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

15.4.3 Control Rod Maloperation (System Malfunction or Operator Error)

This event is covered with evaluation cited in Subsections 15.4.1 and 15.4.2.

15.4.4 Abnormal Startup of Idle Recirculation Pump

15.4.4.1 Identification of Causes and Frequency Classification

15.4.4.1.1 Identification of Causes

This action results directly from the operator's manual action to initiate pump operation. It assumes that the remaining loop is already operating.

15.4.4.1.1.1 Normal Restart of Recirculation Pump at Power

This transient is categorized as an incident of moderate frequency.

15.4.4.1.1.2 Abnormal Startup of Idle Recirculation Pump

This transient is categorized as an incident of moderate frequency.

15.4.4.2 Sequence of Events and Systems Operation

15.4.4.2.1 Sequence of Events

Table 15.4-3 lists the sequence of events for Figure 15.4-6.

15.4.4.2.1.1 Operator Actions

The normal sequence of operator actions expected in starting the idle loop is as follows. The operator should:

(1) Adjust rod pattern as necessary for new power level following idle loop start.

(2) Determine that the idle recirculation pump suction valve is open, the discharge block valves are closed, and the coupler in the idle loop is in the starting position, if not, place them in this configuration.

(3) Readjust flow of the running loop downward to less than half of rated flow.

(4) Determine that the temperature difference between the two loops is no more than 50°F apart.

(5) Start the idle loop pump and adjust flow to match the adjacent loop flow. Monitor reactor power.

(6) Open discharge valve by jogging manual or auto circuitry.

(7) Readjust power, as necessary, to satisfy plant requirements per standard procedure.

Note: The time to do above work is approximately 1/2 hour.

15.4.4.2.2 Systems Operation

This event assumes and takes credit for normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. In particular, credit is taken for high flux scram to terminate the transient. No ESF action occurs as a result of the transient.

15.4.4.2.3 The Effect of Single Failures and Operator Errors

This transient leads to a quick rise in reactor power level. Corrective action first occurs in the high flux trip and being part of the reactor protection system, it is designed to single failure criteria. Therefore, shutdown is assured. Operator errors are not of concern here in view of the fact that automatic shutdown events follow so quickly after the postulated failure. See Appendix 15A for details.

15.4.4.3 Core and System Performance

15.4.4.3.1 Mathematical Model

The nonlinear dynamic model described briefly in Subsection 15.1.1.3.1 is used to simulate this event.

15.4.4.3.2 Input Parameters and Initial Conditions

This analysis has been performed unless otherwise noted with plant conditions tabulated in Table 15.0-2.

One recirculation loop is idle and filled with cold water (100°F). Normal procedure when starting an idle loop with one pump already running requires heating the idle recirculation loop to within 50°F of core inlet temperature prior to loop startup.

The active recirculation loop is operating with about 50% of normal rated diffuser flow going across the active jet pumps.

The core is receiving 38% of its normal rated flow. The remainder of the coolant flows in the reverse direction through the inactive jet pumps.

Reactor power is 55% of NBR power conditions. Normal procedures require startup of an idle loop at a lower power.

The idle recirculation pump suction valve is open, but the pump discharge valve is closed.

The idle pump fluid coupler is at a setting which approximates 50% generator speed demand.

15.4.4.3.3 Results

The transient response to the incorrect startup of a cold, idle recirculation loop is shown in Figure 15.4-6. Shortly after the pump begins to move, a surge in flow from the standard jet pump diffusers causes the core inlet flow to rise sharply.

A short-duration neutron flux peak reaches the flow referenced APRM flux set point at 10 seconds and reactor scram is initiated. The neutron flux peaks at 323% of NB rated. Surface heat flux follows the slower response of the fuel and peaks at 135% of NB of initial value. Nuclear system pressures do not increase significantly

above initial. The water level does not reach the high set point.

15.4.4.3.4 Consideration of Uncertainties

This particular transient is analyzed for an initial power level that is much higher than that expected for the actual event. The much slower thermal response of the fuel mitigates the effects of the rather sharp neutron flux spike and even in this high range of power, no threat to thermal limits is possible.

15.4.4.4 Barrier Performance

No evaluation of barrier performance is required for this event since no significant pressure increases are incurred during this transient. See Figure 15.4-6.

15.4.4.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel. Figure 15.4.4-1 Abnormal Startup of Idle Recirculation Loop Pump.

15.4.5 Recirculation Flow Control Failure with Increasing Flow

15.4.5.1 Identification of Causes and Frequency Classification

15.4.5.1.1 Identification of Causes

Failure of the master flow controller can cause a speed increase for both recirculation pumps. However, both individual speed controllers have error limiters so that this case is less severe than the failure (maximum demand) of one of the M/G set speed controllers. A rapid swing of the coupler is simulated at its maximum rate of 25%/sec.

15.4.5.1.2 Frequency Classification

This transient disturbance is classified as an incident of moderate frequency.

15.4.5.2 Sequence of Events and Systems Operation

15.4.5.2.1 Sequence of Events

Table 15.4-4 lists the sequence of events for Figure 15.4-7.

15.4.5.2.1.1 Identification of Operator Actions

Initial action by the operator will include:

(1) Transfers flow control to manual and reduces flow to minimum.

(2) Identify cause of failure.

Reactor pressure will be controlled as required, depending on whether a restart or cooldown is planned. In general, the corrective action would be to hold reactor pressure and condenser vacuum for restart after the malfunctioning flow controller has been repaired. The following is the sequence of operator actions expected during the course of the event, assuming restart. The operator should

(1) Observe that all rods are in.

(2) Check the reactor water level and maintain above low level (L2) trip to prevent MSIVs from isolating.

(3) Switch the reactor mode switch to the "startup" position.

(4) Continue to maintain vacuum and turbine seals.

(5) Transfer the recirculation flow controller to the manual position and reduce set point to zero.

(6) Survey maintenance requirements and complete the scram report.

(7) Monitor the turbine coastdown and auxiliary systems.

(8) Establish a restart of the reactor per the normal procedure.

NOTE: Time required from first trouble alarm to restart would be approximately 1 hour.

15.4.5.2.2 Systems Operation

The analysis of this transient assumes and takes credit for normal functioning of plant instrumentation and controls, and the reactor protection system. Operation of engineered safeguards is not expected.

15.4.5.2.3 The Effect of Single Failures and Operator Errors

This transient leads to a quick rise in reactor power level. Corrective action first occurs in the high flux trip and being part of the reactor protection system it is designed to single failure criteria. Therefore, shutdown is assured. (See Appendix 15A for details.) Operator errors are not of concern here in view of the fact that automatic shutdown events follow so quickly after the postulated failure.

15.4.5.3 Core and System Performance

15.4.5.3.1 Mathematical Model

The nonlinear dynamic model described briefly in Subsection 15.1.1.3.1 is used to simulate this event.

15.4.5.3.2 Input Parameters and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2. For this event the most severe transient results when initial conditions are established for operation at the low end of the rated flow control rod line. Specifically, this is 65% NB rated power and 50% core flow.

Maximum change in speed control occurs with failure of one of the motorgenerator set speed controllers. A rapid swing of the coupler is simulated at its maximum rate of 25% per second.

15.4.5.3.3 Results

Figure 15.4-7 shows the results of the transient. The changes in nuclear system pressure are not significant with regard to overpressure. Pressure decreases over most of the transient. The rapid increase in core coolant flow causes an increase in neutron flux, which initiates a reactor APRM high flux scram.

The peak neutron flux rise reaches 265% of NBR flux, and the accompanying transient fuel surface heat flux reaches 130% of initial. The MCPR remains above the safety limit of 1.06, and fuel center temperature increases only 407°F. Reactor pressure is discussed in Subsection 15.4.5.4. Therefore, the design basis is satisfied.

15.4.5.3.4 Considerations of Uncertainties

Some uncertainties in void reactivity characteristics, scram time and worth are expected to be more optimistic and will therefore lead to reducing the actual severity over that which is simulated herein.

15.4.5.4 Barrier Performance

This transient results in a very slight increase in reactor vessel pressure as shown in Figure 15.4-7 and therefore represents no threat to the RCPB.

15.4.5.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

15.4.6 Chemical and Volume Control System Malfunctions

Not applicable to BWRs.

15.4.7 Misplaced Bundle Accident

15.4.7.1 Identification of Causes and Frequency Classification

15.4.7.1.1 Identification of Causes

The event discussed in this section is the improper loading of a fuel bundle and subsequent operation of the core. Three errors must occur for this event to take place in the initial core loading. First, a bundle must be misloaded into a wrong position in the core. Second, the bundle which was supposed to be loaded where the mislocation occurred would have to be overlooked and also put in an incorrect location. Third, the misplaced bundles would have to be overlooked during the core verification performed following initial core loading.

15.4.7.1.2 Frequency of Occurrence

This event occurs when a fuel bundle is loaded into the wrong location in the core. It is assumed the bundle is misplaced to the worst possible location, and the plant is operated with the mislocated bundle. This event is categorized as an infrequent incident based on the following data.

Expected Frequency: .004 events/operating cycle

The above number is based upon past experience. The only misloading events that have occurred in the past were in reload cores where only two errors are necessary. Therefore, the frequency of occurrence for initial cores is even lower since three errors must occur concurrently.

15.4.7.2 Sequence of Events and Systems Operation

The postulated sequence of events for the misplaced bundle accident (MBA) is presented in Table 15.4-5.

Fuel loading errors, undetected by in-core instrumentation following fueling operations, may result in undetected reductions in thermal margins during power operations. No detection is assumed, and therefore, no corrective operator action or automatic protection system functioning occurs.

15.4.7.2.1 Effect of Single Failure and Operator Errors

This analysis already represents the worst case (i.e., operation of a misplaced bundle with three SEF or SOE) and there are no further operator errors which can make the event results any worse. It is felt that this section is not applicable to this event. Refer to Appendix 15A for further details.

15.4.7.3 Core and System Performance

15.4.7.3.1 Mathematical Model

A three-dimensional BWR simulator model is used to calculate the core performance resulting from this event. This model is described in detail in Reference 4.3-2 of Section 4.3.

15.4.7.3.2 Input Parameters and Initial Conditions

The initial core consists of bundles with average enrichments that are high, medium, or low with correspondingly different gadolinia-concentrations. The fuel bundle loading error with the most severe consequences occurs at BOC when a low-enriched bundle (which should be loaded at the periphery) is interchanged with a high-enriched bundle located adjacent to a LPRM and predicted to

have the highest LHGR and/or lowest CPR in the core. After the loading error is made and has gone undetected, it is assumed for purposes of conservatism that the operator uses a control pattern which places the limiting bundle in the four bundle array containing the misplaced bundle on design thermal limits, as recorded by LPRM.

As a result of loading the low-enriched bundle in an improper location, the reading of the adjacent LPRM decreases. Consequently, because there are no instruments in the 3 mirror images of this four-bundle-array, the operator believes these arrays are operating at the same power as the instrumented one, when in fact they are not (since no loading error occurred in these quadrants). As a result of placing the instrumented array on limits, the 3 mirror-image arrays exceed the design limit. By replacing the high-enriched bundle with the greatest power peaking by the low-enriched bundle, it is assured that the difference in power peaking between the instrumented and the non-instrumented arrays is maximum, or rather, that the MCPR and MLHGR is the upper bound for this error.

Other input parameters assumed are given in Table 15.4-6 and Figure 15.4-8.

15.4.7.3.3 Results

Results of analyzing the worst fuel bundle loading error are reported in Table 15.4-7. As can be seen, MCPR remains well above the point where boiling transition would be expected to occur, and the MLHGR does not exceed the 1% plastic strain limit for the clad. Therefore, no fuel damage occurs as a result of this event.

15.4.7.3.4 Considerations of Uncertainties

In order to assure the conservatism of this analysis, major input parameters are taken as a worst case, i.e., the bundle is placed in location with the highest LHGR and/or the lowest CPR in the core and the bundle is operating on design thermal limits. This assures that the Δ CPR and the Δ LHGR are the upper bounds for the error.

15.4.7.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since it is a very mild and highly localized event. No perceptable change in the core pressure would be observed.

15.4.7.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

15.4.8 Spectrum of Rod Ejection Assemblies

Not applicable to BWRs.

The BWR has precluded this event by incorporating into its design mechanical equipment which restricts any movement of the control rod drive system assemblies. The control rod drive housing support assemblies are described in Chapter 4.

15.4.9 Control Rod Drop Accident (CRDA)

15.4.9.1 Identification of Causes and Frequency Classification

15.4.9.1.1 Identification of Causes

The control rod drop accident is the result of a postulated event in which a high worth control rod is inserted out-of-sequence into the core. Subsequently, it becomes decoupled from its drive mechanism. The mechanism is withdrawn but the decoupled control rod is assumed to be stuck in place. At a later optimum moment, the control rod suddenly falls free and drops out of the core. This results in the removal of large negative reactivity from the core and results in a localized power excursion.

A more detailed discussion is given in Reference 15.4-1.

15.4.9.1.2 Frequency of Classification

The CRDA is categorized as a limiting fault because it is not expected to occur during the lifetime of the plant; but, if postulated to occur, it has consequences that include the potential for the release of radioactive material from the fuel.

15.4.9.2 Sequence of Events and System Operation

15.4.9.2.1 Sequence of Events

Before the control rod drop accident (CRDA) is possible, the sequence of events presented in Table 15.4-8 must occur. No operator actions are required to terminate this transient.

15.4.9.2.2 Systems Operation

The unlikely set of circumstances, referenced above, makes possible the rapid removal of a control rod. The dropping of the rod results in high reactivity in a small region of the core. For large, loosely coupled cores, this would result in a highly peaked power distribution and subsequent operation of shutdown mechanisms. Significant shifts in the spatial power generation would occur during the course of the excursion.

The Rod Sequence Control System (RSCS) limits the worth of any control rod which could be dropped by regulating the withdrawal sequence. This system prevents the movement of an out-of-sequence rod in the 100 to 75% rod density range, and from the 75% rod density point to the preset power level the RSCS will only allow banked position mode rod withdrawals or insertions. This system is described in Reference 15.4-2 for a typical BWR.

The RSCS is assumed to operate throughout the event. The RWM would provide the same protection as the RSCS if the RSCS was not functioning and the RWM was.

The termination of this excursion is accomplished by automatic safety features of inherent shutdown mechanisms. Therefore, no operator action during the excursion is required. Although other normal plant instrumentation and controls are assumed to function, no credit for their operation is taken in the analysis of this event.

15.4.9.2.3 Effect of Single Failures and Operator Errors

Systems mitigating the consequences of this event are RSCS (or RWM) and APRM scram. The RSCS and RWM are designed as a redundant system network and therefore together provide single failure protection. The APRM scram system is designed to single failure criteria. Therefore, termination of this transient within the limiting results discussed below is assured.

No operator error (in addition to the one that initiates this event) can result in a more limiting case since the reactor protection system will automatically terminate the transient.

See Appendix 15A for further discussion.

15.4.9.3 Core and System Performance

15.4.9.3.1 Mathematical Model

The analytical methods, assumptions and conditions for evaluating the excursion aspects of the control rod drop accident are described in detail in References 15.4-1, 15.4-3, and 15.4-4. They are considered to provide a realistic yet conservative assessment of the associated consequences. The data presented in Reference 15.4-2 shows that the RSCS Banked Position mode reduces the control rod worths to the degree that the detailed analyses presented in References 15.4-1, 15.4-3, and 15.4-4 or the bounding analyses presented in Reference 15.4-5 are not necessary. Compliance checks are instead made to verify that the maximum rod worth does not exceed 1% Δk. If this criteria is not met, then the bounding analysis is performed. The rod worths are determined using the BWR Simulator Model described in Reference 4.3-2 of section 4.3. Detailed evaluations, if necessary, are made using the methods described in References 15.4-1 to 15.4-3.

15.4.9.3.2 Input Parameters and Initial Conditions

The core at the time of rod drop accident is assumed to be at the point in cycle which results in the highest control rod worth, to contain no xenon, to be in a hotstartup condition, and to have the control rods in sequence A at 50% rod density (groups 1-4 withdrawn). Removing xenon, which competes well for neutron absorptions, increases the fractional absorptions, or worth, of the control rods. The 50% control rod density ("black and white" rod pattern), which nominally occurs at the hot-startup

condition, ensures that withdrawal on the next rod results in the maximum increment of reactivity.

Since the maximum incremental rod worth is maintained at very low values, the postulated CRDA cannot result in peak enthalpies in excess of 280 calories per gram for any plant condition. The data presented in Subsection 15.4.9.3.3 show the maximum control rod worth. Other input parameters and initial conditions are given in Table 15.4-9.

15.4.9.3.3 Results

The radiological evaluations are based on the assumed failure of 770 fuel rods. The number of rods which exceed the damage threshold is less than 770 for all plant operating conditions or core exposure provided the peak enthalpy is less than the 280 cal/gm design limit.

The results of the compliance-check calculation, as shown in the Table 15.4-10, indicate that the maximum incremental rod worth is well below the worth required to cause a CRDA which would result in 280 cal/gm peak fuel enthalpy (see Reference 15.4-1). The conclusion is that the 280 cal/gm design limit is not exceeded and the assumed failure of 770 pins for the radiological evaluation is conservative.

15.4.9.4 Barrier Performance

An evaluation of the barrier performance was not made for this accident since this is a highly localized event with no significant change in the gross core temperature or pressure.

15.4.9.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(1) The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10CFR100 guidelines. This analysis is referred to as the "Design Basis Analysis".

(2) The second analysis is based on assumptions considered to provide a realistic conservative estimate of

radiological consequences. This analysis is referred to as the "Realistic Analysis."

A schematic of the leakage path is shown in Figure 15.4-9.

Specific parametric valves used for the design basis and the realistic analyses are presented in Table 15.4-16.

15.4.9.5.1 Design Basis Analysis

The design basis analysis is based on the NRC's Standard Review Plan 15.4.9 (Reference 15.4-6). The specific models, assumptions and the program used for computer evaluation are described in Reference 15.4-7.

It is assumed that 10 percent of the halogens and 10 percent of the noble gases contained in rods that experience cladding damage are released from the fuel. From the rods that exceed 280 cal/g, 50 percent of the halogens and 100 percent of the noble gases are released. Of those fission products released from the fuel, 90 percent of the halogens and 0 percent of the noble gases are absorbed by the reactor water. The remaining activity is released to the condenser prior to isolation valve closure.

Assuming a partition factor of 10 in the condenser for iodines, the activity airborne in the condenser is presented in Table 15.4-11.

The fission product activity released to the environment is dependent upon the activity airborne in the condenser, the condenser leak rate, and the turbine building leak rate. For the purpose of this analysis it is assumed that the condenser leak rate is 1 percent per day and the turbine building leak rate is infinite. Based on the airborne activity presented in the previous subsections and the above leakage rates, the noble gas and iodine releases to the environment are presented in Table 15.4-12.

15.4.9.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are described in Reference 15.4-8.

The following assumptions are used in calculating fission product activity release from the fuel:

a) The reactor has been operating at design power for 1000 days until 30 minutes prior to the accident. When translated into actual plant operation, this assumption means that the reactor was shut down from design power, taken critical, and brought to the initial temperature conditions within 30 minutes of the departure from design power. The 30 minute time represents a conservative estimate of the shortest period in which the required plant changes could be accomplished and defines the decay time to be applied to the fission product inventory calculations.

b) An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity in a failed fuel rod is assumed to be released. These percentages are consistent with actual measurements made during defective fuel experiments (Reference 15.4-9).

c) The fraction of solid fission product activity available for release of the fuel is negligible.

d) The fission products produced during the nuclear excursion are neglected.

The following assumptions are used in calculating the amount of fission product activity transported from the reactor vessel to the main condenser:

a) The recirculation flow rate is 25 percent of rated, and the steam flow to the condenser is 5 percent of rated. The 25 percent recirculation flow and 5 percent steam flow are the maximum flow rates compatible with the maximum fuel damage. The 5 percent steam flow rate is greater than that which would be in effect at the reactor power level assumed in the initial conditions for the accident. This assumption is conservative because it results in the transport of more fission products through the steamlines than would be expected. Because of the relatively long fuel-to-coolant heat transfer time constant, steam flow is not significantly affected by the increased core heat generation within the time required for the main steamline isolation valves to achieve full closure.

b) The main steamline isolation valves are assumed to receive an automatic closure signal 0.5 second after detection of high radiation in the main steamlines and to be fully closed at 5 seconds from the receipt of the closure signal. The automatic closure signal originates from the main steamline radiation monitors. The total time required to isolate the main steamlines (5.5 seconds) combined with the assumptions, dictates the

total amount of fission product activity transported to the condenser before the steamlines are isolated.

c) All of the noble gas activity is assumed to be released to the steam space of the reactor vessel.

d) The mass ratio of the halogen concentration in steam to that of the water is assumed to be 2 percent.

e) Fission product platecut is neglected in the reactor vessel, main steam lines, turbine, and condenser.

Of those fission products released from the fuel and transferred to the condenser, it is assumed that 100 percent of the noble gases are airborne in the condenser. The iodine activity airborne in the condenser is a function of the partition factor, volume of air, and volume of water. A partition factor of 100 is assumed in condenser for iodine activity. By using the above conditions, the activity airborne in the condenser is presented in Table 15.4-13.

The following assumptions and conditions are used to evaluate the activity released to the environment:

a) The leak rate out of the condenser is 0.5 percent per day of the combined condenser and turbine free volume.

b) The activity released from the condenser becomes airborne in the turbine building. The turbine building ventilation rate is seven air changes per day.

c) No filtration or plateout of iodines occurs in the building prior to release to the atmosphere.

Based on the above assumptions, the fission product release to the environment is presented in Table 15.4-14.

The offsite individual exposures for the conservative and the realistic cases were presented in Table 15.4-15 for comparison.

15.4.10 References

15.4-1 R. C. Stirn, et al., "Rod Drop Accident Analysis for Large BWRs", March 1976 (NEDO-10527).

15.4-2 C. J. Paone, "Bank Position Withdrawal Sequence", September 1976 (NEDO-21231).

15.4-3 R. C. Stirn, et al., "Rod Drop Accident Analysis for Large BWRs", July 1972 Supplement 1 (NEDO-10527).

15.4-4 R. C. Stirn, et al., "Rod Drop Accident Analysis for Large BWRs", January 1973 Supplement 2 (NEDO-10527).

15.4-5 "GE BWR Generic Reload Application for 8x8 Fuel" (NEDO-20360).

15.4-6 USNRC Standard Review Plan, NUREG-75/037, Washington, D.C., November 24, 1975.

15.4-7 Stancavage, P. P. and E. J. Morgan, "Conservative Radiological Accident Evaluation - The CONACOl Code", NEDO-21142, March 1976.

15.4-8 Nguyen, D., "Realistic Accident Analysis for General Electric Boiling Water Reactor", The RELAC Code and User's Guide. NEDO-2002. To be issued.

15.4-9 Horton N. R.,Williams W. A., and Holtzclaw K. W., "Analytical Methods for Evaluating the Radiological Aspects of General Electric Boiling Water Reactors," APED-5756, March 1969.

QUESTION 211.112:

Since the reclassification of the generator and turbine trip without bypass transients has not been accepted by the staff and is still under generic review, reanalyze the above events for determination of the operating limit MCPR in which the results would not violate the safety limit MCPR of 1.06. Also, it is our position that the limiting transient be reanalyzed with the ODYN code.

RESPONSE:

The review of the ODYN[1] code has yet to be completed by the staff. Currently, the staff review has not concluded review of the adequacy of the margin inherent in the proposed ODYN licensing basis. Additionally, it appears further review may be necessary in the area of input parameters. In light of the incomplete nature of the review it is not prudent to reanalyze with the ODYN code at this time.

In a similar manner, it would not be prudent to reanalyze the reclassified events with the current REDY licensing basis to determine the effect of these events on operating limits at this time, since the ODYN model is in the final review stages and the Staff has indicated REDY based limits would not be accepted following ODYN approval. Reanalyses should be deferred until complete approval of ODYN is available and the ODYN licensing basis is established.

It should be noted that, although the generic review of reclassification has not been complete, that it remains the proposed licensing basis for this application. Any evaluations of the effect on these events on operating limits would be for information for the staff to judge the magnitude of the change in margin due to reclassification.

1 NEDO-24154, Volume 1, 2, NEDE24154-P Volume 3, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors", dated October 1978. Submitted to NRC Attn.: O.D. Parr 12/15/78, Letter from J. F. Quirk.

QUESTION 211.180:

The narrative on page 15.4-13 discussing the "abnormal startup of an idle recirculation pump" transient states, "The water level does not reach either the high or low level set points." Table 15.4.3. indicates a low level trip occurs 22.0 seconds after pump start. Figure 15.4-6 indicates a low level trip occurs approximately 23.5 seconds after pump start. Further:

a) Table 15.4-6 indicates a low level alarm 10.5 seconds after pump start while Figure 15.4-6 indicates this alarm occurs about 11.5 seconds after the pump starts.

b) Table 15.4-6 indicates vessel level beginning to stabilize 50.0 seconds after the pump starts. Figure 15.4-6 shows no such indication.

Resolve these discrepancies.

RESPONSE:

The sequence in Table 15.4-3 starts out with a scram at 10 seconds following the improper pump start. Figure 15.4-6 confirms this. At 23.5 seconds (rather than 22) level falls to L3 which also issues a redundant scram signal to a system which has already scrammed. It is the intent of Table 15.4-3 has been modified.

a) Table 15.4-4 indicates L4 near 11 seconds. This is verified by Figure 15.4-6.

b) Table 15.4-4 indicates that vessel level is beginning to stabilize at 50 seconds. This appears to be correct. Actually, level recovered from L3 at about 41 seconds and from 30 to 40 seconds level is changing at the rate of 2.5 in/sec. From 50 to 60 seconds level rate is definitely flattening out under normal feedwater level control.

QUESTION 211.261:

GE calculations performed for decrease in reactor coolant temperature (Section 15.1) and for reactor pressure increase (Section 15.2) events using the proposed ODYN licensing basis model (NEDO-24154) have shown that in some cases a more limiting CPR is predicted than by the current REDY licensing bases model (NEDO-10802). Based on a letter to Glen C. Sherwood dated 1/23/80 from Richard P. Denise, the staff's ODYN licensing position is that GE can proceed with ODYN analysis of transients described in Chapter 15 of licensing application Safety Analysis Reports. Provide an ODYN analysis of the applicable events listed in Tables 2-1 and 2-2 of NEDO-24154-P.

RESPONSE:

The final resolution of details for the application fo ODYN calculations in the transient licensing process has not yet been achieved. Generic efforts are underway to determine implementation. Reanalysis (utilizing ODYN) for key pressurization events will necessarily follow the generic resolution. No major change in transient margins is anticipated, as indicated by preliminary, generic analyses.

TABLE 423.28-1

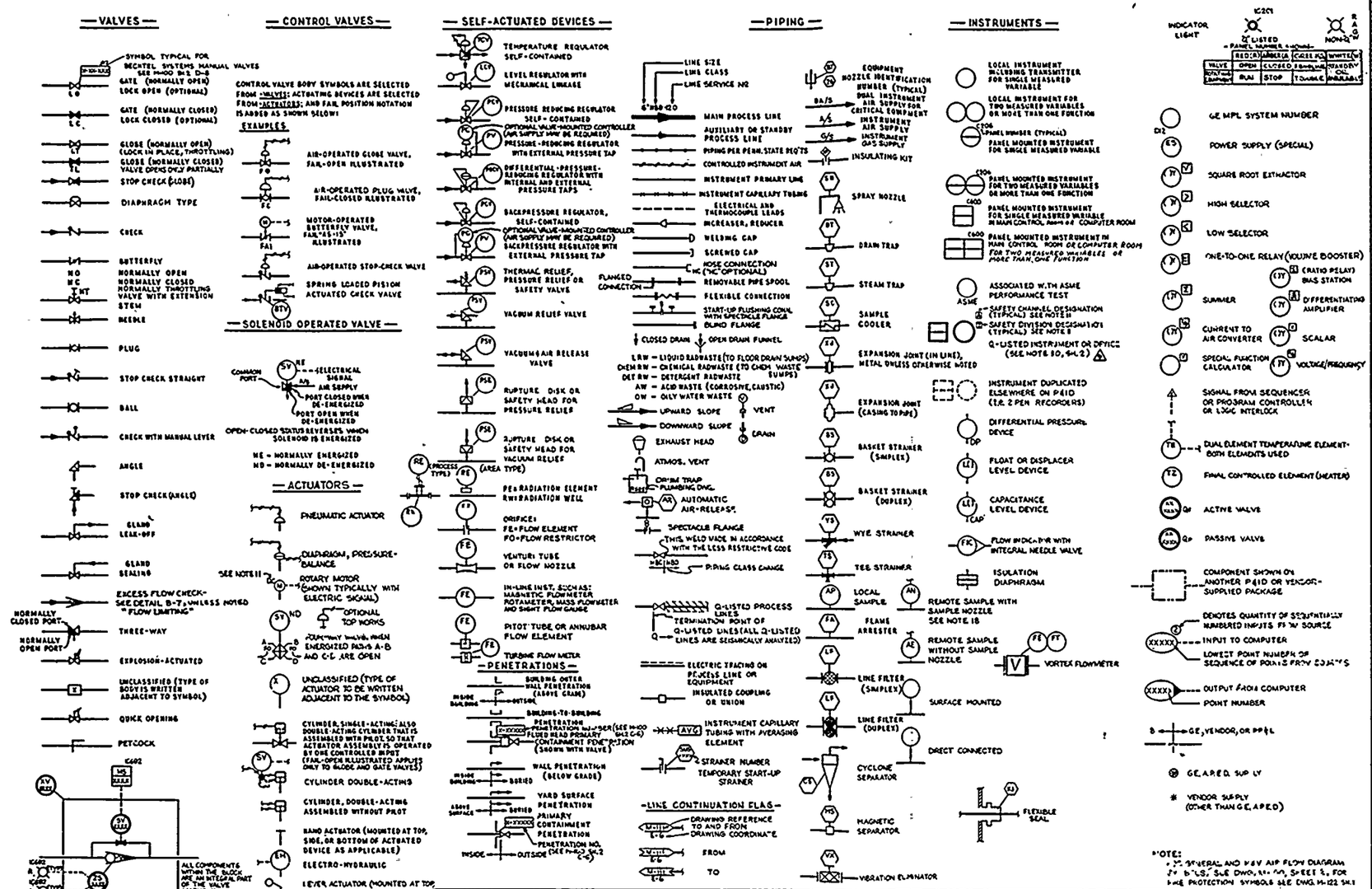
SYSTEM	VALVE NO.	PREOP. NO.	INST. AIR OR PRI. CONT. INST. GAS
Fire Protection	XV-12248,49 XV-02248 XV-02215	P13	Inst. Air
RBCCW	HV-11315	P14	Inst. Air
RB HVAC	HD17534A,B,C,D,E,F,H A11* HD17502A,B; HD17514A,B, A11* HD17564A,B; HD17524A,B A11 HD17576A,B; HD17586A,B A11* HD17508A,B Both* HD17651, BDID17603A,B BDID 17604A,B; BDID 17605A,B BDID 17606A,B; BDID 17609A,B BDID 17652A,B; BDID 17653A,B BDID 17659A,B; BDID 17667A,B BDID 17668A,B; BDID 17669A,B BDID 17670A,B; BDID 1761A,B BDID 17674A,B; BDID 17675A,B	P34.1	Inst. Air

TABLE 423.28-1

SYSTEM	VALVE NO.	PREOP. NO.	INST. AIR OR PRI. CONT. INST. GAS
RWCU	HV-14506A,B; 14507A,B HV-14508A,B; 14510A,B HV-14511A,B; 14512A,B HV-14513A,B; 14514A,B HV-14566A,B; 14522 HV-14523, 14528, 14516 HV-14518, 14519, 14520 HV-14521, G33-1F033	P61.1	Inst. Air
Liquid Radwaste	HV-16108A1, HV-16116A1 HV-16108A2, HV-16116A2 Both*	P69.1	Inst. Air
Containment Recirculation	HV-17521,23,24,22,25 All* HV-15704,05,14 All* HV-15703,13	P73.1	Inst. Air

TABLE 423.28-1

SYSTEM	VALVE NO.	PREOP. NO.	INST. AIR OR PRI. CONT. INST. GAS
Control Structure HVAC	HDM-07802A,B Both* HDM-07833A,B; HDM-07824A2,B2 HDM-07824 A4,B4; HDM-07881A,B HDM-07872A,B; HDM-07873A,B A11* TV-07813A,B TV-08602A,B	P30.1 P30.2	Inst. Air
Feedwater	10604A,B,C 10640, 10641 14107A,B 10650 10606A,B,C 10604A,B,C 10663A1,A2,B1,B2,C1,C2 10664A,B,C	P45.1,2	Inst. Air



FIRST LETTER		SECOND & SUCCEEDING LETTERS																					
MEASURED VARIABLE	SYMBOL FOR MEASURED VARIABLES	DISPLAY DEVICES						CONTROLLING DEVICES						SENSING DEVICES				LOCAL OBSERVATION GLASS	TEST CONNECTION	CALCULATOR RELAY OR CONVERTER (BLIND)
		INDICATING	RECORDING	INTEGRATING INDICATING (SEE NOTE 6)	ALARM (SEE NOTE 10) LOW	ALARM (SEE NOTE 10) HIGH	ALARM (SEE NOTE 10) HIGH-LOW	INDICATING	RECORDING	BLIND	CONTROL VALVE	SELF ACTUATED VALVE	DAMPER	SWITCH (SEE NOTE 4)	PRIMARY ELEMENT	BLIND TRANSMITTER	INDICATING TRANSMITTER			
TYPICAL SYMBOL	( )	( )I	( )R	( )QI	( )AL	( )AL	( )AL	( )IC	( )RC	( )C	( )V	( )CV	( )D	)S(	( )E	( )T	( )IT	( )G	( )P	( )Y
ANALYSIS (SEE NOTE 15)	A	AI	AR		AAL	AAH	AAHL	AIC	ARC	AC	AV			AS( )	AE	AT	AIT		AP	AY
CONDUCTIVITY	C	CI	CR		CAL	CAH	CAHL	CIC	CRC		CV			CS( )	CE	CT	CIT		CP	CY
DENSITY	D	DI	DR		DAL	DAH	DAHL	DIC	DRC		DV			DS( )	DE	DT	DIT		DP	DY
FLOW	F	FI	FR	FQI	FAL	FAH	FAHL	FIC	FRC	FC	FV	FCV	FO	FS( )	FE	FT	FIT	FG	FP	FY
HAND	H							HIC		HC	HV		HO	HS(5)						HY
TIME	K	KI	KR	KQI	KAL	KAH	KAHL	KIC	KRC	KC	KV	KCV	KO	KS( )		KT	KIT			
LEVEL	L	LI	LR		LAL	LAH	LAHL	LIC	LRC	LC	LV	LCV		LS( )	LE	LT	LIT	LG	LP	LY
LEVEL DIFFERENTIAL	LD	LDI	LDR		LDAL	LDAH	LDAHL	LDIC	LDRC	LDC	LDV	LDCV		LDS(						
MOISTURE or HUMIDITY	M	MI	MR		MAL	MAH	MAHL	MIC	MRC	MC	MV		MO	MS ( )	ME	MT	MIT			MY
NEUTRON FLUX	N	NI	NR	NQI	NAL	NAH	NAHL	NIC	NRC	NC				NS ( )	NE	NT	NIT		NP	
FLOW DIFFERENTIAL	FD	FDI	FDR		FDAL	FDAH	FDAHL	FDIC	FDRC	FDC				FDS( )		FDT	FDIT			
QUANTITY OR EVENT	Q	QI	QR	QQI	QAL	QAH	QAHL	QIC	QRC	QC	QV			QS ( )		QT	QIT			
PRESSURE	P	PI	PR		PAL	PAH	PAHL	PIC	PRC	PC	PV	PCV		PS ( )	PE	PT	PIT		PP	PY
PRESSURE DIFFERENTIAL	PD	PDI	PDR		PDAL	PDAH	PDAHL	PDIC	PDRC	PDC	PDV	PDCV	PDO	PDS ( )		PDT	PDIT			
RADIATION (NUCLEAR)	R	RI	RR	RQI	RAL	RAH	RAHL	RIC	RRC	RC				RS ( )	RE	RT	RIT		RP	RY
SPEED	S	SI	SR		SAL	SAH	SAHL	SIC	SRC	SC				SS ( )	SE	ST	SIT			SY
TEMPERATURE	T	TI	TR		TAL	TAH	TAHL	TIC	TRC	TC	TV	TCV	TO	TS ( )	TE	TT	TIT		TP	TY
															(SEE NOTE 8)				(SEE NOTE 8)	
TEMPERATURE DIFFERENTIAL	TD	TDI	TDR		TDAL	TDAH	TDAHL	TDIC	TDRC	TDC	TDV	TDCV		TDS ( )						TDY
VIBRATION	V	VI	VR		VAL	VAH	VAHL	VIC	VRC	VC	VV			VS ( )	VE	VT	VIT			VY
WEIGHT FACTOR	W	WI	WR	WQI	WAL	WAH	WAHL	WIC	WRC	WC	WV			WS ( )	WE	WT	WIT			WY
UNCLASSIFIED (SEE NOTE 5)	X	XI	XR	XQI	XAL	XAH	XAHL	XIC	XRC	XC	XV		XO	XS ( )	XE	XT	XIT		XP	XY
POSITION	Z	ZI	ZR		ZAL	ZAH	ZAHL	ZIC	ZRC	ZC	ZV			ZS ( )	ZE	ZT	ZIT			ZY

M-100	LEGEND AND SYMBOLS, SHEETS 1,2 & 3
M-101	MAIN STEAM
M-102	EXTRACTION STEAM
M-103	VENTS & DRAINS, HTRS. 1, 2 & DRAIN COOLER
M-104	VENTS & DRAINS, HTRS. 3, 4 & 5
M-105	CONDENSATE
M-106	FEEDWATER
M-107	AIR REMOVAL & SEALING STEAM
M-108	CONDENSATE & REFUELING WATER STORAGE
M-109	SERVICE WATER
M-110	SERVICE WATER
M-111	EMERGENCY SERVICE WATER, SHEETS 1 & 2
M-112	RHR SERVICE WATER
M-113	REACTOR BUILDING CLOSED COOLING WATER
M-114	TURBINE BUILDING CLOSED COOLING WATER
M-115	CIRCULATING WATER
M-116	CONDENSATE DEMINERALIZER, SHEETS 1 & 2
M-117	RAW WATER TREATMENT SHTS. 1 & 2
M-118	MAKE-UP DEMINERALIZER
M-119	LUBE OIL, SHEETS 1 & 2
M-120	DIESEL OIL STORAGE & TRANSFER
M-121	AUXILIARY STEAM
M-122	FIRE PROTECTION, SHEETS 1, 2, 3, & 4
M-123	PROCESS SAMPLING, SHEETS 1, 2, 3 & 4
M-124	CHLORINATION
M-125	COMPRESSED AIR, SHEETS 1, 2 & 3
M-126	CONTAINMENT INSTRUMENT GAS
M-127	FEED PUMP TURBINE STEAM, SHEETS 1 & 2
M-128	MAKEUP WATER SUPPLY SYSTEM
M-129	PROCESS VALVE STEM LEAKOFF COLLECTION
M-130	ASME TEST
M-131	GASEOUS RADWASTE RECOMBINER CLOSED CLG WATER
M-132	ACID INJECTION FOR THE CIRCULATING WATER SYSTEM
M-133	HYDROGEN STORAGE
M-134	DIESEL ENGINE AUXILIARIES
M-135	CANCELLED
M-136	PRIMARY COOLANT DEGASIFIER PACKAGE
M-137	AREA RADIATION MONITORING
M-138	COOLING TOWER BLOWDOWN TREATMENT
M-139	NSW LEAKAGE CONTROL SYSTEM
M-140	REACTOR RECIRC MOTOR-GENERATOR SET
M-141	NUCLEAR BOILER [illegible]
M-142	NUCLEAR BOILER VESSEL INSTRUMENTATION
M-143	REACTOR RECIRCULATION
M-144	REACTOR WATER CLEAN-UP
M-145	CLEAN-UP FILTER-DEMINERALIZER
M-146	CONTROL ROD DRIVE - PART A
M-147	CONTROL ROD DRIVE - PART B
M-148	STANDBY LIQUID CONTROL
M-149	REACTOR CORE ISOLATION COOLING
M-150	RCIC TURBINE-PUMP
M-151	RESIDUAL HEAT REMOVAL
M-152	CORE SPRAY
M-153	FUEL POOL COOLING & CLEAN-UP
M-154	FUEL POOL FILTER-DEMINERALIZER
M-155	HIGH PRESSURE COOLANT INJECTION
M-156	RCIC TURBINE-PUMP
M-157	CONTAINMENT ATMOSPHERE CONTROL, SHEETS 1 & 2
M-158	CANCELLED
M-159	PRIMARY CONTAINMENT LEAKAGE RATE TESTING
M-160	MISC DRAINAGE SYSTEMS
M-161	LIQUID RADWASTE COLLECTION SHEETS 1 & 2
M-162	LIQUID RADWASTE PROCESSING, SHEETS 1 & 2
M-163	LIQUID RADWASTE CHEMICAL PROCESSING
M-164	LIQUID RADWASTE LAUNDRY PROCESSING
M-165	BUILDING VENT STACK MONITORING SYSTEM
M-166	SOLID RADWASTE COLLECTION
M-167	RADWASTE SOLIDIFICATION "
M-168	
M-169	OFF GAS RECOMBINER SHTS 1 & 2
M-170	CANCELLED
M-171	AMBIENT TEMPERATURE CHARCOAL OFFGAS TREATMENT SYSTEM
M-172	
M-173	CIRC WATER PUMP HOUSE AIR FLOW DIAGRAM
M-174	TURBINE BLDG. AIR FLOW DIAGRAM
M-175	REACTOR BLDG. AIR FLOW DIAGRAM ZONE III
M-176	REACTOR BLDG AIR FLOW DIAGRAM ZONE I
M-177	DRYWELL AIR FLOW DIAGRAM
M-178	CONTROL STRUCTURE AIR FLOW DIAGRAM
M-179	RADWASTE BUILDING AIR FLOW DIAGRAM, SHEETS 1 & 2
M-180	CANCELLED
M-181	MISC. BLDGS AIR FLOW DIAGRAM
M-182	DIESEL GEN. & ESSW PUMP HOUSE AIR FLOW DIAGRAM
M-183	MISC HV & AC EQUIPMENT DRAINAGE SYST
M-184	HGN, SGHS & SCC AIR FLOW DIAGRAM
M-185	CANCELLED
M-186	CONTROL STRUCTURE CHILLED WATER
M-187	REACTOR BLDG. CHILLED WATER, SHEETS 1 & 2
M-188	TURBINE BUILDING CHILLED WATER
M-189	RADWASTE BUILDING CHILLED WATER
M-190	HGN, SGHS & SCC REFRIGERANT
M-191	
M-192	
M-193	
M-194	
M-195	
M-196	
M-197	

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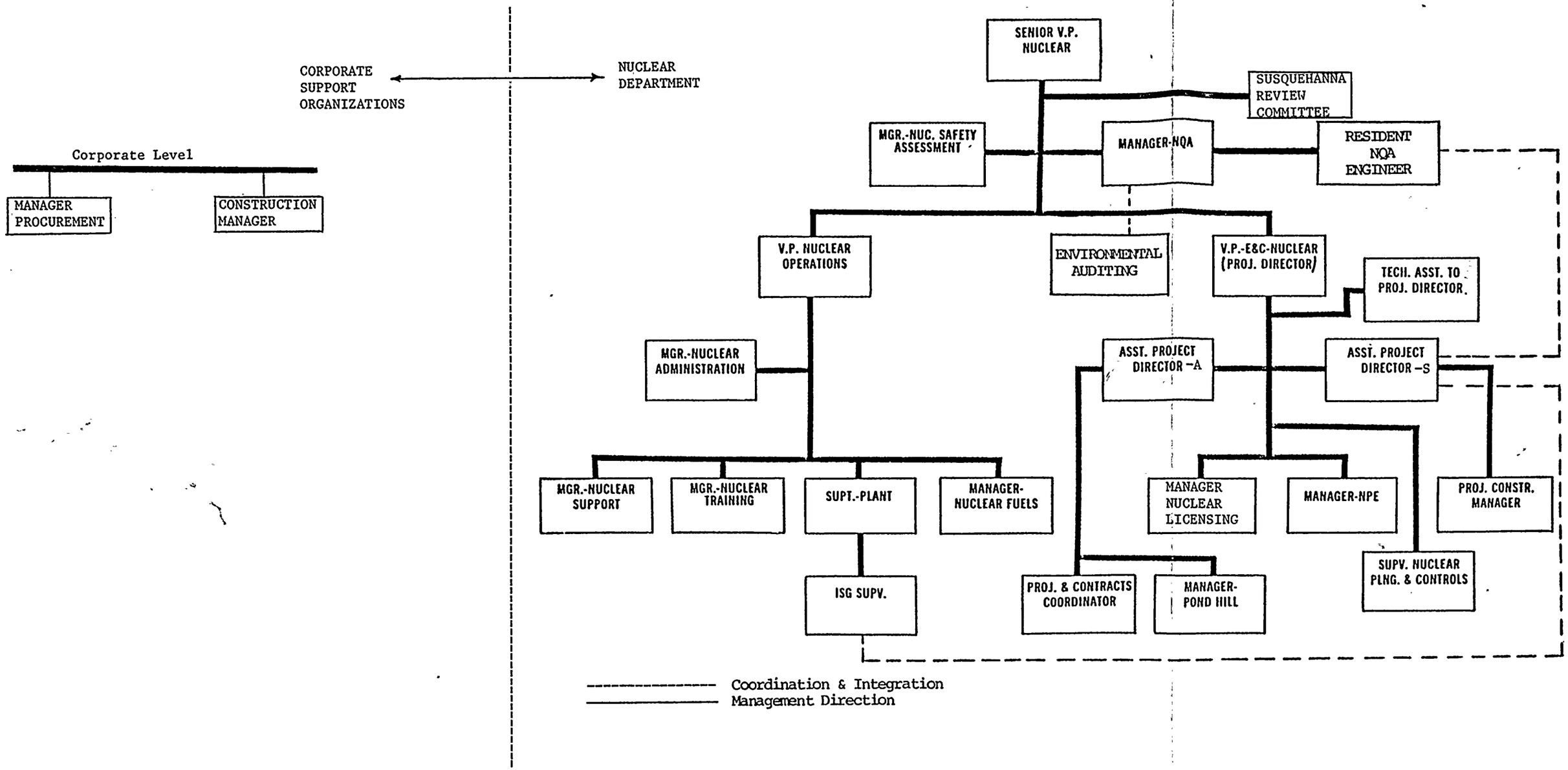
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

P&ID
LEGEND & SYMBOLS

FIGURE 1.8–2a



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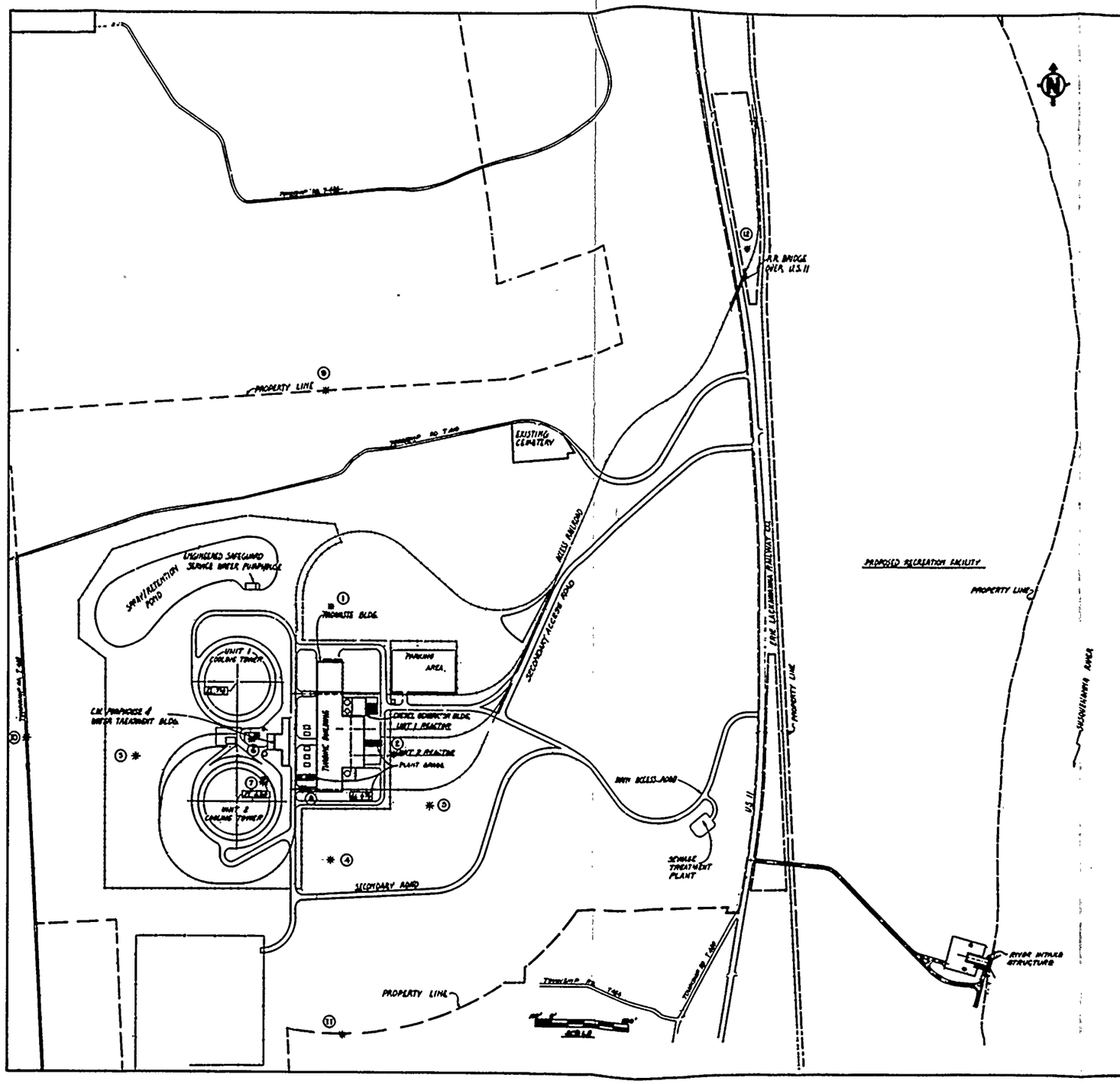
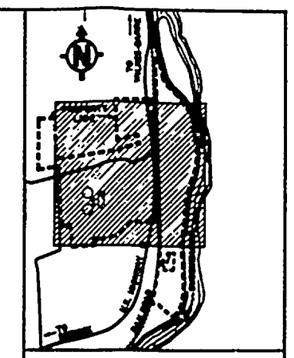
NUCLEAR DEPARTMENT
MANAGEMENT ORGANIZATION

FIGURE 17.2-2





KEY PLAN

SCALE

ESTIMATED TURBINE SHINE DOSE RATES (1) (2) (3)

POINT	DESCRIPTION	REM/YR
1	GENERAL YARD AREA	$6.7\text{-}2^{(4)}$
2	GENERAL YARD AREA	1.0-1
3	GENERAL YARD AREA	2.8-2
4	GENERAL YARD AREA	4.3-2
5	GENERAL YARD AREA	1.3-2
6	GENERAL YARD AREA	1.2-1
7	TOP OF COOLING TOWER	5.7-1
8	FIELD OFFICE	8.8-2
9	NORTH SITE BOUNDARY	3.7-3
10	WEST SITE BOUNDARY	3.5-3
11	SOUTH SITE BOUNDARY	5.6-3
12	VISITORS' CENTER	2.8-8

(1) POINTS 1 THROUGH 8 ARE BASED ON CONTINOUOS OCCUPANCY (8760 HOURS PER YEAR), AND A 100 PER CENT PLANT CAPACITY FACTOR WITH UNIT 1 ONLY IN OPERATION.

(2) POINTS 9 THROUGH 11 ARE BASED ON CONTINUOUS OCCUPANCY (8760 HOURS PER YEAR), AND AN 80 PER CENT PLANT CAPACITY FACTOR WITH BOTH UNITS IN OPERATION.

(3) POINT 12 IS BASED ON 8 HOURS PER YEAR OCCUPANCY, AND A 100 PER CENT PLANT CAPACITY FACTOR WITH BOTH UNITS IN OPERATION.

(4) $6.7\text{-}2 = 10^{-2}$

SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

ESTIMATED TURBINE SHINE DOSE RATES

FIGURE 12.4-1