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**HAZARDS SUMMARY REPORT
for the
ARMY PACKAGE POWER REACTOR
SM-1**

TASK XVII

**Edited By
S. S. Rosen**

Issued May 1960

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Contract No. AT (30-3) -326**

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PREFACE

This report is an updated and revised version of the original SM-1 (APPR-1) Hazards Summary Report. The original report was issued July 1955, almost two years before construction of the SM-1 plant was completed. During that time interval there were numerous design changes. Consequently, the original report does not accurately describe the plant as built. This revision is written after the SM-1 has been operated 3 years. It describes the as-built plant and includes plant modifications and experience obtained during the first 3 years of SM-1 operation.

ACKNOWLEDGEMENTS

A large portion of this revision was taken from the original SM-1 (APPR-1) Hazards Summary Report. A large amount of new information appearing in this revision, particularly that concerned with operating experience, was contributed by Alco's operations group at Fort Belvoir. The U. S. Army Reactor Branch offered many helpful suggestions and contributed much information in the general preparation of this report.

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SUMMARY

The SM-1 is described and the various hazards are reviewed. Because of the reactor's location near the nation's capital, containment is of the utmost importance. The maximum energy release in any possible accident is 7.4 million Btu's which is completely contained within a 7/8 in. thick steel cylindrical shell with hemispherical ends. The vapor container is 60 ft. high and 32 ft. in diameter and is lined on the inside with 2 ft. of reinforced concrete which provides missile protection and is part of the secondary shield.

All possible nuclear excursions are reviewed. The energy from any of these is insignificant compared to the stored energy in the water.

The maximum credible accident is caused by the reactor running constantly at its maximum power of 10 Mw and through an extremely unlikely sequence of failures, causing the temperature of the water in the primary and secondary systems to rise to saturation; whereupon a rupture occurs releasing the stored energy of 7.4 million Btu's into the vapor container. If the reactor core melts during the accident, a maximum of 1.5×10^7 curies of activity is released into the vapor container.

While it is highly improbable for a rupture of the vapor container to occur except by sabotage or bombing, the hazards to the surrounding area are discussed in the event of such a rupture occurring simultaneously with the maximum credible accident.

The SM-1 has been in operation for about 3 years, during which time much operating experience has been accumulated and much nuclear and system data has been obtained through both normal operations and the Research and Development programs conducted. These data are discussed in relation to their effect on the hazards evaluation of the SM-1 and include: core nuclear data, primary and secondary system thermal data, radiological safety, dose rates at various positions in the primary system both during operation and after shutdown and their effects on maintenance, experience with radioactive waste disposal, area monitoring, experience with the use of trainees in plant operation, and plant modifications accomplished during this period.

During 3 years of operation, SM-1 has proved to be an extremely stable and safe system. No incidents occurred during this time that resulted in any significant injuries, radiation exposures or environmental contamination.

A plan is included for continued operation of the SM-1 facility after July 1, 1960, when SM-1 will become the responsibility of the U.S. Army Corps of Engineers.

CHAPTER I - INTRODUCTION

A. GENERAL

The Army Package Power Reactor (now known as the SM-1) was designed and constructed by Alco Products, Inc. under contract with the Army Reactors Branch of the A. E. C. (Contract No. AT (11-1)-318). The SM-1 is a prototype of a reactor designed to meet the requirements and site conditions of a remote military base.

Since completion of construction in April 1957, the SM-1 has been operated under AEC contract by Alco Products, Inc. Contract administration was initially under AEC's Schenectady Operations Office; in 1958 this responsibility was transferred to the AEC New York Operations Office (NYOO). In each case, the AEC operations manager designated the Chief, Nuclear Power Field Office (NPFO) at Fort Belvoir as the AEC's on-site representative for supervision of the SM-1 operating contract. Plant operation has been performed by a military crew under technical supervision of the AEC's contractor, Alco Products, Inc. Overall technical responsibility for the SM-1 has been exercised by the Chief, ANPP, with staff supervision through his staff in AEC Headquarters.

The guidelines for future SM-1 operations are as follows:

1. The SM-1 will continue to be operated under AEC supervision through FY 1960 to assure the complete accomplishment of AEC responsibilities under the original DOD-AEC agreements.
2. On July 1, 1960, the Corps of Engineers, as the agent of the DA and DOD, will assume responsibility for control, operation, safety, and property accountability for the SM-1.
3. The basis for transfer of the SM-1 from AEC to Corps of Engineers control will be as contained in Section 91b of the Atomic Energy Act of 1954. Also see the joint AEC-DOD "Policy for Operation of Military Power Reactors," dated November 8, 1954.
4. Within the Corps of Engineers, SM-1 responsibilities will be carried out by the Special Assistant to the Chief of Engineers for Nuclear Power (SANP), who will exercise staff supervision through the Nuclear Power Division, OCE. Operation of the SM-1 will be performed under direct supervision of the Chief, NPFO, Fort Belvoir, Virginia.
5. After July 1, 1960, the main objectives of the SM-1 operation will be to train operators for military field nuclear power plants and to perform Research and Development.

6. Research and Development tasks to be performed on the SM-1 after July 1, 1960 will be as specified by Office of Asst. Director (Army Reactors) Division of Reactor Development (ARM) and will include but not be limited to those required to:
 - a. Extend knowledge of operating characteristics of the SM-1.
 - b. Support design, construction, operation, and maintenance of future military plants, including prototypes thereof.
7. SM-1 operation and maintenance after July 1, 1960 will be performed by a Government crew under the supervision of the Chief, NPFO. Contractor services will be utilized as required for technical consultation on operation and maintenance matters and on special maintenance tasks. In addition, the Chief, NPFO, may contract for special technical assistance for the execution of specified R&D tasks and testing.

B. DESCRIPTION

The SM-1 is a 10 thermal mw pressurized water reactor delivering 1925 net kilowatts of electricity with 2.4 in. of mercury back pressure (85°F. condenser cooling water). When lower temperature condenser water is available, back pressure can be reduced, permitting delivery of 2000 net kw at 1.8 in. of mercury back pressure. The fuel elements are similar to those in the MTR but are made of stainless steel rather than aluminum, and in addition to the fissionable material contain a burnout poison in the form of boron.

The reactor operates at a pressure of 1200 psia and outlet temperature of 450°F. at full power. Two primary coolant pumps in parallel are provided either of which will provide a flow rate of 3860 gpm. The water flows from the reactor to a steam generator, where heat is transferred to the secondary (steam) system, and is then returned to the reactor through the pump. This entire primary loop is installed inside a vapor container 32 ft. in diameter and 60 ft. high as shown in the cutaway view in Fig. I-1. The enclosure will contain the energy released from all the steam generated by flashing of the superheated primary and secondary system water volumes when the maximum amount of heat has been stored in these volumes.

The turbine generator room is shown in Fig. I-1 just outside the vapor container with the control room on the second floor behind it. To the left of the vapor container may be seen the spent fuel pit where used fuel elements are stored. The rest of the building contains offices, laboratories, shops, etc.

C. SITE

The reactor is located at Fort Belvoir, Virginia within the bounds of the Engineer Research and Development Laboratory, and bordering on Gunston Cove, as shown in Fig. I-2. The nearest living quarters are 2630 ft. from the site and are residential dwellings in the village of Fairfax, Va.

The location of Fort Belvoir with respect to Washington and surrounding area is shown in Fig. I-3. Metropolitan Washington has a population of 1,460,000, practically all of which lies within a 20 mi. radius of the site. The most important activities in the nation are centered within the 20 mi. radius shown. The tremendous importance of complete containment of the SM-1 in the event of an accident is recognized.

The population distribution and percentage frequency of wind direction near the site are shown in Figs. I-4, 5 and 6. The figures are self explanatory. Populations are based on the 1950 census, and the wind directions are based on Washington National Airport hourly observations from January 1945 through December 1952. Fig. I-4 illustrates the above relationship for all wind conditions; Fig. I-5 when precipitation is falling (washout); and Fig. I-6 when the inversion base is below 500 ft. (night time condition). Data for the latter figure are from daily readings taken at 0300 GCT.

The meteorology, climatology, geology, and hydrology of the site are summarized in Appendix "A".

D. Design Data

Following is a summary of design data on the SM-1. More complete descriptions of individual components listed here may be found in subsequent sections of the report, along with some of the design considerations involved.

FIG. I-1 CUTAWAY VIEW OF THE SM-1

FIG. I-2 MAP OF THE FT. BELVOIR AREA

FIG. I-3 MAP OF THE AREA INCLUDING WASHINGTON, D.C.

POPULATION DISTRIBUTION AND PERCENTAGE FREQUENCY
OF WIND DIRECTION

FIG. I - 4

CALM 3.5%

DURING 1953 THERE WERE 153 DAYS HAVING SIGNIFICANT PRECIPITATION.

POPULATION DISTRIBUTION AND PERCENTAGE FREQUENCY
OF WIND DIRECTION DURING PRECIPITATION

FIG. I - 5

POPULATION DISTRIBUTION AND PERCENTAGE FREQUENCY
OF WIND DIRECTION WHEN INVERSION BASE IS BELOW 500 FEET.

FIG. I - 6

1. Overall Plant Performance

Thermal power developed in reactor	kw	10,740
	Btu/hr	3.68×10^7
Electric power generated	kw(2.4 in. Hg)	2145
Net electric power delivered	kw(2.4 in. Hg)	1925
Power required for auxiliaries	kw(2.4 in. Hg)	220
Thermal efficiency of net electric power generation	%(2.4 in. Hg)	18
Power density of reactor core	kw/liter	77

2. Reactor Data

Core

Average Diameter	in.	22.2
Height	in.	22.0
Volume of core	liters	139.5
	cu. in.	8513
Fuel plates per fixed fuel assembly		18
Number of fixed fuel assemblies		38
Fuel Plates per control rod fuel assembly		16
Number of control rod fuel assemblies		7
Total number of fuel plates		796
Dimensions of fuel assemblies (overall)		
Thickness	in.	<u>Fixed</u> 2.863
Width	in.	<u>Control Rod</u> 2.624
Length	in.	2.844
		33.625
		26.625

Control Rods

Type - Modified square "basket" to fit fuel space in lattice; upper section contains absorber material - lower section contains fuel sub-assembly. Actuating rack rigidly attached to bottom of basket.

Overall length including rack and cap	in.	117-3/4
Number		7
Travel	in.	22
Weight of rod	lb	72
Acceleration of rod after release	ft/sec ²	7.7
Maximum distance for rod to drop	in.	22

Thermal Data of Reactor at Full Power

Operating pressure in reactor	psia	1200
Coolant inlet temperature at reactor	°F	427.7
Coolant outlet temperature at reactor	°F	448
Properties of coolant at 440°F		
Density	lb/ft ³	52.25
Change in density per °F	lb/ft ³ -°F	0.046
Viscosity	lb/ft-hr	0.297
Thermal conductivity	Btu/hr-ft ² -°F/ft	0.374
Specific heat	Btu/°F-lb	1.103
Coolant flow through core	gpm lb/hr	3862 1.62 x 10 ⁶
Number of flow passes through reactor		1
Flow area in core	ft ²	2.083
Velocity in core passages (average)	fps	4.14
Design heat output	Btu/hr	3.68 x 10 ⁷
Heat transfer area	ft ² (start of life, EqXe, 440°F (end of life, EqXe, 440°F	565.8 600.6
Average heat flux	Btu/hr-ft ² (start of life, EqXe, 440°F (end of life, EqXe, 440°F	64,962 61,206

Actual peak-to-average heat flux ratio		4:1
Ratio of maximum to average heat flux in any one channel		2.2:1
Maximum bulk water temperature, hottest channel		489
Reynolds number in core (average)		55, 500
Film coefficient of heat transfer (average) Btu/hr-ft ² -°F		2400
Maximum surface temperature	°F	574.3
Boiling temperature at 1200 psia	°F	567.2
Maximum metal temperature	°F	595

3. Primary System Equipment

Pressure Vessel. ASTM-A-212, Grade B, clad with 304 stainless steel

Inside diameter	in.	47-1/2
Wall thickness	in.	2-3/4
Thickness of cladding, min.	in.	1/4
Design stress (ASME Code)	psi	17, 500
Overall length of vessel	in.	144-5/8
Thickness of head	in.	2-3/4
Diameter of head	in.	40-1/4
Diameter of opening at top of vessel	in.	28
Inside Diameter of thermal shield	in.	34-7/8
Thickness of thermal shield	in.	2
Insulation thickness	in.	4

Primary Coolant Pumps, centrifugal, canned rotor

Operating head of pump	ft	28
Operating temperature at suction	°F	427.7
Hydraulic horsepower at 3860 gpm, hot cold	hp	23.5
	hp	28.4
Number of pumps normally operated		1
Number of pumps installed		2

Steam Generator, vertical, double pass, U-tubes

Tube side fluid: primary coolant

Shell side fluid: boiling water and steam

Materials: Tubes, partition, shrouds and baffle:
304 L stainless steel.

Shell, tube sheet, nozzles: Type A-212B
carbon steel clad with type 304 stainless steel.

Design Pressure

Tube Side	psia	1600
Shell Side	psia	500

Design Temperatures

Tube Side	°F	650
Shell Side	°F	650

Operating Pressures

Tube Side	psia	1200
Shell side, no load full load	psia	422
	psia	213

Full load heat transfer	Btu	3.68×10^7
-------------------------	-----	--------------------

Heat Transfer surface:

Steam generating region	ft ²	884
Superheat region	ft ²	201
Number of tubes.-		
Steam generating region		326
Superheat region		44
Diameter of tubes, OD	in.	0.75
Tube wall thickness	in.	0.065
Velocity in tubes, steam generating region	fps	11.65
superheat region		9.95
Reynolds number in tubes		933000
Overall heat transfer coefficient	Btu/hr-ft ² °F	495
Total head loss on tube side	ft of hot fluid	16.7
Full load steam flow	lb/hr	36,585
Feedwater temperature	°F	255.8
Steam quality at entrance to superheater	%	99.7
Superheat	°F	20
Steam pressure at full load	psia	213
Insulation thickness	in.	4

Pressurizer

Materials, SA-212B-FBQ Type 304
Stainless steel clad

Length, overall	in.	102
Diameter, ID	in.	45
Wall thickness	in.	1.25

Thickness of cladding	in.	0.125
Heaters	kw (total)	37
Insulation thickness	in.	4
Design pressure	psia	1600
Design temperature	°F	650
Pipe size, schedule 80	in.	4

Primary Coolant Piping

Material, 304 stainless		
Pipe size schedule 80	in.	12
Diameter OD	in.	12.75
Wall thickness	in.	0.687
Diameter ID	in.	11.376
Maximum allowable pressure (ASA-B31.1-1951)	psi	1500
Insulation thickness	in.	4

4. Control Rod Drive Mechanism

Total travel	in.	22
Rod speed, either direction	in./min	3.14
Fineness of position control		0.01 $\pm$.01
Motor - 1/8 hp, 440 v, 3 ph., 60 cy., 1080 rpm		
Clutch - Warner SF 400 size Power supplied from AC line through rectifier		
Capacity - 0 rpm	ft-lb	20
Time to release	m sec.	60
Location of clutch, on low speed Shaft outboard of seal and position indicator.		

Reduction gearing-totally enclosed,
multi stage planetary, integral with
motor.

ratio

2250:1

Seal type, mechanical breakdown, Labyrinth

Average leakage

lb/hr

2-1/2 - 3

Operating friction

in. -lb

1.4

Break away friction, max.

in. -lb

12

Rack and pinion, pitch

20

Material, 410 stainless steel

Face width

in.

0.75

5. Primary Blowdown and Primary Make Up System

Demineralizer system, cartridge type,
single column, mixed bed.

Number installed

2

Number required for operation

1

Capacity, each

gph

200

Effluent impurities

ppm

0.5 approx.

Inlet temperature, max.

°F

190

Design pressure

psi

100

Primary coolant feed pump, duplex type

Number installed

2

Number required for operation

1

Capacity, each

gpm

1.5

Discharge pressure

psig

1500

Seal leakage return pumps

Capacity

gpm

1.5

Discharge pressure max.

psig

200

Motor size	hp	1/2
Number installed		2
Number required for operation		1

6. Shield Design

Primary Shield:-

Inner radius	in.	32-1/4
Outer radius	in.	80-1/2
Height	in.	248-1/4
Inner wall-steel Thickness of each ring	in.	2
Spacing between rings	in.	1
Height, inner ring	in.	105
Water-Specific gravity min.		0.97

Top Shield

Water thickness	in.	144
Water-specific gravity min.		0.97

Secondary Shield:-

Vapor Container		
Steel shell thickness	in.	0.875
Concrete wall thickness	in.	24
Specific gravity	g/cc	2.3
Additional concrete shielding Thickness	in.	36
Height above grade level	ft	27

<u>Fuel element shielding, height of water above fuel element transfer tube opening.</u>	ft	12.3
--	----	------

Dose rates		
Tolerance dose rate for 40 hr wk	mr/hr	7.5

Control room, reactor operation	mr/hr	design 1.5	measured << 1
Above vapor container, reactor operation	mr/hr	150	< 1
Surface of primary shield, 24 hr. after shutdown	mr/hr	1.0	1.5
during operation	r/hr	-	3.3
Adjacent to primary piping or steam gen. 24 hr. after shutdown	mr/hr	22	65-330
during operation	r/hr	-	17

7. Steam System

Turbine generator: condensing turbine,
gear driven generator

		1
Steam to throttle	psia	213
Exhaust pressure	in. Hg	2.4
Rating, at 0.80 power factor	kw	2005
Voltage	v	4160
Frequency	cps	60
Exciter, direct connected		1
Generator, enclosed, air cooled with integral air cooler		1
Extraction nozzle, at 3030 lb/hr.	psia	31.6
Steam to throttle, full load	lb/hr	34,570
Turbine efficiency, full load	%	65.5
Generator efficiency, full load	%	98.5
Gears, bearings and windage efficiency	%	98.5
Voltage controls, automatic		1
Overspeed trip, operates	rpm	6050
Low steam pressure	psia	145

Condenser horizontal, shell and
tube type, two-pass, divided flow
construction

		1
Heat transfer	Btu/hr	27.6×10^6
Steam condensed, full load	lb/hr	30,980
Condensed coolant temperature, max.	$^{\circ}\text{F}$	96
Tubes, 18 gage, number		936
Tubes OD	in.	7/8
Design pressure, tube side	psig	60
shell side	in./Hg	Full vacuum

Air removal equipment

Twin jet, two-stage air ejectors

Number installed		2
Number required for operation		1
Operating steam pressure	psia	150
Steam required, full load, one set	lb/hr	240
Dry air evacuated, full load	lb/hr	18
Total load evacuated, air and water vapor	lb/hr	60

Inter & after condensers, each

		1
Number of tubes, each condenser		60
Size of tubes, 18 gage OD	in.	5/8
Cooling water flow rate, min.	gpm	20

Feedwater heater, shell and U-tube type, closed

		1
Number of tubes		56
Size of tubes OD	in.	5/8
Feedwater flow rate	gpm	75
Feedwater inlet temperature	$^{\circ}\text{F}$	118.2
outlet temperature	$^{\circ}\text{F}$	255.8

Shell design pressure	psig	160
temperature	°F	450
Tube design temperature	°F	300
<u>Evaporator, two-pass, de-aerating, U-tube type</u>		1
Evaporator feed pump discharge	psig	46
Feed pump motor	hp	5
Steam inlet pressure	psig	175
Vapor outlet pressure	psig	15.2
temperature	°F	250
Distillate solids	ppm	1
Evaporator capacity, total	lb/hr	2150
Pre-heater bleed	lb/hr	400
Net capacity	lb/hr	1750
Normal full load capacity	lb/hr	1050
<u>Boiler feed pumps:</u> - Two motor driven, One turbine driven:		
Number installed		3
Number required for operation		1
Speed	rpm	3550
Capacity, each	gpm	75
Head at 75 gpm	ft-H ₂ O	650
Suction temperature	°F	108.7
Pump efficiency	%	66
Motor, rated each	hp	20
Rated voltage 3-phase	v	440
<u>Auxiliary boiler feed pump</u>		
Turbine driven, centrifugal type		
Speed	rpm	3550
Capacity	gpm	75

Head	ft-H ₂ O	650
Water temperature, suction	°F	108.7
Steam inlet pressure	psig	185.3
Turbine back pressure	psig	5.
<u>Circulating water pump, condenser</u>		
Number installed		2
Number required, normal operation		1
Type vertical, single-stage centrifugal, motor driven		
Speed	rpm	1185
Capacity, each	gpm	4500
Head	ft-H ₂ O	50
Inlet temperature	°F	32-96
Discharge pressure	psig	16-20
<u>Condensate recirculating water pumps</u>		
Type vertical, 3-stage centrifugal, motor driven.		
Number provided		2
Number required, normal operation		1
Motor rated, each	hp	10
Pump discharge pressure at 50 gpm	psig	90
<u>Blowdown cooling water pumps</u>		
Type horizontal, single-stage centrifugal, motor driven		
Number provided		2
Number required, normal operation		1
Motor rated, each	hp	5
Pump discharge pressure	psig	45

Cooling Water Booster Pump

Type horizontal, single-stage centrifugal,
motor driven

Number provided		2
Number required, normal operation		1
Motor rated, each		5
Pump discharge pressure	psig	35

Air Compressors

Number installed		2
Number required normal operation		1 (main)
Type (main), horizontal, double acting, single-stage, reciprocating, bolt driven		
Motor rating	hp	15
Compressor rating, at 100 psig	cfm	50
Type (standby), two-stage, single acting, reciprocating, "V" unit belt driven		
Motor rating	hp	5
Compressor rating at 100 psig	cfm	18.3

8. Power Plant Building (Main)

Overall dimension	ft	27 x 68 x 47
Dimension of control room and storage wing	ft	18 x 22 x 35
Crane capacity	tons	10

9. Scram and Alarm Settings

A. Plant Malfunctions Causing Reactor Scram

<u>Cause</u>	<u>Scram</u>
Period, short	3 sec.
Pressurizer pressure, low	1040 ₊ 15 psig
Pressurizer pressure, high	1455 ₊ 30 psig
Primary coolant flow, low	3500 gpm
Primary coolant temperature, high	480°F
Reactor flux, high	130%
Reactor power level above zero power run safety limit	200 watts
Steam generator pressure, high	550 psig
Steam generator temperature, high	480°F
Manual Scram	by operator

B. Primary System Malfunctions Causing Alarm

<u>Cause</u>	<u>Alarm occurs at</u>
BF ₃ Channel (#1) Log Count rate reorder	off scale
BF ₃ Channel (#2) Log Count rate reorder	off scale
Ion Chamber voltage, low	300 v
Log N and Period Amplifier	Trouble in amplifier
Period, short	10 sec.
Pressurizer level, high	22.5 in.
Pressurizer level, low	16 in.
Pressurizer pressure, high	1325 ₊ 15 psig
Pressurizer pressure, low	1140 ₊ psig

Pressurizer pressure, low, scram	Switch in by-pass position
Primary blowdown coolant pressure high	55 psig
Primary blowdown coolant radiation level, high	7.5 mr/hr
Primary blowdown to demineralizer conductivity, high	5 μ mho.
Primary coolant flow, low	3700 gpm
Primary coolant temperature, high	460°F
Primary make-up tank level, low	2700 gal
Primary make-up tank level, high	4500 gal
Primary make-up tank pressure, low	10 psi
Safety amplifier #1	Trouble in amplifier
Safety amplifier #2	Trouble in amplifier
Seal leakage tank level, high	22 in.
Seal leakage tank level, low	8 in.
Vapor container pressure, high	20 psia

C. Secondary System Malfunction Causing Alarm Only

<u>Cause</u>	<u>Alarm occurs at</u>
Condensate conductivity, high	5 μ mho
Condensate radiation level, high	7.5 mr/hr
Condenser hot well level, high	3 in. above normal
Condenser hot well level, low	3 in. below normal
Distilled water tank level, high	4500 gal
Distilled water tank level, low	2800 gal
Evaporator shell level, high	4 in. above normal
Evaporator shell level, low	4 in. below normal

Feedwater heater shell level, high	2 in. above normal
Feedwater heater shell level, low	2 in. below normal
Service water tank level, high	4500 gal
Service water tank level, low	3500 gal
Steam generator blowdown radiation level	10 mr/hr
Steam generator level, high	4 in. above normal
Steam generator level, low	4 in. below normal
Steam generator, steam from, high temperature	445°F
Steam generator pressure, high	435 $\pm$ 15 psig

D. Electrical System Malfunctions Causing Alarm Only

<u>Cause</u>	<u>Alarm Occurs At</u>
Battery charging motor generator set	Breaker trip
Bearing oil pressure low	8 psig
Emergency lighting on	Breaker closing
Generator air temperature high	52°C
* 10% generator differential currents or generator ground	Generator auxiliary lockout trip
* Line over-currents 200%-0.87 sec	Line auxiliary lockout trip
Primary coolant starter circuits de- energized	Breaker open - pc pumps, heaters or rod drives
* Station service feeder over-currents 193%- 2.4 sec 1200% instantaneous	St. serv. #1 auxiliary lockout trip St. serv. #2 auxiliary lockout trip

Station service motors over current

Station service throwover

*Station service undervoltage 99 v slow close,
51 v 1.1 second close

VEPCO input low

Breaker trip

Automatic throwover
station service buss

Operation of buss A
or buss B under
voltage relay auxiliary

0.5 mw

* These items initiate breaker trip, are not alarm only

E. Miscellaneous Malfunctions Causing Alarm Only

Cause

Alarm Occurs At

Chlorine gas pressure, low

20 psig

Chlorinization room (loss of vacuum)

3 psig control pressure

Control air pressure low

70 psig

Control room radiation level, high

7.5 mr/hr

Demineralizer area radiation level, high

7.5 mr/hr

Hot waste tank pit level, high

12 in.

Hot waste tank pressure, high

12 lb

Hot waste tank pressure, low

2 in. vacuum

Injector water pressure, low

20 psig

Seal pit discharge radiation level, high

7.5 mr/hr

Spent fuel pit area radiation level, high

7.5 mr/hr

Spent fuel pit level, low

24 in. below pit top

Stack effluent radiation level, high

7.5 mr/hr

Vapor container entrance radiation, level

7.5 mr/hr

Vapor container interior radiation, level

75 r/hr

Vapor container manhole water level, low

18 in. below top of
stand pipe

Vapor container sump level, high

3 ft 3 in. from top

CHAPTER II - REACTOR

A. CORE

1.0 GENERAL DESCRIPTION

The SM-1 core consists of a 7 x 7 array of fixed and control rod fuel elements with the four corner elements missing as shown in Fig. II-1. The core consists of 38 fixed fuel elements and seven control rod assemblies. Fig. II-1 shows a plan view of the core layout. The fuel elements contain uranium highly enriched in the U-235 isotope in the form of UO_2 . The fuel elements consists of fuel plates containing a dispersion of UO_2 and B_4C in a stainless steel matrix clad with 5 miles of stainless steel. The fixed fuel plates are arranged in sub-assemblies containing 18 fuel plates brazed into side plates as shown in Fig. II-2a. The control rods consist of a fuel element and absorber section inserted in a tube such that insertion of the absorber section displaces a fuel element as shown in Fig. II-3. The control rod fuel elements consist of 16 fuel plates per element as shown in Fig. II-2b. The absorber section is shown in Fig. II-2c.

2.0 HISTORY OF SM-1 CORE I

The SM-1 Core I first reached power in April 1957. At the end of 2/3 core life (approximately 10.5 MWYR) the following changes were made:

1. Absorbers containing boron were replaced by absorbers containing Eu_2O_3 in the five rods normally inserted in the core (rods 1, 2, 3, 4 and c).
2. One fixed element⁽⁵⁷⁾ in a position adjacent to the reflector was replaced by a new Core I element
3. One control rod element was replaced by a new Core II control rod element.

The removed core components were studied under water in the upper shield tank of the SM-1 with the aid of a periscope. Both of the removed fuel elements and two of the absorber sections were then sent to ORNL for hot cell and metallographic examination.

FIG. II-1 REACTOR CORE CROSS SECTION

II-5

Fig. II-2b
Control Rod Fuel Element Core II

PIN
— A9-13-2019
I REQUIRED

— ASSEMBLED SECTION MUST FIT WITHIN
A THEORETICAL STRAIGHT BOX OF THESE
DIMENSIONS 26 $\frac{1}{8}$ LONG WITHOUT INTERFERENCE

II-9

CONTROL ROD ASSY - 72[#] (ESTIMATED WEIGHT)

Fig. II-3
Control Rod Assembly Core II

Work to date is reported in ORNL-2907 (8) Although a number of suspected cracks were reported in the preliminary examination at the reactor site, close examination of both fuel elements at the ORNL hot cells revealed the suspected cracks in the cladding to be only superficial scratches. No evidence of cracking was found in the fuel elements. Results of dimensional measurements on both fuel elements revealed no evidence of gross dimensional changes. A plate rippling effect was noted, but it was generally well within manufacturing tolerances, except for a limited area on the high burn-up side of the stationary fuel element. Plate spacing and fuel plate thickness measurements also showed no evidence of gross dimensional changes. Burn-up analysis and metallographic examinations are now in progress.

To date, the only deleterious effect found was some corrosive attack on the braze metal fillets bonding the fuel plates to the side plates. However, it was insufficient to affect operational integrity. The remainder of the original SM-1 Core I stainless steel fuel elements have operated satisfactorily for more than 16 MWYR of service. Continued use of the stainless steel dispersion type of fuel element is definitely recommended, based on the satisfactory performance of both fuel elements and test specimens.

Post irradiation examination of two B_4C in iron dispersion absorbers showed loss of bonding and cracking at the high-flux end of the plates. The damage was not unexpected, since similar failures occurred with experimental test coupons. Loss of core-to-cladding bonding, along with subsequent gross swelling from gas accumulation, was shown to occur at burnups above about 17 a/o B-10 in the experimental coupons. The results of the examination of the SM-1 absorbers are in general agreement with the experimental results. Only one absorber plate having a maximum burnup of 14.5 a/o B-10 showed no evidence of loss of core-to-cladding bonding. The remaining seven plates, subjected to maximum burnups of greater than 18 a/o B-10, all showed loss of bonding. However, the failure of the boron absorbers did not affect reactor operation or maintenance up to the time of removal.

The failures were due to gas pressure resulting from the $n-\alpha$ reaction of the B-10. Subsequent absorber design for SM-1 Core II utilizes Eu_2O_3 , as the absorber. The $n-\gamma$ reactions characteristic of Eu and its daughters will not result in the formation of gas, thus eliminating the cause of failure of the absorbers used in Core I. Tests of a series of miniature samples have been initiated to obtain information on the effects of irradiation on Eu_2O_3 dispersions in the Type 304 stainless steel. No failures have been experienced to date.

Five of the europium oxide bearing absorbers originally fabricated by ORNL for use in SM-1 Core II were inserted in the SM-1 Core I to replace the boron containing absorbers at the time of core inspection. To date, these europium absorbers have accumulated approximately 6 MWYR of service, with no operational difficulties. It is anticipated that europium oxide absorbers can be used for two core lives because of the successive formation of high cross section daughters under irradiation.

3.0 SM-1 CORE I AND II DIFFERENCES

The SM-1 Core I control rod fuel elements employed Haynes combs to suppress high thermal neutron peaking that extends into the top of the control rod fuel element. Since this type of flux suppressor is undesirable because of activity buildup from its high cobalt content, the SM-1 Core II control rod fuel elements will employ flux suppressors consisting of Eu_2O_3 dispersed in stainless steel within the fuel plate (Fig.II-2b). Measurements indicate the internal suppressors adequately suppress the flux. The absorber sections of Case I contain boron highly enriched in the B-10 isotope in an iron matrix clad with stainless steel whereas Core II (Fig.II-2c) will contain europium in the form of Eu_2O_3 dispersed in a stainless steel matrix and clad with stainless steel. Measurements indicate negligible difference between the worth of the Eu and B-10 rods.

The core support structure (Fig.II-11) contains the source tube and a 3 x 3 x 0.5 in. beryllium block. Case I used a Po-Be startup source with an initial strength of 15 curies. A 45-curie Po-Be source plus an Sb-Be source designed to produce a neutron source equivalent to the 45-curie source at saturation will be employed with Core II. Fig.II-4 shows a plot of neutron production as a function of operating time for the Core II source. This change in source strength will increase the startup count rate to greater than 10 cps.

4.0 NUCLEAR CHARACTERISTICS

The nuclear characteristics of SM-1 Core I are well established by measurements made during the burnout of Core I. Since SM-1 Core II is identical to SM-1 Core I in U-235 and B-10 loading the nuclear characteristics are expected to be practically identical. The measurements made on Core I apply directly to Core II. Measurements of temperature coefficients at several periods during the life of Core I are shown in Fig.II-5. The values are averages of coefficients evaluated in increments of approximately 25°F.

FIGURE II-4

No significant change in the value of the temperature coefficient at 440°F with core life is indicated. Measurement at 7.2, 9.1 and 12.1 MWYR indicate a value of -3.5 cents/°F at 440°F. The integrated reactivity change from 70°F to 440°F evaluated from the temperature coefficient is 6.63 dollars.

The five-rod bank calibration is shown in Fig. II-6. Measurements include points obtained several times during core life. Data obtained at 12.1 MWYR clearly indicate an increase in bank worth over that expected from a calibration in a uniformly burned out core. This is believed to be due to the shielding of the top of the core due to the bank insertion. Fig. II-7 shows the integrated five-rod bank worth as a function of bank position.

The total reactivity change as a function of the five-rod bank position is shown in Fig. II-8. The curve was obtained using an effective delayed to prompt neutron ratio, β , of 0.0073.

Fig. II-9 shows the five-rod bank position as a function of MWYR, the energy release, for peak Xe (440°F), equilibrium Xe (440°F), no Xe (440°F) and no Xe (70°F).

The pressure-temperature relationship using known values of temperature coefficients yielded values for the pressure coefficient. The relationship is based on the assumption that the temperature coefficient is caused entirely by the change in the density of the moderator. The measured value for the pressure coefficient at 115°F was $1.05 \pm .05$ cents per 100 psi. The computed values are 1.15 cents per 100 psi at 104°F and 2.9 cents per 100 psi at 440°F. The computations were based on a pressure of 100 atmospheres.

The plot of pressure coefficient as a function of pressure is shown in Fig. II-10. The measurements were made by maintaining criticality with a calibrated rod while changing the pressure. Integral measurements were made by quickly increasing the pressure and measuring the resulting period.

FIGURE II - 6 FIVE ROD BANK CALIBRATION

SM-1 CORE I

SYMBOL	SOURCE OF DATA	MWYR
○	ROD C VS TEMP (90°F - 400°F)	0
□	ROD A	6.1
△	ROD A	7.2
⊗	XENON	7.2
x	SUBCRITICAL	7.2
▽	ROD A	9.1
⊞	ROD C	9.1
⊕	XENON	12.1
⊞	ROD A	12.1
④	TEMPERATURE	12.1

FIGURE II-7 INTEGRATED FIVE ROD BANK WORTH AS A

FUNCTION OF BANK POSITION

SM-1 CORE I

FIGURE II-8 INTEGRATED REACTIVITY COEFFICIENT AS A

FUNCTION OF FIVE ROD BANK POSITION

SM-1 CORE I

FIG. II-9
FIVE ROD BANK POSITION AS A FUNCTION OF
ENERGY RELEASE SM-1 CORE1

FIGURE II-10 PRESSURE COEFFICIENT (NO XE; 115°F; 4.79 MWY)

TABLE II-1
SM-1 CORE I AND II NUCLEAR CHARACTERISTICS

1.0 Geometry (Dimensions of core & component parts)

1.1 <u>Overall</u>		<u>Core I</u>	<u>Core II</u>
Configuration		7 x 7 with corners missing	7 x 7 with corners missing
Equivalent diameter	(in.)	22. 2	22. 2
Active core height	(in.)	21. 75	21. 75
Total no. of elements		45	45
1.2 <u>Fuel Elements (Fixed)</u>			
No. of fixed elements		38	38
Active length	(in.)	21. 75	21. 75
Total length	(in.)	33. 625	33. 625
Cross section	(in.)	2. 863 x 2. 844	2. 863 x 2. 844
Shipping length	(in.)	33. 625	33. 625
Lift handle on assembly		Dowel pin on upper end box	Dowel pin on upper end box
1.3 <u>Fuel Elements (Control rod)</u>			
No. of control rod elements		7	7
Active length	(in.)	21. 75	21. 125
Total length	(in.)	26. 44	27. 25
Cross section	(in.)	2. 624 x 2. 619	2. 624 x 2. 619
Shipping length	(in.)	26. 625	26. 625
Lift handle on assembly		Handle on upper end	Handle on upper end

1.4 <u>Fuel Plates (Fixed)</u>	<u>Core I</u>	<u>Core II</u>
Type of fuel	UO ₂ (Geneva)	UO ₂ (Geneva)
No. of plates/element	18	18
Size of fuel plates (in.)	0.030 x 2.778 x 23	0.030 x 2.778 x 23
Type of encasement	304L SS	304 L SS (low cobalt)
Wall thickness of encasement (in.)	0.005 ± 0.001	0.005 ± 0.001
Meat width (in.)	2.500	2.500
Meat thickness (in.)	0.020	0.020
Active volume per plate (cu. in.)	1.1	1.1
Size of fuel particles (microns)	44 to 88	44 to 88

1.5 <u>Fuel Plates (Control rod)</u>		
Type of fuel	UO ₂ (Geneva)	UO ₂ (Geneva)
No. of plates/element	16	16
Size of plates (in.)	0.030 x 2.56 x 23	0.030 x 2.56 x 23
Type of encasement	304L SS	304L SS (low cobalt)
Wall thickness of encasement (in.)	0.005 ± 0.001	0.005 ± 0.001
Meat width (in.)	2.281	2.281
Meat thickness (in.)	0.020	0.020
Active volume per plate (cu. in.)	0.96	0.96
Size of fuel particles (microns)	44 to 88	44 to 88
Total cooling area of plate (sq.in.)	1588	1588

2.0 Composition and Loading

2.1 Fuel Element (Fixed)

Core I

Core II

Weight of U^{235} per element (gm)	515.16	515.16
Weight of UO_2 per element (gm)	630.36	631.98
Weight of SS per element (gm)*	4.263	4.355
Weight of B_4C per element (gm)	2.626	2.8215
Weight of B^{10} per element (gm)	0.3605	0.4163
Weight of B^N per element (gm)	1.9926	2.2747

2.2 Fuel Element (Control rod)

Weight of U^{235} per element (gm)	417.76	401.12
Weight of UO_2 per element (gm)	512.16	493.12
Weight of SS per element (gm)*	6,704	6703
Weight of B_4C per element (gm)/	2.132	2.0486
Weight of B^{10} per element (gm)/	0.2926	0.2813
Weight of B^N per element (gm)/	1.6176	1.5549
Total weight of U^{235} in core (kg)	22.50	22.38
Total weight of B^{10} in core (gm)/	15.75	17.79
Enrichment of U^{235} (%)	~93	~93
Total weight of SS in core (kg)	208.92	212.41
Total weight of water in core (kg)	111.08	111.08

* Includes encasement

/ Best estimate of actual boron content in fabricated plates and core.

2.3 Fuel Plate (Fixed)

Core I

Core II

Weight of U^{235} per plate	(gm)	28.62	28.62
Weight of UO_2 per plate	(gm)	35.02	35.11
Weight of SS per plate	(gm)	99.80	104.9
Weight of B_4C per plate	(gm)	0.1459	0.15675
Weight of B^{10} per plate	(gm)	0.02003	0.023126
Weight of B^N per plate	(gm)	0.1107	0.12637

2.4 Fuel Plate (Control Rod)

Weight of U^{235} per plate	(gm)	26.11	25.36	25.07
Weight of UO_2 per plate	(gm)	32.01	111	30.82
Weight of SS per plate	(gm)	91.56	88.93	91.51
Weight of B_4C per plate	(gm)	0.1332	.1274	0.12804
Weight of B^{10} per plate	(gm)	0.01829	.01776	0.01758
Weight of B^N per plate	(gm)	0.1011	.0972	0.09718
Weight of Eu per plate	(gm)	—		1.00

3.0 Nuclear Characteristics

Core life	(MWYR)	16.4	16.4
Heat power	(Mw)	10	10
Time of full power operation (days)		600	600
Average power density (watts/cc)		77	77
Expected total energy release	(MWYR)	16.4	16.4
Average fuel burnup	(at 16.4 MWYR)	0.37	0.37
Maximum fuel burnup	(at 16.4 MWYR)	0.80	0.80

Core I

17-11 =

	<u>Core I</u>	<u>Core II</u>
Initial core reactivity at 68°F (%)	15.4	15.4
Initial core reactivity at 440°F (%)	10.4	10.4
Initial core reactivity at Eq. Xe 440°F (%)	8.2	8.2
Initial bank position at 68°F (in.)	3.7	3.7
Initial bank position at 440°F (in.)	6.7	6.7
Initial bank position at Eq. Xe 440°F (in.)	8.3	8.3
Thermal neutron flux at 0 MWYR, 440°F (n/cm ² /sec)	1.4 x 10 ¹³	1.4 x 10 ¹³
Thermal neutron flux at 15 MWYR, 440°F (n/cm ² /sec)	2.1 x 10 ¹³	2.1 x 10 ¹³
Ratio of centerline power to average power (radially)	1.48	1.48
Ratio of centerline power to average power (axially)	1.89	1.89
Power peak to average at (radially)	1.49	1.49
startup, eq Xe, 440°F (axially)	1.71	1.71
Percent thermal fissions (at startup)	0.75	0.75
below 0.25 ev (at 15 MWYR)	0.81	0.81
Weight of U ²³⁵ at end of life (kg)	14.3	14.3
Weight of B ¹⁰ at end of life (gm)	2.1	2.1
Burnup of control rod material (abs /orig. atom)	0.3	0.3
Temperature coefficient at 68°F (°F) ⁻¹	-0.26 x 10 ⁻⁴	-0.26 x 10 ⁻⁴

	<u>Core I</u>	<u>Core II</u>
Temperature coefficient at 440°F (°F) ⁻¹	-2.2 x 10 ⁻⁴	-2.2 x 10 ⁻⁴
Center line rod worth 68°F (%)	4.1	4.1
Center line rod worth at 440°F (%)	4.1	4.1
Five rod bank worth at 68°F (%)	19.0	19.0
Five rod bank worth at 440°F (%)	19.2	19.2

B. CORE SUPPORT STRUCTURE

The functions of the core support structure, Fig. II-11, are:

1. Locate and orient the stationary fuel elements.
2. Enclose the core so that the entire primary coolant flow is directed through it.
3. Provide the upper guides for the control rod.
4. Provide the dashpot for deceleration of the control rod during scram.

The core support structure consists of an upper mounting flange, upper and lower grid plates, a skirt, four tie rods, pinion bearing support carrier and seven pinion bearing supports. The upper grid plate, in the form of four hinged doors, and the lower grid plate locate, orient and hold the stationary fuel elements. The upper grid plate also provides the upper guides for the control rods. The skirt, which encloses the entire core, is sandwiched between the upper mounting flange and the lower grid plate. The photoneutron source is mounted on the skirt.

The upper mounting flange, lower grid plate and pinion bearing support carrier are fastened together by four tie rods. Bolted to the carrier are the seven pinion bearing supports. The upper part of these supports provides a dashpot cylinder for decelerating the control rod during scram.

The complete core support structure is assembled in the shop and lowered through the upper vessel opening and bolted in place. It requires no additional adjustments after it has been set in the reactor vessel.

C. REACTOR VESSEL

The reactor vessel, Fig. II-12, contains the active core, its supporting structure and thermal shielding. It consists basically of a cylindrical vessel 47-1/2 in. ID, approximately 5-1/2 ft. long, with a 22 in. ID extension approximately 4 ft. long on the bottom. The latter part of the structure houses the control rod racks and their associated pinions and support structures. Wall thicknesses are 2-3/4 and 1-1/2 in. total respectively.

The lower end of the vessel is closed by welding an ellipsoidal head. The upper end of the vessel is closed by a bolted and gasketed dished cover 2-3/4 in. thick. Total height including cover is 13 ft. 6 in. The opening at the upper end of the vessel is 28 in. in diameter to permit insertion of the fully assembled core structure.

Primary water enters and leaves the vessel by two stainless steel nozzles located above the core and near the top of the large cylindrical section. Seven tubes penetrate the smaller diameter cylindrical lower section at right angles to permit connection of the control rod drives. A mounting flange ties these tubes together 41 in. from the centerline of the vessel. The seals and drive mechanisms are bolted to this flange.

Structural support for the pressure vessel is by means of a ring attached to the outside diameter of the cylindrical section just below the inlet and outlet pipes. This ring rests on a support ring welded to the inner steel shielding ring which in turn rests on a concrete structure in the bottom of the vapor container.

The pressure shell and control rod extension is surrounded by thermal insulation 3-5/8 in. thick, which in turn is contained in a water-tight stainless steel shell. Additional insulation is applied to the removable cover at the top of the pressure shell. Provisions are made inside the pressure shell for support of the core structure support and thermal shield which also serves as a flow divider to direct cooling water flow to the core.

The pressure shell is designed and fabricated in accordance with applicable sections of ASME Code, Section VIII, covering Unfired Pressure Vessels, Edition of 1952. The design conditions are based on a maximum

Fig. II-12
Reactor Vessel

temperature of 650°F, and a maximum allowable working pressure of 1600 psig. The temperature and pressure chosen are somewhat higher than usual design practice, but it is felt desirable to design the major structural elements on a conservative basis, partly for safety reasons and partly to permit a margin for variations in operating conditions in order to make the first unit as versatile as possible within reasonable structural and economic limitations.

The material used in construction of the shell of the SM-1 reactor vessel was ASTM A-212 B carbon steel plate, clad with 1/4-in. type 304 stainless steel. At the time of material selection, preliminary evaluation of available data indicated that ASTM A-212 B would be capable of withstanding exposure of $4-5 \times 10^{19}$ nvt, based upon a design life of 20 yr.

A literature survey of all available data was conducted as a part of the SM-2 Core and Vessel program with the objective of evaluating the effects of neutron irradiation on the properties of commercially available steels. This work was reported in APAE 61. (27) The data available were found to consist mainly of sub-size Izod impact specimen results. Some full size Charpy data and some tensile specimen data are also available, but NDT (Nil Ductility Test) data after irradiation are practically non-existent. Although the NDT concept is the most accurate method of predicting brittle behavior, the scarcity of post-irradiation data at the present time makes it necessary to base evaluation upon the most significant means available, i. e., sub and full size impact specimen data.

Industrial experience with non-nuclear pressure vessels has indicated that transition temperatures and transition temperature increases, determined from impact data, offer a valuable tool in predicting brittle or non-brittle behavior. Evaluation of transition temperature increases as a result of irradiation, together with associated property changes such as decrease in maximum energy absorption, offers a reliable means of prediction of reactor vessel performance. Since transition temperature is defined as that temperature characteristic of change from desirable plastic behavior to undesirable brittle behavior, attainment of temperature above transition, with a sufficient margin of safety assured by correlation with the NDT concept, allows conservative prediction of ductile behavior.

The literature noted above indicated that an increase in transition temperature of about 20°F could be expected for A-212 B after exposure to 7×10^{18} nvt, corresponding to 3 yr operation at 60% load factor. Similarly, an increase of about 175°F can be predicted after exposure to 5×10^{19} nvt, corresponding to 20 yr operation at 60% load factor. These predictions of transition temperature increases are conservative because no credit was allowed for the possible beneficial effect of elevated vessel temperature during exposure.

The increase in transition temperature suffered by the SM-1 reactor shell materials as a result of exposure to Core I is practically negligible. The increase expected after 20 yr of operation, however, is of sufficient magnitude that brittle failure might be expected under adverse conditions, unless planned precautions are taken. Brittle failure could conceivably result if stress sufficient to cause localized yielding in an area containing notch defects were applied when the vessel material was in the temperature region of brittle behavior.

This problem will be avoided by the use of the present special startup and shutdown procedures which are designed to prevent application of significant stresses while the vessel temperature is in a low temperature range possible conducive to brittle failure. Ductile behavior would still be expected at vessel operating temperature after 20 yr. of operation.

Based on evaluation of available data, satisfactory performance of the SM-1 reactor vessel for at least 20 yr. can be predicted, provided that adherence to established start-up and shutdown procedures is maintained

D. CONTROL ROD ASSEMBLY

The control rod assembly, Fig. II-3, consists of a square tube housing a fuel element and absorber. The lower end of the tube is welded to the decelerating piston. The rod drive rack is oriented and bolted to this piston. The upper end of the tube contains the cap assembly which holds the fuel element and absorber sections in place. The control rod runs vertically thru the core. It is guided at the top by guide rods in the top grid of the core structure support and at the bottom of the pinion and rack backup roller.

The fuel element is of similar design to the stationary fuel element mirror end adapters. A lifting handle is provided at the upper end for removing and loading of the element.

The absorber section, Fig. II-3, which is located above the fuel element in the tube, consists of four rolled and clad plates welded together at the corners to form a square box. A pin is provided in the upper end for removal and reloading this element.

The control rod cap Fig. II-3, is designed to achieve two main objectives: (1) to make it impossible for the cap to ever come free during rod operation, and (2) to prevent possible misapplication of either the absorber section or the fuel element, or both, during reloading. A special installation tool cannot

be removed from the cap unless the fuel element and the absorber section are in place, in the proper order and position and the cap securely locked. Additional weight has been added in the cap to provide sufficient weight in the complete control rod assembly to insure adequate acceleration during scram. An additional positive driving force is available through the use of the overrunning clutch in the rod drive in case of rod sticking.

A decelerating piston is provided at the lower end of the control rod tube to cushion the shock of too rapid deceleration at the bottom of travel when the rods are scrammed. Holes have been provided for in the piston and the upper section of the tube to provide proper flow of coolant through the rod.

E. CONTROL ROD DRIVES

The control rod drives have been designed to provide as positive a mechanism as possible. The drives are located at the bottom of the pressure shell. They are raised to increase activity and lowered to reduce reactor output. Thus, gravity can be used as the major motivating force for emergency insertion of the control rods to scram the reactor. Control rod travel under motor control is 3.14 in. per min.

The drive train, Fig. II-13, consists of a rack attached to the bottom of each control rod, a pinion meshing with this rack and a drive shaft attached to each pinion extending outward through enclosing tubes where connection is made through a magnetic clutch to a motor and reduction gear drive.

Penetration of the pinion shaft through the pressure-containing wall is by means of a mechanical breakdown labyrinth seal. This unit consists of a series of floating rings surrounding the shaft to be sealed, which provide a very close-clearance annular leakage path from the pressure volume to the outside. There is a small controlled amount of leakage as a result of this close clearance. The average leakage per seal is 2-1/2-3 lb. per hr. The construction of the unit is such that virtually no rubbing occurs between the sealing elements, so that wear is extremely small.

Make-up water for the primary system is introduced at the high pressure end of the seal. This arrangement reduces the problem of disposal of such seal leakage as does occur, since this leakage is largely confined to uncontaminated water which is then purified and returned to the primary make-up tank.

Provision is made in the design to remove and replace the complete seal unit without draining the primary system, thus minimizing the hazards associated with handling this large quantity of contaminated water, and making this replacement a simple, routine maintenance procedure. Working area in which this operation and other maintenance operations on drive motors, etc., are performed is shielded for adequate protection of maintenance personnel. Tests at SM-1 have demonstrated that the drive can be changed in 46 min with the system at full pressure (5)

The actuating mechanism comprises two major assemblies for each control rod. One of these, consisting of the rack and pinion, associated bearings and spherical valve, operated in the primary system water and is inside the pressure vessel. The other, connected to the first through the seal discussed above, includes the drive motor with integral brake and reduction gear box, electric magnetic and overrunning clutches, rod position indicator and limit switches. These units are outside the pressure vessel, but within the vapor container. They are of conventional design and are conventionally lubricated, inspected and maintained.

The integral brake is used to permit fine positioning of the rods by eliminating any possible overrun caused by coasting of the drive gear train. Operating experience has shown that overrun is negligible due to the large gear reduction. During rod motion with the brake disconnected, a small amount of anticipation on the part of the operator is required to stop the rods at a particular point. However, plant safety is not affected by the amount of overrun experienced at SM-1.

Emergency insertion of the control rods is performed by de-energizing the magnetic clutch. This unit is a standard commercial item manufactured by the Warner Electric Clutch and Brake Company, and has a torque rating approximately three times that necessary to operate the rods. Release time of the clutch is 0.060 sec max. The total weight of the control rod assembly is approximately 72 lb.

The drive motor is connected into the system in such a way as to start rotating in the down direction when the scram signal is given. If for any reason the rod should fail to fall of its own weight, the full motor torque is applied through the overrunning clutch to break it free so that, once free, it can fall by gravity. This overrunning clutch transmits motion directly through mechanical connection and will operate without slippage. Also in the scram condition the magnetic clutch is bypassed and the cam type overrunning clutch is used.

Rod position indication is provided by means of two synchros per drive which are geared to the pinion shaft on the pinion side of the clutch. Upper and lower limit switches are actuated by a traverse nut which is driven by a threaded shaft geared to the pinion shaft on the pinion side of the clutch. This system facilitates inspection and maintenance. The alternative of inserting the synchros in the reactor vessel is less reliable because of the necessity for sealing numerous electrical penetrations and prohibiting maintenance unless the primary system is completely shut down.

The instrument drive and the clutch drive are individually complete assemblies easily removable without disturbing adjoining drives. Maintenance problems will be at a minimum since all parts are completely interchangeable.

F. PRIMARY SHIELD

Fig. II-14 is an elevation drawing of the reactor and primary shield with a plan view in Fig. II-15. Outside of the pressure vessel are 3-5/8 in. of insulation held in place by a 1/4-in. shell of steel. After a small air gap there is the inner wall of the shielding vessel which is a 2-in. thick steel cylinder. This inner wall of the shield tank also supports the pressure vessel below the inlet and outlet pipes. The shield tank is 48-1/4 in. thick and is filled with water. Seven steel cylinders 2-in. thick are arranged concentrically around the inner wall of the tank, with 1 in. of water between adjacent cylinders. These seven layers of steel, plus the 2-in. inner wall of the tank, give a total of 16 in. of steel radially around the reactor outside of the pressure vessel and constitute the primary gamma shield.

Above the pressure vessel, the inner wall of the shield tank forms a well, 12 ft. deep, which is filled with water. Around the inlet and outlet pipes to the reactor pressure vessel there are 4-in. of insulation which is a void in the shield. Since the pipes are located above the core and turn at right angles within the shield tank, direct streaming of radiation through the annular voids cannot occur. To prevent scattered radiation from penetrating the shield, a ring of steel 7-in. thick is located around the pipe close to the pressure vessel as shown in Fig. II-15.

G. REACTOR INSTRUMENTS

Five ionization chambers and two BF₃ counters are used on the SM-1. The chambers are mounted next to the inner wall of the shield tank as shown in Fig. II-15. In the first critical experiment at SM-1, neutron counters were mounted temporarily within the pressure vessel next to the core.

Fig. II-14
Primary Shield - Elevation

FIG. II-15 - REACTOR AND PRIMARY SHIELD - PLAN VIEW

There are three uncompensated ionization chambers which serve to actuate the safety level scram system. One compensated ionization chamber is connected to the log N and period meters. The second compensated ionization chamber provides a signal to the linear power channel.

The control and safety circuitry is described in greater detail in Chapter VI.

CHAPTER III - POWER PLANT

The SM-1 is a nuclear power plant capable of producing 1855 net kw (at 80 percent power factor) of electrical energy. The plant consists of two systems; the primary or pressurized water system and the secondary or steam system, together with auxiliaries required to insure proper functioning of the two systems as an integrated plant to produce power. The fundamental flow diagram is shown on Fig. III-1, and the heat balance on Fig. III-2a, 2b and 2c. The primary system equipment; namely, the reactor, pressurizer primary coolant pumps and steam generator located in the vapor container are shown on Fig. III-3 and III-4.

A. PRIMARY SYSTEM

The primary system consists of the following: reactor core with control rods, reactor vessel to enclose the core and provide attachments for the control rod drives, two primary coolant circulating pumps, the tube side of the steam generator, a pressurizer with electric heaters rated at 37 kw total heat output. These components are connected by the primary coolant piping system which contains the necessary instrumentation to transmit flow, pressure and temperature data to the control room. A primary blow-down system and primary make-up system is provided for purifying the primary coolant water. The high pressure-high temperature portion of the system is enclosed in a vapor-tight pressure vessel.

The water in the primary system is maintained at 1200 psia to preclude boiling in the reactor. This water enters the reactor at 427.7°F and leaves at 448°F when operating at full load. The electric heaters in the pressurizer maintain pressure at 1200 psia by keeping the pressurizer water at 568°F which generates a steam bubble in the top section of the pressurizer. Over-pressure protection of the primary system is provided by a safety valve located on top of the pressurizer and set to relieve at 1500 psia. The water level in the pressurizer may be automatically controlled by the pressurizer level controller which actuates the controls of the primary make-up pump to increase or decrease pump speed. The quantity of make-up water required is determined by the amount of primary blowdown which is manually controlled.

The primary system water is circulated through the reactor and steam generator at a rate of 3860 gpm by either of two primary coolant pumps. The primary coolant pumps are of the canned motor type to provide zero seal leakage at pump shaft. The motor windings are indirectly cooled by condensate cooling water circulating through coils located in the stator. An external safety trip shuts down the pump in case of failure of the cooling water supply. This in turn scrams the reactor.

The two-way valve or Y valve is a specially designed unit which connects the outlets of the two primary coolant pumps to the primary piping system. This valve automatically permits flow from whichever pump is operating. It also permits a back flow estimated at 5% through the inoperative pump to keep it at operating temperature. This avoids thermal shock to the inoperative pump when it is placed in operation and prevents a slug of relatively cold water entering the reactor core from the inoperative leg in case this pump is started.

The steam generator is the vertical-type heat exchanger with a bolted head located on the bottom end to enclose the tube ends. It has a U-tube bundle of stainless steel Type 304 through which the primary coolant is circulated. On the shell side of the tubes a shroud divides the tubes into a vaporizer section and a superheater section. Feedwater enters the shell through a 2-1/2 in. pipe connection in the side of the generator, mixes with the fluid around the shroud, and flows down and under the shroud into the vaporizer section. Heat transferred through the tube walls from the primary coolant vaporizes the feedwater. The vapor flows upward through an external centrifugal moisture separator. A dip tube returns the separated moisture to the vaporizer section. The dry steam then flows through the superheater section. The superheated steam flows under the superheater shroud and out of the steam generator through an 8-in. pipe outlet. A 1-in. drain connection at the bottom of the superheater section is provided to remove any condensate that may collect in the superheater.

A safety valve set to relieve at 500 psia protects the steam generator and the high pressure section of the secondary system against over-pressure in the system. The safety valve is located in the main steam pipe within the vapor container.

The discharge from the steam generator safety valve and also from the safety valve on the pressurizer is exhausted within the vapor container. Water and other liquids discharged within the vapor container are collected in the rod drive pit where they may be pumped out by the sump pump to the demineralizer room for disposal or storage in the hot waste tank if the radioactivity is excessive.

All incoming water lines are provided with a check valve and a manually operated shut-off valve. The outgoing water and steam lines (and including the hydrogen line to the seal leakage tank) are equipped with an electropneumatic trip valve paired with a manually operated shut-off valve. The check valves, trip valves and manual shut-off valves are located outside and near to the wall of the vapor container. The trip valves are automatically closed by a 5 psi pressure increase within the vapor container. The trip valves are fail-safe type and will trip close on loss of electric power. The check valves and trip valves are needed to seal the vapor container in the event of the rupture of a vessel or a nuclear incident to prevent radioactive material from entering the

FIG. III-1 FUNDAMENTAL FLOW DIAGRAM SM-1

NET STATION HEAT RATE:
 EXCLUDING AIR COND. = $\frac{36,159,000}{2000} = 18,080$ BTU PER KW HR
 CORRECTED FOR 20 KW
 AIR COND. LOAD = $\frac{36,159,000}{2020} = 17,900$ BTU PER KW HR

FIG. III-2a HEAT BALANCE RUN 1

FIG. III-2b HEAT BALANCE RUN 2

NET STATION HEAT RATE:
EXCLUDING AIR COND. = $\frac{36,304 \times 10^6}{1900 \text{ KW}}$ BTU/HR = 19,107 BTU KW-HR

CORRECTED FOR 20 KW
AIR COND. LOAD = $\frac{36,304 \times 10^6}{1920}$ = 18,908 BTU KW-HR

FIG. III-2c HEAT BALANCE RUN 3

ELEVATION

VAPOR CONTAINER ARRANGEMENT

SM-1 PLANT

Fig. III-3

Equipment Arrangement in Vapor Container - Elevation

PLAN VIEW

VAPOR CONTAINER ARRANGEMENT

SM-1 PLANT

Fig. III-4

Equipment Arrangement in Vapor Con-
tainer - Plan

plant area through the piping system. The manual shut-off valve provides a back-up protection in case a valve does not close leak tight.

B. PRESSURIZER

The pressurizer consists of a 27.6 cu-ft capacity carbon steel spherical steam dome welded on top of a 16-in. ID carbon steel vertical cylinder. The inside diameter of the steam dome is 45 in. and its wall thickness is 1-3/8 in. (without cladding). The vertical cylinder is closed at the bottom by a steel disc; stud bolted to 4-1/8 in. carbon steel flange. The steel disc contains the penetrations for 30 electric heaters with total heat output of 37kw. The heater elements are wired in "Y" in groups of three for 3 phase power. Five groups of heaters are supplied power from Buss No. 1 and the other five groups of heaters are supplied power from Buss No. 2. Ten control switches on the operator's console operate the heaters for fine or coarse control or "off" position.

A 4-in. nominal diameter pipe interconnects the pressurizer and the primary coolant line. The connecting line penetrates the pressurizer near the bottom of its cylindrical section. Pressure produced in the steam dome of the pressurizer is exerted on the primary system through the connecting pipe.

The heater units are enclosed in tubes of stainless steel Type 304. All the internal surfaces of the pressurizer are clad with stainless steel Type 304 with a thickness of 1/4-in. and a minimum of 1/8-in. The entire unit and the connecting piping is covered with 4-in. of thermal insulation.

Instrumentation is provided which transmits signals to the control room to indicate or record pressurizer water level, vapor temperature in the steam dome and pressurizer pressure.

The pressurizer steam volume provides for expansion and contraction of the primary water with load changes as a function of the steam volume and heater capacity. The steam volume also provides a variable water level for control of the primary make-up pump. The pressurizer volume contains sufficient steam and liquid to absorb the fluctuations of the primary liquid volume without the excessive pressure fluctuations and without dropping the water level below the top of the heater elements.

The variations in volume of the primary liquid result from changes in its average temperature, due to abrupt changes of electrical load on the power plant. Primary system pressure changes have been measured for a loss of load from rated power and for a change from 10% to 100% power in 70 sec. As reported in Reference 5, the pressure changes were +90 psi and -25 psi respectively for these transients.

Thorough provisions for oxygen removal from the primary water, by hydrogen injection into the primary make-up tank, assure that at equilibrium conditions, the water in the pressurizer will be fully up to the saturation temperature for the normal system pressure, resulting in immediate steaming when the pressure drops.

Fundamentally, the primary loop pressure is restored, following a severe positive load transient by the action of the reactor in returning the average coolant temperature to normal, which it does in 2.5 min.⁽²⁶⁾

During a dip in pressure, the heaters are automatically turned on by their pressure sensing controls, in two stages. Thus, they necessarily contribute to, and accelerate, the return to normal pressure. In the first stage, two of the heaters will be turned on automatically when the pressure drops to 1200 psia, and in the second stage, the rest of the bank will be turned on when and if the pressure reaches 1150 psia.

This second stage pressure setting is obviously below the minimum pressure caused by the maximum load transient. This fact, and the short period of the low pressure swings, together mean that the heaters will not contribute significantly to the control of the minimum pressure reached.

A cool-down line connection was added to the pressurizer to reduce the time required for cool down.* Primary make-up water is introduced into the top of the pressurizer to provide cooling and to circulate water through the pressurizer to remove the heat energy. This reduces the cool-down time and provides more time for training of operating personnel.

C. PRIMARY BLOWDOWN AND MAKE-UP SYSTEM

A continuous blowdown method is used to purify the primary coolant water. The blowdown is first cooled to below 140°F by the primary cooler using river water for cooling. Next to the cooler outlet is a motor operated control valve manually controlled from the operator's console to establish the blowdown rate. Then the blowdown flows out of the vapor container through the shut-off valve and the trip valve. The primary blowdown and the cooling water is continuously monitored for radioactivity. a 3-way valve on each system may automatically divert each to the hot waste tank in the event of excessive radioactivity. The primary blowdown flows through one of the primary blowdown demineralizers and then through a micro-metallic filter to remove any resin carryover from the demineralizer. From the

* The pressurizer cool-down line was approved for operation by NYOO, per letter from Chief, NPFO to Alco Products, Inc. dated November 19, 1959. Test data is presented in a letter report from H. L. Weinberg, Alco Products, Inc. to USAEC, NYOO dated October 22, 1959.

filter the blowdown flows to the retention chamber of the primary make-up tank, for radioactive decay for 8 hrs. It flows out of the retention chamber and is ready for re-use.

Instrumentation is provided for local and distant indication of primary blowdown pressure, temperature, flow, conductivity and radioactivity.

Diversion and other losses of primary water are made up by diverting condensate cooling water through the primary blowdown demineralizer and then to the primary make-up tank.

A safety valve set to release at 75 psig protects the blowdown system from overpressure. The exhaust from the safety valve is to the hot waste tank.

The primary make-up system uses an adjustable stroke, variable speed, positive displacement, duplex pump to supply make-up water to the primary coolant system. There are two similar pumps installed with one in operation and the other on standby. The primary make-up tank provides a positive suction head to the primary make-up pumps. The speed of the pump is controlled by the pressurizer level controller which generates a signal to increase pump speed as the pressurizer level falls below normal level. At normal pressurizer liquid level the pump speed is decreased to a minimum which supplies the minimum quantity of make-up water. However, the primary make-up controls cannot lower the level in the pressurizer and a high level alarm on the pressurizer alerts the operator to take corrective action by increasing the primary blowdown or stopping the pump. In case of equipment failure, a low level alarm is provided.

The primary pump discharge water flows through the make-up line, check valve, shut-off valve and into the vapor container. There the flow divides and part is piped directly into the primary coolant line downstream from the primary coolant pumps. The remainder of the make-up water is piped to the seven shaft seals of the control rod drives. The reactor vessel is penetrated at seven points for the control rod drive shafts. The make-up water line pressure is greater than the primary system pressure and the flow is into the pressure vessel and serves as make-up water. Leakage from the low pressure side of the rod drive seal flows to a seal leakage collecting tank which is intermittently pumped out by one of the two canned motor pumps (one is used for standby). The seal leakage pumps are automatically controlled by a high and low level control switch. The seal leakage water is pumped into the primary blowdown line downstream from the motor operated blowdown control valve. Since the seal leakage water is mostly primary make-up water, it is relatively pure containing little radioactivity.

The seal leakage tank and the primary make-up tank are provided with a blanket of hydrogen gas. This prevents the stored water from absorbing

oxygen from the air, and also the water absorbs the hydrogen gas which combines with oxygen in the water by radiolytic action in the reactor core. This reduces the oxygen content of the primary coolant to a very low level which reduces oxygen corrosion within the primary system. To protect the hydrogen systems from overpressure, safety valves are used. One safety valve is located on top of the primary make-up tank and the one for a seal leakage tank is located on the hydrogen gas supply manifold located outside of the vapor container. The exhaust from the safety valves is through a flame arrestor. In case the discharging gas is ignited, the flames will not carry-over into the tank and cause an explosion. A hydrogen gas leak inside the vapor container could be a hazard, and gas leaks outside and the hydrogen system must be closely watched for leaks.

D. SECONDARY SYSTEM

The principle components of the secondary system are the steam system, condenser cooling water system, condensate circulating cooling water and the make-up water system.

The design full load operating conditions of the steam system are: the feedwater is supplied to one of the boiler feedwater pumps from the condenser hot well at 108°F temperature and at 2.4-in. Hg absolute pressure. The boiler feed pump discharge flows through the inter-condenser and after-condenser of the air ejector, the feedwater heater, the feedwater control valve, the check valve and into the steam generator at a temperature of 256°F.

A minimum flow line at the boiler feed pump discharge prevents operating the pump at no flow. A recirculating line downstream from the air ejector after-condenser provides cooling water from the ejector condensers at low flow or no flow of feedwater to the steam generator. The minimum flow line and recirculating line flow is returned to the hot well. Steam at 31.6 psia is automatically extracted from the third stage of the turbine to supply heat to the feedwater heater. Steam exhaust from the evaporator is used to supply heat to the feedwater heater in which case the extraction steam line check valve automatically closes to prevent the flow of evaporator exhaust steam to the turbine.

A continuous blowdown is used to reduce the impurities of the water in the steam generator. The blowdown rate is 200lb/hr (about 0.42 gpm) and is manually controlled. The blowdown is cooled, then continuously monitored for radioactivity and discharged to the seal pit. In case of excessive radioactivity the blowdown is manually diverted to the hot waste tank.

Steam is generated at a rate of 36,585 lb/hr with a pressure of 213 psia, and a temperature of 425°F.

Steam enters the turbine at a rate of 34,570 lb/hr at a pressure of 205

psia, and at a temperature of 417°F. The exhaust steam from the turbine passes directly to the condenser shell where it condenses and collects in the hot well for re-use.

Two pressure control valves connected in parallel in the main steam line limit the pressure supplied to the turbine to 205 psia. The two control valves are located outside of the vapor container and are manually adjusted.

Instrumentation is provided to indicate and transmit to the control room temperature, flow, pressure and conductivity of the boiler feedwater and steam. Automatic water level controllers are provided for the hot well and steam generator.

The turbine throttle valve is manually opened and trip closed. The trip-close feature is used to shut down the turbine for the following conditions: automatic or manual scram, turbine overspeed trip, and generator differential relay trip. The turbine throttle valve is trip closed by releasing the oil pressure from the power cylinder of the throttle valve. This reduction in oil pressure operates a 3-way valve to admit instrument air pressure to the power piston on the non-return check valve on the steam extraction line. This positively closes the check valve and prevents the back flow of the steam and water in the shell side of the feedwater heater from entering the turbine.

1. Condenser Cooling Water System

One of two water pumps situated in Gunston Cove draws river water from a screened water pit at the rate of 4350 gpm to provide cooling water for the main condenser, booster water cooling system and the blowdown cooling water system. The main condenser cooling water divides into two parts, each part cooling one bank of tubes in the divided flow condenser. This arrangement permits one side of the condenser to be shut down for cleaning tubes while the other side condenses the exhaust steam at reduced load. The cooling water from the condenser flows into a seal pit and then is returned to Gunston Cove. The water entering the seal pit is continuously monitored for radioactivity. In case of excessive radioactivity, the plant would shut down. A chlorine solution is introduced into the cooling water line upstream from the circulating water pump. The chlorine solution is automatically injected once every 24 hrs. This prevents the growth of algae in the cooling water systems.

2. Condensate Cooling Water System

The primary coolant pump motors and the shield tank water are cooled by condensate taken from the hot well by one of the two condensate recirculating pumps. The condensate cooling water flows in two parallel loops to primary system components. One loop supplies cooling water to the primary coolant pump motors with the condensate flowing in series through the pump motor cooling coils. The other loop supplies cooling water to the

cooling coils around the bottom of the pressure vessel and then to the shield water cooling coils, located in the shield water surrounding the pressure vessel. Both loops recombine and then discharge to the condenser hot well. A back-pressure valve is installed in the return line at the condenser inlet to maintain a pressure of 5 psig in the system. This is to prevent the water from boiling and thus reduce its heat removing ability. The condensate cooling water is continuously monitored for radioactivity.

3. Secondary Make-Up Water System

An evaporator is used to distill service water for secondary make-up water. A 5000-gal. service water tank is used to store the potable water obtained from the Fort Belvoir water mains. The evaporator is a horizontal shell and tube-type with a deaerating pre-heater. The evaporator feed pump takes a suction from the service water tank and discharges to the pre-heater where the incoming water cascades down a series of trays. The water is heated by the steam vapor in the evaporator shell which reduces the oxygen content of the water. The non-condensable gases collecting in the pre-heater are vented to atmosphere. Heat to the evaporator heating coils is supplied from the main steam line and is manually controlled. The exhaust steam from the evaporator enters the shell side of the feedwater heater to heat the boiler feedwater. The condensate drain from the feedwater heater shell is to the hot well. As the hot well over-fills from this condensate from the evaporator, a diversion valve on the boiler feedwater discharge line opens and allows the distilled water to flow to the distilled water tank. The valve is automatically controlled by the high level controller in the hot well. The level in the evaporator shell is automatically controlled. A continuous blow-down is used to reduce the impurities in the evaporator. The conductivity of the distilled steam exhaust is continuously monitored. The conductivity must be 3.5 μ mho. or less for the evaporator steam to be satisfactory for make-up water.

E. AUXILIARIES

1. Station Service Power

The 460-v. station service power on two busses is taken from two delta-Y connected transformers which are connected by feeders to the 4160-v. busses. Each feeder, if necessary, is capable of meeting all station service needs. In the event of a short circuit in either feeder, relay operated air circuit breakers automatically isolate the defective circuit and place the entire load of both busses on one feeder. In most cases, station service equipment is provided in duplicate, one on each 460-v. bus to further safeguard continuous operation. Each station service component is protected by an individual air circuit breaker. Station service power is drawn from the Virginia Electric Power Company for plant start-up.

2. Compressed Air System

Compressed air for the operation of pneumatic instruments and controls throughout the plant is supplied by two air compressors, a main air receiver and an emergency air receiver. The main air compressor is water cooled by coolant delivered by the cooling water booster pumps with an emergency cooling connection to the service water system. The auxiliary compressor is air cooled and serves as a standby unit, automatically starting when the after-cooler air pressure drops to 70 psig. The main air receiver of approximately 20 cu ft capacity is located outside the turbine room on the pump room roof. The emergency receiver is provided to supply the trip valves on the shield tank cooling water outlet line and the steam dump line to the condenser during an emergency shutdown of the plant. Sufficient air is contained within this vessel for 30 min after which these two trip valves will close.

3. Auxiliary Lube Oil Pumps

Two auxiliary lube oil pumps are provided; one is electrically driven and the other is steam driven. The electrical auxiliary lube oil pump is placed in operation prior to introducing steam to the turbine. The steam driven auxiliary pump serves as an emergency stand-by pump in the event of the failure of the electric auxiliary pump to operate. The electrical auxiliary pump serves as the normal stand-by pump for the turbine generator.

4. Blowdown Cooling Water

The blowdown cooling water system includes two motor-driven centrifugal pumps and the necessary piping for supplying coolant to the primary blowdown cooler, vapor container, space cooler and various sample coolers. The pumps take suction from the main condenser cooling water inlet line at a point upstream of the main condenser.

5. Cooling Water Booster System

The cooling water booster system includes two pumps and the necessary piping to supply coolant to the lube oil coolers and generator air cooler. The pumps take suction from the main condenser cooling water inlet line upstream of the main condenser.

6. Chlorination System

Condenser circulating water is periodically treated with a dilute chlorine solution to prevent algae growth within the circulating water system. The system which performs this function draws chlorine gas from one or more chlorine bottles through a manifold into a chlorinizer where a concentrated chlorine solution is produced. This solution is drawn from the reservoir by an injector powered by river water from a booster water pump. The injector mixes the chlorine solution with the injection water, diluting it, and

forcing the solution to a nozzle located in the circulating water pump discharge line upstream of the main condenser and auxiliary cooling water system piping. Service water is utilized to operate the chlorine gas pressure control valve. An automatic timer determines the amount of chlorine solution injected by alternately starting and stopping the chlorinizer system controls.

CHAPTER IV - VAPOR CONTAINER

A. STRUCTURE - (Fig. IV-1)

The vapor container is an enclosure of concrete and steel construction in which the entire primary system is housed. It consists of a cylindrical steel outer shell, with hemispherical ends, and a steel inner lining. The space between the inner lining and outer shell is filled with concrete. A secondary concrete shield, outside the vapor container, surrounds all but the top dome of the unit. The vapor container is capable of withstanding the pressure which would be developed by the release into steam of the entire volume of primary and secondary system water. The enclosure shields the surrounding area from direct radiation, and in the event of a failure of any part of the high-pressure primary system, it will contain resulting airborne radioactivity or missiles.

The vapor container outer shell was designed to withstand a pressure of 66 psig and a temperature of 293°F, which are the maximum values anticipated following a maximum credible accident. A pressure of 66 psig would result in a fiber stress of 16,400 psi in the vapor container outer shell. The material used in the outer shell, ASTM Specification A 201 Grade B, has a minimum yield strength of 32,000 psi.

1. Vapor Container Integrity

After erection the integrity of the vapor container was tested in three phases as follows:

- a. All final welds were completely radiographed and found satisfactory.
- b. The second step was a hydrostatic test of 6 hr. duration. The initial pressure at the extreme top of the vapor container was 76 psig and at the end of 6 hr. was 77.5 psig in the same location. During this period, the water temperature was $74 \pm 5^{\circ}\text{F}$. Three complete visual checks were made during the 6 hr. period. A slight leak was found in the top manhole gasket seal; however, this was tightened and did not leak during the helium leak test. After insertion of all reactor components, this manhole was seal welded. The only other leak noted was at the ground level access manhole gasket seal. The gaskets on the inner and outer doors of the access manhole were replaced. Since that time minor seeping of water has been observed infrequently at the outer door. The inner door cannot be observed during operation but leakage therefrom, which would be indicated by a water level alarm located on a 3/4 in. standpipe several feet above the top of the

Fig. IV-1

Vapor Container

manhole, has not been indicated to date.

- c. The final phase consisted of a helium leak test of all welded joints using a Consolidated Engineering Leak Detector, Model 24-101A. Two leak detectors calibrated for minimum detectable leaks of 2×10^{-6} cc/sec and 0.86×10^{-6} cc/sec. respectively, were used and were re-calibrated about every 30 min. during use. A 36 in. circulating fan was operated in the containment vessel to assure mixing of air and helium and a uniform temperature. At no time during the test was any leak detected.

Since the initial testing of the vapor container, electrical wires have been pulled through five of the spare conduits, all of which were tested during the original construction. Each of these five new electrical cable installations was helium leak tested at 180 psi with pure helium and was found satisfactory. The only other modification to the containment vessel was the installation of a 1 in. valve on the 6 in. sump drain line. The 6 in. drain line and valve terminate outside the containment vessel and were tested during the original construction.

The three-phase testing of the containment vessel demonstrated essentially zero leakage at the time of construction. The nature of the subsequent modifications to the vessel gives no evidence that its integrity has changed.

B. ACCESS OPENINGS (Fig. IV-2 and 3)

Entrance to the vapor container is made through a double door access opening at the lower plant floor level. When the reactor is in operation, the doors are bolted closed. The space between the double doors is filled with water, thereby providing adequate shield protection against radiation streaming.

A manhole with a 6.5 ft. clear opening was provided in the top of the container through which major pieces of apparatus were introduced into the vapor container during construction of the primary system. After all of the equipment was installed, the manhole door, made of 2-1/2 in. steel plate, was bolted closed and seal welded to insure leak tightness.

C. PIPE PENETRATIONS (Fig. IV-4 and 5)

All piping and conduit penetrating the vapor container wall is offset to prevent radiation streaming. Pipes carrying incoming fluids are fitted with check valves located outside the vapor container wall to prevent the

Fig. IV-2

Vapor Container - Top Access Opening Detail

Fig. IV-4
Vapor Container Penetration Details

Fig. IV-5 Vapor Container Air Duct Penetration

escape of radioactive fluid. Pipes carrying outgoing fluids are fitted with pneumatically operated trip valves outside the vapor container which will close in the event that pressure in the vapor container rises to 5 psi above atmospheric pressure. The vapor container wall is also penetrated by a tubular steel chute for transfer of spent fuel elements from the reactor to a spent fuel pit outside the vapor container. This tube is fitted with a plug valve physically located in a portion of the tube outside the vapor container in order to ensure pressure integrity of the container in case of damage to the tube inside the container. Air connections from a ventilating system, used only during shutdown, also penetrate the vapor container wall. Gate valves and flanges with blank inserts prevent the spread of airborne radioactivity outside the vapor container during operation.

D. TRIP VALVES

The pneumatically operated trip valves which provide seal protection in the event of damage to piping in the vapor container are closed by spring action. They are held in the open position for normal plant operation by an air supply of 60 to 100 psi. A pressure rise of 5 psi in the vapor container opens the pressure switch relay-actuated contacts, interrupting current through the solenoid valve coil in each trip valve air supply line. This loss of current closes the solenoid valves, interrupting the air pressure at each valve, which then closes by spring action. Any trip valve may subsequently be opened by depressing a push-button which energizes the coil of the solenoid valve, restoring air supply to the trip valve when the pressure drops below 5 psig. The trip valve trip and reset circuit is powered from the emergency DC system to ensure continuous operation in event of power failure.

E. COOLING AND VENTILATION

A space cooler has been provided for controlling the ambient temperature within the vapor container during normal operation. The normal operating temperature within the vapor container is 115°F with the maximum allowable temperature being 125°F.

The space cooler is rated at 75,000 Btu/hr maximum at maximum air condition.

Cooling water is received from the blowdown cooling water line and the blowdown cooling water system must be in operation before the space cooler can operate.

A system of spray heads adequately protected from missiles is installed at the top of the container to provide cooling to prevent a secondary pressure rise following a nuclear incident.

The ventilating system for the vapor container is designed to be used for recirculating the air in the vapor container or for ventilation by taking suction in the vapor container and discharging to atmosphere through the stack. A filter is provided in the suction line which is 99.7% efficient and will remove particles 0.3 microns or greater in diameter.

F. MISSILE PROTECTION

A carefully study of missile penetration of the container structure was made in cooperation with Bureau of Yards and Docks, U.S. Navy, at Boston, and the Army Ordnance at Aberdeen Proving Grounds. A variety of missiles of various sizes, shapes, and masses were calculated as shown in Appendix B.

Missile protection for the body of the vapor container is provided by concrete between the inner and outer shells. Protection for the steam and water lines, and the ventilation duct is shown in the detail design of these features referred to previously.

G. SPENT FUEL PIT SYSTEM

A concrete pit, lined with 18-gage stainless steel and filled with water, is located outside the west side of the vapor container to receive and store spent reactor fuel elements. The pit is 23 ft 6 in. deep, 11 ft long and 9 ft wide. (See Fig. IV-6).

1. Fuel Element Storage

On the floor of the pit, 80 stainless steel pipes, 3 ft high and of 5 in. dia., are placed vertically in rows to store the fuel elements. Each pipe is capable of holding one element. The pipes are plated with cadmium to assure a subcritical system. Four 1/2 in. holes near the bottom of each pipe permit thermal circulation of water around the fuel element inside the pipe. A movable platform rolls along along tracks on two sides of the top of the pit to facilitate placing of fuel elements in the rows of storage pipes. A hinged aluminum cover closes the top of the pit. An aluminum ladder leads to the bottom of the pit for cleaning and maintenance. A lead-shielded shipping container may be lowered to the bottom of the pit for removal of fuel elements.

2. Water Shield

The water in the pit shields the surroundings from the residual radiation of the spent fuel. The normal water level is 19 ft. 6 in. above the top of the fuel storage pipes, or more than twice the minimum

Fig. IV-6
Fuel Chute and Spent Fuel Storage Pit

Fig. IV-7

Spent Fuel Pit Plug and Transfer Tube

of 9 ft. 6 in. necessary to reduce the dose rate to tolerance. The water level in the pit is maintained at the same height as the water in the primary shield tank to prevent flow of primary shield water into the pit when the chute is unsealed during fuel transfer.

3. Fuel Chute (Fig. IV-7)

A 6 5/8-in. OD stainless steel pipe runs from inside the vapor container, near the top of the pressure vessel, through the vapor container wall, and into the bottom of the spent fuel pit. The pipe serves as a chute for the transfer of spent fuel elements from the reactor to the pit. When not in use, the chute is sealed with a molded rubber compression element plug. The plug consists of two metal assemblies, the plug cone and the seal compressor, separated by the rubber seal. The lug on the plug cone is inserted in the lug slot of the chute wall and prevents axial motion of the cone. The lug lock on the seal compressor plate positions the compressor and through the compressor stop, positions the plug cone to engage the lug and seat the cone on the cone rest. As the shaft is turned in the direction to SEAL, the shaft thread advances into the plug cone until the compressor stop prevents further motion. The shaft disc carries the seal compressor toward the cone, compressing the plug and causing the rubber to expand and seal the chute. Unsealing and removal of the plug is accomplished in the reverse manner. The plug is located outside the vapor container in the event a rupture of the chute occurs inside the container. The chute penetration of the container wall is backed up by a lead shield ring, 9 in. thick, to prevent radiation streaming past the chute during the fuel transfer operation. The chute enters a sloped hopper at the bottom of the spent fuel pit which receives each fuel element as it is lowered through the chute.

4. Recirculating Pump

A pump and suitable piping is provided to fill the pit and to recirculate the water for cooling, filtering or demineralization.

CHAPTER V - SHIELDING

A. GENERAL CONSIDERATION

In designing the SM-1 shielding the objective has been to provide sufficient shielding to meet accepted practices of the Atomic Energy Commission program regarding radiation levels for operating personnel. Under normal operating conditions, levels for personnel continuously exposed during working hours are restricted to a fraction of the accepted permissible level of 300 mr per week. In certain areas, however, above-tolerance levels are permitted where infrequent access is required for periods of controlled short duration. Taking advantage of this fact permits somewhat greater flexibility of design and operation but without increased hazards to operating personnel.

While it is desired to achieve as much of a prototype in the shield design as possible, one basic compromise was introduced by locating the facility at Fort Belvoir. In a location such as Fort Belvoir, it is necessary to provide a facility which can be approached at any time and from any angle without persons being unwittingly exposed to excessive radiation levels. Thus the radiation existing at any point around the building is no higher than the permissible continuous exposure level. This not only makes it possible for modifications to the facility to be considered at a later date with impunity, but it also renders unnecessary a close policing of visitors to prevent them from wandering into areas where radiation exposure may be of some consequence.

A primary shield is provided around the reactor pressure vessel which reduces the gamma rays from the core to a level comparable to the intensity of those gammas arising in the activated water external to the shield. This primary shield also provides the necessary attenuation for capture gamma rays and reduces the neutron level to a point where no significant activation of equipment outside the shield occurs.

The dose rate within the vapor container ranges from 1 to 20 r/hr depending upon position. The dose rate adjacent to the primary piping is about 17 r/hr which arises from the N-16 activity in the primary water. The core contributes about 3 r/hr through the primary shield. In addition to 2 ft of concrete in the liner of the vapor container, there are 3 additional ft of concrete as high as the ceiling of the control room. The secondary shield provided by this 5 ft of concrete reduces the dose rate to less than 1 mr/hr in the control room.

The principal source of gamma rays in the primary water is from the $^{16}_0\text{n}, \text{p} \text{ } ^{16}_7\text{N}$ reaction. Since the half life of $^{16}_7\text{N}$ is only 7 sec, access may

be had to the vapor container within a reasonable period of time after shutdown. The shutdown activity in the primary coolant water arises from activated impurities in the system, which result from pick-up of corrosion products and impurities in the make-up water. The shutdown activity is discussed in greater detail in Chapter XI.

B. SECONDARY SHIELD

The secondary shield is described in Table V-1. It was designed such that radiation from the primary coolant would give not more than one-tenth of laboratory tolerance for a 40 hr week at any point outside the vapor container up to the full height of the power plant building. The calculated dose rate at a position outside the vapor container due to the important primary coolant components is indicated in Table V-2. The dose rates on the outside surface of the vapor container walls were measured and found to be much less than 1 mr/hr during full power operation. The dose rate just above the spent fuel pit from sources inside the vapor container will be greater than two-tenths tolerance, but it will not exceed tolerance during reactor operation.

Penetrations of the secondary shield by the steam line and several smaller lines are shielded by placing sufficient material around the openings to compensate for the concrete removed. The steam line is oriented so as not to be in a direct line with any major radiation source.

The neutron source from the decay of nitrogen-17 produced by the $^{16}\text{O}(n,p)^{17}\text{N}$ reaction is lower by several orders of magnitude than the gamma source from N-16. These neutrons are readily attenuated to a negligible dose by the 5 ft of concrete.

C. PRIMARY SHIELD

The structural details for the primary shield are described in Section II-D. Tables V-3 and V-4 indicate respectively the shielding materials available from the core to the outside of the primary shield in the horizontal and vertical direction.

With the reactor operating at 10 Mw the dose rate from neutrons and gammas at various points in the radial and axial direction is given in Tables V-5 and V-6. The values of the gamma dose rates were measured during SM-1 Research and Development Task VI. ⁽²⁾ The dose from thermal neutrons outside a concrete shield is always much less than the fast neutron dose. The control room, the point for which data is given, is the nearest normally occupied location to the reactor.

TABLE V-1
SECONDARY SHIELDING

<u>Description</u>	<u>Material</u>	<u>Outer Radius In.</u>	<u>Thickness In.</u>
Vapor Container Lining	Steel	204.1	0.1
Vapor Container Structure	Concrete	224.1	24.0
Vapor Container Wall	Steel	225.0	0.9
Secondary Shield*	Concrete	261.0	36.0

* - To a height of 27 ft above ground level.

TABLE V-2
**CALCULATED DOSE RATE OUTSIDE VAPOR CONTAINER FROM PRIMARY
COOLANT ACTIVITY DURING REACTOR OPERATION**

<u>Source</u>	<u>Diameter In.</u>	<u>Length In.</u>	<u>Distance from Center to Outside of Shield Near Control Room Ft</u>
Piping	11	144	18
Steam Generator Bundle	44	144	22
Steam Generator Channel	33	18	22

<u>Source</u>	<u>Gamma Flux Outside Shield Photons/Cm²-Sec.</u>	<u>Gamma Dose Rate Outside Shield mr/hr</u>
Piping	40	0.31
Steam Generator Bundle	43	0.33
Steam Generator Channel	16	0.12
Total	99	0.76
Measured Value		<< 1

TABLE V-3
DESCRIPTION OF REACTOR SHIELD - RADIAL

<u>Description</u>	<u>Material</u>	<u>Outer Radius</u> <u>In.</u>	<u>Thickness</u> <u>In.</u>
Core	- - -	11.1	- -
Reflector	Primary Water	17.5	6.4
Thermal Shield	Stainless Steel ⁽¹⁾	19.5	2.0
Inlet Passage	Primary Water	23.75	4.25
Pressure Vessel	Steel	26.5	2.75
Insulation	Glass Wool (2)	30.125	3.625
Insulation Cladding	Steel	30.375	0.25
Clearance Space	Void	32.25	1.875
Vessel Support and Shield Tank Wall	Steel	34.25	2.0
1st Cooling Passage	Shield Water	35.25	1.0
1st Shield Ring	Steel	37.25	2.0
2nd Cooling Passage	Shield Water	38.5	1.25
2nd Shield Ring	Steel	40.5	2.0
3rd Cooling Passage	Shield Water	41.5	1.0
3rd Shield Ring	Steel	43.5	2.0
4th Cooling Passage	Shield Water	44.5	1.0
4th Shield Ring	Steel	46.5	2.0
5th Cooling Passage	Shield Water	47.5	1.0
5th Shield Ring	Steel	49.5	2.0
6th Cooling Passage	Shield Water	50.5	1.0
6th Shield Ring	Steel	52.5	2.0
7th Cooling Passage	Shield Water	53.5	1.0
7th Shield Ring	Steel	55.5	2.0
Neutron Shield	Shield Water	80.0	24.5
Shield Tank Outer Wall	Steel	80.5	0.5

(1) Considered as steel for shielding purposes.

(2) Considered as void for shielding purposes.

TABLE V-4
DESCRIPTION OF REACTOR SHIELD - VERTICAL

<u>Description</u>	<u>Material</u>	<u>Distance from Center of Core To Outer Surface In.</u>	<u>Thickness In.</u>
Core	-	11.0	-
Header	Primary Water	59	48
Pressure Vessel Cover	Steel	61.75	2.75
Insulation	Glass Wool (1)	68	6.25
Insulation Cladding	Steel	68.5	0.5
Neutron and Gamma Shield	Shield Water	202.5	134

(1) Considered as void for shielding purposes.

TABLE V-5
DOSE RATE FROM REACTOR DURING FULL POWER OPERATION - RADIAL

<u>Location</u>	<u>Gamma Dose Rate mr/hr</u>	<u>Fast Neutron Dose Rate mrep/hr</u>
Reactor Surface	$4.6 \times 10^{11*}$	$1.8 \times 10^{11+}$
Outside Shield Tank	$3.3 \times 10^{3*}$	$2.6 \times 10^{1+}$
Inside Vapor Container	$(1 \text{ to } 20) \times 10^{3*}$	4.1+
Outside Vapor Container	$\ll 1^*$	$4.6 \times 10^{-6+}$
One-Tenth Tolerance	0.75	7.5×10^{-2}

* Values measured during R & D Task VI(2).

+ Calculated Value.

TABLE V-6
DOSE RATE FROM REACTOR DURING FULL POWER OPERATION-VERTICAL

<u>Location</u>	<u>Gamma Dose Rate</u> <u>mr/hr</u>	<u>Fast Neutron Dose Rate</u> <u>mrep/hr</u>
Reactor Surface	$4.6 \times 10^{11*}$	$1.8 \times 10^{11+}$
Above Water	232*	$7.1 \times 10^{-2+}$
Inside Vapor Container	100-500*	$7.0 \times 10^{-3+}$
Outside Vapor Container	<1*	$2.8 \times 10^{-5+}$
10 times Tolerance	75	7.5

+ Calculated Value.

* Values measured during R & D Task VI⁽²⁾

The inlet and outlet lines are surrounded by 4-in. annuli containing insulation, which offers low resistance to gamma penetration. Although neither pipe is in direct line with the core, a considerable amount of scattered radiation and a significant quantity of direct radiation can escape through the annuli. To prevent this streaming, blocks of steel as shown in Fig. II-14 & 15 are added. These rings effectively stop open paths in all directions.

The inlet and outlet lines both penetrate the pressure vessel at a vertical position above the top of the reactor core. Therefore, any radiation that may stream along the piping axis through the primary shield enters the pipe at an angle to its axis and then has to undergo at least one scattering to change its direction. A scattering reaction greatly reduces both the magnitude and the energy of the gamma rays to a level where they are easily stopped by the water in the pipe and the steel in the pipe wall. This is evidenced by the fact that measurements on the surface of the primary shield tank do not show any "hot spots" in the areas where the primary coolant lines penetrate the pressure vessel or the shield tank outer wall.⁽⁵⁾

D. SHIELDING AFTER SHUTDOWN

The gamma dose outside the primary shield has been calculated for various times after shutdown following continuous operation at full power for the various periods specified. The results are shown in Table V-7. These dose rates were also measured as part of SM-1 R & D Task VI⁽²⁾ and the calculations and measurements agree very well. The dose which can be expected in case the shield tank water has been lost is also listed.

The core is assumed to have a uniform volume source strength. A linear buildup factor has been used as being conservative, but reasonably correct for the low energy of fission product gammas.

The activity level outside various parts of the primary coolant system due to the buildup of radioactive corrosion products is discussed fully in Chapter XI. These dose rates range from about 25 to almost 3000 mr/hr depending upon position and time after shutdown.

E. SHIELDING DURING SPENT FUEL ELEMENT TRANSFER AND STORAGE

Spent fuel elements must be shielded during removal from the core, transfer to the spent fuel element storage pit, and storage in the pit.

Table V-8 indicates the dose rate expected from one used fuel element 24 hrs after shutdown from 1000 hrs of continuous operation, with various thicknesses of water shielding. These dose rates are estimated by extrapolating measured values obtained during SM-1 R & D Task VII. (4)

Lead shielding surrounds the fuel element transfer tube for the short length between the primary shield tank and the vapor container wall. A cylindrical segment 9 in. thick will reduce the dose to about 150 mr/hr at the fuel handling position for the short time that the fuel element is in that portion of the transfer tube.

The water level in the spent fuel storage pit must be the same as in the top shield tank when the transfer tube is open. Thus when the water level in the top shield tank is lowered prior to starting up after changing fuel elements, the water level in the outside pit will be lower also. However, this leaves more than 16 ft of water above the fuel elements in the storage pit. This is completely adequate, as Table V-8 indicates.

TABLE V-7
DOSE RATE OUTSIDE RADIAL PRIMARY SHIELD
FROM FISSION PRODUCT ACTIVITY

	Time After Shutdown Hours	Dose Rate mr/hr		
		Infinite Operation	1000 Hr. Operation	100 Hr. Operation
Regular Shield	5	1.4*	0.9+	0.4+
	12	1.2*	0.7+	0.3+
	24	1.0*	0.6+	0.2+
No Water In Shield	5	111+	76+	33+
	12	98+	63+	24+
	24	84+	54+	15+

* Measured values

+ Calculated values

TABLE V-8
DOSE RATE ABOVE WATER FOR A USED FUEL ELEMENT*

The fuel element is removed 24 hrs after shutdown from infinite operation.

<u>Thickness of Water</u> <u>ft</u>	<u>Dose Rate</u> <u>mr/hr</u>
5	123
6	13.5
7	1.5
8	0.16

* These values were obtained by extrapolating measured data reported in APAE Memo 206⁽⁴⁾

CHAPTER VI - OPERATION & CONTROL

A. CONTROL FEATURES

Because of its strong negative temperature coefficient, the SM-1 is a very stable reactor while operating at power. The instrumentation and controls have been designed to take full advantage of this inherent stability. With the temperature coefficient of $-2.2 \times 10^{-4} \text{ }^{\circ}\text{F}^{-1}$ no additional control mechanism is needed to override rapid transients or power excursions.

The general arrangement of the instrumentation and controls in the control room is shown in Fig. VI-1 a, b, c, d and e. The three safety chambers are set to drop the rods at 130% of design power, and the period meter will cause a scram if the reactor period becomes less than three sec. The circuitry is extremely fast and the rods will start to drop in about 60 msec. Details of the operation of the nuclear instrumentation are given in the following section.

In addition to the nuclear safety circuits, there will be certain limits in the sensing elements in the primary loop which will initiate a scram.

The flow rate in the primary loop is 3860 gpm and if this flow drops to 3500 gpm while the reactor is at power, a scram is initiated.

The reactor cannot be operated without 60% of design flow. However an interlock, with a lock type switch is provided so that the reactor may be operated without flow up to 200 W before a scram occurs. The design pressure in the system is 1200 psi, and a 1455 psi pressure causes a scram. If the reactor outlet temperature goes 30°F above the design point of 450°F ., there will be a scram. A deliberate back leakage of water is allowed through the standby pump in the primary loop so that the temperature of the water in this dead leg is essentially the same as that in the flowing loop. However, if the temperature in the dead leg drops more than 10°F . below that in the flowing loop, the standby pump can not be started without shutting down the reactor.

FIG. VI-1a Control Room SM-1

FIG. VI-1b Control Console

LEGEND FOR FIGURE VI-1b

1. Annunciated point nameplate over indicator lights
2. Common alarm signal light -- red
3. Test indicator lights
4. Test panel nameplates
5. Test buttons
6. Annunciator panel switch
7. Acknowledge switches
8. Safety rod A position indicators
9. Safety rod B position indicators
10. Regulating rod position indicators
11. Shim rod No. 1 position indicators
12. Shim rod No. 2 position indicators
13. Shim rod No. 3 position indicators
14. Shim rod No. 4 position indicators
15. Rod position upper limit light -- red
16. Rod position lower limit light -- green
17. Reactor ON-OFF switch
18. Power ON-OFF switch
19. Scram pushbutton
20. Scram circuit reset pushbutton
21. Shim rod No. 4 individual drive control switch
22. Shim rod No. 3 individual drive control switch
23. Shim rod No. 2 individual drive control switch
24. Shim rod gang drive control switch
25. Shim rod No. 1 individual drive control switch
26. Regulating rod drive control switch
27. Safety rod B individual drive control switch
28. Safety rod gang drive control switch
29. Safety rod A individual drive control switch
30. Intercommunication system controls
31. Range selector switch-linear power channel
32. Regulating rod servo-control set point deviation
33. Pressure gauge (Remote parameters indicator)
34. Pressurizer heater individual control switches
35. Pressurizer heater individual control switches
36. Primary blowdown valve control switch
37. Transfer lock switch
38. Motorized steam valve close button
39. Primary coolant pump No. 2 control switch
40. Primary coolant pump No. 1 control switch
41. Parameter selector switch
42. Parameter selector switch
43. Heater gross control pushbuttons
44. Heater fine control pushbuttons
45. Regulating rod AUTO-MANUAL selector switch
46. Regulating rod servo-control set point adjustment
47. Signal light test button
48. Motorized steam valve position light
49. P. C. blowdown trip valve controls
50. Secondary blowdown trip valve controls
51. 400 cycle indicator light

FIG. VI-1c Electrical Section Control Panel

LEGEND FOR FIGURE VI-1c

1. Synchroscope Voltage - SM-1
2. Synchroscope Voltage - VEPCO
3. Synchroscope
4. Generator Phase A and amps
5. Generator Kw
6. Excitation - Amps
7. Line Phase A Amps
8. Generator Phase B Amps
9. Generator Kvar
10. Excitation - Volts
11. Line Phase B Amps
12. Generator Phase C Amps
13. Generator Volts
14. Generator Frequency
15. Line Phase C Amps
16. Generator Temperature
17. Turbine Speed rpm
18. Receiver Signal
19. Line Kw
20. Temperature Meter Transfer SW
21. Temperature Meter Transfer SW
22. Generator Phase Volts Switch
23. Governor Motor
24. Load Limit Control
25. Synchronizing Breaker D
26. Synchronizing Breaker B
27. Voltage Adjust-regulator
28. Breaker D
29. Breaker B
30. Exciter Breaker
31. Breaker E
32. Breaker C
33. Breaker A
34. Synchronizing Key Breaker E
35. Synchronizing Key Breaker C
36. Selsyn Transmitter
37. Generator Differential Trip
38. Generator Exciter Field
39. Turbine Trip Reactor Scram
40. Throttle Valve Trip Solenoid Point
41. Test Block
42. Test Block
43. Test Block
44. Test Block
45. Transfer #1-amps
46. Generator Watt Meter
47. Transfer #2-amps
48. 480 V-Bus #1 amps
49. 480 V-Bus #2 amps
50. Transfer #1 ammeter switch
51. 480 V-Bus #1-AC Volts
52. Transfer #2 ammeter switch
53. 480 V-Bus #2-AC Volts
54. Synchronizing Breaker F
55. Synchronizing Breaker H
56. Synchronizing Breaker G
57. Breaker F
58. Breaker H
59. Breaker G
60. Station Service #2 Lockout Relay
61. Station Service #1 Lockout Relay
62. Line Lockout Relay
63. Test Block
64. Test Block
65. Test Block
66. Test Block
67. Generator - MG Set Amps
68. Generator Battery Volts
69. Battery amps
70. Ground Detector
71. Field Rheo-Gen. MG Set
72. Gen. Cont. MG Set
73. Batt MG Set Volts
74. Motor Control MG Set
75. Circulating Water Pump #1
76. Circulating Water Pump #2
77. Boiler Feed Pump #1
78. Boiler Feed Pump #2
79. Cond. Recirculating Cooling Pump #1
80. Cond. Recirculating Cooling Pump #2
81. Cooling Water Booster Pump #1
82. Cooling Water Booster Pump #2
83. Primary Makeup Pump #1
84. Primary Makeup Pump #2
85. Blowdown Cooling Water Pump #1
86. Blowdown Cooling Water Pump #2
87. Decontamination Vent Fan
88. Blank
89. Evaporator Feed Pump
90. Blank
91. Breaker J
92. Breaker K
93. Fort Belvoir Load Meter

FIG. VI-1d Graphic Section Control Panel

LEGEND FOR FIGURE VI-1d

1. Exciter	46. Steam Dump Manual Control Station
2. Generator	47. Flow
3. Turbine	48. Pressure
4. Steam Generator	49. Air Loading Station
5. Vapor Container	50. Hot Waste Tank Vent
6. Pressurizer	51. Conductivity Selector
7. Seal Pit	52. Evaporator Overhead Conductivity
8. Turbine Throttle Trip Valve	53. Open to Stack
9. Feedwater Control Valve	54. Trip Valve #2 Steam from Generator to Turbine
10. Radiation Monitor	55. Trip Valve #1 Steam from Generator to Turbine
11. Condenser	56. Trip Valve Drain from Steam Generator
12. Primary Coolant Pump No. 1	57. Trip Valve Blowdown from Steam Generator
13. Rod Seals	58. Trip Valve Cooling Water Return from Shield Tank
14. Reactor Pressure Vessel	59. Trip Valve River H ₂ O from Blowdown Cooler
15. P. C. Blowdown Cooler	60. Trip Valve Primary Coolant Blowdown
16. Three Way Valve	61. Trip Valve Steam to Boiler Feed Turbine Pump and Condenser
17. Level Control Valve	62. Spare
18. Boiler Feed Pumps	63. Spare
19. No. 1	64. Trip Valve from Seal Leakage Tank
20. No. 2	65. Steam Generator Level Controller
21. Emergency	66. Name Plate
22. Cond. Recirc. Cooling Pump No. 1	
23. Cond. Recirc. Cooling Pump No. 2	
24. Air Ejector	
25. Feedwater Heater	
26. Primary Coolant Pump No. 2	
27. Seal Leakage Return Pumps	
28. No. 1	
29. No. 2	
30. Primary Coolant Blowdown Valve	
31. Hot Waste Storage	
32. Three Way Valve	
33. Radiation Monitor	
34. Service Water Tank Supply	
35. Evaporator Feed Pump	
36. Blowdown Cooling Water Pumps	
37. No. 1	
37A No. 2	
38. Evaporator	
39. Variable-Volume P. C. Make-up Pumps	
40. No. 1	
41. No. 2	
42. Demineralizer A	
43. Demineralizer B	
44. Supply from Circulating Water Pump Discharge	
45. Spare Switch	
	A. Conductivity Recorder
	B. Distilled H ₂ O Tank (Level Indicator)
	C. Service H ₂ O Tank (Level Indicator)
	D. Condenser Pressure (Recorder)
	E. Steam Pressure (Recorder)
	F. Steam Temperature (Recorder)
	G. Feedwater Flow (Recorder)
	H. Steam Flow (Recorder)
	J. Steam Generator Level (Recorder)
	K. Vapor Container Pressure (Recorder)
	L. System Pressure (Recorder)
	M. Pressurizer Level (Recorder)
	N. P. C. Flow (Recorder)
	P. ΔT P. C. Pump (Indicator)
	Q. Temp. Differential across Reactor (Indicator)
	R. P. C. Blowdown Flow (Recorder)
	S. Primary Coolant Make-up Tank (level indicator)

NUCLEAR PANEL

RADIATION PANEL

FIG. VI-1e Nuclear and Radiation Sections Control Panel

B. INSTRUMENTATION CONTROLS & SAFETY SYSTEM

1. General Approach

The SM-1 system of nuclear instrumentation and controls is essentially a modification of the basic ORNL type of circuitry. It is fundamentally a conventional, vacuum-tube and relay system, employing standard and conventional nuclear-channel functions, parts and components. In addition to the nuclear controls and instrumentation proper, a safety system of the same class of circuitry is provided. Into this system, information from both the nuclear channels and a number of process parameters is fed. When the value of any of these parameters exceeds prescribed limits in a dangerous direction, automatic actions are effected to guarantee safety of reactor, plant and personnel from excessive nuclear excursions, overheating, overpressuring, and too-low coolant flows.

The instrumentation system is designed so that any malfunction or failure of the instrumentation itself will be in the safe direction. Malfunction or failure of instrumentation equipment of prime importance to plant safety will cause the reactor to scram and will alarm. Malfunction of equipment of lesser importance will cause an alarm to sound so that corrective measures may be taken by the plant operator.

2. System Description

a. System-Channel Ranges

The SM-1 system of nuclear instrumentation and controls is fully shown block-diagramwise in Fig. VI-2, 3a and b. Fig. VI-2 illustrates the startup- and intermediate-range channels. Fig. VI-3a illustrates the power-range channels, together with interlocks with the startup and intermediate-range channels and the more important process parameters. Fig. VI-3b illustrates the two-out-of-three coincidence circuits for "fast" and "slow" scrams in a functional diagram. Fig. VI-4 illustrates the approximate number of decades for each of these channels covered. The decade reference level is the design full-power level of reactor operation, i. e., approximately 10 thermal Mw. The startup-range channels are shown to cover the region of about $10^{-8}\%$ of reactor power level (corresponding to approximately 1 count per second cps) to about $10^{-4}\%$ (corresponding to approximately 10^4 cps).

FIG. VI-2
FUNCTIONAL BLOCK DIAGRAM
NUCLEAR INSTRUMENTATION
SM-1

FIG. VI-3a
FUNCTIONAL BLOCK
DIAGRAM SM-I
SAFETY SYSTEM

FIG. VI-3b

TWO-OUT-OF-THREE
COINCIDENCE SCRAM
DETAIL SM-1

FIGURE VI - 4
NUCLEAR INSTRUMENTATION
MEASURING RANGES

The intermediate range channels begin at the latter point (i. e. approx. $10^{-4}\%$, corresponding to 5×10^{-11} detector amperes) and extend to a point beyond 200% power level. The power or safety-level channels begin at approximately 1% of reactor power level and extend to a point beyond 200% power level. Two decades of overlaps are obtained between startup and intermediate channels by proper adjustment of the pulse height discriminator. Thus the entire range of levels is covered with overlap from source ranges to a point well beyond full reactor-power level. In the following discussion, the three figures referenced above should be referred to as required for an understanding of the system.

b. Neutron-Detector Considerations

The reactor power level is proportional to the neutron flux. Neutron-caused pulse-counting techniques are presently required in the startup regions and neutron-current measuring techniques above the startup level. Specifically, BF_3 (proportional) counters are employed in the startup region; gamma-compensated ion chambers in the intermediate region; and non-compensated ion chambers are employed in the safety (power)-level region where the gammas are "swamped out" by the neutrons and become an insignificant matter.

All of these various detectors are arrayed circularly (and approximately equidistantly) in seven instrument wells surrounding the reactor pressure vessel. Physically, these dry wells are positioned between the first two steel shield rings and extend downward well below reactor level and upward to the platform, where the detectors may be examined, removed or changed by removing flush cover plates. Thus all the detectors may be positioned as required to afford attenuation of neutron flux to desired levels bearing a known relationship to accurately - measureable parameters (e. g. thermal parameters) at power levels.

This consideration is, of course, more important in the higher-ranged channels than in the startup-region channels, where the objective is less that of measuring the reactor power level accurately than that of following it reasonably smoothly, as the control rods are withdrawn and the count rate rises above source levels. The detectors proper are constructed of materials which are of low activation, such

that (a) secondary emissions or "pileups" of an undesired nature have minimum effect on the apparent measured power level and (b) handling of them can be accomplished without unwarranted and unwieldy shielding problems.

c. Startup-Range Channels

(1) Description

Both channels are alike. Pulses taken from BF_3 (proportional) counters are preamplified in the vapor container. Final amplification is at the Control Room Nuclear Panel. The shaped-pulse output of the linear amplifier is counted on a scaler (which also supplies high voltage to the BF_3 counters). The shaped output of the linear amplifier is taken through a log diode, further amplified; and the resulting count rate is both indicated and recorded as a current. A recorder interlock, set at a minimum count rate of 2 cps, assures that neutrons are available before the initial pulling of the control rods.

(2) Channel Pre-Checkout & Calibrations

In the SM-1, the startup channels are pre-checked for both operability and reasonable linearity at the source-range end of their ranges by means of withdrawing safety rods A & B. Since these rods are not needed to insure sub-criticality, they are normally pulled out almost fully. Actual tests run on SM-1 show an increase of 50% on one channel and 15% on the other. The multiplication will depend on the location of the detector tube relative to the core and source. Therefore, this is an indication of both the operability and the reasonable linearity of the startup channels before the pulling of the other rods which are actually required to become critical.

d. Intermediate-Range Channels

(1) Description

These channels are of the same ranges, with respect to reactor power level, but of different circuitry and purpose. Both operate from electrically-adjusted, gamma-compensated ion chambers which receive their high-voltage supplies from a specially-made chassis in the Control Room Nuclear

Panel. Beyond these points in common (which govern their ranges) the two channels are totally different. The gamma compensation is required and effective in only approximately the two lower decades of both these channels (corresponding roughly to $10^{-4}\%$ to $10^{-2}\%$ of full reactor power level). A description of the two channels follows.

(a) Linear Channel

This channel operates through a range selector switch (a conventional rotary attenuator in principle) to achieve its linear nature. The signal, through the selected switch position, is first amplified, then both indicated and recorded. At the present time, the channel is used for no direct control function and controls only through an operator. However, an associated servo controller is capable of controlling the reactor by adjustment of the control rod (c) position. Since, however, the SM-1 pressurized-water reactor has a very high negative-temperature coefficient, the coefficient (alone) is capable of controlling the reactor power level safely through all possible load transients above a very small power level. The servo controller has been installed but is not being used in the SM-1. The primary function of the linear channel is to give the operator the only available wide-ranged source of linear information relative to the reactor power level.

(b) Log N & Period Channel

The two principal functions of this channel are to give the operator wide-ranged source of information relative to reactor power level (not requiring deliberate range selection) and to provide period protection for the reactor.

The chamber current is taken through a log diode to extract the log of the signal, effectively compressing the signal range in a meaningful and predetermined manner. The log signal is both indicated and recorded.

A portion of the log signal developed is differentiated to determine period information, i. e. specifically a signal whose value is proportional to the rate of rise

or drop-off of reactor power level which always follows a theoretically exponential curve. To check the accuracy of the period data derived, it is thus only necessary to track the 80-sec decay curve of neutron flux after a scram. (This curve may also be followed on down to source levels on the startup channels, when desired, as a further check on their accuracy -- particularly in their upper-decade levels.) The extracted rate information is recorded on a period recorder and indicated on a meter.

Four interlocks operate from this channel: (1) in the "zero power" position of the reactor-mode switch, a recorder control scrams the reactor at a prescribed (but adjustable power level; (2) the BF_3 counters on the startup channels are disconnected from their high-voltage supplies and grounded for protection; (3) the rods are prevented from further withdrawal after the reactor period is shorter than 10 sec (positive); and (4) the reactor is scrammed when a period shorter than 3 sec (positive) is attained. No negative-period interlocks are necessary for any known purpose.

e. Power-Range Channels

(1) Description

These nuclear channels are connected to operate on a 2-out-of-3-coincidence type of redundancy basis to improve system reliability. This is to say, two out of three channels must agree that a scram is necessary before the scram is actually effected. All operate from non-compensated ion chambers on the theory that at all higher power levels the combined neutrons and prompt gammas (both of which are proportional to reactor power level) essentially "swamp out" all residual gammas. This is in contrast with the case of the startup channels and the first two decades of the intermediate channels, since residual gammas can (at various times) be very troublesome in the lowest six decades of operation.

However, in the power range of operation, the signal-to-noise ratio is very good and causes very few operational problems. Two scram speeds are provided by this circuitry -- both a so-called "fast" scram and a "slow" scram.

This refers to the electronic speeds only -- and, for most practical purposes, it is therefore substantially meaningless. The reason is that most of the scram delay is not in the electronic, but in the mechanical portion of the scram system. The latter is the sum of de-energization time of the clutches, their face-separation time intervals and sufficient rod-dropping time to insert adequate negative reactivity into the reactor to assure its quick shutdown (or scram).

The "fast" scrams, however, operate through the grid circuit of the clutch-holding safety-amplifier output tubes, whereas the "slow" scrams operate through the clutch power supplies. This effects a slight additional delay, primarily because of the electrical energy storage in these supplies. The nuclear parameters (power level and period) operate through the "fast" scram circuit; all non-nuclear parameters (including manual) operate through the "slow" scram circuit. The total delay rod-drop oscillographic tests indicate that the SM-1 rods will show perceptible motion in 50-70 msec after scram initiation.

(2) Nuclear & Non-Nuclear Scrams & Interlocks

As stated, all non-nuclear scrams operate through the so-called "slow" scram circuit by interrupting the clutch power supplies in the safety amplifiers. Besides the nuclear parameters (power level and period, described above), the following non-nuclear parameters currently will effect scram of the reactor:

1. Primary system pressure (high or low)
2. Primary system flow (low)
3. Low flow of cooling water to primary coolant pumps
4. Primary systems temperature (high)
5. Secondary Steam Temperature (high)
6. Secondary steam pressure (high)
7. Primary coolant pumps (off) -- routine operation (Reactor-Mode Switch)
8. ΔT Between primary pump-leg outlets (high)

C. PRELIMINARY TESTING

All primary system components were tested in accordance with the ASME code for Unfired Pressure Vessels and all other applicable codes as a minimum requirement. Stringent cleanliness requirements and a 125 rms interior surface finish was maintained.

Each component was hydrostatically tested and in addition each vessel was leak checked using the helium mass spectrometer technique. No leaks were found in any component even at gasketed joints.

Each component was shipped with all openings sealed with expendable rubber plugs and filled with inert gas to 4-5 psig; to insure cleanliness on arrival at the site.

All valves were pressure tested by the manufacturer prior to shipment.

Major components of the secondary system were shop tested prior to shipment to insure proper operation.

The condenser was hydrostatically tested and given a Freon leak test prior to shipment. Before installation under the turbine pedestal a second Freon leak test was made to insure leak tightness.

To assure absolute cleanliness in the primary system, all equipment has been cleaned either in the vendor's shop or during actual construction. Rust and imbedded foreign material was removed with a stainless steel brush. Oil was removed with methyl alcohol-soaked rags.

In-place cleaning of the primary system components was performed, by installing a temporary line from the lower control rod penetration at the bottom of the pressure vessel to the top of the pressurizer. A 20 gpm pump was inserted into this line to recirculate the water from the pressure vessel through the pressurizer and back to the pressure vessel.

A cartridge-type filter was placed in the temporary line between the pressurizer and the reactor vessel. A 25 micron filter was used to remove dirt and grit from the primary system during the first 8 hr of running. A second filter, of 10 microns, was then placed in the line and the system was operated until no additional foreign material could be removed by the filter, a period of 24 hr.

After the above cleaning it was deemed unnecessary to add any chemicals to clean the primary system.

The secondary system components were cleaned according to standard practice. The condenser hotwell, evaporator, feedwater heater and other auxiliary items were cleaned by an alkaline boil-out procedure.

The steam generator was cleaned by circulating a mild detergent through it and the auxiliary piping.

Upon completion of installation of the vessels, the piping was welded in and the joints radiographed. The vessels, piping and valves were then subjected to a helium mass spectrometer leak test. A hydrostatic test of the system was performed at 1.5 times design pressure.

Erection of all secondary system equipment was made with careful check for alignment and support. Direction of rotation of all rotating equipment was checked to prevent damage to equipment. Piping was hydrostatically tested in accordance with ASME code requirements.

Upon completion of component and system testing a portable oil fired steam generator was attached to the secondary system and used to bring the primary system to temperature and pressure for the non-nuclear tests.

All instrumentation was checked for proper operation at this time.

During the non-nuclear testing a larger portable oil-fired steam generator was installed to permit operation of the turbine generator set; thus a check out of operation of the secondary system and its instrumentation at low generator loading was performed.

After the preliminary testing was completed a non-nuclear test program was conducted. The reactor core was loaded with a set of dummy fuel elements and the primary system was brought to operating temperature and pressure.

The system was operated from March 16 to March 22 and from March 29 to April 3 to demonstrate the system's integrity.

The following operations were accomplished during the testing period.

1. Demonstration of loading and unloading the reactor core.
2. Demonstration of changing a rod drive seal.
3. Demonstration of techniques in maintaining control of primary water purity.
4. Testing of the control rod drives under operating conditions.
5. Determination of the effects on the fuel elements of short time operation at temperature.

Upon completion of the non-nuclear testing the shield was filled with water and the top cap removed from the pressure vessel. The dummy fuel elements were removed and replaced by uranium bearing fuel elements. The fuel bearing control rods were installed and observed for proper operation.

The circuits of all nuclear instrumentation were checked by simulating a chamber signal at a point as close to the sensing element as possible. This method was also used to check out nuclear scrams and interlocks.

D. INITIAL CRITICALITY & ZERO POWER EXPERIMENTS

After the system had been completely checked out and shown to be operating satisfactorily, the reactor was loaded with fuel. The fuel elements were added one by one with a measurement of the subcritical multiplication after each element was added. The BF_3 counters were mounted temporarily next to the reactor core within the pressure vessel. The reactor first became critical with 17 fuel elements, control rod C at 16.50 in. and the remainder of the rods full out. (5) As additional fuel was added, the position of the rods were noted, giving a calibration of rod worth in terms of fuel addition. The rods were also calibrated against reactor period. Additional zero power experiments were then performed to determine temperature and pressure coefficients. The results are reported in APAE-18. (5)

E. POWER OPERATION

The cap was then placed on the pressure vessel and the pressure in the primary system brought up to 1200 psi. With the reactor held critical, the system was heated to 450°F . and the temperature coefficient of reactivity measured as the temperature increased. The external heating was then stopped and the reactor power elevated to about 100 Kw. The vapor container was sealed at that time. The primary pumps were then started and after synchronizing with VEPCO and adjusting the governor to its maximum value the nuclear power was increased in steps to 10 Mw by adjusting the load limiting device. The xenon buildup was measured and the system was observed closely for local boiling in the reactor. After checking the performance of all equipment, operation of the reactor was then routine.

1. 700-Hour Test

The 700-hr performance test was started at 2030 June 2, 1957 and concluded on 0030 July 2, 1957. During this 700-hr period the plant was inoperative 13 hr and 38 min. Of this down time

only 7 hr and 24 min was due to malfunctioning of SM-1 equipment. This down time was a result of an error in the calibration of the steam generator level indicator; recalibration of the instrument corrected the trouble.

During this test SM-1 operated so as to meet all design requirements proving itself to be a reliable nuclear power plant.

A complete discussion of the startup tests and the 700-hr plant performance tests will be found in APAE 18. (5)

F. SHUTDOWN & HOT FUEL ELEMENT HANDLING

When the time comes to change fuel elements, the reactor will be shut down and allowed to cool off for about 24 hr. The air in the vapor container will be changed by blowing out through the stack. While no airborne activity is expected, a monitor is installed in the stack as a precaution. The vapor container will be entered through the access door and the lid of the pressure vessel removed underwater with special tools. The fuel elements will be removed from the core one by one and passed through the discharge chute to the spent fuel pit for storage (See Fig. III-3). All operations are to be performed with the fuel elements under at least 9.5 ft of water.

New fuel elements will then be added, the pressure vessel will be capped, and the reactor will then be ready for further operation.

G. TEMPERATURE TRANSIENTS FOLLOWING PUMP FAILURE

The reactor behavior immediately following pump failure is considered to be the loss of flow problem. The period beginning about five sec. after pump failure is referred to as the decay heat removal problem.

The accident assumed is failure of the primary coolant pump due to electrical or mechanical malfunction. Primary coolant continues to circulate for a short time after the accident due to the momentum of the water, but it eventually (in 6 to 8 sec) decreases to an approximately steady state level determined by natural thermal convection.

The flow coastdown can be represented by:

$$(R/R_0) = \left(\frac{1}{1 + bt} \right)^{1.25}$$

where

$b \approx 0.60$ as determined by experiment. (6)

1. Loss of Flow Problems

At the initial startup of SM-1, the behavior of the reactor following shutoff of the primary pump was investigated. ⁽⁵⁾ This test indicated satisfactory performance during the period from loss of flow until scram a short time later. As part of the SM-1 R & D project the behavior during loss of flow has been measured for long scram delays.

Test 600⁽⁶⁾, loss of Primary Coolant Pump, was conducted on the SM-1 with no detectable adverse effects. The pump was shut off with delays of 1, 5, and 15 sec before the reactor was scrammed at various initial power levels up to and including 10 Mw. An analysis using the ATBAC Code for the IBM 704⁽⁷⁾ shows that for reactors of this type, the minimum burnout ratio occurs within the first second.

To determine if a rupture of the fuel plates had occurred, samples of the primary coolant blowdown were analyzed after each run to detect any increase in fission product activity. None was found. This was considered proof that in all cases investigated, the negative temperature coefficient drove the reactor power down rapidly enough to prevent burnout of the fuel plates. Fig. VI-5 shows the normalized power for the 10 Mw run.

After the test was completed, the reactor was operated for over a month (until the end of core life) with no evidence of sticking or jamming of the control rods. Hence, it has been assumed that no warpage or deformation of the fuel plates or control rods was experienced as a result of the test.

Thus, for at least a 15-sec period, the SM-1 can undergo a loss of flow accident without any adverse affects.

2. Decay Heat Removal

The SM-1 primary system was designed so that upon loss of flow, the reactor would scram and removal of the afterheat would be accomplished by thermal convection.

A natural convection loop is established by using a steam generator with vertical U-tubes at a higher elevation than the reactor vessel.

For test purposes (refer APAE 18), ⁽⁵⁾ the primary coolant pump was shut off, and the reactor scrammed at succeeding higher power levels. In the final test, the reactor was run at 10 Mw

FIG. VI - 5
NORMALIZED POWER COASTDOWN
VS.
TIME AFTER PUMP FAILURE $P_0 = 10 \text{ MW}$

for 2 hr, the primary coolant pump was then shut down and the reactor scrammed. In order to detect boiling, the reactor was brought critical at a very low power as soon as possible after the scram. The neutron level recorders were observed to see if any detectable void formation was occurring in the core. The results of this test are shown in Fig. VI-6 and 7.

The trace of the reactor outlet temperature shows that the primary system temperature began to rise after about 20 min and continued to rise for about 30 min due to decay heating. At this point, the system heat losses became the dominant factor and the temperatures began to drop. Since the log N chart shows no effect of boiling past this point of maximum primary temperatures, it can be concluded that no boiling occurs in the core. The slight oscillation indicated in the neutron level (Fig. VI-6) was compared with previous traces taken at low power and found to be typical of the instrumentation. Thus it was concluded that the SM-1 will cool itself without detectable boiling following a primary coolant pump failure and scram.

H. CONTINUOUS 5 ROD BANK WITHDRAWAL AT POWER

The continuous five rod bank withdrawal accident has been investigated by analog computer methods using a lumped model that neglects high power level, high temperature and short period scrams.

The analog model shows that for a withdrawal rate of 3 in. per min, corresponding to a $dk/dt = 0.0008$ per sec, the period has decreased to the period scram setting of 3 sec after 8 sec of withdrawal.

However, the power level scram setting is 13 Mw. This value is reached after 3.5 sec of withdrawal.

Thus the reactor should scram after 3.5 sec of withdrawal due to high power level. Assuming failure of this scram, the power would continue to increase, until the period decreased to 3 sec when the short period scram should be initiated.

I. RAPID LOAD TRANSIENTS

In experimental as well as actual operation situations, the SM-1 has proven to be extremely stable and to give excellent response to sudden changes in load demand.

Fig. VI-6 Emergency Cooling Log N Chart

Fig. VI-1 Emergency Cooling Reactor Outlet Temperature

After the excellent responses made by the SM-1 during sudden drops to station load from full power were noted, it was decided to subject the plant to a complete loss of load by tripping the turbine off the line.

Tripping the turbine throttle valve removes all load from the steam generator with the exception of the turbine driven lube oil pump. Since the reactor continued to produce power, the primary coolant temperatures started to rise. The coolant temperature increase drove the power down by means of the negative temperature coefficient of reactivity, which in turn caused the power to increase again, causing the coolant temperature to increase again. The increasing primary temperature decreased the reactivity and diminished the power output.

After about 76 min, relatively cold water was added to the steam generator to calibrate the steam generator level indicator. (See Fig. VI-8.) The cold water caused a decrease in primary system temperature causing another out of phase damped oscillation in power and coolant temperature.

During this transient, the reactor did not scram and the control rods were not moved. The plant went through the complete loss of load transient with no operator adjustment, thereby exhibiting the plant's self regulatory capabilities.

Reactor behavior following a sudden increase in load demand was tested by increasing the generator electrical load from 225 Kw to 2050 Kw in approximately 75 sec. As in the loss of load tests, no operator adjustments were made during the transient.

At the beginning of the transient, the reactor heat output was about 2 Mw, corresponding to the station electrical load of 225 Kw. The power output of the reactor rose very nearly as fast as the plant demand with very little overshoot. (See Fig. VI-9 and 10.) A more complete description of the effect of sudden load transients can be found in APAE 18. (5)

Fig. VI-8
Log N. Chart, Loss of Load Test

Fig. VI-9 Generator Wattmeter Increase in Load Test

Fig. VI-10 Log N Chart Increase in Load Test

CHAPTER VII - NUCLEAR EXCURSIONS

A. POSITIVE REACTIVITY AVAILABLE

The SM-1 was designed to operate at 100% of its maximum power of 10 Mw 1-1/2 yr without changing fuel elements. To achieve this 15 MWYR life time* with a minimum of control rods, use is made of a burnout poison. The critical mass at the end of the 16.4 MWYR cycle is 14.3 kg of U-235. The cycle is started with a loading of 22.5 kg of U-235 and is poisoned with 15.8 gm of B-10. With this boron loading, the peak in reactivity occurs at the beginning of core life as illustrated in Fig. VII-1. The lower curve in the figure shows that change in multiplication at the operating temperature of 440°F and equilibrium Xe with a maximum excess reactivity of 8.2% at the beginning of the core life. If the reactor is allowed to cool down to room temperature at any time during the cycle the resultant multiplication can be obtained from the upper curve in the figure. The maximum excess reactivity at room temperature with no Xe is 15.4%. It is necessary to provide control rods in the reactor sufficient to override the 15.4% excess multiplication. Seven rods are used with a total worth of about 25%. During normal operation, rods A and B are fully withdrawn and the reactor is controlled with the 5 rod bank which is worth 19.0% at room temperature.

B. STARTUP SAFETY SYSTEM

During normal operation, the startup of the SM-1 is accomplished by the removal of shim rods A and B followed by removal of the remaining 5 rods. There are various methods used to prevent shorter reactor periods and high powers from being reached.

The first method is the 10 sec rod motion stop. In the event that the period is less than or equal to 10 sec, rod motion is automatically stopped. The rods cannot be withdrawn until the period increased to a value of 14 sec or greater.

In the event that the 10 sec rod motion stop method fails, the rods can be withdrawn until the period decreases to three sec. For reactor periods of three sec or less, the reactor is scrammed.

If both the period and rod motion stop methods fail, the core power will continue to rise until it is shutdown by the 13 Mw power scram or by an inherent shutdown mechanism such as void formation in the moderator.

* Actual measured Core I life is 16.4 MWYR

FIG VII-1
 REACTIVITY AS A FUNCTION
 OF ENERGY RELEASE MWYR
 SM-1 CORE I

C. BANK POSITION

The cold clean critical bank position at the initial startup of the SM-1 Core I and II is 3.7 in. for the five rod bank (see Fig. VII-1 and 5.4 in. for the seven rod bank.

D. REACTIVITY ADDITION RATE

The reactivity addition rate is determined by the rod withdrawal speed and the differential rod worth.

The maximum rod withdrawal speed is 3 in./min. The maximum differential rod worth for the seven rod bank is 2.46%/in. and 1.88%/in for the five rod bank. The maximum differential rod worth occurs at 5.3 in. for both the seven and five rod bank.

During the normal startup operation the seven rods would not be withdrawn as a bank however, since it is possible for all the rods to be withdrawn at the same time, the startup calculations were performed using the reactivity addition rates of the seven rod bank.

The seven rod bank worth per in. is 2.46% ρ or 0.0252 δk . The maximum reactivity addition rate is

$$R = (0.0252 \frac{\delta k}{\text{in.}}) (3 \frac{\text{in.}}{\text{min}}) = 1.3 \times 10^{-3} \frac{\delta k}{\text{sec}}$$

E. NEUTRON LIFETIME

The neutron lifetime was calculated from two group perturbation theory methods as

$$\ell = \frac{L_f}{V_1 \Sigma_1} + \frac{L_t}{V_2 \Sigma_2}$$

where

$$\frac{1}{V_1 \Sigma_1} \approx \frac{2}{V_2 \xi \Sigma_s}$$

The notation used is the following

L_f	fast leakage
L_t	thermal leakage
Σ_1	fast removal macroscopic cross section - cm^{-1}
V_1	fast group velocity - cm/sec
Σ_2	thermal absorption microscopic cross section - cm^{-1}
V_2	thermal group velocity - cm/sec
ξ	average logarithmic energy charge per collision
Σ_s	macroscopic scattering cross section evaluated at average thermal energy - cm^{-1}

Using thermal constants evaluated at the hardened neutron energy of 0.0331 ev, the calculated value for the lifetime is

$$\ell = 1.7 \times 10^{-5} \approx 2 \times 10^{-5} \text{ sec}$$

F. EFFECTIVE DELAYED NEUTRON FRACTION

The effective delayed neutron fraction used in the calculations was

$$\beta_{\text{eff}} = 0.0073$$

G. POWER AND PERIOD DURING CONTINUOUS ROD WITHDRAWAL

The effect of rod withdrawal on the reactor period and power may be found by using published analog studies.⁽¹⁰⁾ Figure VII-2 illustrates the neutrons flux level and period as a function of time after the reactor is critical. The analog studies assume that there are no temperature or void formation effects on the reactivity. For the case illustrated in Fig. VII-2, the reactivity addition rate is $10^{-3} \frac{\delta k}{\text{sec}}$, the neutron lifetime is $2 \times 10^{-5} \text{ sec}$,

$\beta_{\text{eff}} = 0.0073$ with five delayed neutron groups, and the initial k_{eff} is 0.8.

The largest source of error is the reactivity addition rate of $10^{-3} \frac{\delta k}{\text{sec}}$ which was used in the calculations since the maximum addition rate calculated in Section D was $1.3 \times 10^{-3} \frac{\delta k}{\text{sec}}$. In order to compensate for the smaller reactivity addition rate used in the analog studies, all values of period used in the calculations were reduced by a factor of 1.3

MODEL

DATE

FIG. VII-2

PERIOD AND RELATIVE NEUTRON RESPONSE

T_0

SEVEN ROD BANK WITHDRAWAL AT STARTUP

while the corresponding δk was increased by a factor 1.3.

The 10 sec. rod motion stop is reached 0.2 sec. after the reactor is critical. If rod motion is stopped, the period increased to an asymptotic value of

$$T = \frac{\bar{\ell}}{K_{\text{eff}} - 1} = \frac{0.0928}{(0.2)(1.3 \times 10^{-3})} = 357 \text{ sec.}$$

where $\bar{\ell}$ is the average neutron lifetime and $K_{\text{eff}} - 1$ is the excess K_{eff} . The flux level reached is approximately $10^5 \text{ cm}^{-2} \text{ sec}^{-1}$ which corresponds to a power of 0.1 watts.

In the event of failure of the period scram and rod motion stop, the core power will continue to increase until the power scram or some inherent shutdown mechanism such as void shutdown stops the excursion.

Neglecting the effects of temperature, the power level scram is reached 7.7 sec. after the reactor is critical. The reciprocal period at that time 36 sec^{-1} . The control rod bank would be inserted after the $60 \times 10^{-3} \text{ sec.}$ clutch delay. During that time the core power will continue to rise. If the period stays the same as the period reached at the level scram, the power will increase as

$$P = P_s e^{\tau/T}$$

where τ = clutch delay time

T = period reached at level scram

P_s = level scram

For the continuous rod withdrawal problem, $\tau = 60 \times 10^{-3} \text{ sec.}$; $(T)^{-1} = 36 \text{ sec}^{-1}$; $P_s = 13 \text{ Mw.}$ The power reached at the beginning of rod motion is

$$P = (13) \exp (60 \times 10^{-3}) (36) = 113 \text{ Mw}$$

H. PEAK POWER

The method used to calculate the peak power after the start of rod motion is presented by MacPhee⁽¹¹⁾. The differential equation which describes the response of core power for a constant δk acceleration is

$$\frac{1}{N} \frac{dN}{dt} = \frac{\delta k_p}{\ell} - \beta - \frac{at^2}{2\ell}$$

The ratio peak to initial power or neutron level is

$$\log_e \frac{N_m}{N_i} = \frac{2 \sqrt{2} (\delta k_p - \beta)^{3/2}}{3 \ell \sqrt{a}}$$

The notation used is the following:

N_m = Maximum power or neutron level

N_i = Power or initial neutron level at the beginning of rod motion

a = δk acceleration in $\delta k \text{ sec}^{-2}$

ℓ = neutron lifetime $2 \times 10^{-5} \text{ sec}$

β = effective delayed neutron fraction 0.0073

δk_p = excess positive $k_{\text{eff}} = k_{\text{eff}} - 1$

The δk acceleration rate a was determined by assuming a differential rod worth of 0.015 $\delta k / \text{in.}$ and an average rod acceleration of 92 in. / sec^2 . The δk acceleration is

$$a = (0.015 \frac{\delta k}{\text{in.}}) (92 \frac{\text{in.}}{\text{sec}^2}) = 1.38 \frac{\delta k}{\text{sec}^2}$$

The power scram is actuated 7.7 sec. after criticality is achieved. The excess positive k_{eff} is then

$$\delta k_p = (7.7 \text{ sec}) (1.3 \times 10^{-3} \frac{\delta k}{\text{sec}}) = 0.01$$

Using a neutron lifetime of $2 \times 10^{-5} \text{ sec}$ and an effective β of 0.0073,

$$\ln \frac{N_M}{N_i} = 5.61 \quad \text{or} \quad \frac{N_M}{N_i} = e^{-5.61} = 270$$

Since the power at the beginning of rod motion is 113 Mw, the peak power, assuming the effects of temperature and void formation are not present, is

$$N_M = (270) (113) = 3.0 \times 10^4 \text{ Mw}$$

I. INHERENT SHUTDOWN MECHANISMS

In the event of failure of the control system during startup, the power will continue to rise until an inherent shutdown mechanism stops the excursion⁽¹²⁾.

Recent tests of SM-1 Core I elements at the SPERT facility demonstrate the inherent shutdown mechanisms for short periods. The first mechanism to act is void formation. The second is fuel plate melting. There are additional shutdown mechanisms which are important for longer periods but are not believed important for short periods. The principal ones reached before fuel plate melting are:

1. Fuel plate expansion. The expansion of the fuel plates when heated forces some water out of the core.
2. Radiolytic gas production. The formation of radiolytic gas in the moderator produces a small amount of void. The change in moderator volume is small and the effect is not significant.
3. Nuclear heating of moderator. Heating and the resultant expansion of the moderator by neutron and gamma-ray interactions decrease reactivity.
4. Fuel plate heating of moderator. This effect is unimportant for any short period excursions because of the time delay before the moderator is heated significantly.

J. COMPARISON OF CALCULATION AND EXPERIMENT

The method of Golian was used to determine the magnitude of the power excursions⁽¹³⁾. The constants used in the Golian calculations are presented in Table VII-1 and the nomenclature is defined in APAE Memo 177⁽¹³⁾. The validity of the Golian method was checked by comparison of experimental results obtained from the SPERT facility⁽²⁴⁾. The comparison of peak power reached in an excursion is illustrated in Fig. VII-3 for the P-18/19 SPERT core and the 45 element SM-1 core. The P-18/19 core is composed of 19 SM-1 Core I type elements. The SPERT elements and the SM-1 elements are made to identical specifications. It was assumed that the peak power is proportional to the fuel volume, and the SPERT peak power values were multiplied by a factor of 45/19 to correct for the total core volume difference. The Golian predictions of peak power are less than the experimental SPERT data. For short periods, the Golian predictions give better agreement with SPERT data than for long periods. On the other hand, Golian predictions of clad coolant interface temperatures are in good agreement with SPERT data. SPERT data indicates plate damage when the reactor is put on an inverse period of 200 sec⁻¹. The plate temperature was about 300°C. The Golian predicts (see Fig. VII-4) a temperature of 280°C for an inverse period of 200 sec⁻¹.

K. EFFECTS OF INHERENT SHUTDOWN MECHANISMS ON SM-1 BEHAVIOR

In the event of failure of the 10 sec. rod motion stop, the 3 sec. period scram, and the 13 Mw power scram, the excursion will be stopped by an inherent shutdown mechanism. The period reached at the power trip level is

FIG. VII - 3
SM-1 PEAK POWER
VS
INVERSE PERIOD

TABLE VII - 1
CONSTANTS USED IN GOLIAN CALCULATIONS

Symbol	Item	Units	Conditions			
			68°F 15 psi	68°F 1400 psi	440°F 1200 psi	440°F 1400 psi
a	Fuel half thickness	cm	0.024765	0.024765	0.024765	0.024765
b-a	Cladding thickness	cm	0.0127	0.0127	0.0127	0.0127
c-b	Coolant channel half thickness	cm	0.16979	0.16979	0.16979	0.16979
$V_f(-1\%)$	Water void fraction % required to introduce -1% of reactivity	%	3.11	3.11	3.11	3.11
h_{fg}	latent heat of vaporization	cal/gm	539.1	319.3	339.8	319.3
f_{vol}	total active fuel region volume	cm	1.4299×10^3	1.4299×10^3	1.4299×10^3	1.4299×10^3
l^*	mean neutron life	sec	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}
$H_3(b, 0)$	initial temperature of coolant	°C	20	20	226.67	226.67
$H_1(0, 0)$	initial temperature of fuel at center line	°C	20	20	298.89	298.89
$H_3(b, t_B)$	boiling temperature of coolant	°C	100	310	299.3	310
$H_1(0, t')$	temperature of center line at end of excursion (may be melting point)	°C	1399	1399	1399	1399
$H_3(b, t')$	clad coolant interface temperature	°C	1399	1399	1399	1399

TABLE VII - 1 (Cont'd)
CONSTANTS USED IN GOLIAN CALCULATIONS

<u>Symbol</u>	<u>Item</u>	<u>Units</u>	<u>Conditions</u>			
			68°F <u>15 psi</u>	68°F <u>1400 psi</u>	440°F <u>1200 psi</u>	440°F <u>1400 psi</u>
K ₁	coefficient of thermal		0.03012	0.03012	0.03985	0.03985
K ₂	conductivity for fuel	cal	0.03638	0.03638	0.04299	0.04299
K ₃ ^W	region, cladding, coolant	sec-cm ² _{oC}	1.47x10 ⁻³	1.414x10 ⁻³	1.545x10 ⁻³	1.549x10 ⁻³
K ₃ ^S	(water and steam) respectively	cm	5.66x10 ⁻⁵	1.025x10 ⁻⁴	1.025x10 ⁻⁴	1.025x10 ⁻⁴
ρ ₁	density of fuel region, clad- ding, coolant (water and steam) respectively	gm/cm ³	7.957	7.957	7.957	7.957
ρ ₂			7.85	7.85	7.85	7.85
ρ _{3W}			0.9982	1.00	0.8379	0.8395
ρ _{3S}			4.34x10 ⁻⁴	0.05315	0.0413	0.05315
C ₁	thermal capacity of fuel region, cladding, coolant (water and steam) respectively	cal/gm-°C	0.12	0.12	0.12	0.12
C ₂			0.12	0.12	0.12	0.12
C ₃ ^W			0.9982	0.9926	1.100	1.098
C ₃ ^S			0.44	0.44	1.42	1.62

FIG VII-4

SM-1

CLAD COOLANT INTERFACE TEMPERATURE VS INVERSE PERIOD VOID SHUTDOWN

FIG. VII -5
SM-1

COMPARISON OF PEAK POWER
SHUTDOWN MECHANISM-VOID FORMATION

used as input for the Golian analysis to determine the peak power. This assumption can be justified since the power is increasing by about a factor of 10 at the power trip level every 0.1 sec. while the change in period is small.

The inverse period attained at the power trip level during cold start-up is 36 sec^{-1} . From Fig. VII-2 it can be seen that 36 sec^{-1} is a reasonable value for the asymptotic period for a continuous rod withdrawal incident. If the startup were accomplished with core at atmospheric pressure, the power reached before shutdown by void would be greater than 62 Mw (Fig. VII-5). In the event the core were pressurized during a cold startup, the power reached before void shutdown would be 360 Mw. The peak power reached after a power scram was calculated in the previous section as $3.0 \times 10^4 \text{ Mw}$. Consequently, the power scram does not protect the SM-1 during the initial cold startup.

L. MAXIMUM ENERGY RELEASE

The maximum energy release was determined by calculating the shortest period that may be attained before the core is shutdown by centerline melting. From Fig. VII-6, it is seen that the curves of energy release for centerline melting and void shutdown intersect at an inverse period of about 50 sec^{-1} with a corresponding energy release of 23 Mw - sec. or $2.2 \times 10^4 \text{ Btu}$. This is much less than the total energy storage in the water system of $7.4 \times 10^6 \text{ Btu}$. From Fig. VII-7, the peak power at an inverse period of 50 sec^{-1} is $5.5 \times 10^2 \text{ Mw}$.

M. EFFECT OF DROPPED FUEL ELEMENT

The effect of dropping a central fuel element into the critical SM-1 cold core was evaluated. Calculations show that the Δk between a central fixed element and a central water hole is 0.008. In the event a fuel element is dropped into the hole, the reactor would be put on a period

$$T = \frac{\ell}{\beta k_e - \beta k_e} = 0.033 \text{ sec}$$

From Fig. VII-3 and 4, the core would be shut down by void at a peak power of 50 Mw, the total energy release would be 18 Mw - sec. and the fuel centerline temperature would rise to 210°C .

N. POSSIBILITY OF ROD EJECTION

It can be postulated that with the system at 1200 psia pressure, the outlet pipe ruptures at the exit from the shield tank. (See Fig. II-14). If the incident should occur at room temperature a momentary upward impulse is

FIG.VII-7

SM-1

PEAK POWER VS INVERSE PERIOD

INITIAL PRESSURE-1400PSI

INITIAL TEMPERATURE-68°F

SHUTDOWN MECHANISMS-VOID AND CENTERLINE
MELTING

exerted on the rods lifting them upward 0.031 in. , and introducing a 0.076% reactivity increase which quickly subsides as the rods fall back into place (See Appendix C-1).

If the rupture occurs at the operating temperature of 450°F, the internal energy of the primary water will expel the water into the vapor container and as this primary water rushes out, it will exert an upward force on the control rods.

For higher temperatures than 450°F, this upward force would be even greater. As is described in the next chapter, it is conceivable that the temperature of the primary circuit can go as high as 596°F. Accordingly, the reactor behavior following a primary system rupture at this temperature has been calculated (See Appendix C-2) with the results illustrated in Fig. VII-8.

It is assumed that all rods start from their most reactive position with a rate of addition of reactivity 2.46% per in. of travel. The rods are accelerated upward and the effect of this rod ejection on the multiplication is shown in the upper curve. However, steam bubbles form rapidly. The calculations show that the density of the steam water mixture in the chamber above the core is always less than in the chamber below the core. For conservatism, it is assumed that the steam formation within the core is the same as in the lower chamber. The effect of this water expulsion from the core on a multiplication is shown on the lower curve of Fig. VII-8, with the net effect that the reactor is driven subcritical. It is expected that nearly all of the water will be expelled as the quasi-equilibrium condition within the vapor container requires an expansion on a volume basis of approximately 180 times.

O. SUMMARY OF RESULTS OF NUCLEAR EXCURSIONS

Several possible operating conditions have been studied to determine if any unusual or extreme conditions result. Both the calculated and experimental response of the SM-1 system to control rod movement and load demand changes expected during normal operation shows the system to be highly self-regulating.

The startup behavior of the SM-1 has been investigated to determine whether the scram system will protect the core from damage. The conclusions reached are that the 10 sec period rod motion stop and the period scram methods will protect the system. However, due to the long clutch delay time, the power scram setting does not protect the SM-1 during a cold startup accident.

The maximum possible energy release due to a nuclear excursion is 2.2×10^4 Btu which is much less than the total energy storage in the system of 7.4×10^6 Btu at the postulated maximum credible accident conditions.

FIG. VII-8 REACTOR BEHAVIOR FOLLOWING A PRIMARY SYSTEM RUPTURE AT 596°F.

CHAPTER VIII - CONTAINMENT OF THE MAXIMUM CREDIBLE ACCIDENT

A. DEFINITION

In Chapter VII it was shown that a nuclear excursion of sufficient intensity to completely melt the fuel plates in the core releases about 22,000 Btu. However, no mechanism for initiating such an excursion exists, except by deliberate sabotage.

By contrast with this there are 5,000,000 Btu stored as heat energy in the primary and secondary water systems when the reactor is operating at steady state full power. Thus the containment problem centers around the containment of the energy stored in the water system. The maximum credible accident can then be defined as a rupture of the primary and secondary systems at a time when abnormal operating conditions have permitted the energy stored in each of these important components to increase to the maximum attainable level.

B. CONDITIONS INITIATING THE MAXIMUM CREDIBLE ACCIDENT

A series of events and failures of equipment can be postulated as in the following paragraphs which will develop the "maximum credible" accident. It is assumed that the first failure is that the operator leaves his post for a period of a few minutes with the reactor and power plant operating. Shortly thereafter a fault in the electrical system develops which causes the turbine and generator to lose their load. This happens at a time when the average system temperature has been adjusted to the high side of the average temperature by the operator.

The resultant sudden loss of load causes the reactor temperature to start to climb while awaiting the effect of the temperature coefficient to bring the reactor power level down to the new equilibrium value. Before this inherent stability can make itself felt, a second failure is now postulated. For some unknown reason, the servo comes on, causing the rod to withdraw at its design rate of 3 in. per min.

Under these conditions the reactor will continue to run at a constant power level of 10 Mw. This follows from the fact that the rate of removal of a rod corresponds on an average to approximately 0.6% per min increase in reactivity, which for the central rod becomes more nearly 0.8% due to the greater effectiveness of the control rod. With a temperature coefficient of $-2.2 \times 10^{-4}/^{\circ}\text{F}$, a rate of rise of about 40°F per min will compensate for the rate of removal of the rods. The heat capacities of the primary and secondary systems and the heat transfer characteristics are such that at a

10 Mw power level with no external means of heat removal, a rate of rise of 45°F per min will establish itself. Accordingly, for this incident it is postulated that the reactivity decrease by rise in temperature is just cancelled by the reactivity increase by rod withdrawal. In reality this delicate balance would not be achieved, but slow departure from the initial level would occur.

In determining the time relationships discussed in the following paragraphs the system has been reduced to a simple model, consisting of a heat source (reactor), primary heat sink (primary system fluid) and a second heat sink (secondary fluid inside of the steam generator), which is deriving its heat from the primary heat sink (see Appendix D). The time involved is long enough that with 17 sec circulation time in the primary system, essential equilibrium will exist throughout the system, yet short enough that the amount of heat transfer to the mass of metal in the system was considered to be negligible.

C. SEQUENCE OF FAILURES

In Fig. VIII-1 there is shown the sequence of events which transpires subsequent to these two initial failures plotted as a function of time starting the instant the turbine throttle closes. The significant failures which must occur for the series of events to extend to the maximum credible accident are shown at their proper positions in the time sequence. After approximately 25 sec have elapsed, the rod withdrawal is started. At the end of the 39th sec the reactor outlet temperature reaches 480°F (10°F above the average temperature plotted in the diagram), which is the level at which the temperature scram should cause shutdown of the reactor. This signal fails so that the temperature and pressure continue to rise.

At the end of 45 sec the primary system pressure has reached a value of 1455 psi due to the rising temperature of the primary system water. This signal is also presumed to fail in its mission to shut the reactor down.

At the end of 51 sec the primary system pressure has reached 1500 psi, which is the value at which the primary relief valve is set to open. For the accident to reach its worst proportions, it is necessary for this relief valve to function in its proper manner. The relief valve discharges into the vapor container, thereby conserving within the system the energy which is so released from the primary system. The operation of the valve in its normal manner permits the reactor to continue operation at 10 Mw appreciably longer than would be the case if the relief valve were to stick in the closed position. If the pressure continues to rise adiabatically, the system pressure would reach 2000 psi at the end of 1 min and 20 sec. This is approximately 2 min and 40 sec earlier than occurs in the sequence that follows opening of the valve in its proper fashion.

FIG. VIII-1 SEQUENCE OF EVENTS LEADING TO MAXIMUM CREDIBLE ACCIDENT

At the end of 1 min and 9 sec the pressure in the secondary system would reach 500 psi, which is considered an upper limit by which time the relief valve should be opened and operating. While activation of this is not automatically tied in to the scram signal, it can certainly be considered an alarm, which should recall the operator to the control system for shutdown of the reactor. Since there are nearly 3 min yet remaining before the energy reaches its maximum value, the operator would have adequate time to return to the control room and take action to prevent further departure. This valve is assumed to stick shut, permitting the secondary system pressure to continue to rise.

At the end of 1 min and 27 sec the secondary system will have reached a temperature of 480°F, which is 40°F over the maximum that can be attained with the reactor operating in a normal fashion. Failure of the scram on this limiting value will permit the reactor power to continue unaltered, and temperatures and pressures to climb. At the end of 1 min and 38 sec the secondary pressure will reach a value of 600 psi which exceeds the maximum under normal operating conditions by approximately 175 psi. A scram signal actuated by this pressure should shut the reactor down, but it is assumed that this also fails.

At the end of 1 min and 46 sec the primary system average temperature has reached a value of 510°F. The change in density of the primary coolant causes displacement of 10 cu ft of water from the primary system into the vapor container. The rate of this displacement so far exceeds the possible rate of removal for normal purge purposes that the level within the pressurizer has risen well above the normal design limits. This is the last possible point for scrambling the reactor. Failure on the part of this scram signal will permit continuation of reactor operation at the 10 Mw level without any control by the operator.

D FINAL RUPTURE OF THE SYSTEM

The pressures and temperatures of both the primary and secondary systems continue to rise in accordance with the curves shown in Fig. VIII-1. At the end of the fourth minutes after the turbine throttle was closed, the primary system has reached saturation temperature at 1500 psi, the relief valve setting. With no means of relief for the secondary system the pressure and temperature in that part of the system have also risen to saturation value at 1500 psi. The curve of stored energy shows that at this point a total of 7.4×10^6 Btu are stored in the system. No further energy can be accumulated because the reactor power level will inherently cut back as the system reaches saturation temperature, and steam formation takes place within the reactor core. Since the reactor has approached this point at a slow rate it is anticipated that generation of steam will occur in an orderly fashion and reduce the power level of the reactor to a level consistent with the rate at which heat is leaking

from the system through the thermal insulation. When this condition exists the reactor has succeeded in storing the maximum amount of energy that can be stored.

With the conditions just described, the maximum accident is triggered by the abrupt failure of a thermometer well in the primary system. The energy imparted to the thermometer well and its location are such that the well becomes a missile which ruptures the shell on the secondary side of the steam generator. The stored energies of both the primary and secondary systems are released within the vapor container. While several seconds or perhaps a few minutes could elapse before all of the superheated water could escape from both systems, the rate is still sufficiently rapid that it may be considered as instantaneous release of this stored energy. The pressure in the container rises essentially instantaneously to a pressure peak, which is determined by the relation between total stored energy and volumes of water and vapor container. Once this peak has been reached, the time relationship of pressure and temperature is the resultant of heat leakage through the vapor container walls and into the primary shield water plus the additional heat released from the metal parts (approximately one-half million Btu) and from the decay of equilibrium fission products (reaching a total of approximately fourteen million Btu at the end of 48 hr).

E PRESSURE-TIME RELATIONSHIP WITHIN VAPOR CONTAINER

In the long time aspect of the containment problem (36 to 48 hr after rupture, the heat capacity of the primary shield and of the vapor container structure is of major consequence. In Fig. VIII-2 the pressure-time relationship existing within the container is shown for 48 hr following the rupture. Supporting data is presented in Appendix E. The initial peak upon rapid release of the stored energy is dissipated within approximately the first hour after the rupture has occurred. During this hour the heat stored in the nearly 50,000 lb of primary system metal is released to the water, while at the same time heat leakage to the walls of the container and to the primary shield mass is taking place. At the end of the hour the metal parts are essentially in equilibrium with the surrounding water. During this hour and for nearly 48 hr after the rupture the rate of heat released from the decaying fission products can be taken into account. An additional fourteen million Btu will be released in 48 hr from this source alone. Fortunately the time interval is great enough that the effect of the large bulk of water in the primary shield and the heat capacity of the vapor container make themselves felt. The result of these competing factors is a secondary pressure peak of minor proportions as shown in Fig. VIII-2. Water sprayed into the vapor container by a manually controlled emergency system to provide a large additional heat sink results in further reduction of the secondary peak as shown.

F. REQUIREMENTS FOR CONTAINMENT

The actual design of the vapor container to confine the results of the accident can follow no prescribed guide lines. While a vessel of the characteristics described in Chapter IV could be considered as an unfired pressure vessel designing to this standard is felt to be unnecessarily conservative.

FIG. VIII - 2 PRESSURE VS. TIME IN VAPOR CONTAINER AFTER A MAXIMUM CREDIBLE ACCIDENT

The objective of the pressure vessel code is to provide a container which can be used continuously under operating conditions, whereas this vessel is to be used but once under the conditions of maximum credible accident. In the light of this major condition in utilization it is believed logical to base the design upon permitting the stress in the containment to approach 80% of the yield point of the structural material (32,000 psi minimum for the steel used in the design).

The design of vapor container proposed for this installation has been analyzed to develop the relationship between the amount of heat released in an accident and the resultant stress in the containment vessel. Fig. VIII-3 is the curve of stress vs. heat release from which it can be seen that a total of 9.5×10^6 Btu can be released essentially instantaneously within the vapor container without exceeding 80% of the yield point of the containment material. Under these conditions the internal pressure will amount to approximately 90 to 95 psi. Since the maximum credible accident only releases 7.4×10^6 Btu the pressure rises only to about 65 psi which produces a stress of 18,000 psi- only 56% of the minimum yield strength.

In calculating this stress and equilibrium pressure no allowance was taken for heat leakage into the walls of the vapor container or into the mass of water which constitutes the primary reactor shield. While this is conservative it was not believed to be unduly so since the release of pressure from the primary and secondary systems can occur in a matter of 20 to 30 sec, during which time the amount of heat transferred to the primary shield and the container would obviously be insignificant. While it is true that rupture of the primary shield tank would release 150,000 to 160,000 lb of water to mix with the superheated water, it cannot be guaranteed that such rupture would occur. Lacking this assurance conservatism indicates that it cannot be relied upon as a major source of energy absorption for the immediate storage problem following rupture of the system.

G. Variations from the Maximum Credible Accident

The accident described in the previous paragraphs is the worst that can be postulated with credulity. Eight failures in proper sequence must occur to attain the maximum energy release of 7.4×10^6 Btu. Variations from the runaway described can be considered. If the runaway occurs at a faster rate so that the reactor gets on a fast period, both the period and level scrams will be actuated. In addition, on a fast excursion, there is a much higher probability that boiling will occur in the core which will reduce the reactivity. In any event no more than the maximum energy of 7.4×10^6 Btu can be stored in the system.

If the runaway occurs at a slower rate than that described, again no more than 7.4×10^6 Btu can be stored in the system. In this case the probability is increased that the operator can shut down the plant before the final system rupture occurs.

FIG. VII-3 STRESS VS. HEAT RELEASE IN VAPOR CONTAINER

Attention is called to a nuclear excursion which might result from expulsion of the control rods as described in the previous chapter. While the calculations indicate that for a cold reactor the energy released would be considerably larger than the vapor container could confine, this is not considered as a credible accident because no mechanism has been found which would permit expulsion of the rods under this condition. The only source of energy is that stored in the system pressurizer, but at the most this would manifest itself as a shock wave traversing the fuel element in a period of 400 microseconds. The impact given to the rods is sufficient only to give a minor and temporary increase in reactivity.

The possibility of expulsion of the rods with the reactor hot is increased because of the energy stored within the reactor coolant. An abrupt rupture, causing release of pressure in the exit plenum chamber would permit establishment of a pressure drop across the control elements in such a direction as to propel them from the reactor. As discussed in the previous chapter and in Appendix C-2, the pressure drop which furnishes the driving force for expelling the rods also causes steam formation at a rate so fast that the net change in reactivity is negative.

CHAPTER IX CONSEQUENCES OF A CONTAINED MAXIMUM CREDIBLE ACCIDENT

A. ACTIVITY AVAILABLE IN REACTOR CORE

The radiological hazard to the surroundings in the event of a fission product release is dependent upon the activity available in the reactor core. The determination of the radioactivity built up in the reactor is based upon the conservative assumption that the reactor has been operated at full power (10 Mw) continuously for 1.5 yr.

The total accumulated gamma activity was calculated for four energy groups as a function of time after shutdown utilizing the tabulations and cruves of fission product production by Blomeke and Todd (14). The results of the calculation are presented in the curves of Fig. IX-1 and IX-2.

The total accumulated fission product power as a function of time after shutdown from full power operation was calculated from the nomographs of Clancy and Stehn. (15) The results of this calculation are presented in the curve of Fig. IX-3.

B. FUEL PLATE MELTING AFTER LOSS OF COOLANT

The accident considered was the type in which a rupture, occurring in the primary system, causes a reduction in coolant level to the lower grid plate exposing the entire length of the core. Free convection through the channels is therefore not considered.

1. Previous Core Meltdown Studies

In APAE No. 49⁽¹⁶⁾ a core meltdown study was performed for the PM-2A. That study closely paralleled the core meltdown calculation for the SM-1A which was documented in APAE No. 13, Supplement I. (17) Briefly, these two studies were based on the following heat balance:

$$(1) \left(\begin{array}{l} \text{Time rate of change} \\ \text{of internal energy} \\ \text{in the fuel elements} \end{array} \right) = \left(\begin{array}{l} \text{Rate of} \\ \text{decay energy} \\ \text{generation} \end{array} \right) + \left(\begin{array}{l} \text{Energy} \\ \text{Loss} \end{array} \right)$$

The energy balance was solved stepwise with time with the heat generation term taking suitable time averaged values across each of brief time intervals and for each of four radial cylindrical regions. The losses considered were radiation and conduction. Tempera-

ture vs. time curves were then established for the regions and the time of melting was thereby determined. Details are given in AP Note 154. (18)

The moderate conservatism used in PM-2A and SM-1A calculations render the results obtained therefrom quite reliable (referring to the calculated values of time at which the melting point is reached). The justification for performing an abbreviated calculation on SM-1 core meltdown is therefore established provided that such a calculation yields results in consistent agreement with the more detailed SM-1A and PM-2A determinations described above.

2. SM-1 Calculation Model

The core is treated as a lumped system and no heat losses are considered. Therefore the loss term in relation (1) is neglected and the energy balance is formulated as

$$(2) \quad WC (T - T_0)_t = \int_0^t P(t) dt$$

The decay energy as shown in Fig. IX-3 is approximately fitted by an analytic function of time within the period of interest, its form being

$$(3) \quad P(t) = 948 t^{-0.242} \text{ Btu/sec}$$

$$(4) \quad \therefore \int_0^t P(t) dt = \int_0^t 948 \xi^{-0.242} d\xi = 1250 t^{0.758} \text{ Btu}$$

Substitution from (4) into (2) and simplifying gives

$$(5) \quad (T - T_0)_t = 18.18 t^{0.758} \text{ } ^\circ\text{F}$$

for $W = 459.5 \text{ lb steel in core}$

$$C = 0.15 \text{ Btu/lb } ^\circ\text{F}$$

Calculations indicate that only 80% of the total decay energy generated in the core is released there, producing heat. This reduction is included in equation (5).

3. Core Meltdown Results

Fig. IX-4 is a plot of equation (5) with T_0 conservatively set at 600°F . The graph indicates that the SM-1 elements are at their melting point, T_m , at 480 sec after loss of coolant.

4. Comparison with Previous Meltdown Studies

The results of section 3 are consistent with the previous studies cited in section 1. The SM-1A core melts at 150 sec. after shutdown; the PM-2A core melts at 187 sec. The SM-1A is a 20.2 mw core of 796 plates; the PM-2A operates at 10 mw from 656 plates. Therefore the average power densities in these two cores are 25 kw per plate and 15 kw per plate respectively. The 10 mw SM-1, containing 796 plates has a power density of 12.5 kw per plate. The value obtained of 480 sec after shutdown for the core to begin melting is therefore consistent with previous calculations.

C. SHIELDING PROBLEMS IN THE EVENT OF A FISSION PRODUCT RELEASE DURING A CONTAINED MAXIMUM CREDIBLE ACCIDENT

Following the maximum credible accident, the reactor core may be devoid of water. Once the moderator has been expelled the chain reaction dies off immediately but the plates will continue to heat up because of fission product afterheat. As was shown in the previous section the core will start to melt within 480 sec after the coolant loss. Because of this, fission products could be released into the vapor container.

Experiments have been performed at ORNL (19) to determine the release of fission products from SM-1 type fuel plates on meltdown. The experimental data indicates that a maximum of 16.8% of the gross gamma activity is released when an SM-1 type fuel plate is melted in a steam atmosphere. It will therefore be conservatively assumed that 20% of the gross gamma activity will be released from the SM-1 core upon melting down after the maximum credible accident.

During the destruction experiment on the Borax I core the total energy release was 125 mw-sec, (20) which is about 5 times the maximum expected release for a nuclear excursion of the SM-1. Core material, consisting of fuel plates and structural parts was distributed within a 350 ft radius of the original reactor location. Although the measurements lacked precision, they showed that only a small fraction of the core material left the site in the form of air-borne particles. This indicates that the activity release during an excursion should not be any worse than the 20% fission product release postulated for a core meltdown due to fission product afterheat.

FIG. IX - 4

AVERAGE CORE TEMPERATURE
VS.

TIME AFTER LOSS OF COOLANT

The fission products that escape from melted or vaporized fuel plates are assumed to remain airborne and diffuse throughout the free volume of the vapor container, forming a large homogenous source of gamma radiation. The volumetric source strength within the vapor container as a function of time after the incident is computed by multiplying the total gamma activity of the core (from Fig. IX-1 and IX-2) by the fraction of this activity released (20%) and dividing the result by the free volume of the vapor container. The dose rates at the outside surface of the vapor container side and dome are then computed as a function of time after the incident. The results are shown in Fig. IX-5.

The dose rate at the side of the vapor container will be a maximum of 5.7 r/hr immediately after the incident and will decay to less than 100 mr/hr during the first month. This will allow about 4 hrs for evacuation of the area before the maximum emergency dose of 25 r is accumulated. However evacuation should require much less than 1 hr and therefore the maximum dose accumulation should be less than 5 r.

CHAPTER X - HAZARD TO SURROUNDING AREA IN THE EVENT OF A CATASTROPHE

A. EVENTS LEADING TO A CATASTROPHE

It has been shown in preceding chapters that the vapor container of the SM-1 will contain an accident resulting from any credible sequence of failures. Yet to be considered are the consequences of the simultaneous occurrences of an accident releasing a large quantity of fission products and a rupture of the vapor container by sabotage, aerial bombing, or natural catastrophe such as fire, windstorm, flood and earthquake. The only inflammable material of significance is the hydrogen used to inhibit corrosion in the primary system. If the entire contents of a hydrogen cylinder were drained into the vapor container and ignited the energy released would be only 71,500 Btu. Wind velocities up to 100 mph have been encountered in the Washington area but the vapor container will withstand wind velocities much higher than this. The vapor container itself would withstand a flood and its lowest elevation is 14 ft above the normal level of Gunston Cove. No seismic disturbances of sufficient intensity to cause structural damage are on record in the Washington area.

Accordingly, the only reasonable events which can lead to rupture of the vapor container are sabotage or aerial bombing.

B. TYPES OF ACCIDENT POSTULATED

This reactor is located at Fort Belvoir, Virginia, approximately 17 mi south-southwest of Washington, D.C. A review of the climatology of the site is presented in Appendix A-1. The following facts should be kept in mind. The wind direction at the proposed site is from the south and south-southwest approximately 20% of the time; precipitation can be expected in the Washington area about 120 days per year; and stable atmospheric conditions may be expected 40 to 60% of the time, mostly during the nighttime hours.

The hazard calculations are based on the long time (1.5 yr) operation of the reactor at a power level of 10 Mw.

In the event of a maximum credible accident the core will melt after 8 min due to loss of coolant, or in the event of a nuclear excursion, the core will melt essentially instantaneously. The expected fission product release due to these accidents will be 20% of the total core inventory and will amount to 5×10^6 curies of gamma emitters and 1×10^7 curies (14) of beta emitters, 1 sec after the reactor shuts down from full power.

The products are then assumed to be released instantaneously as a "puff" cloud to the atmosphere through a break in the vapor container.

C. METHOD OF EVALUATING HAZARDS

Hazards due to the fission product cloud that is released to the atmosphere are evaluated by employing the methods and nomographs of J. J. Fitzgerald, which are generally based on O. G. Sutton's atmospheric diffusion equations for the release of puff clouds of radioactive material. A complete description of the method and equations are reported in KAPL-1045 (21). The calculations are conservative, as they assume that all the released fission products remain airborne and are expelled to the atmosphere outside the vapor container.

D. EVALUATION OF HAZARDS

Hazards that are considered may be divided into two major categories. The first is concerned with hazards due to external radiation, and the second category is concerned with hazards due to internal radiation from inhaled sources.

The external radiation hazard arises from two sources. The first is due to radiation directly from the passing cloud and the second is from rainout and/or fallout of activity from the cloud which will contaminate the area over which it occurs.

1. Total Integrated External Exposures

External exposures resulting from the passage of the cloud containing fission products is given in Fig. X-1 and X-2.

The integrated exposure is the summation of both the gamma and beta radiation. Results are given for four different meteorological conditions, ranging from a large lapse to a large inversion, and two different wind velocities of 2 meters per sec and 5 meters per sec. These calculations assume there is no cloud rise.

Doses from the passing fission product cloud were computed by assuming 0.5 Mev/disintegration. This method does not consider air scatter, which tends to increase the exposure, but assumes there is no fallout or cloud rise, which would tend to decrease the exposure from the cloud. The effects of air scattering and absorption are negligible and can be neglected when the cloud and the receptor are close (within 20 meters), and in most treatments of the problem air scattering is neglected for all distances. (25) Finally, it should be noted that these estimates are based on the irradiation of a body by a finite cloud of fission products, rather than the more conservative infinite cloud method.

FIG. X-1

FIG. X - 2

Total integrated external exposures resulting from fallout an/or rainout, as a function of distance from the origin of the accident, are given in Fig. X-3 and X-4. These curves are similarly given for four different weather conditions and two different wind velocities. The fundamental assumption is that a constant fraction of radio-activity in the cloud falls out per unit distance traveled by the cloud, and that for each distance this fall-out rate is what would give the maximum dose at that point. This calculation is very conservative since assumed fall-out rate exceeds a rate one would normally expect even during a heavy rain.

Hazards due to passage of the radioactive cloud and rainout and fallout depend on exposures that are experienced. The lethal limit to personnel is the LD₅₀/30 days which is 400 rem. A maximum permissible dose of 25 rem is assumed for establishing the radius of major hazard, provided that an exposure of this magnitude is received by an individual only once during his lifetime. An area would have to be evacuated for a lengthy period if fallout were greater than 7.5 rem for a 10-hr period. If fallout were greater than 1.8×10^{-2} rem for a 10-hr period, the area would have to be temporarily evacuated. The different hazard areas for various meteorological conditions are given in Table X-1.

TABLE X-1
RADII OF HAZARD AREAS FOR EXTERNAL DOSAGE (METERS)

A. <u>Wind Vel. = 2 m/sec</u>				
	<u>Large Lapse</u>	<u>Zero or Small Lapse</u>	<u>Modified Inversion</u>	<u>Large Inversion</u>
1. Lethal Radius	0	0	0	0
2. Maximum Permissible Emergency Dose Radius	8	10	10	10
3. Lengthy Evacuation Radius	41	44	45	46
4. Temporary Evacuation Radius	1300	1700	2400	3200
B. <u>Wind Vel. = 5 m/sec</u>				
1. Lethal Radius	0	0	0	0
2. Maximum Permissible Emergency Dose Radius	11	12	12	12
3. Lengthy Evacuation Radius	50	52	54	55
4. Temporary Evacuation Radius	1550	2150	2900	4200

FIG. X-3

RAINOUT AND FALLOUT VS.

DISTANCE FROM ACCIDENT

2. Total Integrated Internal Exposures

The total integrated internal dose due to the inhalation of the fission products that were assumed to be released instantaneously after 1.5 yrs of operation at 10 Mw power, was calculated. In this evaluation it was assumed that the cloud does not rise, the distribution of fission products in the cloud is a Gaussian function of the distance from the center of the cloud, the exposed person remains in the center path of the cloud at the distance specified, and each critical organ dose is computed from the beginning of inhalation to infinite time. Calculations were performed to obtain the doses received in various critical organs (lungs, thyroid and bones), for one day, and 60 day exposure times.

Criteria for internal doses to critical organs are given in Table X-2 for exposures of one and 60 days. The lethal dose for these organs is equivalent to the whole body dose of 400 rem; the emergency dose is equivalent to the whole body dose of 25 rem; and the evacuation dose is equivalent to a dose rate of 0.3 rem per week. It should be pointed out that these doses are received over an extended period of time, and action may be taken to increase the excretion rate of the fission products and thereby reduce the indicated total dose.

From these calculations the radii of the various hazard areas are obtained and are given in Table X-3, for receiving lethal, emergency and evacuation doses for the different meteorological conditions.

TABLE X-2
CRITERIA FOR INTERNAL DOSES IN REM⁽²²⁾

	<u>Lung</u>		<u>Thyroid</u>		<u>Bone</u>	
	<u>One Day</u>	<u>60 Days</u>	<u>One Day</u>	<u>60 Days</u>	<u>One Day</u>	<u>60 Days</u>
Lethal Dose	200	2800	2100	6800	50	250
Emergency Dose	4	79	42	141	1	10
Evacuation Dose	0.043	2.6	0.086	5.2	0.08	4.8

TABLE X-3
RADI OF HAZARD AREAS FOR INTERNAL DOSAGE (METERS)

Wind Velocity = 2 m/sec

Organ	Type of Dose	Large Lapse $n = 1/5$		Small or Zero Lapse $n = 1/4$		Modified Inversion $n = 1/3$		Large Inversion $n = 1/2$	
		1 day	60 day	1 day	60 day	1 day	60 day	1 day	60 day
Lung	Lethal	14	0	15	0	15	0	15	0
	Emergency	170	0	220	0	275	0	350	0
	Evacuation	1100	0	1800	0	3400	0	6000	0
Thyroid	Lethal	0	0	0	0	0	0	0	0
	Emergency	245	700	605	1700	1600	4500	6800	19,000
	Evacuation	15,500	5,800	38,000	14,000	110,000	39,000	630,000	185,000
Bone	Lethal	0	290	0	570	0	760	0	2350
	Emergency	1000	2000	2200	4250	5,600	7,800	16,000	30,000
	Evacuation	1,170	2,900	2550	5800	6,400	12,500	19,000	48,000

Wind Velocity = 5 m/sec

Lung	Lethal	18	0	18	0	18	0	18	0
	Emergency	190	0	250	0	320	0	410	0
	Evacuation	1300	0	2100	0	3900	0	7600	0
Thyroid	Lethal	0	0	0	0	0	0	0	0
	Emergency	70	400	220	940	440	1500	2350	12,000
	Evacuation	10,500	3800	27,000	9,500	78,000	27,500	500,000	160,000
Bone	Lethal	0	84	0	225	0	505	0	1650
	Emergency	650	1500	1500	3200	3800	7600	13,000	28,500
	Evacuation	780	2050	1750	4800	4400	11,500	15,500	45,000

E. RESULTS

In examining the results of dosage computations for the various methods of irradiation, it can be seen that the limiting hazard results from the inhalation of the fission products. Values greater than the maximum permissible exposure occur within a radius of approximately 1 to 19 mi, depending on weather conditions, for a wind velocity of 2 meters per sec, and at approximately 1 to 18 miles for a wind velocity of 5 meters per sec. It was shown that no hazard severe enough to result in evacuation of an area would exist beyond a radius of 115 mi for the most adverse possible weather conditions.

CHAPTER XI RADIATION HAZARDS AND EXPERIENCE WITH PLANT OPERATION, AND TRAINING

A. INTRODUCTION

All pressurized water reactors produce radiation levels from the primary gammas and neutrons as a result of the fission process and from N-16 in the primary coolant which follow the operating power level of the reactor. The primary shielding of the reactor and the vapor container reduce these radiation levels in normally accessible areas adjacent to the vapor container to less than the maximum permissible levels. There is no access to the vapor container during reactor operation. (See Chapter V for typical radiation levels in the vapor container during operation.) This section of the report gives the data concerning the source of typical radiation levels, and hazards associated with the operation, research and development, and training at the SM-1 during full power (excluding the vapor container) and after reactor shutdown. Procedures to be used to avoid over exposure of personnel are also included.

B. SOURCES OF ACTIVITY

As part of the SM-1 Research and Development Program, the extent and source of the buildup of radiation and activity levels in the SM-1 primary system after shutdown have been investigated. It was realized that the radiation could increase to such levels that reactor operations and maintenance could become seriously hindered or hazardous. The investigation included the measurement of activity found in the primary coolant and deposited on primary system surfaces. Details of the experimental techniques and procedures used, as well as discussions on the cause and amount of activity found, are given in APAE Memo 180(28), and APAE No. 's 44(29), 50(30), and 51(3).

As a result of the R & D Programs it was found that the observed activity was due to radioactive corrosion products and fission products in the coolant or deposited on the walls of the primary loop. Radioactive corrosion products are formed by neutron bombardment of: (1) core cladding and structural material (304 ss); (2) other materials exposed to a high neutron flux (Haynes 25 flux suppressors); or (3) possibly by out-of-flux corrosion products (304 ss) deposited on in-flux surfaces. The radioactive corrosion products are assumed to be released to the coolant and subsequently lost to the purification system, deposited on the system walls, or remain in the coolant. The mechanisms of nuclide activation, transport, and final disposition are complex and have not been completely defined. Mathematical calculations based on the above assumption have been moderately successful in accounting for the

measured activity. The degree of induced activity buildup is a function of several parameters: the extent of corrosion of the primary system materials; nuclear characteristics of materials exposed to the primary coolant; chemistry conditions of the coolant; and primary system geometry. The extent to which each parameter effects activity buildup and transport is unknown. This results in difficulties when attempting to predict shutdown radiation levels in the primary system.

It was shown that most of the fission product nuclides must enter the coolant via a cladding defect. As in the case of active corrosion products, they are lost to the purification system, deposit on the system walls or remain in the coolant.

The nuclides formed in the reactor can be conveniently divided into two types: (1) short-lived nuclides, i. e., those with a half-life of 24 hr or less; and (2) long-lived nuclides, i. e., those with half-life greater than 24 hr. The short-lived nuclides affect operations in the demineralizer room when at full power and also any requirements for quick entry into the vapor container. Long-lived nuclides primarily affect any lengthy maintenance requirements in the vapor container.

C. HAZARD CONTROL EXPERIENCE

Control of exposure to personnel and release of activity to the environment is accomplished through procedures as stated in Chapter V of the Operation SM-1 Semi-mobile, Medium powered, Pressurized-water, Nuclear Power Plant Manual. All maximum permissible levels of exposure and release to environment used are compatible with Title 10 Chapter I Part 20 of the Congressional Federal Register and the NCRP recommendations covered in NBS Handbook 69 and 59 including supplements.

1. Personnel Monitoring

Film badges are issued to each individual entering the fenced area of the SM-1 site except visitors on guided tours. The tour guide is issued an extra badge which will represent the exposure of the group. The film badges are processed by the Lexington Signal Dept., U.S. Army Signal Corps on a bi-weekly schedule. The badge is constructed so as to include metal absorbers which permit the determination of type and energy of radiation encountered.

Self reading pocket meters are issued to each individual entering a radiation area. Maintenance personnel log their dose at the conclusion of the work period, while laboratory personnel log their doses daily. The daily exposure limit is 50 mr.

The number of personnel assigned permanent film badges since the beginning of reactor operation in June 1957 is as follows:

<u>Calendar Year</u>	<u>Number of Employees Monitored</u>
1957	95
1958	111
1959	196

<u>Dose in Rem</u>		<u>No. of Employees</u>		
<u>From</u>	<u>to Less Than</u>	<u>1957</u>	<u>1958</u>	<u>1959</u>
0	1	95	101	180
1	2		9	10
2	3		1	2
3	4			4
4	5			

No SM-1 personnel have received any radiation exposures in excess of those stated in the above table. During 3 years of operation, there have been no radiation overexposures. An overexposure is defined as a dose of greater than 900 mr to the whole body in any seven consecutive days. A distinction should be made between overexposure and technical overexposure, the latter being a dose larger than the pertinent maximum permissible weekly dose of 300 mr but less than 900 mr. There have been five technical overexposures at the SM-1 as of January 1, 1960. These exposures (310 mr, 330 mr, 490 mr, 800 mr, and 880 mr) were investigated and appropriate corrective action was taken.

2. Area Monitoring and Control of Access

Radiological instrumentation is provided to measure the radiation level at key points so that overexposure of personnel is prevented. In addition, radiation dosimeters are required when personnel enter the plant and radiation areas.

a. Location and Type of Monitoring Instruments

The monitors indicate and record the gamma radiation level in five plant areas. Each station is recorded on the multi-point recorder on the control room radiation panel over the range from 0.01 mr/hr to 10 mr/hr. One exception is the ion chamber located inside the vapor container, which has a range of 100 to 100,000 mr/hr. An audible alarm is activated

in the control room when the radiation levels exceeds 7.5 mr/hr except in the vapor container where the alarm is set at 75 r/hr. The three ion chambers located in and about the vapor container are also sensitive to thermal neutrons. The recorder indicates the combined effects of gamma rays and thermal neutrons at these points. The monitors are located as follows:

1. Control Room - At the northernmost point on the vapor container wall, about 4 ft. above the floor.
2. Vapor Container Interior - Inside the upper portion of the vapor container, about 3 ft above the floor grating, on a railing opposite the access opening.
3. Vapor Container Entrance - Outside of the water-filled access hatch on the ground floor level.
4. Spent Fuel Pit - Under the fixed pit cover plate, next to the vapor container wall.
5. Demineralizer Room - Near the doorway just inside the demineralizing equipment room.

Typical radiation levels in these and other areas are given in Table XI-1.

b. Control of Access to Radiation Areas

All work in radiation areas or on potentially contaminated equipment is performed under and in conformance with the requirements of a radiation work permit. Permanently established radiation areas are shown in Fig. XI-1. Temporary radiation areas are established as required to control the spread of contamination and exposure to personnel.

The radiation work permit requires the evaluation by health physics of the external dose-rates present and of the potential contamination available. Health physics specifies the protective clothing and equipment required by personnel and the time allowed in the radiation area.

After evaluation and approval by area supervision, work may proceed. A total of 1061 radiation work permits were used in the 3 yr of operation.

TABLE XI-1
GENERAL AREA RADIATION LEVEL EXPERIENCE

<u>Area</u>	<u>During Reactor Operation</u>	<u>During Shut Down</u>	<u>Comments</u>
Vapor Container Interior	10-20 r/hr	< 100 mr/hr	Monitored by Jordan Detector. Radiation Work Permit Control during shutdown.
Work and Conden- ser Area (Vapor Container Entrance)	< 1 mr/hr	< 1 mr/hr	During shutdown portions of work area are estab- lished as entry control of vapor container. Moni- tored by Jordan chamber.
Health Physics and Chemistry Lab.	1-20 mr/hr	1-20 mr/hr	Permanent High Radiation Area.
Control Room	≤ 1 mr/hr	≤ 1 mr/hr	Monitored by Jordan area Monitor.
Turbine Area	< 1 mr/hr	< 1 mr/hr	
Demineralizer Room	50-2000 mr/hr	50 - 2000 mr/hr	Permanent High Radiation Area; Monitored by Jordan area Monitor.
Tankage Area	0-250 mr/hr	0-250 mr/hr	Permanent High Radiation Area
Solid Waste Storage Bldg.	0-1 r/hr	0-1 r/hr	Permanent Radiation Area
Spent Fuel Pit	None External	None External

FIG. XI-1 SM-1 PLANT LAYOUT

3. Waste Disposal

The accumulation discharge and/or shipment of radioactive wastes requires specific considerations and controls to assure that the gaseous and liquid waste release to the uncontrolled environment or storage and shipment as solid wastes does not create any radiation hazard to the general public.

a. Liquid Waste Disposal

The philosophy of liquid waste disposal at the SM-1 is to collect all radioactive waste, retain it until short lived activity has decayed and then dilute and disperse the material to Gunston Cove under controlled conditions. The waters of Gunston empty into the Potomac River at approximately 17 mi south of Washington, D. C.

Circulating water pumps at Gunston Cove deliver river water to the secondary system condenser. These pumps have a capacity of 4000 gpm and after passing through the condenser, this water is returned to the seal pit where it is used for diluting liquid radioactive wastes being discharged into the river.

The liquid waste storage facilities consist of two 250-gal laboratory waste storage tanks and one 5000-gal hot waste tank. The two laboratory waste tanks are designated "A" and "B" and each is provided with remote level indicators calibrated to read from 0 to 47 in. Drain lines from the hot sink in the laboratory and the sample drain in the demineralizer room drain to these tanks. One tank is used at a time leaving the other tank for standby or holding contaminated waste until its activity has decayed sufficiently to permit discharge. In addition, the washing machine discharge may be directed to the laboratory waste tanks. At the present time, the SM-1 building is being expanded to provide additional space for R & D, routine plant chemistry and health physics efforts. As part of this program modifications of the waste collection system are being made. All laundry waste will be collected in the hot waste storage tank before release. This change will reduce difficulties encountered from volume restrictions on the 250-gal waste tanks.

The hot waste tank receives radioactive water from several sources. Blowdown cooling water from the blowdown cooler and space cooler is normally discharged to the seal pit; however, should its activity be above tolerances a radiation monitor automatically operates a three-way control valve to divert the water to the hot waste tank. Primary coolant blowdown, seal leakage, secondary system blowdown and demineralizer room trench drains may also be diverted to the hot waste tank if necessary.

The total quantity of liquid radioactive waste discharged to the Potomac River since the beginning of SM-1 reactor operation is presented in Table XI-2.

TABLE XI-2
TOTAL RADIOACTIVITY RELEASED FROM HOLDUP TANKS

<u>Year</u>	<u>Gallons of Waste</u>	<u>Activity in Millicuries</u>
1957	11,818	8.4
1958	11,270	42.0
1959	27,340	68.0

The variation in the apparent annual average concentration of radioactivity discharged is due to the variations in the sources of the water and the types of operation involved in the R & D program. Typically, large quantities of low activity level water were discharged to the radioactive waste storage tank during the reactor head removal, replacement, and startup. Sampling of primary water for radioactivity analysis creates high specific activity in small volumes. However, discharge levels from the seal pit were maintained within accepted limits.

An effluent monitor is installed in the seal pit discharge line to continuously monitor and sample all water leaving the site. Samples of all liquid wastes are evaluated quantitatively in the laboratory to determine alpha and beta-gamma activities prior to discharge. The maximum permissible activity concentrations recommended by the National Bureau of Standards and Code of Federal Regulations are not exceeded; further restrictive limits are followed. The maximum permissible activity concentration in effluents is:

<u>Activity</u>	<u>Water ($\mu\text{c/cc}$)</u>
Alpha	1×10^{-7}
Beta-Gamma	1×10^{-7}

b. Stack Effluents

The airborne activity enters the main plant stack from the demineralizer room where primary water samples are taken. Additional airborne activity may enter the stack from the two laboratory waste tanks and the hot waste tank.

The stack air monitor detects and records the radiation level of beta-gamma airborne material and actuates a warning system when the activity in the air leaving the plant approaches potentially hazardous levels. In operation, a blower pulls air from the stack into a shielded detection chamber where particulate activity is collected on filter paper. A Geiger-Mueller tube operates a counting rate meter continuously indicating the beta-gamma activity deposited on the paper. The instrument has three ranges and an automatic range-changing mechanism. When activity reaches 2000 c/m, an amber blinking light comes on and an alarm sounds. At high range (10,000 c/m), a red blinking light comes on and an intermittent alarm is actuated. The activity level is continuously recorded on the control room monitoring recorder.

During normal operating conditions there is no release of airborne effluents. The vapor container is completely closed and there are no off-gasing systems required. During reactor shutdown the vapor container is ventilated with the effluent ultimately going to the plant stack, where it is monitored. The ventilation system provides for 8 complete changes of air per hr. The venting system is designed to allow passage of the air through a system of high efficiency filters to remove particulate matter under an arrangement permitting a single or multiple pass of the same air through the filters. The capacity of the ventilating system allows a discharge rate of 4000 cu ft per min. Based on this discharge rate, the average stack effluent activity does not exceed $1 \times 10^{-9} \mu \text{ c/cc}$

The only sources of airborne activity are from primary system sampling used for routine health physics, chemistry and R & D studies. This activity is observed in the immediate vicinity of the open sample point upstream of the demineralizers. Particulate activity collected on Hollingsworth and Vose filter paper showed the following isotopes: ++

<u>Isotope</u>	<u>Level ($\mu \text{ c/cc}$)</u>
Rb ⁸⁸	5×10^{-8}
Rb ⁸⁹	9×10^{-9}
Sr ⁹⁰	Cs 137*
Cs ¹³⁷ (Calculated from Rb ⁸⁹)	1×10^{-14}
Cs ¹³⁸	1×10^{-8}
Ba ¹³⁹	Cs ¹³⁸ *

* Concluded by comparing the half-lives of the respective gaseous precursors.

++ These data were tabulated during the period from Oct. 1958 to Jan. 1959.

The isotopes Cs¹³⁷, Sr⁹⁰ and Ba¹³⁹ were not detected experimentally. Their presence was calculated from the half-lives of their gaseous precursors and assumed transit time of 40 sec from core to sample point. The gaseous activity levels in the primary system water are given in the following table. ++

Isotope	Gaseous Nuclides in Primary Water dpm/ml
Kr ^{85m}	1.2 x 10 ⁴
Kr ^{87*}	
Kr ⁸⁸	2.0 x 10 ⁵
Xe ¹³⁵	7.5 x 10 ³

* No quantitative data possible because of interference from other isotopes.

Although only four nuclides of the fission product gases were identified and quantitatively analyzed, it is assumed that all fission product gases are present. During sampling of the primary system the gaseous activity level does not exceed that of the particulate activity. The total quantity of radioactive airborne material does not exceed one tenth of the amount permitted per day based on an average permissible effluent concentration of $1 \times 10^{-9} \mu \text{ c/cc}$ of air.

c. Solid Waste

Under normal operating conditions, the bulk of solid radioactive waste is accumulated as a result of plant health physics and chemical laboratory activities. The waste consists mainly of discarded or broken contaminated glassware, planchettes and paper towels. During maintenance shutdowns there may be such things as contaminated water filters, tools and replacement parts. These articles are collected in metal waste containers provided for this purpose. The level of radiation is not permitted to exceed 20 mr/hr at the surface of the container. Articles exceeding this level are handled separately to prevent any further exposure to personnel. Previous to June 1, 1959 all solid wastes were packaged in 5 gal steel cans which were sent to an outside contractor, Radiation Service Company, who undertook final disposal. Since that time the Army Chemical Corps has established procedures for the disposal of wastes. The quantities of solid waste for the first 3 yr of operation are given in Table XI-3.

++ These data were tabulated during the period from Oct. 1958 to Jan. 1959.

TABLE XI-3
SOLID RADIOACTIVE WASTE

<u>Date</u>	<u>Quantity</u>
8 July 1957	5 - 5 gal containers
31 July 1957	9 - 5 gal containers
4 Oct. 1957	5 - 5 gal containers
26 March 1958	10 - 5 gal containers
28 August 1958	8 - 5 gal containers
7 October 1958	8 - 5 gal containers
10 Dec. 1958	9 - 5 gal containers
13 May 1959	16 - 5 gal containers
11 August 1959	24 - 5 gal containers
23 Oct. 1959	48 - 5 gal containers
23 Oct. 1959	5 -55 gal containers
6 April 1960	21 -55 gal containers

The SM-1 utilizes a low pressure blowdown system for purification of the primary system. A mixed-bed ion-exchange demineralizer is used for concentrating and controlling the buildup of radionuclides in the primary system coolant. During operation these demineralizers are contained in a lead shield for radiation protection to personnel. Upon exhaustion of the ion-exchange columns a lead cover is placed into position over the demineralizer and bolted in place. The demineralizer and cask are then checked to insure compliance with ICC regulations and transferred to the Army Chemical Corps for disposal. Frequent radiation checks are made on the cask to insure that ICC limits are not exceeded. Operating experience at the SM-1 has indicated that the design of the cask is more than adequate and the resin is depleted before dose rate levels exceed allowable shipping dose rates. ⁽²⁾

The following demineralizers have been disposed of since the beginning of plant operations.

Demineralizer Shipments

<u>Demineralizer</u>	<u>Date</u>	<u>Receiver</u>
A1	5 Oct. 1957	ORNL
B1	30 June 1959	Army Chemical Corps
A2	11 August 1959	Army Chemical Corps
B2	6 October 1959	Army Chemical Corps
A3	19 April 1960	Army Chemical Corps

d. Spent Fuel Shipment

During the period from March 7 to May 17, 1959, the SM-1 was shut down for research and development program studies. The major objective was to obtain irradiation data on the SM-1 type core for future design and development purposes. At that time, the fixed fuel element from position 57, the control fuel element from position C and the absorbers from position 3 and C were shipped to ORNL for examination. Escorts accompanied the shipments to Oak Ridge for loading the Oak Ridge cask (B. E. Permit No. 712) in the spent fuel pit, and shipment was accomplished without difficulty. Radiation dose rates at the surface of the shipping cask were approximately 40 mr/hr, thus meeting ICC regulations without difficulty.

The Army is procuring special shipping casks which will be available for the shipment of SM-1 spent cores.

4. Environmental Monitoring

The environmental monitoring program is the responsibility of the Sanitary Engineering Branch, Military Engineering Department, ERDL, Corps of Engineers - U.S.A., at Fort Belvoir, Virginia. The information here summarized was taken from "Progress Reports" Nos 10 and 13 of the Environmental Radioactivity at Fort Belvoir and Vicinity. The method of sampling and sample analysis are discussed in the Progress Reports published previously.

The environmental monitoring program consists of periodic radiological analyses of precipitation, dust collected on gummed paper, air particles, SM 1 cooling water, SM-1 waste water discharge, and Gunston Cove sediment. A continuous record was made of the gamma background at the Sanitary Engineering Branch Office located at the ERDL site. The program was initiated in 1955 before the SM-1 went into operation.

Table XI-4 lists the environmental data of the periods July 1, 1958 through October 31, 1958 and July 1, 1959 through October 31, 1959. It should be noted that the high levels of activity present in the dust collected on the gummed paper during July 1, 1958 through October 31, 1958 is attributable to the fallout from the Nuclear Test Firing listed in Table XI-5 conducted by the United States, the United Kingdom, and the USSR. Cessation of the test firings occurred on October 30, 1958 and tests have not yet recommenced. Comparison of other areas of sampling analysis indicate that the SM-1 plant has not contributed to the radiation background level in the environment.

TABLE XI-4
ENVIRONMENTAL DATA OF JULY 1, 1958 THROUGH OCTOBER 31, 1958
AND JULY 1, 1959 THROUGH OCTOBER 31, 1959

	July 1, 1958 thru October 31, 1958		July 1, 1958 thru October 31, 1959	
	Average ($\mu\text{c/ml}$)	Range ($\mu\text{c/ml}$)	Average ($\mu\text{c/ml}$)	Range ($\mu\text{c/ml}$)
Precipitation (filtered)	1.9×10^{-7}	$0.5-3.4 \times 10^{-7}$	1.1×10^{-7}	$0-6.1 \times 10^{-7}$
(unfiltered)	2.6×10^{-7}	$1.0-6.1 \times 10^{-7}$	1.5×10^{-7}	$0.3-5.8 \times 10^{-7}$
SM-1 Inlet Cooling Water	1.8×10^{-8}	$0-6.5 \times 10^{-8}$	1.2×10^{-8}	$0-14 \times 10^{-8}$
SM-1 Outfall Cooling Water	2.0×10^{-8}	$0-5.74 \times 10^{-8}$	1.6×10^{-8}	$0-8.6 \times 10^{-8}$
Potomac River Water (Weekly)	Not sampled during this period.		0.9×10^{-8}	$0-2.4 \times 10^{-8}$
	($\mu\text{c/gm}$)	($\mu\text{c/gm}$)	($\mu\text{c/gm}$)	($\mu\text{c/gm}$)
Potomac River Sediment (Weekly)	0.4×10^{-5}	$0-0.8 \times 10^{-5}$	1.9×10^{-5}	$0-3.2 \times 10^{-5}$
Potomac River Sediment (4 Months)	1.2×10^{-5}	$0-2.5 \times 10^{-5}$	Not sampled during this period.	
Gunston Cove Sediment (Weekly)	1.6×10^{-5}	$0-4.9 \times 10^{-5}$	1.9×10^{-5}	$0-4.8 \times 10^{-5}$
	($\mu\text{c/ft}^2$)	($\mu\text{c/ft}^2$)	($\mu\text{c/ft}^2$)	($\mu\text{c/ft}^2$)
Dust collected on gummed paper	9.3×10^{-5}	$3.3-470 \times 10^{-5}$	12×10^{-5}	$2-38 \times 10^{-5}$

TABLE XI-5
NUCLEAR TEST FIRING SCHEDULE DURING JULY 1, 1958 THROUGH
OCTOBER 31, 1958

<u>Date</u>	<u>Country</u>	<u>No. of Shots</u>
2 July	United States	1
5 July	United States	1
11 July	United States	1
26 July	United States	1
1 August	United States	1
12 August	United States	1
22 August	United Kingdom	1
2 September	United Kingdom	1
11 September	United Kingdom	1
19 September	United States	1
23 September	United Kingdom	1
29 September	United States	1
30 September	USSR	2
2 October	USSR	2
5 October	USSR	1
8 October	United States	1
10 October	United States	1
10 October	USSR	1
12 October	USSR	1
13 October	United States	1
15 October	United States	2
15 October	USSR	1
16 October	United States	1
18 October	United States	1
18 October	USSR	1
19 October	USSR	1
20 October	USSR	1
22 October	USSR	1
22 October	United States	3
24 October	USSR	1
25 October	USSR	1
26 October	United States	2
28 October	United States	1
30 October	United States	1

D. EFFECT OF RADIATION LEVELS ON MAINTENANCE AND OPERATIONS

1. General

After two years of operation, at shutdown approximately 75% of the total radiation level at any one point is due to short-lived nuclides. About 80% of the short-lived radiation level is due to fission products. These are primarily I^{132} , I^{133} , I^{134} , Kr^{88} , and Cs^{138} . The other 20% of short-lived radiation are due to the active corrosion products Mn^{56} and W^{187} . For example, in January 1959 the dose rate on the reactor inlet pipe was 530 mr/hr, extrapolated back to time of reactor shutdown. After 24 hr the dose rate had decreased to 130 mr/hr.

The long-lived dose rate has been found to be due entirely to active corrosion products. The dose rate due to nuclides suspended or dissolved in the coolant is negligible compared to that due to nuclides deposited on the walls of the primary system. The active corrosion products contributing to the dose rate are Co^{60} , Co^{58} , Mn^{54} , Fe^{59} and Cr^{51} . Of these, Co^{60} and Co^{58} account for about 90% of the total. The Co^{60} is formed by a (n, γ) reaction on Co^{59} which is present in Haynes 25 flux suppressors and as an impurity in stainless steel. Co^{58} originates from a (n, p) reaction on Ni^{58} , which is present in stainless steel. Figure XI-2 shows the long-lived dose rate buildup at two selected points. The points are on opposite sides of the steam generator and the dose rate is due to the adherent deposit of activated corrosion products on the tubes. Certain points on the primary system have a static volume subject to the collection of crud. Dose rates at these points have reached 2000 mr/hr, but upon draining the primary system were reduced to about 50 mr/hr. Figure XI-3 shows the radiation levels due to long-lived nuclides on March 9, 1960 at various points in the vapor container.

2. Effect of Radiation Levels on Maintenance

Radiation levels at the SM-1 after two years of operation have not reached such magnitude that maintenance and operations have been seriously hindered. Several special maintenance activities, however, have necessitated careful planning and work scheduling to avoid overexposure to personnel. Since the SM-1 is also used as a training facility, student military personnel are present. Their availability has reduced the possible excessive loss of time or overexposure of regular personnel while performing special maintenance work.

FIG. XI - 2
LONG-LIVED DOSE RATE BUILDUP

FIG. XI-3 LONG-LIVED RADIATION LEVELS IN SM-1 VAPOR CONTAINER, MARCH 8, 1960, 48 HOURS AFTER SHUTDOWN

The effect of radiation levels on maintenance was evidenced in several instances at the SM-1. At times, it was necessary to shield the elbow on the pressurizer line while working on a thermocouple assembly located on the steam generator inlet pipe, about 4 ft from the pressurizer elbow. Shielding was accomplished by wrapping 1/16 in. thick lead sheets around the elbow, thus reducing the dose rate at the thermocouple. Dose rates at the elbow (2 r/hr) were reduced significantly when water from the primary system was drained and the crud accumulated in the elbow was apparently removed from the system.

In July, 1958, the circuit breaker controlling one of the 4000 gpm primary coolant pumps opened. An inspection of the pump electrical wiring and pump interior showed no defects. The radiation level of the impeller and shroud which had been in contact with the primary water was about 2 r/hr on contact. A total time of 74 hr was spent in disassembly, inspection, assembly and testing of the pump. There were no technical overexposures during the entire operation.

After 1 1/2 yr of reactor operation, an inspection of the steam generator water box and tube sheet was performed. Because of anticipated high radiation levels, manpower usage was carefully planned. Personnel involved in the initial operation (removing the lagging and loosening the bolts on the head) wore protective clothing as required by health physics rules. These included cotton gloves, coveralls, shoe covers and protective head gear.

When the work had progressed to the point where there was a possibility of coming in contact with the primary coolant, additional protective clothing was worn. This clothing included rubber coverall, rubber gloves and plastic face masks. In addition, gloves and shoe covers were taped to the arms and legs to assure that no water could get on the body. During the whole operation, personnel wore pocket dosimeters which were checked periodically to assure that the maximum permissible daily dose rate (50 mr/day) was not exceeded without permission. A health physicist was on duty during the entire operation. Dose rate readings at contact on the tube sheet of the steam generator varied from 6.3 r/hr on the inlet side to 2.9 r/hr on the outlet side. There were no technical overexposures during this operation.

In all, 673 manhours were expended in the inspection of the steam generator. This number includes the time expended in obtaining

research and development data. A total of 48 men assisted in the operation. It is difficult to estimate the time necessary for tube or tube sheet repairs. The time to remove and reapply the steam generator head was approximately 24 hr. This is exclusive of time that would be required for performance of any actual repairs. The radiation levels and contamination problems would be expected to increase the repair time compared to that for repair of a conventional power plant boiler.

In March 1959, the pressure vessel was opened to remove and inspect several control rods and fuel elements. During their removal, the inner shield tank water became contaminated with crud particles. The particles apparently were flaked off the control rods or fuel elements. The radiation level above the inner shield tank water, which is usually 1 mr/hr, increased to approximately 360 mr/hr. This level substantially reduced the working period for personnel and also hindered the visual examination. A simple but effective filtering system was used to clean up the activity and crud. A part of the water from the bottom of the shield tank was continuously withdrawn and passed through a cellulose filter, demineralizer, micrometallic filter and then returned to the top of the shield tank. Within 48 hr, the radiation level was reduced to about 10 mr/hr and no problems were encountered afterward.

3. Effect on Operations

Activity in the primary system has affected reactor operations in several areas. Whenever primary samples are taken for chemistry or health physics analysis, proper precautionary measures are followed. Compliance with these measures tends to lengthen the time for sampling beyond that required for similar sampling for chemical analysis in conventional power plants.

Many of the sample locations are in the demineralizer room where radiation levels have increased. This rapid buildup is due to two factors: (1) the demineralizers, which function both as a primary coolant filter and demineralizer, have accumulated a considerable amount of crud; (2) crud has collected in certain tees and stagnant portions or "dead legs" of the purification piping. After one year of operation the general background radiation in the area where the coolant is sampled was about 20 mr/hr. Radiation levels near some of the "dead legs" were approximately 2 r/hr.

Since SM-1 startup, five demineralizers have been replaced, or an average of one about every five months. The radiation level around

a typical unshielded spent demineralizer varies from 75 r/hr to 85 r/hr on contact. The demineralizer is encased in a lead shipping cask and surrounded by a concrete cubicle. Thus, personnel working in the demineralizer room are sufficiently protected from these high radiation levels. Since the top of the demineralizer cask is not in place when the demineralizer is in use, some radiation streams through the roof of the demineralizer room. The maximum radiation level has been measured at 85 mr/hr at waist height in this area. This is an area where access is not normally permitted.

Cleaning of a sintered stainless steel filter located downstream of the demineralizers has caused some difficulty. This filter serves to remove resin fines and small crud particles, with its radiation level at times reaching 6 r/hr. The filter is placed behind lead bricks and cleaned by boiling in water and acid for several hours. The radiation level is generally reduced to 200 mr/hr.

Radioactive particles that have settled on the bottom of the primary makeup tank have increased radiation levels at this point to 200 mr/hr. Since some operating personnel are frequently in this area, proper health physics rules are observed.

Except in the above areas (and in the vapor container) the effect of radiation is not seen in the other portions of the power plant. No detectable dose rate has been found in the control room, turbine room, or electrical room.

E. ACTION TO BE TAKEN TO REDUCE ACTIVITY BUILDUP

It is apparent from the preceding discussion and data that the activity buildup phenomenon is a potential problem area in the operation of pressurized water reactors. The problem is not fully solved. Experience gained in research, development, and operation of these reactors has revealed certain areas which improved design considerations are expected to reduce activity buildup. However, data are still lacking or are limited on the effects caused by modifying other design factors which are believed to affect the activity buildup.

The primary purification system also helps to reduce activity buildup. A portion of the primary coolant is continuously withdrawn and passed through a mixed bed demineralizer. The demineralizer functions both as a filter and ion-exchanger and thus removes radioactive particles dissolved or suspended in the coolant. Radiochemical data shows that approximately 30% of the total activity generated is removed by the demineralizer.

The selection of fuel element cladding and other materials in the core is an important design consideration. Materials that yield long-lived gamma emitting nuclides upon irradiation should be minimized or avoided. For example, the Haynes 25 alloy used as flux suppressors in the SM-1 Core I control rod fuel elements is undesirable from the activity buildup standpoint. This alloy contains 50% cobalt. Although the surface area of the material exposed to the coolant is relatively small, a major part of the Co^{60} in the SM-1 is thought to originate from this alloy. To minimize further buildup of Co^{60} in the SM-1, the flux suppressors for Core II are internal to the fuel plates of the control rod fuel element. The effect of activation of the cobalt in stainless steel must also be considered. Fuel cladding for Core II has less than one-tenth the cobalt impurity that was present in Core I. Thus, activity arising from this source will be reduced during Core II operation.

Decontamination of the steam generator and associated components is planned for late 1960. Decontamination of these components will reduce their radiation level and thus should considerably reduce the long-lived background level in the vapor container.

F. EVALUATION OF HAZARDS ASSOCIATED WITH PLANT OPERATION, TRAINING, AND RESEARCH AND DEVELOPMENT EXPERIMENTS

The multiple objectives of the SM-1 plant require the continual review of radiation hazards to assure safe operation of the plant. The precautions and controls required during routine reactor operation and the collection of operating data are covered in the SM-1 Operating Manual. The radiation hazards associated with the training program are similar to those for normal operation and the trainees are continually under direct supervision of qualified reactor operators. However, the frequent start-up and shutdown of the plant for training creates additional stresses on the plant components and as a result more frequent maintenance of this plant is required than for a continuously operating plant.

Various studies and tests on the SM-1 plant have been carried out during 3 yr of operation under the R & D program. The areas of interest covered are core physics measurements, shielding, water chemistry and radio chemistry, instrumentation, system response studies, and thermal studies. Each test in the program involving plant operation or design modifications was reviewed on an individual basis with respect to associated hazards prior to its approval. The guidelines used in evaluating the hazards were generally:

- a. Compliance of any equipment involved in tests with applicable codes, such as ASME etc.

- b. Evaluation of the magnitude of potential hazards due to failure of involved components.
- c. Evaluation of availability and magnitude of corrective or controlling action in case of failure of components.
- d. Evaluation of the operating procedures and instructions involved.
- e. Evaluation of anticipated radiation levels to which personnel would be exposed during operation of the test.
- f. Method of control of unusual levels of radioactive waste not compatible with present waste disposal systems.
- g. Protective equipment required which is not normally available in routine operations.

On tests involving the energy content of the reactor system, fission product inventory, and/or safety systems and interlocks, special hazards reports were submitted to the AEC for evaluation.

G. EMERGENCY EVACUATION PROCEDURES

Emergency evacuation procedures have been formulated for various types of probable nuclear incidents at the SM-1 (Chapter V of SM-1 Operating Manual). Potential incidents have been designated according to any of three classes, depending upon severity of the situation. A system of warning horns are installed in the area around the SM-1 for prompt evacuation in cases of emergency. Depending on the classification of the incident, evacuation can proceed from the immediate area of the plant to increasing distances from the reactor site, according to predetermined circumferential zones. The evacuation proceeds to an assembly point where personnel are organized and plans can be implemented to further evaluate the incident and initiate necessary emergency procedures and re-entry plans. No incidents have occurred in the first 3 yr of operation which have required initiation of any phase of these procedures.

The SM-1 emergency plan is coordinated with an emergency plan being formulated by the Commanding General, Fort Belvoir, for the entire Fort Belvoir, Virginia area.

CHAPTER XII PLAN FOR OPERATION BY THE ARMY BEGINNING WITH FY 1961

A. INTRODUCTION

On July 1, 1960 the Corps of Engineers, as agent of the Department of the Army (DA) and Department of Defense (DOD) will assume responsibility for control, operation, safety, and property accountability for the SM-1.

After July 1, 1960, the main objective of the SM-1 operation will be to train operators for military field nuclear power plants and to perform Research and Development to include but not be limited to those tasks required to:

1. Extend knowledge of operating characteristic of the SM-1.
2. Support design, construction, operation and maintenance of future military plants, including prototypes thereof.

B. ORGANIZATION FOR MANAGING SM-1 OPERATION

1. General

Within the Corps of Engineers, SM-1 responsibilities will be carried out by the Special Assistant to the Chief of Engineers for Nuclear Power (SANP), who will exercise staff supervision through the Nuclear Power Division, (NPD), Office, Chief of Engineers (OCE).

SM-1 operation and maintenance will be performed by a Government crew under the supervision of the Chief, Nuclear Power Field Office (NPFO) with technical assistance from the Office of Assistant Director (Army Reactors), Division of Reactor Development (ARM). Contractor services will be utilized as required for technical consultation on operation and maintenance matters and on special maintenance tasks. In addition, the Chief, NPFO, may contract for special technical assistance for the execution of specified R & D tasks and testing.

2. Responsibilities Within NPFO, Fort Belvoir, Virginia

a. General

The relationship of the various major elements of NPFO to the operation of the SM-1, the detailed organization of the

operating crew and staff, and support for plant operation are as follows:

- (1) NPFO will be organized as shown in Fig. XII-1 which shows an Operations Branch, a Training Branch, and Operations Support Branch and Administrative Branch.
- (2) The Operations Branch will be responsible for operation and maintenance of the SM-1; for implementation of the overall plant schedule set by the Chief, NPFO; for developing and changing as necessary the maintenance system for the plant; for furnishing advice on operational and maintenance aspects of other ANPP pressurized water plants; for on-the-job and supervisory level training of operation and maintenance personnel; for performance of approved tests; for reporting data from tests, investigations, problems, etc., to appropriate elements of NPFO; for SM-1 property and supply; and for keeping of detailed operational and maintenance records.
- (3) The Training Branch will be responsible for operational training of students, preparation of reports on operational and maintenance experience which relate to MOS fields, doctrine, crew organization, etc.; furnishing advice in specialty fields such as water chemistry, health physics, instrumentation, process control, etc. Operational control of student training will be under the direction of a training coordinator (assigned to the Training Branch but physically located in the SM-1) but actually carried out by the shift supervisors assigned to the Operations Branch.
- (4) The Operations Support Branch (OSB) will provide the NPFO technical capability for support of SM-1 operations and R & D. It will have responsibility and capability (backed up as necessary by Belvoir on-site contractor) for taking immediate action on SM-1 problems and to define problems for solution, by more extensive effort, if necessary, by the AEC and its development contractors. OSB will provide for NPFO coordination, liaison and technical support of SM-1 "in-plant" R & D.
- (5) Administrative Branch, NPFO, will be responsible for:
 - (a) Matters pertaining to budgets and contracts relating to SM-1 operations and for coordinating preparation of reports on operating costs.

FIG. XII-1

ORGANIZATION OF NPFO

- (b) Scheduling, coordinating, and supervising preparations for tours of the SM-1, and assistance in the execution of these tours.
- (c) Personnel actions effected in transfers between the Training and Operations Branches, to field plants, etc; for scheduling of special medical examinations, whole body counts, etc.; and for coordination of all other outside requirements which affect operational availability of personnel.

b. Support for Plant Operations

- (1) Support will be rendered the Operations Branch from three general areas--from within ANPP (primarily from other branches of NPFO), from U. S. Army Engineering Research and Development Laboratory (USAERDL), and from contractor services.
- (2) The Operations Support Branch will render engineering support consistent with its mission as outlined above. Engineering and Technical Support will be provided by ARM as requested by NPD.
- (3) USAERDL support will be continuous in administrative matters. Engineering assistance will be requested from USAERDL, as appropriate
- (4) USAERDL Radiological Isotopes Committee will provide assistance on health physics and radiological safety matters
- (5) Contractor consulting services will be necessary from time to time, as will certain special maintenance support. Provisions will be made for this support under suitable contract arrangements. The scope of such a contract will include.
 - (a) Engineering effort to assist in resolving SM-1 problems except those which will require solution by ARM and its development contractors.
 - (b) The effort required to supplement NPFO personnel in conducting special tests or installation of special equipment.

C. EXPERIENCE WITH AND FUTURE USE OF TRAINEES IN PLANT OPERATION

Plant trainees have been used very successfully in their specialties such as instrument technician, electrician, health physicists, chemistry technicians and mechanics.

Trainees have been used only under direct supervision when operating the plant controls.

Trainees will continue to be used in their specialty fields. The training received when performing actual necessary tasks at the plant provides a sound background as well as building confidence leading to an easy transition to regular plant assignment. Trainees are especially valuable during periods when heavy maintenance or plant modifications are being performed. The trainees are also used very effectively in R & D projects especially in the chemical field. Health physics trainees have been used almost exclusively when round the clock work in radiation areas must be performed. It is not desirable to entrust operations of controls in the control room to trainees without direct supervision.

D. BASIC PRINCIPLES FOR REVISION OF PROCEDURE AND FOR ACCOMPLISHING PLANT MODIFICATIONS

The requests for revising procedures will, in most cases, originate with the operating crews. However, requests may also originate with the Training Branch or with others as a result of plant modification or malfunction review or as a result of the R & D program. The requests for a plant modification will in most cases, originate as a result of the R & D program either as a result of new information developed by the R & D tasks or in order that a specific task may be performed.

Each plant modification or change in operating procedure will be reviewed individually from a hazards point of view prior to its initiation.

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APPENDIX A-1 - METEOROLOGY AND CLIMATOLOGY

1. INTRODUCTION

Fort Belvoir, Va. is located approximately 17 miles SSW of Washington, D.C. just west of the Potomac River and between Dogue and Pohick Creeks. The latter creek flows into Gunston Cove which forms the southern boundary of the military reservation. The location of the Army Package Reactor (SM-1) installation is on the south shore of this relatively small Ft. Belvoir peninsula facing Gunston Cove. The terrain in the immediate vicinity of the reactor location slopes upward steeply from the river's edge, from approximately 2-ft mean sea level (msl) to 130 ft msl in 1/3 of a mile. The entire peninsula is also heavily timbered.

2. SOURCE OF DATA

Although no meteorological data exist for the SM-1 site itself or for Ft. Belvoir, very complete meteorological records have been taken for many years at both the Central Office of the Weather Bureau in Washington and at the Weather Bureau Airport Station at the Washington National Airport (WNA). The latter office is also located along the Potomac River 13 miles NNE of the SM-1 site. Climatological data from the Quantico Marine Corps Air Station (20 miles SW of the site, also along the Potomac) have also been examined.

3. CLIMATOLOGICAL REVIEW

There does not appear to be any significant change in the general meteorology of the area between Ft. Belvoir and Washington, D.C., so that for most engineering purposes, the published data for Washington, D.C., may be used.

Washington, D.C. lies at the western edge of the middle Atlantic coastal plain, about 50 mi east of the Blue Ridge Mountains and 35 mi west of Chesapeake Bay. Elevations range from a few feet above sea level to about 40 ft in parts of the NW section of the city.

Summers are warm and humid and winters mild; generally pleasant weather prevails in the spring and autumn. Coldest weather normally occurs in late January and early February, when average low temperatures are in the upper 20⁰'s and average high temperatures in the middle 40⁰'s. Warmest weather normally occurs during the middle of July with average daily high temperatures in the upper 80⁰'s. The record high temperature of 105.6⁰ occurred on July 20, 1930. The record low temperature of

14.9° below zero occurred on February 11, 1899, just preceding the worst blizzard in Washington's climatic record.

The average annual snowfall is near 20 in. The greatest recorded single fall was 28 in, which occurred in two days, of January 1922. Snowfalls approaching the magnitude of this storm are rare, and snow accumulation of the bad winter storms in Washington is normally nearer 10 in. than 30 in. Also, the 10-in. falls usually melt off rapidly.

The normal annual precipitation is near 41 in. No pronounced wet and dry seasons are evident, rainfall being well distributed throughout the year. The greatest amount for a 24-hr period was 7.31 in., which occurred on August 11-12, 1928. The longest period without appreciable rainfall extended from October 15 to November 11, 1901 - a total of 28 days.

The average date of the latest killing frost in spring is April 10 and the latest recorded date of a killing frost is May 12, 1913. The average date of the first killing frost in the fall is October 28 and the earliest recorded date October 2, 1899. The average length of the growing season is 200 days.

There are thunderstorms in Washington 33 days a year on the average. During summer months they often bring sudden and heavy rain showers and may be attended by damaging winds, hail, or lightning. On June 9, 1929 a violent local thunderstorm with wind gusts up to 100 mph was recorded. Two severe hailstorms have been recorded - one in April 1938 and one in May 1953.

Tornadoes rarely occur, but two severe ones have been recorded - one in April 1923 and one in November 1927.

Tropical disturbances occasionally, during their northward passage, influence Washington's weather - mainly with high winds and heavy rainfall; but extensive damage from this cause is rare.

Floods occur in the Potomac River about once every two years.

4. SURFACE WIND DIRECTION AND SPEED

The hourly observations for an 8-yr period 1945-52 for both Washington National Airport (WNS) and Quantico Marine Station were studied in detail. Tables A-1 and A-2 present the percentage frequency of the wind direction and wind speeds. As is indicated in Table A-1 the wind direction frequency in this area appears to have two maxima, one from the south and one from the northwest. This results from the fact that the prevailing winds are from the south during the summer months and from the northwest during

the winter. These average wind directions should be applicable to the SM-1 site.

It is necessary to examine wind structure during periods of precipitation to consider the effect of washout of possible waste contaminants. Table A-2 presents the percentage frequency of wind direction for WNA during those hours when precipitation is falling (approximately 11% of the time). As may be noted, the prevailing direction in these cases is from the NNE and NE while there is a secondary maximum from the S and SSW.

Wind direction is also important when the lower atmosphere is very stable and atmospheric diffusion is at a minimum. The frequency of this type of condition is discussed later in this report, but the wind direction frequencies are included in Table A-2. The prevailing direction associated with inversion conditions is southerly. It is also noteworthy that nighttime inversions (0300GCT) are accompanied by a larger frequency of calms.

In Table A-3, it will be noted that a larger frequency of weaker winds (3-7 knot interval) have been observed at Quantico than at WNA and vice-versa for the stronger wind class intervals. This difference in wind speed frequencies must be due, in part, to the fact that the anemometer at WNA is somewhat higher above ground level. More than likely, because of the proposed location of the SM-1 site, it is our estimate that a significantly larger frequency of weaker winds should be expected. This estimated frequency is included in table A-2. These weak winds will be very predominant during the night and early morning hours (2200 to 0800). From records at Washington, D.C., the average velocity is 7 mph and may range from an average of 10.7 mph for a windy month to 5.0 mph for a calm month. The maximum speed ever observed which existed for at least 5 min was 53 mph from the NE. An instantaneous gust of 77 mph has been observed once in 74 yrs of record.

5. PRECIPITATION

As indicated in the Climatological Review, the SM-1 area receives approximately 40 in. of precipitation annually which is spread over approximately 120 days. The heaviest precipitation will be observed during the summer and spring months. The maximum amount of precipitation ever recorded in Washington, D.C., (during 74 yrs) over a 24-hr period was 7.3 in; over 12 hrs, 6.2 in; over 3 hrs, 4.1; over 30 min, 2.4 in; and over 10 min, 1.2 in.

The average snowfall is 13.1 ins. which falls on approximately four days during the winter months. The greatest amount ever recorded for a 24-hr period was 25.0 in. and for one storm 28.0 in. The maximum depth

on the ground was 34.2 in.

6. TEMPERATURE

The SM-1 site temperatures range from 0° to 100° over a year. The average temperature is approximately 56° and the average number of degree days is approximately 4300.

7. ATMOSPHERIC STABILITY

To study the diffusion potentiality of the Washington area, the twice daily radiosonde observations (1000 and 2200 EST) from WNA were examined for a 5-yr period (October 1945 through September 1950) to determine the frequency of atmospheric inversions. (An inversion is present if the temperature increases with height, hence diffusion is retarded.) Of 3600 radiosonde observations, an inversion condition was found to be present 8% of the time at 1000 EST and 41% at 2200 EST. The inversion data presented are only for the time of the radiosonde observations. For 2200 EST observations, which are shortly after the onset of the nightly inversion, fewer stable periods are experienced than would be expected during the early morning hours. Undoubtedly, if data were available for 0400 to 0600 EST, frequency of inversion conditions would be much greater.

Hourly visibility observations for both Washington National airport and Quantico were also examined for the 8-yr period 1945-52. On the average, 9% of hourly observations at WNA and 12% at Quantico were reduced to 6 mi or less by fog, which is another indication of stable atmospheric conditions and poor diffusion probabilities. Fog at the SM-1 site may be slightly more than that experienced at Quantico. As with the atmospheric inversion conditions mentioned above, fog will be more frequent during the night and early morning hours.

TABLE A-1
PERCENTAGE FREQUENCY OF WIND DIRECTION

(Based on hourly observations January 1945 - December 1952)

	<u>Washington National Airport</u>	<u>Quantico</u>
N	3.6%	5.7%
NNE	5.5	2.4
NE	5.3	5.8
ENE	3.4	4.0
E	2.3	2.8
ESE	2.2	2.3
SE	3.4	5.8
SSE	6.6	5.8
S	10.1	8.5
SSW	10.9	5.0
SW	4.9	6.0
WSW	3.3	2.9
W	3.9	9.8
WNW	8.1	7.4
NW	10.1	9.7
NNW	7.0	8.1
CALM	9.2	8.0

TABLE A-2
PERCENTAGE FREQUENCY OF WIND DIRECTION

(Based on hourly observations October 1945 through September 1959)

	<u>When Precipitation Is Falling</u>	<u>When Inversion Base Is Below 500 ft. MSL</u>	
		<u>0300 GCT</u>	<u>1500 GCT</u>
N	5.4%	2.0%	1.5%
NNE	12.8	2.1	3.6
NE	12.8	3.1	3.9
ENE	6.7	1.5	0.6
E	3.6	1.1	2.2
ESE	3.3	0.5	4.7
SE	4.0	2.2	3.6
SSE	5.9	6.2	23.7
S	8.1	13.8	15.8
SSW	9.0	21.8	10.4
SW	4.4	10.8	6.8
WSW	2.5	3.8	0.8
W	2.4	3.2	4.3
WNW	3.8	4.2	5.3
NW	6.8	5.9	3.1
NNW	6.8	4.1	6.2
CALM	3.5	14.0	3.6

TABLE A-3
PERCENTAGE FREQUENCY OF WIND SPEED GROUPS (KNOTS)

(Based on hourly observations January 1945 - December 1952)

	<u>Calm</u>	<u>3-7</u>	<u>8-12</u>	<u>13-20</u>	<u>21-30</u>	<u>31-40</u>	<u>40</u>
Washington National Airport	9.2	27.6	33.8	24.7	4.5	0.2	*
Quantico	8.0	52.1	30.8	8.6	5	*	0
Estimate for SM-1 Site	12	60	20	5	3	-	-

*Less than 0.1%

APPENDIX A-2 - GEOLOGY AND HYDROLOGY

1. AREA OF INVESTIGATION

The Fort Belvoir Military Reservation lies in both the Coastal Plain and the Piedmont Plateau Physiographic provinces. The reactor site, however, lies only in the Coastal Plain province. References to the Piedmont Plateau Physiographic province will be limited to subsurface geologic and hydrologic information that might have a bearing on the site.

The reactor site which is about 500 ft square, extends from the water's edge, Gunston Cove, back and up the slope in a northeast direction to an elevation of about 60 ft above mean sea level. This area lies between two small streams that rise a short distance back from Gunston Cove, which joins the main channel of the Potomac River about 500 yds southeast of the site.

2. GEOLOGY

Immediately underlying the site are unconsolidated deposits of Cretaceous Age. These deposits are about 450 to 500 ft thick, and dip to the southeast at a rate of about 30 ft to the mi. Below the unconsolidated strata are granite rocks of the same type as those which crop out a short distance northwest of the area under consideration. These granite rocks comprise the so-called "bedrock", and are of the Pre-Cambrian Age.

The unconsolidated rocks, which are Cretaceous, consist of sands, gravels, and clays. These strata have been correlated with the Potomac Group, which includes three formations - The Patuxent, Arundel, and Patapsco in Maryland. In this area, however, these sediments are represented by near shore, or even terrestrial deposits, and no differentiation of all formations has been possible.

The logs in Table A-4 show the Cretaceous rock penetrated in the drilling of wells at the Fort Belvoir Military Reservation. The logs show the nature of sediments penetrated and the depth to bedrock at these points. The elevation of land surface at these wells is estimated.

3. EARTHQUAKE HISTORY

This is not an area of frequent earthquakes. Only one earthquake is reported in N. H. Heck's Earthquake History of the United States. This earthquake occurred at 5:22 a.m., August 31, 1861. The quake's epicenter was at 38.8° north latitude and 77.0° west longitude. The intensity of this shock was 5 on the Rossi-Forel scale of intensity. A quake with an intensity of 5

on this scale is described as a shock of moderate intensity, generally felt by everyone.

4. SURFACE-WATER HYDROLOGY

The reactor site, as mentioned above, is adjacent to Gunston Cove. Gunston Cove, an inlet of the Potomac River, is formed by the submersion of the lower parts of valleys of the Accotink and Pohick Creeks.

There are no published data on temperatures and chemical quality of the Potomac River water near this site. Oral reports from Interstate Commission on the Potomac River Basin, however, indicate that the temperature of samples of water from the Potomac River between Marshall Hall and Hallowing Point ranged from 32° F to 90° F. The river is tidal, but the chloride content of the water at these same points has not exceeded 20 parts per million. It should be noted, however, that these temperatures and chloride determinations are not the result of a continuous sampling over an extended period of time. They are, rather, isolated determinations made over a period of years, and the conditions under which the samples were taken are not known. The chloride content might be different at different depths and in different parts of the channel. Changes of the river tides also may cause differences in chloride content.

As the Fort Belvoir Military Reservation is between the sampling points for the analysis mentioned above, it is reasonable to expect that the chloride content of the river opposite the reactor site might be comparable to that noted above.

However, another factor to consider in evaluating the temperature and chemical quality of surface water near the proposed reactor site is that Gunston Cove is an inlet of the Potomac River, which receives fresh water from two small streams (Accotink and Pohick Creeks). This freshwater discharge into the cove tends to freshen and/or hold out salt water from the Potomac.

5. GROUND WATER HYDROLOGY

a. Occurrence and Movement of Ground Water

Precipitation falling to the earth's surface either percolates down through the soil, evaporates, or moves along the land surface until it enters surface streams, lakes, or the ocean.

The water that percolates downward into the soil is either intercepted by roots of vegetation or percolates down to the zone of saturation,

the upper surface of which is the water table. When water is added to the zone of saturation it tends to make a high point on the water table. These points of high water table are characteristic of points of recharge, whereas points of discharge such as springs, seeps, or wells are low points on the water table. Ground water moves from point of recharge through the interstices of unconsolidated rocks, and through fractures in consolidated rocks.

In the Fort Belvoir area ground water occurs in both consolidated and unconsolidated rocks. In the area northwest of the reactor site consolidated rocks are relatively impervious, but they have been fractured and ground water occurs and moves through these fractures. The productivity of wells drilled into these rocks is dependent upon the size and number of fractures intercepted. As the size and number of fractures generally decrease, the chances of getting water are very poor at depths of more than about 350 ft in rocks of this type. Yield of such wells is small, and some of them yield little or no water.

At the reactor site the area is underlaid by unconsolidated gravels, sands, and calys for a depth of about 500 ft. In these unconsolidated deposits ground water occurs in the sands and gravels in which the water moves through the interstices between sand grains. In these deposits a properly constructed well drilled into a saturated sand would produce water. The amount of water than can be produced from such a sand is dependent upon the hydrological characteristics of the aquifer. Wells in aquifers of this type usually are more uniform in production than wells in fractured consolidated rocks.

b. Ground Water Use at Fort Belvoir

Little factual information is available regarding ground water in the Fort Belvoir Military Reservation. The water supply for the reservation itself is obtained from the city of Alexandria Water System. Three drilled wells, however, are interconnected with this water supply, but they are used only for emergencies and supplemental supplies. In addition to the three wells mentioned above, the Engineer Research and Development Laboratories have one drilled well, used to supply cooling water to the climatological laboratories.

Table A-5 shows the available information on the records for these four wells. Chemical analysis of water from each well is given in Table A-6.

c. Evaluation of the Site

It is believed that the site, considering the geologic and hydrologic factors, is a reasonably satisfactory one for the construction and operation of a power producing nuclear reactor.

The loading of the reactor is less than 4000 psf. From the only logs available showing subsurface materials, Table A-4, it is seen that in the excavation for this reactor building the material underlying the foundation is capable of supporting 4000 psf of loading.

The earthquake history of this area indicates no potential problem from possible earthquakes. As far as known there is no reason to believe that the occurrence and intensity of earthquake shocks would be different from the past.

The site offers reasonably good facilities for using an abundant source of river water for cooling. The temperature of the water ranges from a minimum of 32°F to a maximum of 96°F.

Another consideration in choosing a site for a reactor is the potential dangers of contamination of water by radioactive material and subsequent dangers to public water supplies downstream and/or down dip from the reactor site. Under normal operations the Health Physics regulations would control any such potential contamination, but such regulations and operation standards do not take into account danger from accidental spillage. The reactor site is very well situated in relation to this problem. In case of an accidental spilling of radioactive material on the ground, its possible course of travel is as follows:

If the radioactive fluid were run overland, due to the inability of the soil to absorb it - a probability in periods of high precipitation or frozen ground - the contaminants would be intercepted by either the small streams northwest or southeast of the site or by Gunston Cove. If intercepted by the small streams the contaminant would still be discharged into Gunston Cove. From Gunston Cove the contamination would move into the Potomac River, and then into Chesapeake Bay. In no case is a public water supply taken from the Potomac River below Gunston Cove. However, there is considerable recreational use made of the lower Potomac and Chesapeake Bay area. Also, damage to a highly developed fishing industry would have to be considered. Whether an accidental spillage of radioactive fluid would contaminate the lower Potomac and Chesapeake Bay area is dependent

upon the type and concentration of the fluid. However, there would be a very large dilution factor in this reach of the stream and bay.

If the radioactive fluid percolates into the soil, rather than running overland to surface streams, the probability of damage to a public supply would be remote. The rate of travel through the soil and sub-surface material would be much slower than surface travel. Ground-water flow is measured in ft per yr, rather than mi per day, as for surface flow. Whether the delay in travel time encountered in the slower flow of ground water would be enough to neutralize the contaminating fluid would, of course, depend upon the nature of the fluid.

Dangerous elements in a radioactive fluid possibly would be absorbed on the clay particles of the underlying formation, or removed from the fluid by base-exchange. If, however, the radioactive elements did reach the water table, they would then move down gradient. Due to the fact that the site itself is on a peninsula-like projection of land on Gunston Cove, the elements probably would be discharged into the Cove itself. In this case, they would then follow the travel course outlined in a preceding paragraph; however, there would be considerable delay in time from the occurrence of the accident until the contamination appeared in the cove.

If the contaminated ground water did not discharge into Gunston Cove, it would slowly continue down dip in the aquifer. Eventually, the contamination might reach a public supply down dip, however, enough information is not now available to evaluate the probability of such an event.

From all considerations of the geology and hydrology, the reactor site in the Engineer Research and Development Laboratories section of Fort Belvoir appears to be as satisfactory a site as could be selected in the metropolitan Washington, D.C. area.

TABLE A-4
DRILLERS LOGS OF WELLS DRILLED AT FORT BELVOIR

Fort Belvoir Well #1
Layne Well #1

<u>Depth in ft</u>	<u>Thickness in ft</u>	<u>Material Penetrated</u>
0	5	Gravel
6	1	Boulders
25	19	Heavy Gravel
31	6	Gravel and sand
44	13	Blue sandy clay
54	10	Fine gray sand
57	3	Brown sandy clay
67	10	Fine brown sand
79	12	Medium sand and gravel
90	11	Gray sand, some clay
100	10	Blue sand, and boulders
108	8	Dark sandy clay
216	108	Brown sandy clay
237	21	Blue sand
253	16	Hard sand and boulders
		Rock to 278', core to 290'

Fort Belvoir Well #2
Sydnor Well #2

5	5	Soft sandy mud and gravel
25	20	Red clay
30	5	Red clay
40	10	Gray clay
43	3	Sticky blue clay
55	12	Sticky gray clay
58	3	Sticky gray clay
75	17	Gray clay and sand, no water
80	5	Gray sandy clay
90	10	Sticky sandy clay
100	10	Sand and clay, no water
102	2	Sand and clay, no water
112	10	Pasty sand and clay
120	8	Blue clay
138	18	Blue clay
143	5	Sandy clay
144	1	Hard streak
158	14	Sticky clay, streaked
175	17	Sticky streaked clay
180	5	Streaked clay, sticky
190	10	Red clay, sticky
200	10	Streaked clay, sticky
214	14	Sticky streaked clay
225	11	Clay with some sand
238	13	Sand and clay
240	2	Sticky blue clay
245	5	Sticky blue clay
265	20	Silty solid clay
280	15	Sandy clay
300	20	Coarse sand and clay
311	11	Coarse sand and clay
317	6	Hard sand and clay
326	9	Hard sand and clay
330	4	Shale rock
333	3	Shale rock

TABLE A-5
DATA ON WELLS AT FORT BELVOIR, VA.

Well Location	#1 Filter Plant	#2 About 0.7 Mi NNE of Filter Plant	#3 About 1.1 Mi NE of White Stone Pt.	#4 ERDL Area Climatological Laboratory
Elevation (in ft)	24	105	95	87
Depth (in ft)	245	322	375	372
Diameter (in in)	18-10-8	10-8	8	10
Depth to Bedrock (in ft)	253	320*(2)	500*(2)	525*(2)
Water Bearing Formations (Elevations)	97-102 231-253	-----	-----	110-135 252-382
Water Level (in ft) (Below Land Surf., Est'd.)	8	96	84	86
Gallons (gpm)	185*(1)	136*(1)	236*(1)	175
Drawdown (in ft)	86	64	74	34
Remarks	Post Well #1 (Layne # 1) Behind Filter Plant	Post Well #2 (Sydnor #2) West of Outdoor Amphitheater	Post Well#3 (Sydnor#4) About. 15 mi. NW of Sewage Disposal Plant #2	Post Well #4 (Sydnor #3)

*(1) 72 Hr Test - March 1945

*(2) Estimated from report test drilling near by

TABLE A-6
CHEMICAL ANALYSIS OF WATER SAMPLES
FROM WELLS AT FORT BELVOIR, VA.

Analyses by Geological Survey, United States Department of the Interior
(parts per million)

Laboratory	49109	49110	49111	49112
Date of Collection 1954	May 18	May 18	May 18	May 18
Silica (SiO ₂)	44	25	30	33
Iron (Fe), dissolved <u>1/</u>	.33	.59	.56	.02
Iron (Fe), total	.59	1.2	1.6	.03
Manganese (Mn), dissolved <u>1/</u>	-	-	-	-
Manganese (Mn), total	-	-	-	-
Calcium (Ca)	2.8	7.1	1.0	.7
Magnesium (Mg)	2.0	6.7	.1	.1
Sodium (Na)	-	-	-	-
Potassium (K) (Calc.)	4.1	26	7.5	9.2
Bicarbonate (HCO ₃)	0	0	0	0
Carbonate (CO ₃)	10	107	14	12
Sulfate (SO ₄)	8.4	11	3.6	4.6
Chloride (Cl)	5.5	2.1	2.8	4.9
Fluoride (F)	.1	.1	.1	.1
Nitrate (NO ₃)	.2	.3	1.6	.5
Dissolved solids				
Sum				
Residue on evaporation at 180° C	79	126	58	61
Hardness at CaCO ₃	16	47	4	2
Non-carbonate <u>2/</u>	8	0	0	0
Specific conductance (micromhos at 25° C)	75.3	177	34.4	39.6
pH	6.3	7.4	6.1	5.8
Color	3	4	35	3
Carbon Dioxide (CO ₂) (Calc.)	8.0	6.7	15	30

For notes 1/ and 2/ see next page.

TABLE A-6 (Cont'd.)

<u>SAMPLE NUMBER</u>	<u>FORT BELVOIR WELL NUMBER</u>	<u>LOCATION see Table A-5</u>
Laboratory No. 49109	#1	Accotink filter plant
Laboratory No. 49110	#2	0.7 mi NNE Filter Plant
Laboratory No. 49111	#3	1.1 mi Northeast White Stone Pt.
Laboratory No. 49112	#4	ERDL area

1/ In solution at time of analysis.

2/ Includes hardness of all polyvalent cations reported.

APPENDIX B - VAPOR CONTAINER MISSILE PENETRATION

1. TYPE OF MISSILE

Selection of the maximum credible missile during a nuclear incident in the vapor container is not subject to precise calculation, but is a matter of judgment. Credible missiles include objects which might become free during a nuclear incident and be propelled by an expanding jet of vapor against the wall of the vapor container.

Several credible missiles were considered:

- a. A 2-in., 1500 lb (test) steel globe valve. This valve weighs 50 lb. The complete projected area of such a valve is 30 sq in. in the bottom-on position. The projected area in the end-on position is approximately 7 sq in., but after the initial impact the bonnet and stem would also resist penetration so that the estimated average penetrating area of this valve is 18 sq in.
- b. A heavy wall thermometer test well, normally welded into the primary coolant circuit, weight of 5 lb, with an area of 1.25 sq in.
- c. A 2-ft length of 2-in. steel bar with a weight of 21 lb and an area of 3 sq in.
- d. A 4-ft length of 2 in. schedule 160 steel pipe with a weight of 30 lbs and an area of 4.4 sq ft.

The most damaging missiles are those which combine maximum weight in minimum area of impact.

The credible missiles are summarized below.

<u>Type</u>	<u>Weight, lb.</u>	<u>Projected Area of Impact, Sq. In.</u>
2-in., 1500-lb steel globe valve	50	18
Thermometer test well	5	1.3
2-in., steel bar, 2-ft long	21	3
2-in., schedule 160 steel pipe 4-ft long	30	4.4

2. VELOCITY OF MISSILE

Missiles can acquire velocity either by being struck by a jet of vapor issuing from the point of fracture and being propelled by that jet, or if the

missile is hollow and contains fluid at the time of rupture, the missile can be self-propelled by the expanding fluid.

The problem becomes one of determining the velocity of the missile and its kinetic energy and establishing whether or not such a missile will penetrate the vapor container wall.

3. SELF-PROPELLED MISSILES

Of the above tabulated credible missiles, the only one which could contain fluid and be self-propelled is the 2-in schedule 160 pipe.

If the pipe were initially filled with 1200 psi, 450°F water, the available energy in the fluid when expanded to final pressure in the vapor container (assumed atmospheric as the worst case) is 38 Btu per lb of fluid. The weight of water per foot of pipe is 0.8 lb and this has a kinetic energy, if all the thermal energy were converted, of 23,000 ft-lb per foot of pipe. Assuming that the pipe, which weighs 7-1/2 lb per ft and the weight of the water is accelerated through the entire travel, the corresponding final velocity is 435 fps. The available maximum travel within the vapor container is approximately 40 ft and such an object would reach a velocity of approximately 435 fps in that distance.

Similarly, if such pipe were filled with 425-psi saturated steam at the instant of detachment, the available heat energy for expansion to atmospheric pressure is 255 Btu per lb. The weight of such fluid in one foot of pipe is 0.17 lb and the kinetic energy of the steam is 30,000 ft-lb per foot of pipe. This results in a velocity of 510 fps if all the thermal energy of the fluid went into developing kinetic energy. Again, such an object would reach a velocity of approximately 510 fps in 40 ft of travel.

Consequently, the velocity of a self-propelled missile has been taken at 500 fps, which is conservative, since it has been assumed that all the energy of a fluid is transferred into kinetic energy of the missile at 100% efficiency.

4. JET-PROPELLED MISSILE

A credible missile upon being detached may be propelled by the jet of escaping fluid and continue in that jet of fluid until it strikes the wall of the vapor container. The jet, which has mass and velocity, imparts impulse to the missile from which the increase in velocity of the missile from an initial state of rest can be determined. With the impulse force known, first the acceleration and then the final velocity of the missile in 40 ft of travel

can be calculated.

The fluid pressure, initially at 1200 psi and 450°F, drops very rapidly to saturation pressure of 423 psia by the expulsion of a few pounds of water at the rupture opening. The 423-psia fluid expands into the vapor container and when expanded to 15 psia reaches a peak velocity a few ft from the orifice of rupture. The missile is propelled by and accelerated in this jet of vapor. After the jet has fully expanded, that is, when the jet has reached vapor container pressure, the assumption is made that the jet continues to propel and accelerate the missile until the missile impinges on the vapor container wall. Since no data have been found which would determine the shape of the jet between its point of maximum expansion and the vapor container wall, it has been assumed that the jet moves at constant velocity with all elements of the jet remaining parallel to each other rather than continuing to expand at some angle. This is conservative. The impulse of the vapor jet acting on the missile provides force to accelerate it. There is neither distance nor time for the missile to reach the velocity of the jet, but it does reach the theoretical velocity calculated below.

5. RATE OF FLOW OF PROPELLING JET

The rate of release of fluid in the jet depends on the area of the rupture and the velocity of the issuing fluid. It has been assumed that a missile of the above type would be associated with an opening approximately 3 in. in diameter as might result from a 2-in. valve or other object tearing out a 2-in. nipple by which it was attached to the primary coolant circuit. Since the size and shape of such a rupture opening is unknown, no coefficient of contraction has been used. Use of a conventional coefficient of 0.7 would indicate a 3.6-in. diameter opening.

In the following:

Subscript m refers to the missile

Subscript f refers to the fluid

With an isentropic expansion between 423 psia initial pressure and atmospheric pressure and a coefficient of velocity of 0.9 due to friction in the orifice, the velocity of the jet will be given by:

$$V = C_v \sqrt{2g \frac{\Delta p}{\rho}}$$

$$\rho = 52 \text{ lb/ft}^3$$

$$V = 243 \text{ ft/sec.}$$

that is, the velocity of the issuing jet near the orifice of rupture is 243 fps.

The area of a 3 in. hole is $A = 0.049$ sq ft and the weight rate of flow of the fluid is given by:

$$W_f = V_f \int A = 629 \text{ lb/sec}$$

that is, fluid issues from the orifice at 629 lb per sec. Neglect of the velocity of the first few pounds of fluid accelerated as the pressure drops from 1200 to 423 psia is more than compensated for by the conservative assumptions of the missile remaining in the jet at 100% efficiency for its full travel.

6. ANGLE OF JET

The weight of fluid actually impacting on the missile decreases after the jet expands starting at 629 lb per sec at the point of rupture and decreasing as the missile moves further and further from the point of rupture, since the total weight of fluid in the jet remains constant, but as the area of the jet increases, the weight of fluid actually striking the missile on its projected area continually decreases. Thus the final velocity of the missile depends upon the assumption as to the angle of the jet.

Data obtained from the U. S. Department of Defense, Ballistics Research Laboratory in paper No. 843, AFSWP No. 768 indicate that the advancing front of a jet of air suddenly released from an orifice expands at an angle of 60° to 70° from the orifice pressure down to atmospheric pressure. On the other hand, in steam turbine practice, nozzles are designed with an angle of 10° to 15° since with a greater angle the jet does not expand fast enough to maintain contact with the walls. With an unconfined and uncontrolled expansion within the vapor container and with the jet issuing from a ragged opening, it is considered that the angle is more likely to approach the greater value above, but calculations have been made both for a 12° and a 60° angle.

The force of the jet acting on the missile, assuming the latter to be a flat plate surface and that it turns the jet through a 90° angle, is given by

$$F = \frac{W_f}{g} (V_f - V_m)$$

This force is a maximum when V_m is 0 at the start and the force becomes 0 when the V_m is equal to V_f (V_m never reaches V_f in the available distance of travel). In the above expression, the drag coefficient of the jet acting on the missile has been assumed equal to unity. For sharp edged objects the drag coefficient is reported as 0.8 to 1.0 and independent of Reynold's Number.

Computations of the impulse of the jet on the missile have been undertaken for the 2-in., 1500-lb valve weighing 50 lb. The computation is one by trial and error. An assumption is made of the final velocity of the missile which permits computing the impulse force from the above expression. Knowing the force, the acceleration of the missile is computed from the expression

$$a_m = \frac{g}{W_m} \times F$$

From the acceleration, the final velocity is computed from the expression

$$V_m = \sqrt{2 a s + V_o^2}$$

where s is the distance traveled.

The result of the computations for the 2-in., 1500-lb (test) valve weighing 50 lb with a projected area of 30 sq in., assuming both a 12° and a 60° angle of expansion for the jet are:

<u>Item</u>	<u>12° Angle</u>	<u>60° Angle</u>
Weight of missile, lb	50	50
Projected area of missile, sq in.	30	30
Final velocity of fluid on expansion from 1200 psia to 15 psia, fps	1, 277	1, 277
Weight rate flow of fluid from orifice, lb per sec	629	629
Distance from orifice at which the jet reaches 15 psia, ft.	7.8	1.43
Velocity of missile when jet has reached atmospheric pressure, fps	480	210
Distance jet travels at constant velocity, ft	32.2	38.7
Final velocity of missile on traveling total distance of 40 ft, fps	700	500

Since both the self-propelled missile and the jet propelled missile with an expansion angle of 60° reach a theoretical velocity of 500 fps, it is considered that this represents a credible velocity. Nevertheless, further calculations have also been made with the 700 fps missile velocity corresponding to 12° jet angle, although this is considered to be greater than a credible velocity.

The calculation of missile velocity has been made only for the 2-in., 1500 lb valves weighing 50 lb. The velocity reached with the other missiles would be less. As stated above, force is proportional to the weight of the missile and acceleration is proportional to force divided by the weight of the missile. Since the weight of the fluid striking the missile is proportional to the projected area, acceleration is, in turn, proportional to the area of the missile divided by the weight of the missile. That missile which has the greatest ratio of area to weight will obtain the greatest acceleration and hence the greatest final velocity. In the following tabulation this ratio is greatest for the 2 in. valve. Hence, other missiles would have a smaller final velocity.

<u>Missile</u>	<u>Ratio of Area to Weight of Missile</u>
2-in. steel valve	.60
Thermometer well	.25
2-in. steel bar	.14
2-in. steel pipe	.15

7. PRIMARY VESSELS

The reactor, pressurizer, and steam generator would be subjected to thrust forces in the event of a rupture of the vessel. For an idealized instantaneous failure (such as the bottom dropping out) which would expose an area at the largest diameter, the maximum vertical thrust was calculated by two methods. One method was to consider the fluid to be ejected in an isentropic expansion. The second method employed the use of shock hydrodynamics in which the jet velocity is calculated by considering the fluid to be accelerated across a rarefaction wave which propagates from the rupture surface. Both methods are in good agreement. The vessels are anchored to resist the thrust forces. Thrust forces calculated by the method of shock hydrodynamics gives about $1.5 (10)^6$ lb thrust for the reactor and steam generator and $2 (10)^5$ lb for the pressurizer. The anchorage is conservatively designed to withstand $1.3 (10)^6$ lb and $3 (10)^5$ lb respectively.

8. MISSILE PENETRATION THROUGH VAPOR CONTAINER

The degree of penetration of the wall of the vapor container has been investigated by several formulations of a semi-empirical nature which predict the penetration of steel and concrete by an object possessing kinetic energy. These have been made available by the U.S. Department of Defense. The first is that used by the Ballistics Research Laboratory. The second is that used by the Navy in their publication Navy Docks Bulletin No. P-51.

The vapor container consists of a 1/8-in. steel plate inner liner, 2 ft of reinforced concrete and a 3/4-in. outer steel shell. The inner liner has been neglected. Manhole covers are 2-1/2-in. steel plate.

The BRL formulation for concrete is

$$P = \frac{6Wd^{1/5}}{d^2} \times \left(\frac{V}{1,000} \right)^{4/3}$$

where

P = penetration of missile, in.

W = weight of missile, lb

d = diameter, in.

V = velocity, fps

BRL recommends that the calculated penetration in concrete be multiplied by a factor of 1.3 to insure no complete penetration due to the cracking of concrete ahead of the point where the missile stops. Since in the vapor container the 2 ft layer of concrete is followed by 3/4-in. steel, this 1.3 factor is not considered absolutely essential.

The penetrations of the several missiles with a missile velocity of 500 fps are:

<u>Missile</u>	<u>2-in. Steel Valve</u>	<u>Thermometer Well</u>	<u>2-in. Steel Bar</u>	<u>2-in. Steel Pipe</u>
Weight, lb	50	5	21	30
Area, sq in.	18	1.3	3	4.4
Diameter, (equiv.), in.	4.8	1.3	2	2.4
<u>If velocity = 500 fps</u>				
penetration, in.	7	7.5	14.5	15
penetration x 1.3, in.	9	10	19	19.5
<u>If velocity = 700 fps</u>				
penetration, in.	10	11	21	22
penetration x 1.3, in.	13	14	27	29

It will be noted that none of the missiles with 500 fps penetrate the concrete nor do the missiles at 700 fps penetrate except when using the 1.3 factor. Navy Docks states that in a compound wall of steel and concrete, 1 in. of steel is equivalent to 12 in. of concrete so that if the complete vapor container wall is considered equivalent to 33 in. of concrete, there is no penetration even with 700 fps missile velocity.

The BRL formulation for penetration of steel is expressed by

$$t^{3/2} = \frac{1/2 m V^2}{K^2 \times 17,400 \times d^{3/2}}$$

where

t = wall thickness, in.

m = mass of the missile, slugs

V = velocity of missile, fps

K = a constant depending on the grade of steel and = 1

d = the diameter of the missile, in.

Using the above formulation, the missile penetrations are:

<u>Missile</u>	<u>2-in. Steel Valve</u>	<u>Thermometer Well</u>	<u>2-in. Steel Bar</u>	<u>2-in. Steel Bar</u>
<u>If Velocity - 500 fps</u>				
Penetration in.	1	.8	1.5	1.5
<u>If Velocity - 700 fps</u>				
Penetration in.	1.5	1.3	2.3	2.3

Thus the 2-1/2-in. steel plate, where there is no concrete, is sufficient to prevent penetration even with 700 fps missile velocity.

The above penetrations may be rechecked with Navy Docks formulation. This is

$$D = K A_p V^2$$

where

D = depth of penetration, ft

K = a coefficient depending on the nature of the concrete and is 0.00799 for mass concrete, 0.00426 for normal reinforced concrete, such as would be used in building construction and is 0.00284 for specially reinforced concrete.

$$V' = \log_{10} \left(1 + \frac{v^2}{215,000} \right)$$

V = the velocity of the missile, fps

A_p = weight of missile per sq ft of projected area, lb per sq ft.

Using the above formulation, the following calculated penetrations result:

<u>Missile</u>	<u>2-in. Steel Valve</u>	<u>Thermometer Well</u>	<u>2-in. Steel Bar</u>	<u>2-in. Steel Pipe</u>
Weight, lb	50	5	21	30
Area, sq in.	18	1.3	3	4.4
A _p , lb per sq ft	400	500	1,000	1,000
<u>If velocity = 500 fps</u>				
Penetration, in.	8	10	19	19
<u>If velocity - 700 fps</u>				
Penetration, in.	12	16	30	30

With a velocity of 500 fps, concrete is not penetrated. With a velocity of 700 fps, which is more than a credible velocity, two of the missiles penetrate the concrete, but considering the total equivalent thickness of steel in concrete of 33 in., there is still no theoretical penetration.

APPENDIX C-1 EJECTION OF CONTROL RODS RESULTING FROM A RUPTURE OF COLD, PRESSURIZED PRIMARY SYSTEM

A rupture of the primary system in the cold, pressurized condition will result in a pressure wave that travels at the speed of sound or faster throughout the primary loop. The pressurizer, containing hot liquid and steam, will gradually expel fluid from the pressurizer, but this will occur over a considerably longer period than is required to relieve the pressure of compressed liquid in the primary loop. Thus, the actual pressure relief will be more gradual than would result from releasing only compressed liquid.

During the time the pressure wave travels the length of the core, there is a net force exerted upward on the control rods. It is conservatively assumed that a pressure wave of 1200 psi travels through the core with the velocity of sound in water, 4800 fps. Actually the pressure wave will not be so great because of the pressurizer influence, and the speed of sound increases significantly with increased pressure differential.

During the 400 microseconds the pressure wave travels through the control element fuel plates, an impulse of 0.82 lb-sec is imparted to the 72-lb control rods, which results in a rise of the control rod, if unrestrained, of 0.0008 in. and a final velocity of 0.35 fps. This velocity will cause a total control rod lift of 0.031 in., as the effect of gravity restores the rod to its initial position. Based on a maximum control rod worth of 2.46 cents/in. this lift corresponds to a 0.076% maximum increase in reactivity from all seven rods, lasting for 30 msec.

APPENDIX C-2 EJECTION OF CONTROL RODS RESULTING FROM A RUPTURE OF THE HOT PRESSURIZED PRIMARY SYSTEM

An investigation has been carried out to determine the upward force which might be applied hydraulically to the control rods following a major rupture of the primary system, and its effect on their position or motion. The concurrent growth of voids in the reactor has been calculated and the net change of reactivity from these opposing influences has been computed. A summary of the results of these computations is shown graphically in Fig. C-1.

Assuming the accident to be a complete, instantaneous rupture of the 12-in. primary line on the discharge side from the reactor, a pressure drop and flow from bottom to top of the reactor would develop, tending to lift the control rods. On the other hand, as this flow builds up in excess of the normal flow through the reactor, steam bubbles will form which will lower and finally completely stop the reactivity. This flashing to steam results from stored energy and is independent of reactor response to the accident. It is assumed that the control rods are de-clutched from the control mechanism at the same initial instant to permit them to fall freely and scram the reactor. However this freedom would also permit their being lifted instead, as soon as the combined effect of pressure gradient, entrainment force and flotation exceed the weight of the rods.

The worst possible condition preceding the accident was taken to be the point in the maximum credible accident sequence when the primary pressure has risen to 1500 psia (the setting of the pressure relief valve) due to overheating of the entire primary water content and compression within the steam dome due to expansion of the water, followed by a further temperature rise to the saturation value of 1500 psia at 596°F. This constitutes a far greater quantity of stored energy than could be reached before rupture if the pressure relief valve failed to open.

Since the time required to reach this 1500 psia and 596°F. point is 3.7 min, compared to 17 sec required for primary fluid to complete its circuit, this initial condition has been taken as applicable for both the upper and lower portions of the reactor vessel. The following values were used:

Liquid volume above core ("B" Chamber), ft ³ -----	27.4
Liquid volume below core ("A" Chamber), ft ³ -----	50.0
Area of opening discharging B (severed 12 in. pipe), ft ² -----	0.71
Area of opening discharging A (Core passages), ft ² actual --	2.08
effective ---	0.90

FIG. C-1 PROGRESS OF VARIABLES AFFECTING REACTIVITY FOLLOWING RUPTURE OF PRIMARY SYSTEM.

The effective core passage area is derived from the calculated pressure drop through the reactor at rated primary circulation, consisting predominantly of frictional loss, by considering the core as an orifice with the characteristic $v^{1.8} \sim 2gH$, where the difference between the exponent 1.8 and the standard value of 2.0 is the variation of friction coefficient with velocity.

Two major premises of this analysis are that:

- a) Conditions and properties of liquid and gas in each chamber will, of their own accord, rigorously follow saturation equilibrium values during the blow-down.
- b) The whole incident will be completed so rapidly that negligible gravitational separation of water and steam can take place. Instead the two will remain together in a fairly homogeneous mixture of liquid and vapor bubbles.

The following nomenclature will be used in explaining the method of analysis:

Symbols

P = Pressure, psia
 T = Temperature, $^{\circ}F$
 v_s = Specific volume, ft^3/lb
 u = Internal energy of fluid, $\frac{Btu}{lb}$
 V = Velocity, ft/sec
 W = Weight, lb

Subscripts, etc.

ℓ = liquid
 g = vapor
 ev = increment of the function gained or lost during evaporation
 m = mixture
 B = "B" chamber (above the core)
 A = "A" chamber (below the core)
 $'$ = mean value for the time or weight increment under calculation

The calculating procedure is as follows:

A. Blowdown Sequence for Each Chamber Treated Independently

1. Prepare from standard steam tables a graph of saturation $P, T, v_{s,g}$ and u_{ev} , all versus u_{ℓ} as the abscissa.

2. For either chamber assume successive weight increments of water (or water-steam mixture) expelled, without reference to the time required for this to take place.
3. Determine the volume thus made available for steam, recognizing that after the first step some of the expelled material will be steam, and that the remaining steam will expand as the pressure decreases.
4. Calculate the weight of liquid which can evaporate into the space made available, the internal energy required to do this, and the resulting reduced value of u_ℓ for the unevaporated liquid remaining in the vessel.
5. By maintaining a running plot of the resulting values of u_ℓ and of %-evaporated - weight versus quantity expelled, and extrapolating to the mid and end-points of each new interval, the effective values of $v_{s,\ell}$, $v_{s,g}$ and u_{ev} needed for each step can be obtained from the steam table chart. The new values of P and T can also be read off this chart and $v_{s,m}$ derived, while percent reduction in density is figured directly from the weight expelled. A trial and error method is used to compute the performance in the first interval.
6. The corresponding graph for the other chamber is prepared by plotting the same derived conditions against values of weight expelled corrected in the ratio of the total volumes of the two chambers.

B. Combined Blowdown of A & B Chambers versus Time

1. Assume intervals of time (10 to 50 msec range)
2. Estimate P' and $v'_{s,m}$ for each chamber for the period.

NOTE. By trial and error for first one or two points.
By extrapolation of running graph thereafter.

3. Calculate velocity for discharge from B:

$$V = \sqrt{2g \cdot 144 (P'_B - 14.7) v'_{s,m} B}$$

4. Calculate velocity for discharge from A to B:

$$V = 1.8 \sqrt{2g \cdot 144 (P'_B - P'_B) v'_{s,m} A}$$

where the radial 1.8 is used instead of 2 to allow for the high L/D ratio and resulting frictional characteristics of the core passages. (See discussion of equivalent area on the preceding page).

The working chart prepared to expedite this and the next step checked closely with independently derived points of pressure drop vs. flow in the pertinent range, using standard methods and Pigott's friction factors, based on water.

5. Calculate volume and weight flow out of B, from A to B, and resultant net for B, for the period.
6. Determine from the previously prepared working charts the new values of P , T , and $v_{s,m}$ for A and for B.
7. By plotting cumulative net weight expelled from B and from A vs. time and extrapolating the lines, obtain these values for midpoint of the next time interval.
8. Tabulate and plot, in addition to the cumulative values of W expelled, P_A , P_B , $(P_A - P_B)$ and percent reduction in density.

C. Resulting Rod Ejection Sequence

1. The effect on the control rod assemblies of the developing pressure gradient and primary coolant flow has been analyzed, taking account of the actual geometry of a complete control rod assembly and of the flow through it. The overall pressure drop between A and B has been broken down into its component fractions, taking account of kinetic and frictional effects. These were multiplied by their appropriate projected areas, to determine the force applied to the rod for each value of overall pressure gradient.
2. Rod displacements have been calculated based on the total equivalent mass of the rods, racks and engaged rotating assembly, acted on by the total force calculated as described on the preceding page, plus the flotation effect. Friction of the rotating assembly has been neglected, since this is the more pessimistic treatment yielding more rapid rod ejection.

The results of this investigation are presented in Chapter VII, and illustrated by Fig. VII-8.

The actual reduction in reactivity will be more rapid than shown for two reasons:

1. The controlling reduction in density will be an integrated mean within the core, roughly 30% of the way from chamber A conditions to those of chamber B. This would amount to a 22.5% reduction, for example, concurrent with 18% in A.

2. The calculations have ignored the continuing heat generation of the core, which would actually accelerate the local boiling and density reduction in the core. Thus 20 Btu/lb working in the range where $u_{ev} = 600$ Btu/lb could produce additional evaporation of 3-1/3% of each pound passing through, or an additional reduction in density of about 3%.

APPENDIX D - 10 MW CONSTANT POWER RUNAWAY

In determining the time dependence of the important variables of the reactor power plant system during the events leading up to the maximum credible accident, a simple analytical model can be employed because of the relatively long period of time involved. The more important of the simplifying assumptions may be summarized as follows:

1. With a primary loop circulation time of less than 20 sec. and an overall excursion measured in terms of minutes, the primary system can be considered to be at a uniform average temperature.
2. The reactor, primary loop and the secondary loop may be simplified to a three element system, such that the heat developed in the reactor is transmitted to the primary heat sink. Part of this heat energy is accumulated in the primary heat sink and part of it is transmitted to the secondary heat sink, which is the water existing in the steam generator.
3. The heat absorbing capacity of the metallic structure can be neglected since the time for the excursion is short enough that the amount of heat transferred into the metal will be insignificant. Further, the heat so stored is not available for the pressure peak developed after the rupture, but will be released over a considerable time interval after the rupture.
4. The heat transfer coefficient of the steam generator is constant throughout the incident.

With these simplifying assumptions the heat balance of the system at any instant may be written as:

$$q = W_p C_p \dot{T}_p + W_s C_s \dot{T}_s \quad (D-1)$$

where	C = Specific heat of water	Btu/lb-°F
	q = Reactor power	Btu/sec.
	W = Weight of water	lb
	T = Water temperature	°F

and subscripts p and s refer to the primary and secondary systems respectively. Dotted quantities indicate derivatives with respect to time.

The primary and secondary temperatures may be related through the heat transfer characteristics of the steam generator. Since it has been assumed that the characteristics of the steam generator are constant during the incident,

$$W_s C_s \dot{T}_s = hA (T_p - T_s) \quad (D-2)$$

where A = Heat transfer area of steam generator ft^2
 h = Heat transfer coefficient of steam generator $\text{Btu/ft}^2\text{-}^\circ\text{F-sec}$

From this relationship it is possible to determine T_p in terms of the first and second time derivatives of T_s . Substituting these values in equation (D-1), the differential equation for the system may be written as:

$$\frac{\ddot{T}_s + ha (W_p C_p + W_s C_s) \dot{T}_s}{(W_p C_p) (W_s C_s)} = \frac{hA}{(W_p C_p) (W_s C_s)} q \quad (D-3)$$

For the case of a constant power excursion $q = \text{constant}$, and the solution to equation (D-3) is:

$$T_s = T_{is} + \frac{(W_p C_p)(W_s C_s)}{hA(W_p C_p + W_s C_s)} \left[\frac{hA(T_{ip} - T_{is})}{W_s C_s} - \frac{q}{W_p C_p + W_s C_s} \right] \left[1 - e^{-\frac{hA(W_p C_p + W_s C_s)t}{(W_p C_p)(W_s C_s)}} \right] + \frac{qt}{W_p C_p + W_s C_s} \quad (D-4)$$

where the subscript i refers to the initial value.

The primary temperature can then be found from equation (D-2) to be:

$$T_p = \frac{W_s C_s T_s}{hA} + T_s$$

$$= \frac{W_s C_s}{hA} \left[\frac{hA(T_{ip} - T_{is})}{W_s C_s} - \frac{q}{W_p C_p + W_s C_s} \right] e^{-\frac{hA(W_p C_p + W_s C_s)t}{(W_p C_p)(W_s C_s)}} + \frac{q W_s C_s}{hA(W_p C_p + W_s C_s)} + T_s$$

For the particular conditions of the SM-1, when the power level of the reactor remains constant during the incident, these equations reduce to:

$$T_s = 415.5 - 33.5e^{-0.069t} + 0.75t$$

and $T_p = 430.5 + 11.5e^{-0.069t} + 0.75t$

The performance of the system has also been investigated for the case in which reactor power varies with time, the manner of this variation having an important effect upon the time response of the system. In view of the strong negative temperature coefficient of the system, a fair approximation to reality can be obtained by assuming a linear variation of power with time. Thus the reactor power may be written as:

$$q = q_0 + Qt$$

where Q = Rate of increase of reactor power.

The differential equation of the system becomes:

$$\ddot{T} + B\dot{T} = q_0 + Qt$$

The solution to which may be written as:

$$T = Ce^{-Bt} + \frac{Qt^2}{2B} + \left(\frac{q_0}{B} - \frac{Q}{B^2} \right) t + \left(\frac{Q}{B^3} - \frac{q_0}{B^2} + D \right)$$

where C and D are constants of integration. B , q_0 and Q refer to the system parameters and operation characteristics.

Numerical results were obtained for the SM-1 system for linear power increases with rates of rise as high as 2000 kw/sec. Simulator results have shown even without the inclusion of a negative temperature coefficient, local boiling will limit the rate of power rise to very low values so that the constant power case is considered to be a satisfactory approximation.

APPENDIX E - PRESSURE IN VAPOR CONTAINER FOLLOWING A MAXIMUM CREDIBLE ACCIDENT

The pressure-time relationship in the vapor container following a maximum credible accident has been calculated and the results shown in Chapter VIII, Fig. VIII-2. The following is a summary of the basic assumptions and design characteristics and the methods of calculation used.

A. ASSUMPTIONS AND DESIGN CHARACTERISTICS

1. Total energy released immediately following rupture is 7.32×10^6 Btu. This release is essentially instantaneous. This value varies slightly from the value of 7.4×10^6 Btu used in the discussion of Chapter VIII because it was arrived at by entirely independent calculations by a subcontractor who used a slightly different method in the detail calculations which resulted in the curve of Fig. VIII-2. Agreement is within less than 1%, however, which is considered to be good. The following values were used:

Net volume of vapor container (corrected for internal)

Construction	37,000 ft ³	
Total volume of primary circuit	200 ft ³	
Total volume of secondary circuit	138 ft ³	
	<u>Wt. - Lb.</u>	<u>u-10⁶ Btu</u>
Water in primary system at instant rupture (1500 psia., 596°F)	8500	5.14
Water expelled to vapor container from primary system through pressure relief valve (Av. Temp. 1450°F)	431	0.26
Steam expelled from pressurizer	69	0.08
Water in secondary system	2900	1.75
Steam in secondary system	82	0.09

2. Afterheat released by fission product decay during 48 hrs following rupture is as given in Fig. IX-3. The various effects operating during this period are:

- (a) Release of decay heat from fission products.
- (b) Release of heat stored in hot metal parts.
- (c) Absorption of heat by primary shield water and structure. This is accomplished by heat transfer through the walls and free water surface of the shield tank, realistic heat transfer values being used. The shield tank was assumed not to fail and release its contents into the vapor container. Heat absorption of concrete shielding outside the vapor container is ignored.
- (d) The water spray is turned on (manual emergency system) approximately 2 hrs after the rupture and continued for 46 hrs at 20 gpm.

B. CALCULATION METHODS

1. Conservation of internal energy was used within the system under consideration.
2. The pressure in the vapor container was calculated by trial and error, assuming a final temperature and pressure, calculating volumes of steam and water and repeating until the total volume equals that available in the vapor container. Because of the rapidity of the pressure and temperature rise, the initial peak pressure is based only on energy stored in the steam and water released.
3. The heat released by hot metal parts takes place by boiling off the water remaining in the primary system. The metal parts are assumed to be at the initial operating temperature (450°F) since the time of rise from such conditions to those at the moment of the rupture is too short to increase the temperature of this large mass of metal (46,000 lb) appreciably. This heat release takes place in approximately 1/2-hr. and amounts to approximately 1.2×10^6 Btu.
4. The heat is absorbed by the 2 ft concrete lining of the vapor container and is the major effect which causes the pressure to fall rapidly after the initial peak. This effect continues during the later phases of heat release from the metal parts and fission products. The heat absorbing capacity of the concrete was

calculated by determining, at the end of each interval of elapsed time, the depth of penetration of the heat wave and the average temperature rise of the concrete. Density of the concrete was taken as 150 lb/ft^3 and its specific heat as $0.22 \text{ Btu/lb} - ^\circ\text{F}$.

5. Overall rates of heat transfer to the shield tank follow those used in previous calculations of a similar problem. They were assumed to fall no lower than $50 \text{ Btu/hr-ft}^2 - ^\circ\text{F}$. When the temperature difference between the shield tank water and the surrounding vapor became small, the use of the rate equation was discontinued and only the heat balance equation was used.
6. Spray water at 75°F was assumed available for the final 46 hr of the 48-hr interval studied. Its heat absorption was calculated by a heat balance equation, assuming thermal equilibrium of the water and vapor.
7. Net interval energy was calculated at the end of several time intervals out to the 48-hr elapsed time. Total pressure was established at each time interval by trial and error as in the case of the initial peak.

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