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NUCLEAR POWER PLANT

MARTIN
NUCLEAR DIVISION

DISCLAIMER

MND-M-1853

PM-1 NUCLEAR POWERPLANT

HAZARDS SUMMARY EVALUATION

MND-M-1853

(First Submission)

October 15, 1959

The Martin Company
Nuclear Division

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INTRODUCTION

The PM-1 Nuclear Power Plant will provide the electrical power and space heating required for the Operations Area of the Sundance Air Force Station six miles northwest of Sundance, Wyoming.

The Sundance Air Force Station is a radar surveillance installation. The radar equipment, with its support facilities and the PM-1 Power Plant, will be located atop Warren Peak, which is the highest mountain in the area. The administrative facilities and base housing will be located approximately four miles from Warren Peak.

The remoteness of the Warren Peak site and the presence of the Sundance Air Force Station provide very favorable conditions for a service test of the PM-1 design.

The PM-1 is a small, pressurized-water, nuclear power plant. It is designed to produce one thousand kilowatts of electrical power and seven million Btu per hour of low-pressure steam for space heating. The PM-1 is of modular design, capable of being transported by aircraft.

This Hazards Summary Report presents a detailed description of the Warren Peak site, together with a description of the PM-1 Nuclear Power Plant. The potential hazards of operating the PM-1 at this site are evaluated.

In the hazards evaluation, consideration is given to the protection of plant-operating personnel through the conservative design of radiation shielding and through the use of detailed protective procedures. The evaluation of the shielding design takes into account radiation conditions during normal operation as well as during maintenance and refueling operations. The outline of the operating procedures presents methods of controlling the plant and personnel during normal and emergency periods.

In addition, an evaluation of the hazards to the general public is presented. This evaluation considers the effects of a hypothetical accident occurring within the PM-1. The hypothetical accident postulates a series of failures believed beyond the realm of probability. Of particular concern was the resulting radiation level at the town of Sundance, which is the largest population group within a 15-mile radius of Warren Peak.

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CONCLUSIONS

An evaluation has been made of the potential hazards associated with the operation of the PM-1 Nuclear Power Plant at Warren Peak, Wyoming. It is shown that the plant operating personnel are protected from radiation hazards through properly designed radiation shielding. The use of detailed operating and maintenance procedures provides a further means of significantly reducing any potential hazards.

The analysis of the hazards to the general public following an extreme accident are shown to be within generally accepted tolerances. In particular, the population of Sundance, Wyoming is sufficiently distant so as not to be jeopardized.

As a result of these evaluations, it is believed that the PM-1 Nuclear Power Plant can be operated at the Warren Peak site without undue risk to the general public or to the Sundance Air Force Station personnel.

I. PM-1 NUCLEAR POWER PLANT DESIGN SUMMARY

Reactor Design Characteristics

1. Overall Performance Data

Pressurizer water, nominal operating pressure, (psia)	1300
Average core coolant temperature, nominal, (°F)	463
Reactor thermal power, nominal, (mw)	9.35
Core life, nominal, (mw yr)	18.7

2. Core Design Characteristics

Geometry, right circular cylinder (approximately)	
Diameter, average, (in.)	23
Active length, (in.)	30
Overall length, (in.)	33-1/8
Core structural material	Stainless steel
Fuel element data, tubular, cermet type	
Outside diameter, (in.)	0.500
Inside diameter, (in.)	0.416
Clad thickness, (in.)	0.006
Clad material	AISI Type 348 Stainless Steel Co and Ta controlled
Pitch, triangular, (in.)	0.665
Number of fuel tubes	700 to 750
U-235 inventory, (kg)	26.7
U-235 burnup, (kg)	9.0
Control rods--number	6
--shape	Y

3. Core Heat Transfer Characteristics

Heat flux, (Btu/ft^2 hr)	
Average	70,000

Systems Design

1. General Plant

Steam generator power output, nominal, (mw)	9.35
Steam pressure, full power, minimum, (psia) (saturated)	300

2. Main Coolant System

Number of coolant loops	1
Coolant flow rate, (gpm)	1900
Main coolant pumps	1
Steam generators	
Number of units	1
Design pressure, approximately, (psi)	600
Type	Vertical with integral steam drum and separators
Temperature, primary inlet, full power, approximately, (°F)	481
Temperature, primary outlet, full power, approximately, (°F)	445
Temperature, steam side outlet, full power, (°F)	417

3. Pressurizing and Pressure Relief System

Number of pressurizers	1
Type	Steam
Temperature, normal, (°F)	577
Pressure, normal, (psia)	1300
Pressure element (decreasing)	Water spray head
Pressure element (increasing)	Electric immersion heaters

4. Coolant Purification and Sampling System

Number of purification loops	1
Purification device	Ion exchange resin
Maintenance provisions	Cartridge type, 1-yr life

5. Primary Shield Water System

Primary shield water cooler	Air blast type
Purification loop	Ion exchange resin
Maintenance provisions	Cartridge type, 1-yr life

Secondary Systems

1. General Plant

Steam flow at full power, approximately, (lb/hr)	35,000
Steam conditions at full power, 300 psia, dry and saturated, (°F)	417
Feedwater flow at full power, approximately, (lb/hr)	35,500
Rated gross electrical output, (kw at 0.8 pf)	1250
Net electrical output, (kw at 0.8 pf)	1000
Line voltage (4-wire wye)	4160/2400
Cycles	60
Phase	3
Auxiliary equipment voltage	480
Process heat (6609 lb/hr of 35 psia dry and saturated steam--Btu/hr)	7×10^6
Design elevation, (ft)	6500
Auxiliary power, approximately, (kw)	135

2. Turbine Generator Set

Type	Single extraction turbine
Throttle flow, full power, approximately, (lb/hr)	26,600
Throttle pressure, (psia)	290
Turbine steam exhaust conditions, full power	
Pressure, (in. Hg ab)	9
Turbine speed, approximately, (rpm)	7500
Generator rating, (kva)	1562.5
Generator rating, (kw at 0.8 pf)	1250
Generator type	Salient pole
Generator speed, (rpm)	1200

3. Condenser System

Number of units	2
Type	Direct air-to-steam

4. Feedwater System

Deaerator	
Type	Atomizing
Feedwater design flow, approximately, (lb/hr)	40,000

Design pressure, (psia)	50
Storage, (min)	5
Boiler feed pumps	
Number	2
Driver	One steam driven One electrically driven
Closed feedwater heaters	
Number	1
Type	Tube and shell, horizontal

5. Auxiliaries

Evaporator, reboiler	
Capacity, (lb/hr of 35 psia steam)	7500
Design pressure, (psia)	65
Feedwater storage tank	
Capacity, approximately, (gal)	2750
Turbine steam bypass system	
Type	Manual
Auxiliary generator unit	
Type	Hi-speed diesel
Number	1
Capacity, (kw)	200
Electrical characteristics	480 v, 60 cps, 3ϕ
Emergency power	
D-c power source	Batteries
A-c power source	3-unit MG set

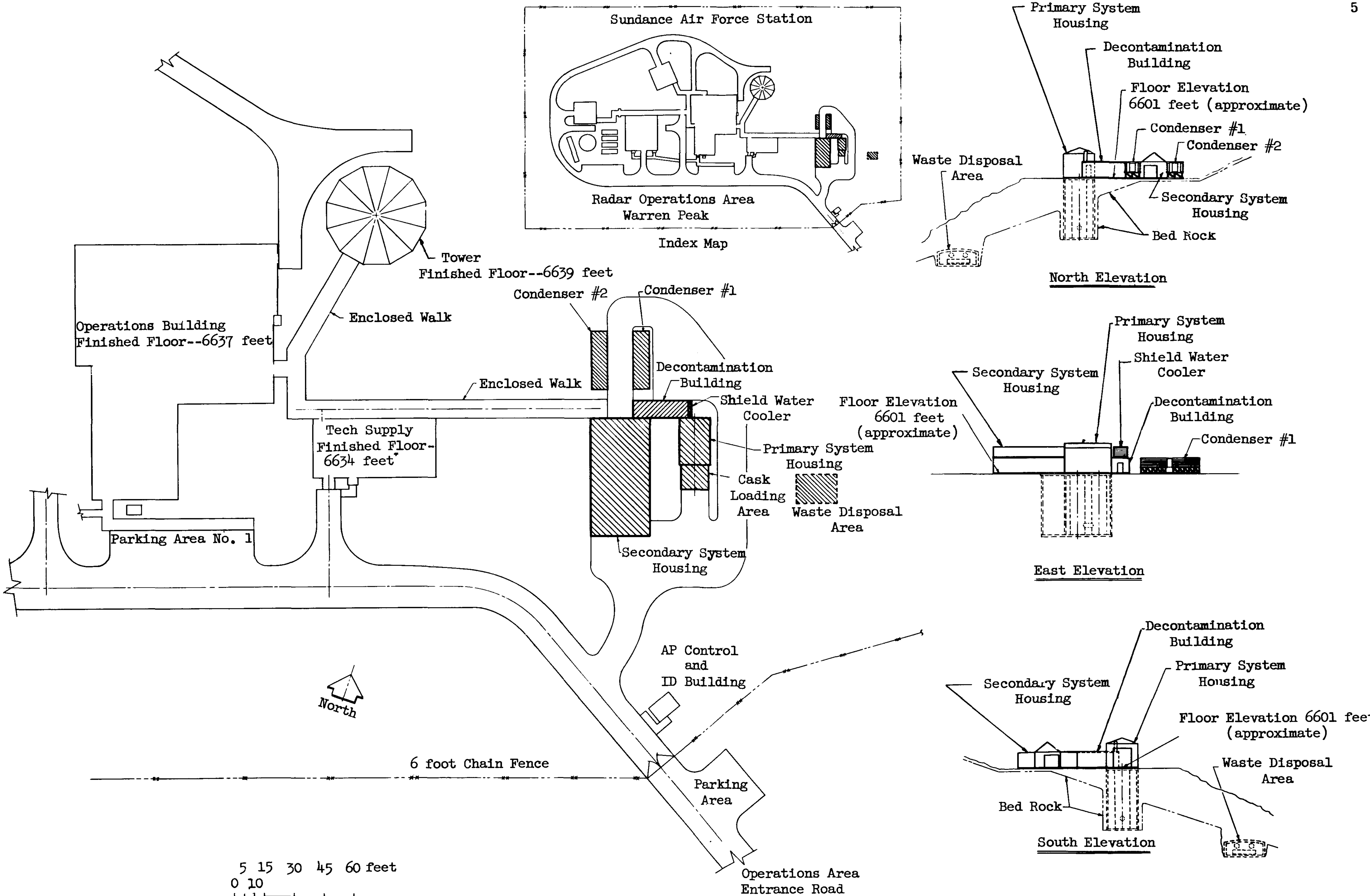


Fig. I-1. Site Layout and Elevations--PM-1 Power Plant

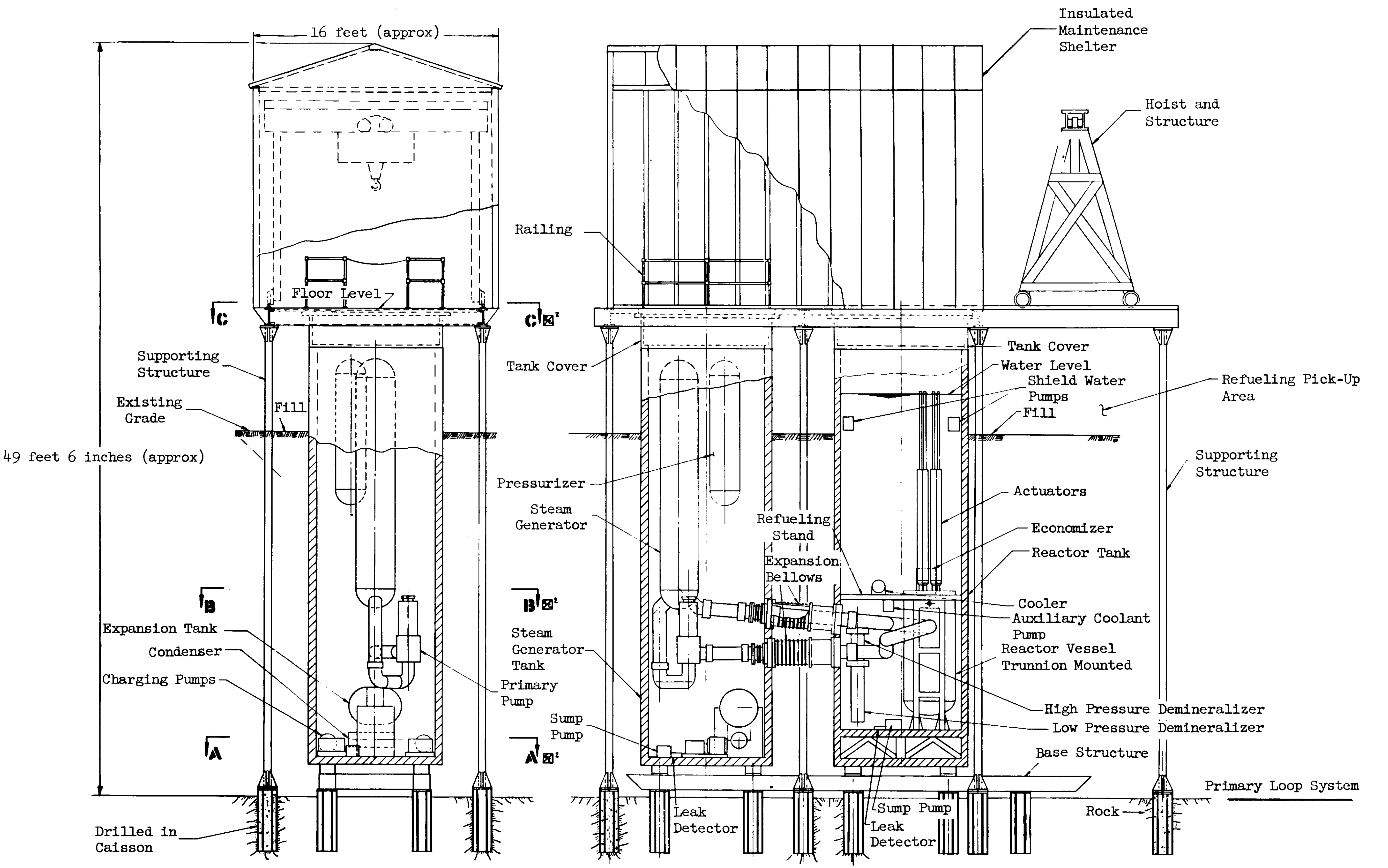


Fig. I-2. Plant Elevations--PM-1 Primary System and Housing

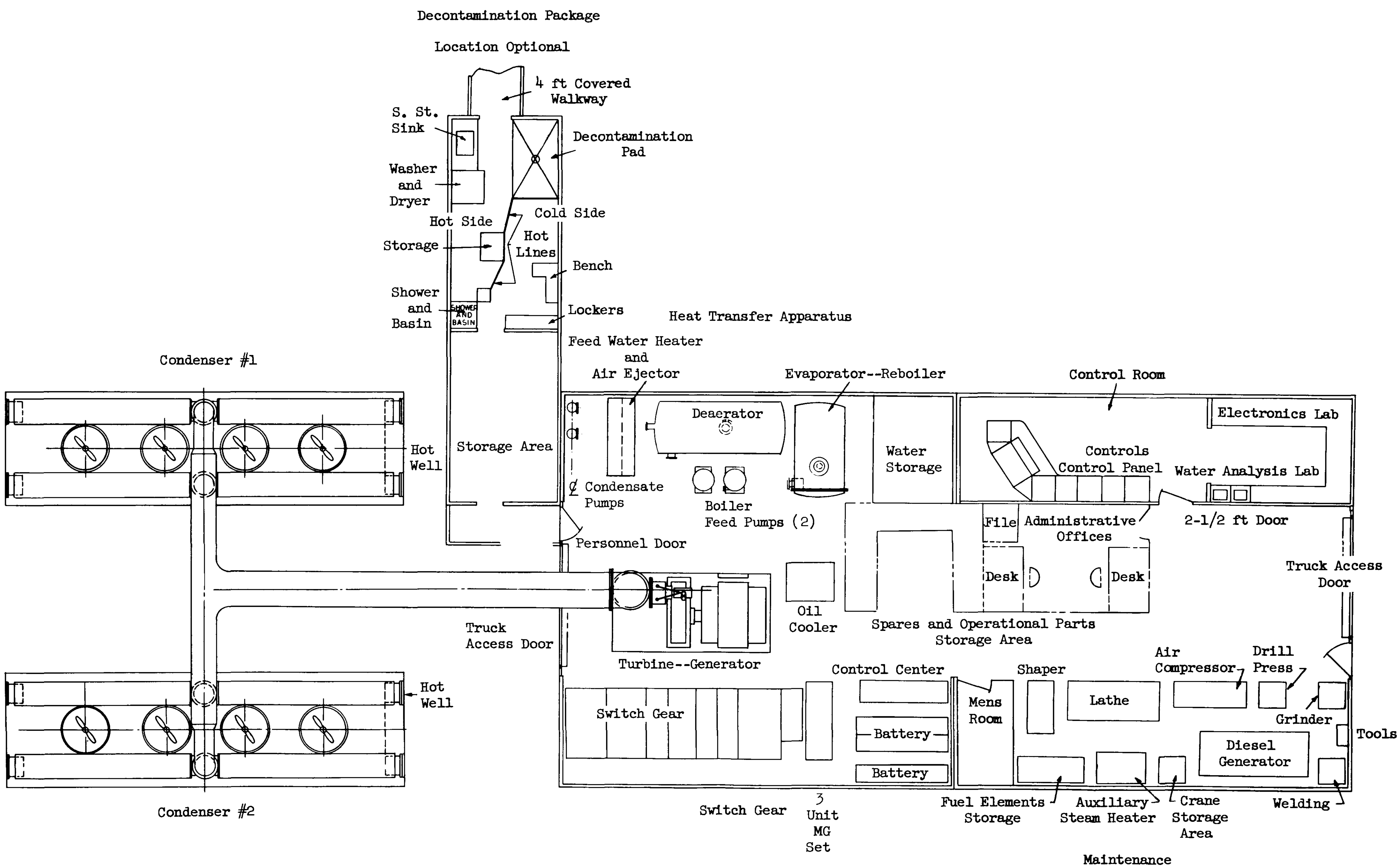


Fig. I-3. Plant Layout, Secondary System--PM-1 Power Plant

II. SITE ENVIRONMENT

The proposed PM-1 Nuclear Power Plant site is in a remote region of the Black Hills in Eastern Wyoming. The PM-1 will supply power and space heating for the Sundance Air Force Station radar installation to be built atop Warren Peak. Warren Peak is the highest mountain in the immediate area with its summit at an elevation of approximately 6600 feet. The site is characterized by its isolation, high elevation, high wind flow rate, upland hydrology, solid aseismic foundation and restrictive land usage. Aside from a limited number of military personnel, the nearest offsite population concentration is at Sundance, Wyoming, six miles southeast of the site. The population of Sundance is approximately 1000. The site, therefore, has many favorable features for a nuclear power plant installation.

A. SITE LOCATION

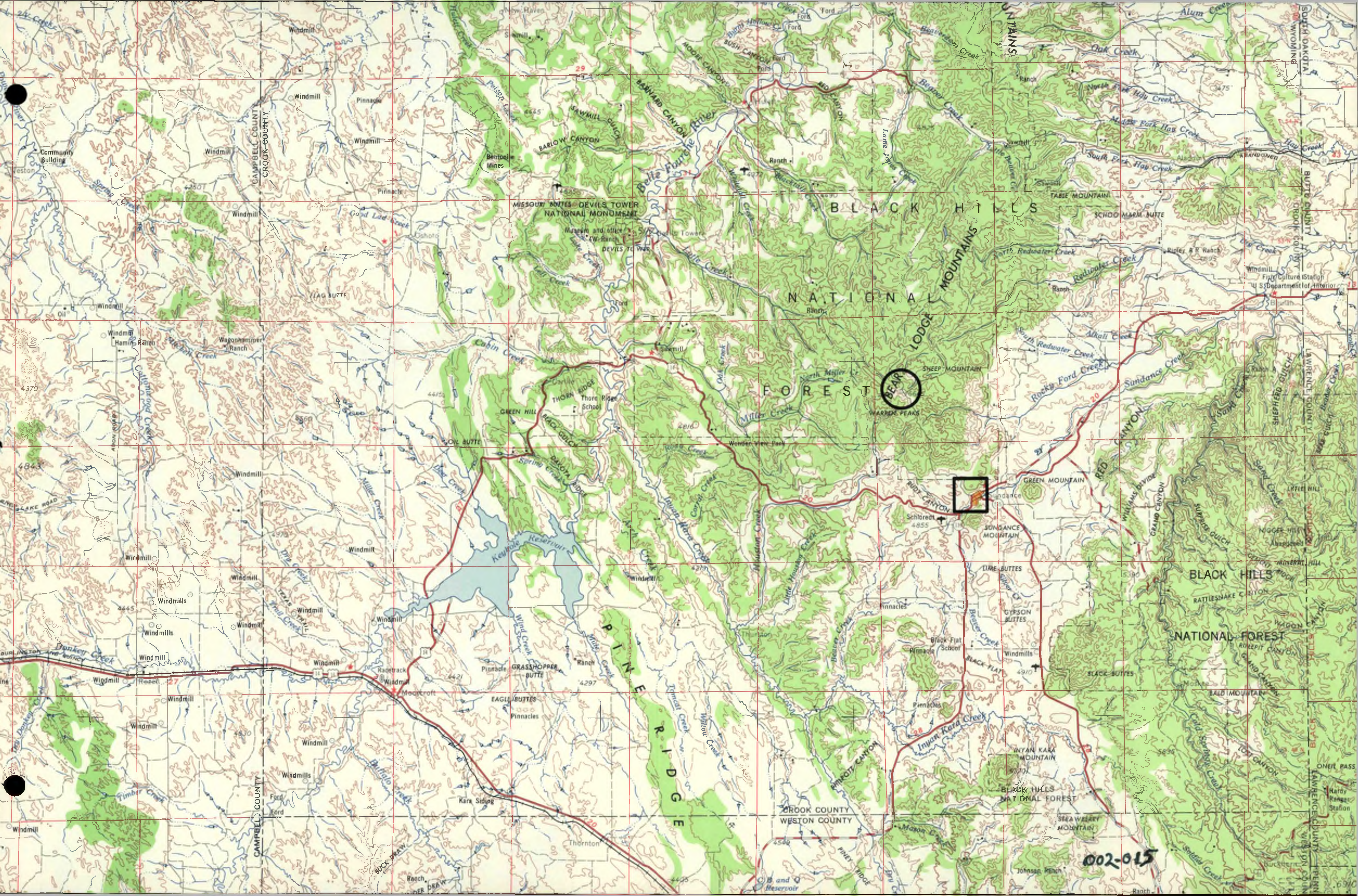
1. Region

The PM-1 site is in the Black Hills area of Wyoming, 16 miles west of the South Dakota state line. The Black Hills are approximately 100 miles long and 50 miles wide and rise to an elevation of several thousand feet above the plain. The northwest sector of the Black Hills is known as the Bear Lodge Range and is separated from the main uplift by the Red Canyon. Figure II-1 shows the regional location of the PM-1 site.

2. Area

The Sundance Air Force Station will consist of a radar installation on Warren Peak, six miles northwest of the town of Sundance, Wyoming within the Black Hills National Forest. The PM-1 will be constructed on the existing roadbed east of the peak and will be approximately 85 feet from the nearest structure in the radar complex. The exact location is 200 feet east-southeast of an existing fire lookout tower. This location will allow the PM-1 to be operationally integrated with the radar complex, construction of which is scheduled for the near future.

Fig. II-1. Regional Location of PM-1 Site
(opposite page)



BLACK HILLS
NATIONAL
FOREST
BEAR LODGE MOUNTAINS
DEVILS TOWER NATIONAL MONUMENT
MISSOURI BUTTES
Belle Fourche River
Keyhole Reservoir
PINE RIDGE
BLACK HILLS NATIONAL FOREST
BLACK HILLS NATIONAL FOREST
CROOK COUNTY
WESTON COUNTY
CAMPBELL COUNTY
CROOK COUNTY
SOUTH DAKOTA
WYOMING
BUTTE COUNTY
LAWRENCE COUNTY
Moorcroft
Sundance
SUNDANCE MOUNTAIN
GREEN MOUNTAIN
WARREN PEAKS
SHEEP MOUNTAIN
INYAN KARA MOUNTAIN
Inyan Kara Creek
Miller Creek
Redwater Creek
Sundance Creek
Donkey Creek
Beaver Creek
Hay Creek
Oak Creek
Alum Creek
BARLOW CANYON
BARNARD CANYON
RED CANYON
CABIN CREEK
TABLE MOUNTAIN
SCHOOLMARM BUTTE
LIME BUTTES
GYPSUM BUTTES
BLACK FLAT
BLACK BUTTES
BALD MOUNTAIN
EAGLE BUTTES
GRASSHOPPER BUTTE
OIL BUTTE
THORN RIDGE
GREEN HILL
Wonder View Peak
Kara Siding
Thornton
B and G Reservoir
Mason Draw
Windmill
Ranch
Pinnacles

002-015

B. GEOGRAPHY

1. Topography

The Bear Lodge Mountains are the most extensive of the igneous intrusions which make up the Black Hills. The main intrusion is a laccolithic dome, somewhat elongated from northwest to southeast and marked by prominent elevations and numerous canyons which create fairly rugged topography. These mountains rise abruptly as steep rocky slopes. The summit elevation at Warren Peak is 6656 feet, which is approximately 2000 feet above the surrounding plains. The surrounding foothills are characterized by an irregular, rolling surface periodically incised by deep canyons and branch draws; beyond lie the rolling hills of the plains interspersed with mesas and buttes.

2. Land Usage

Land usage in the Bear Lodge Mountains area is limited to sporadic lumbering, summer cattle grazing, intermittent mineral exploration, agricultural activity on the fringes of the mountains and summer recreational activity. Land utilization is restricted by the topography, climate and water resources of the area.

Lumber activity.- Much of the Bear Lodge Mountains area is heavily forested in Ponderosa Pine, Aspen and Scrub Oak. However, the area within approximately a one-half mile radius of Warren Peak is treeless. Some lumbering is permitted under government control in the Black Hills National Forest, but there is no evidence of any sizable lumbering operation in the Bear Lodge Mountains area.

Cattle activity.- Grazing permits are issued for the Bear Lodge Mountains area, depending upon range conditions. The grazing is limited to four summer months from the middle of May to the middle of September. During the 1959 grazing season, the allotment for the Warren Peak region (the area within an 8-mile radius of the Peak) was approximately 1000 head of cattle. It is estimated that 300 cattle grazed within a 2-mile radius of the Peak. Some grazing activity is expected on a "year-round" basis on the Sundance Plain.

Agricultural activity.- Arable lands are scattered throughout the Sundance area; however, not all are favorable for farming, especially in high-altitude areas. The limited rainfall in the valleys and plains is another factor restricting extensive agriculture. There is a limited amount of farming where water is available and topographic conditions are favorable. Wheat and hay are the chief crops.

Mineral activity.- The mineral resources of the Bear Lodge Mountains in the vicinity of Warren Peak do not appear to be of economic value. The area has a history of considerable prospecting and mineral exploration dating back to 1875, but the commercial activity has either been abortive or on a very small scale. Exploration in the area still continues without the discovery of significant commercial deposits.

There is both copper and gold mineralization in the area, in the form of greatly disseminated faint fracture and joint deposition. There is an abandoned mine, one mile north of Warren Peak, which has been explored for copper occurring there in thin veinlets and seams in the parent trachyte. A second copper outcrop has been exposed by prospecting approximately one-half mile north of the Peak. There are three gold prospects southeast of the Peak in the igneous rock and one old mine in Rudy Canyon in the Deadwood formation. There have been no large-scale commercial operations in the area.

Even though sporadic mineral prospecting in the Warren Peak area will continue, there is little probability of significant commercial operations.

Recreational activity.- Recreation in the Warren Peak area is limited to tourist activity in the summer and hunting in the fall. Since recreational facilities are limited, the influx of people is not extensive. The transient summer population within a 2-mile radius of Warren Peak is estimated to be about five individuals at any given time.

3. Access

Sundance.- The town of Sundance is served from the east and west by U.S. Highway 14, which is a 2-lane, hard-surfaced, medium-duty, all-weather road. State Highway 116 serves Sundance from Upton, 28 miles to the south.

Site.- The present road from Sundance to Warren Peak is a light-duty, dry-weather dirt road, with a lightly graveled surface. This road is eight miles long. In wet weather, the portion at the higher elevations is somewhat better than the lower portion as a result of faster runoff. Present planning calls for reworking this road during construction of the Air Force Station.

This road continues northward from Warren Peak for 15 miles where it joins with Wyoming State Highway 24. Most of the other roads in the area are fire trails. These roads are best suited for 4-wheel-drive vehicles, although other vehicles can pass in dry weather.

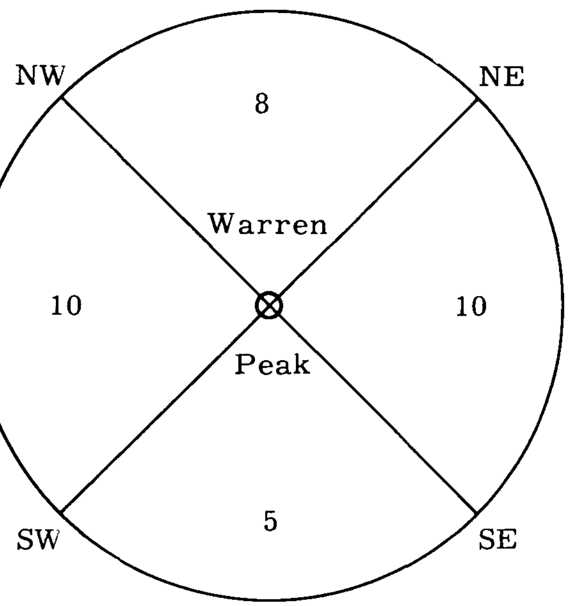
C. POPULATION DENSITY AND DISTRIBUTION

1. Military

Military personnel will be concentrated in two areas which are significant to a hazards evaluation. The radar station at Warren Peak will be manned by approximately 25 people. The Cantonment Area, located four miles south-southeast of Warren Peak, will house about 195 people, consisting of military personnel and their families.

2. Civilian

The town of Sundance, six miles southeast of Warren Peak, is the largest population center within a 15-mile radius. Its population was 893 according to the 1950 census, with recent local estimates being close to 1000. The rural population is limited to scattered farm homes and ranches, with about 33 people living within a 5-mile radius of Warren Peak, as indicated in the following diagram.



The summer months attract transient people, such as campers, fishermen, prospectors, cattlemen, etc., who utilize the resources of the national forest. It is difficult to estimate the number of transients in the area for any given period, but it is believed to be extremely small because of the poor roads and limited facilities. Five or six picnic tables are provided at Reuter Springs, two and one-half miles from Warren Peak toward Sundance.

D. CLIMATE

1. Available Climatological Data

The meteorological data available for this region are limited. The major source closest to Warren Peak is a climatological substation at Sundance, which has recorded temperature, precipitation and wind direction for periods of more than 40 years. However, the data from this station are not considered entirely representative of Warren Peak.

The following general climatic description for the town of Sundance is somewhat different from that expected for Warren Peak since Sundance is 2000 feet lower in elevation and somewhat sheltered from the prevailing winds.

Considering the normal lapse rate of temperature with altitude, a 6° F difference in temperature between Sundance and Warren Peak would be expected. The observed temperature difference is about 10° F which is consistent with the dry adiabatic lapse rate which occurs in the free air in this region. This difference may be indicative of a lack of connecting air currents between the two points. Connecting air currents would decrease the temperature difference to the usual value of 6° F.

2. General Climatology

The climate of Sundance is semiarid, with long, cold winters and short, hot summers. The hottest days of the summer are marked by very low humidity. There are few summer nights when the temperature remains above 60° F. During July, the warmest month, temperatures of 90° F or above occur frequently, but the average minimum is in the middle fifties.

The winters are long and cold with January being the coldest month. During all of the winter months, more than 55% of the possible sunshine is received. The clear skies and dry air provide a moderating influence during sunlight hours by permitting greater absorption of solar radiation. The extreme winter weather results from outbreaks of cold Canadian air moving southeastward from the Rocky Mountains, across the plains to the Black Hills. The initial onslaught of Arctic air is usually accompanied by strong northerly or northwesterly winds and drifting snow. The coldest nights are associated with clear skies and very light winds, which are ideal conditions for a strong temperature inversion. Advection inversions could occur frequently during the winter months when the Arctic air masses slide up the slopes of the Bear Lodge Mountains, forcing the warmer air to a higher level and creating a temporary but very stable condition.

Freezing temperatures have occurred as late in the cold season as June 25 and as early as August 28. However, the average date of the last freezing temperature in the spring is May 26 and of the first freezing temperature in the fall is September 18. These dates are expected to be considerably altered with the increased elevation and exposed position of Warren Peak. Here freezing temperatures could conceivably occur throughout the entire year.

More than half of the annual precipitation in Sundance falls during the period from April through July. The three winter months constitute the period with least moisture. Amounts of snowfall are quite large but the water content of the snow is usually low. During the months of March and April, however, precipitation often begins as rain mixed with snow and turns to heavy, wet snow. These snowstorms are frequently accompanied by strong winds and drifting. On the average, March has more snow than any other month.

3. Significant Meteorological Conditions

Wind flow.- A 30-year period of record for the town of Sundance shows the prevailing wind to be from the southwest. During 80% of the time, the wind blows from a quadrant from southwest through northwest. Figure II-2 shows a wind rose for the town of Sundance. The chart shows wind direction only since velocity data are not available.

Some limited data from Warren Peak show the same general prevailing direction but with much stronger influence from the northeast and southeast at the expense of the westerly component. These data represent summer observations taken by the U.S. Forest Service fire lookout at Warren Peak. The wind rose depicting these data is shown in Fig. II-3.

In any analysis of these data, it is important to take into account the relative positions of Warren Peak and Sundance. While Warren Peak is the highest point in the immediate area and very much exposed, Sundance is located in a valley between two mountains and somewhat sheltered. This valley runs in a southwest-to-northeast direction and it is possible that a funneling of winds through the valley is experienced, although this is not obvious from the data. However, simultaneous observations from the Peak and Sundance frequently show winds from different directions.

The only wind velocity data for this area were also taken by the fire lookout on Warren Peak. Observations were made five times daily during the summer fire seasons of 1955 and 1956. The mean wind speed for the observations taken from July 1 through September 19, 1955 was 13.8 miles per hour. The following year, observations taken from June 3

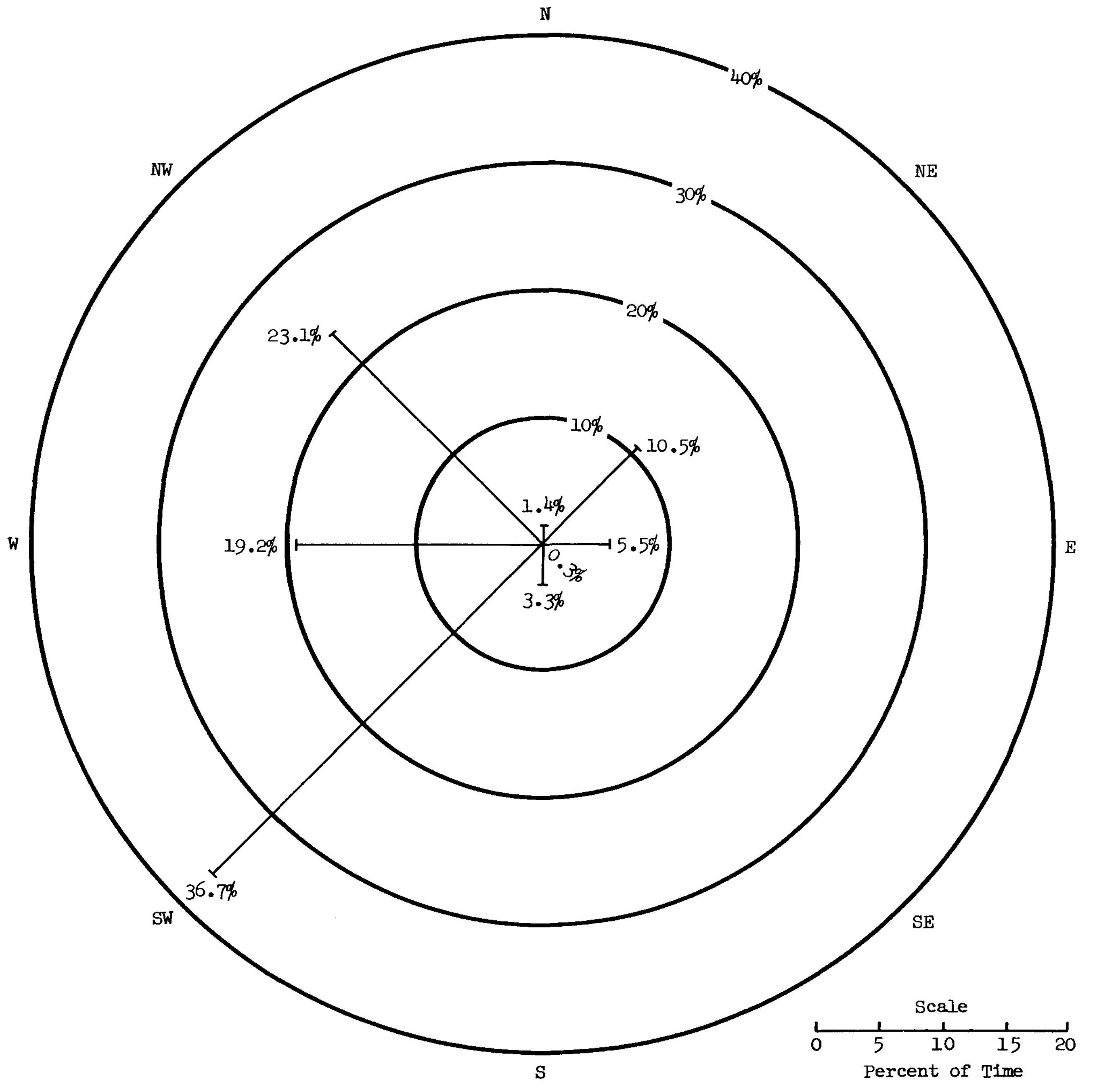
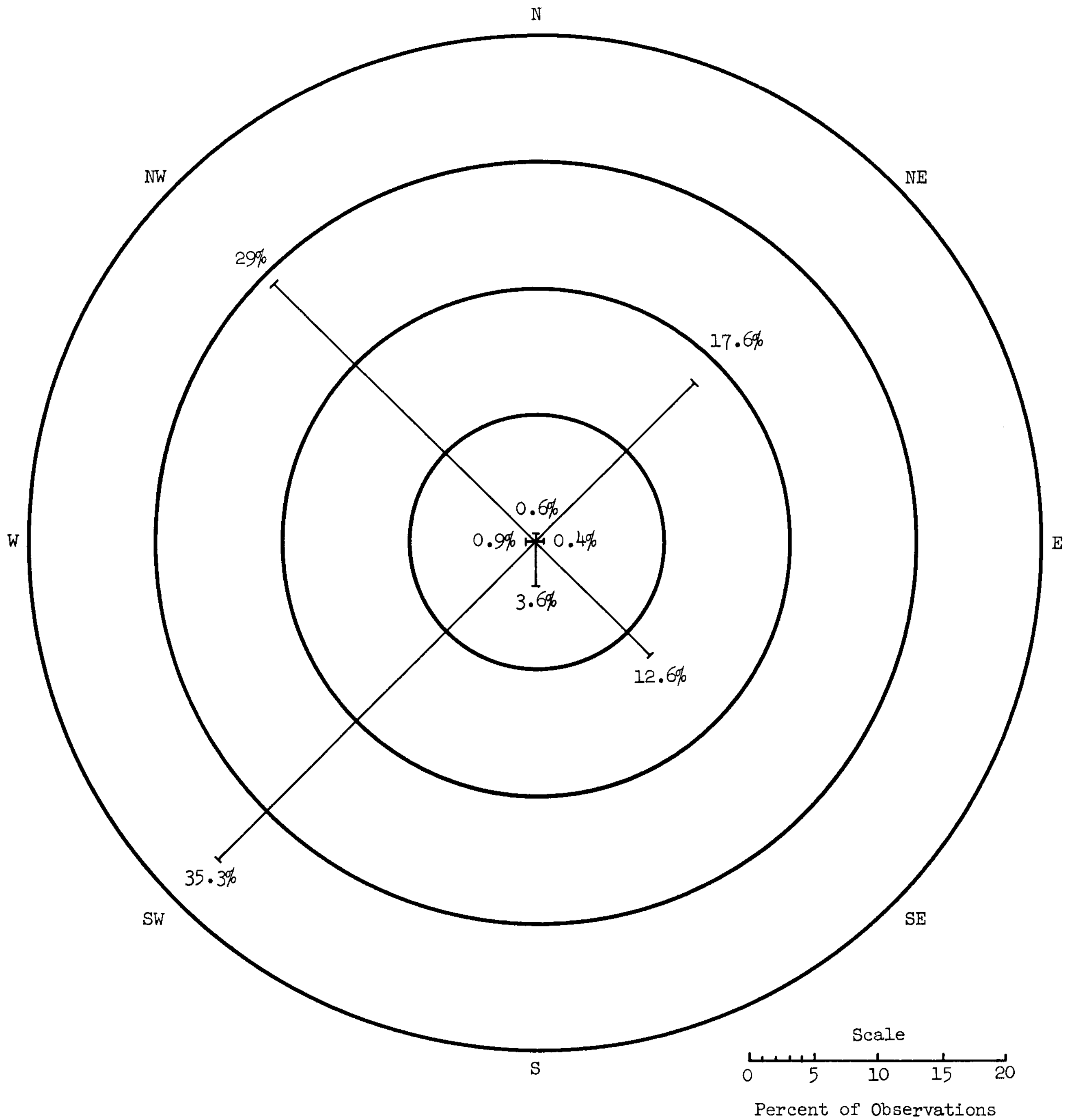


Fig. II-2. Percent of Time with Monthly Mean Wind Direction from Direction Given Sundance, Wyoming (30-year record)



Wind Direction

1014 Summertime Observations Taken over a Two-Year Period.
Observations Taken at 8 AM, 10 AM, 1 PM, 3 PM and 5 PM.
Data Recorded by U.S. Forest Service Fire Lookout at Warren Peak.

Fig. II-3. Percent of Total Observations with Wind from a Given Direction at Warren Peak through October 23 showed a mean of 15 miles per hour. A high of 50 miles per hour and a low of 5 miles per hour were recorded for a span of 1000 observations taken during these two seasons. Although no definite conclusions can be drawn from these data, they do tend to indicate that wind speeds will generally be rather high, with only occasional calm conditions expected. These will occur during extremely cold winter nights with clear skies.

Estimates have been made of the mean wind speeds for both winter and summer periods. These estimates were made by local observers, such as forestry personnel, who utilize a knowledge of wind conditions in their occupations. Their estimates indicate a mean wind speed of 15 miles per hour for the summer period and 18 miles per hour for the winter.

Precipitation.- The means and extremes of precipitation for the town of Sundance are as follows:

Mean annual precipitation (48 years)	18.5 inches
Maximum year	27.81 inches (1922)
Minimum year	11.58 inches (1954)
Mean annual number of days with 0.01 inch or more	86 days
Mean annual snowfall	79 inches
Maximum snowfall in one winter	168.7 inches (1916 to 1917)

The precipitation and snowfall data listed for Sundance are not considered representative of Warren Peak. Empirical estimates for Warren Peak are:

Mean annual precipitation	23 inches
Mean annual snowfall	110 inches

Temperature.- The means and extremes of temperature for 45 years of record at the town of Sundance are as follows:

Minimum of record	-42° F (February 1936)
Mean (January)	19° F
Mean annual	43° F
Mean (July)	68° F
Maximum of record	105° F (July 1936)

Although these data are not considered directly applicable to Warren Peak, some comparison between the temperatures of the two locations is essential. A comparison of the mean maximum temperatures for two months of the year, shown on the chart below, represents the only data available for such a comparison. Although the data are not directly comparable, this chart provides some indication of the temperature differences experienced between Sundance and Warren Peak.

Sundance		Warren Peak	
(45-year mean)	1955	1956	2-year mean
July 82.6° F	July 72.6° F	68.3° F	70.4° F
August 81.6° F	August 72.7° F	68.5° F	70.6° F

Even though not conclusive, these data do tend to support the anticipated dry adiabatic lapse rate of approximately 5.5 degrees per 1000 feet for this area.

Since the lookout tower is manned only during the summer fire season, no winter temperatures have been recorded for Warren Peak. The record low temperature for Sundance is -42° F, which occurred in February 1936. Somewhat colder extreme temperatures can be anticipated at Warren Peak.

4. Severe Weather Conditions

This area is subject to blizzards, hailstorms, rain storms and high winds. Generally, severe storms do not occur until early January, with only a few blizzards each winter. Cold waves are not extremely frequent in this area but are often accompanied by sudden temperature falls of 20 to 40 degrees, with equally sudden, and sometimes greater, rises in temperature after passage of the front.

E. GEOLOGY

The PM-1 site in the Bear Lodge Mountains is underlain by the igneous core of a laccolith. The intrusion, fed by a columnar pipe of igneous rock penetrating the basement structure, is lenticular in geometry and was at one time overlaid by metamorphic and sedimentary rocks. The crest of the upwarp was subsequently eroded, exposing the igneous core in the Warren Peak area. The sedimentary and metamorphic rocks surround the igneous mass on the flanks of the uplift at lower elevations.

1. Petrography and Stratigraphy

The PM-1 site is on monzonite-syenite porphyry of the Tertiary Age. The igneous core is transected by dikes of granite and pegmatite which follow the general fracture pattern of the uplift. The porphyry contains phenocrysts of feldspar and ferromagnesian minerals and a ground mass of similar fine-ground material. The porphyry is of three general physical types:

(1) Unweathered porphyry beyond the zone of chemical leaching. This has a higher content of ferromagnesian phenocrysts than the two other types.

(2) Porphyry which is subject to insitu weathering, in a zone above the fresh rock. This has lost many of the ferromagnesian constituents due to preferential leaching.

(3) Weathered mantle, making up the overburden which consists of remnants of porphyry, kaolin, muscovite and limonite. The overburden on Warren Peak is from one to five feet thick.

The sedimentary and metamorphic rocks on the flanks of the uplift consist in ascending order of Cambrian schists, Ordovician limestone, Carboniferous limestone and sandstone, Triassic mudstone and Tertiary gravels. The Spearfish formation of Triassic Age makes up the floor of the nearby lowlands and consists of red mudstone and gypsum. These rocks are of little importance except for their hydrologic characteristics.

2. Structure

The Bear Lodge Mountains are part of the Black Hills uplift and were formed in Tertiary time through igneous orogenic activity. To the east of the PM-1 site there is a shallow syncline occupied by the Spearfish redbeds on the surface. South of the site the Sundance area is occupied by horizontal beds of Triassic Spearfish and the Sundance Formation of Jurassic Age.

The igneous rocks of the site area exhibit characteristic discontinuous fracturing and jointing. No visible faults are present at the site. Faulting is in evidence at Sheep Mountain three miles to the northeast of the site. There are no active faults in the region immediately surrounding the site.

F. HYDROLOGY

1. Subsurface Hydrology

The entire central portion of the Bear Lodge Mountain laccolith is probably not underlaid by groundwater. However, the subsurface hydrologic regime is actually controlled by discontinuous fracture and joint zones which are recharged by precipitation. Since these zones are essentially vertical in attitude and discontinuous in nature, they are somewhat difficult to trace except for springs issuing on the terminal outcrop of the fracture zone. Since fracturing of the porphyry is somewhat limited and the parent rock is impervious, there is a consequent limitation in the subsurface water resources of the area.

Several springs are present in the site area. The planned water source for the radar station will be Hutchins Spring, which is one-half mile northwest of Warren Peak on the Lytle Creek watershed. Its flow rate is about 6 gallons per minute. Table II-1 shows a water analysis from Hutchins Spring.

TABLE II-1

Chemical Analysis of Water
Hutchins Spring, Warren Peak
Sundance, Wyoming

Date sampled: July 26, 1958

Analysis	Parts per Million
Suspended Solids	0.0
Dissolved Solids (Residue at 103° C)	51.0
Alkalinity to Phenolphthalein as CaCO3	0.0
Alkalinity to Methyl Orange as CaCO3 (Total Alkalinity)	4.3
Total Hardness as CaCO3	10.3
Calcium (Ca)	1.4
Magnesium (Mg)	1.7
Alkalies as Sodium (Na)	3.3
Iron (Fe)	0.1
Aluminum (Al)	0.0
Manganese (Mn)	0.0
Sulfate (SO4)	5.0
Chloride (Cl)	1.4
Nitrate (NO3)	0.3
Bicarbonate (HCO3)	5.5
Carbonate (CO3)	0.0
Silica (SlO2)	15.4

ph 6.0

Sodium, parts per million	2.0
Potassium, parts per million	1.3

Other springs in the area include Reuter Spring which is 2-1/2 miles south-southeast of the site and Davis and Cole Springs, located 1-1/2 miles northeast of the site on Beaver Creek. These springs each produce about 10 gallons per minute. There are numerous springs at greater distances from Warren Peak. A few of these have been improved and cattle watering tanks installed. Several of the springs are shown in Fig. II-4.

Subsurface transport of potential contaminants released at the site would probably be at an infinitesimal rate. The impermeability of the parent rock, thinness of fracture zones, discontinuity of these zones and the effect of filtration would certainly restrict migration. The nearest point of recharge of regional aquifers such as the Minnelusa sandstone and Pohasapa limestone is one mile, providing a significant distance for migration. Should the charging area of nearby aquifers become contaminated, there is great distance between the point of recharge and the point of potable water withdrawal further down the slopes.

2. Surface Hydrology

The amount of flowing water in the Warren Peak area is very limited and most of the draws, valleys and canyons are dry except for runoff from rain or melting snow. None of the streams have a continuous flow from head to mouth; many appear at intervals in springs or bottom seeps which supply waters that flow for various distances, then are either evaporated or pass underground. Often the water in a depression will consist of a series of small pools, generally created by beaver dams.

The streams in the area, which are reported to have some "year-round" flow, are: Beaver, Ogden, Rocky Ford, Houston, Lost Houston, Miller, Lytle and Whitelaw Creeks. With Warren Peak the highest point in the area, these streams radiate out from this apex as shown in Fig. II-4.

G. SEISMOLOGY

The seismic record in Table II-2 was derived from "Earthquake History of the United States, Part I", published by the U.S. Coast and Geodetic Survey. This list includes all recorded earthquakes through the year 1956, and goes back to 1699 in South Dakota and 1852 in Wyoming. This record indicates that no past earthquake epicenter is present within a 50-mile radius of the PM-1 site, and that no major earthquake has occurred within a 300-mile radius. However, several earthquakes of moderate intensity and extent are listed within a 300-mile radius, and some minor activity has occurred from time to time.

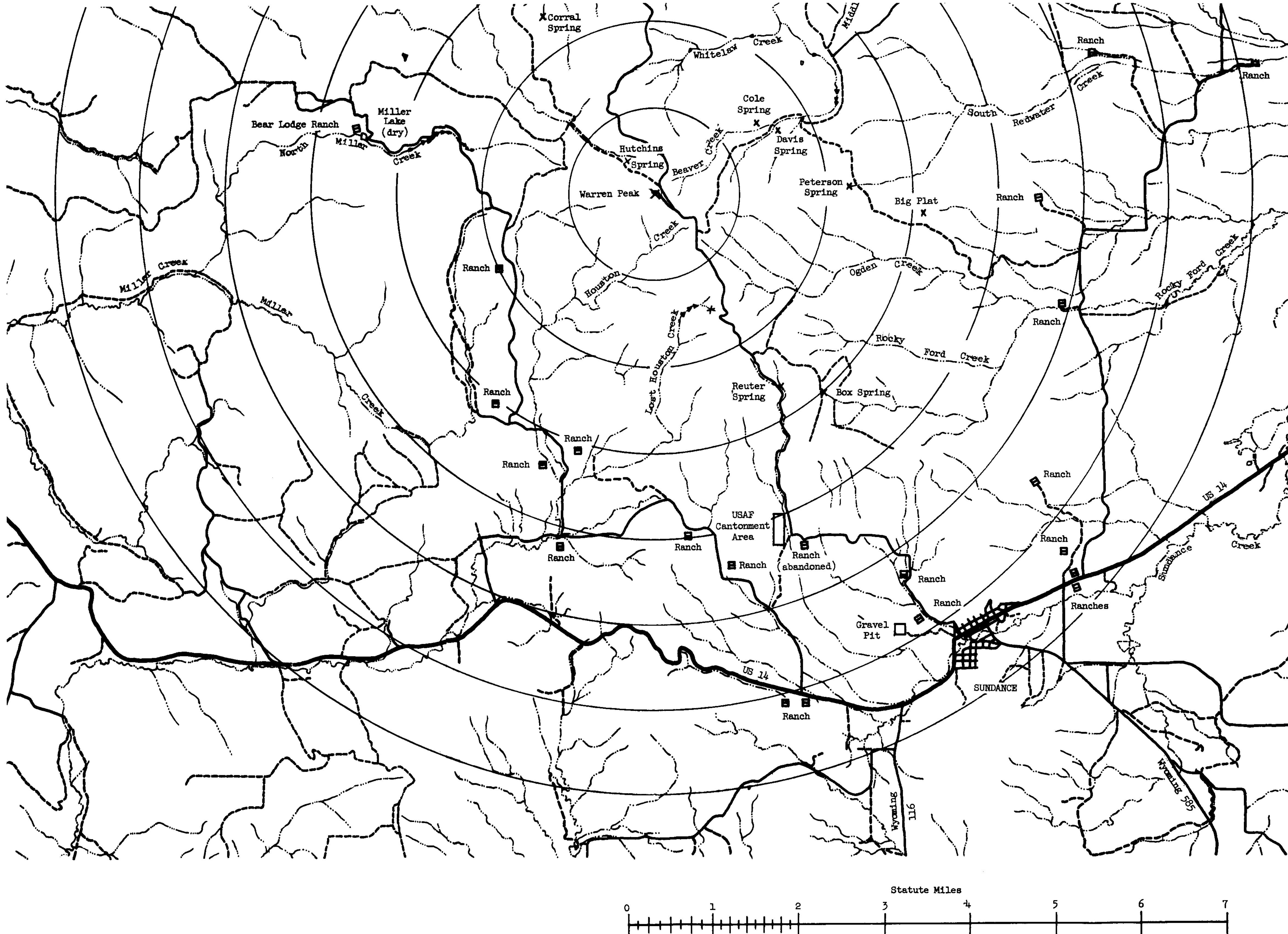


Fig. II-4. PM-1 Plant Site Drainage Map

Figure II-5 shows the location of all the seismic activity in the region with respect to Warren Peak. The details about these earthquakes are listed in Table II-2 in order of their distance from the PM-1 site.

The 1959 earthquake at West Yellowstone, Montanna was probably one of the most severe this area has experienced in recorded history. According to U.S. Coast and Geodetic Survey, it had an intensity of 10 on the modified Mercalli scale. The total area affected was estimated to be about 550,000 square miles, which extends to a 420-mile radius. Warren Peak is 325 miles from this epicenter. However, earthquakes equal to or less than intensity 10 in the Yellowstone seismic zone are not considered potentially detrimental to the PM-1 site. The rigid granitic base upon which the site is located makes it particularly favorable to resistance of shocks of even greater magnitude and somewhat closer range. The Seismic Probability Map of the United States, contained in the Uniform Building Code of the Pacific Coast Building Officials Conference, designates the eastern third of Wyoming as a "zero seismic zone." The seismic design criteria for the PM-1 Nuclear Power Plant has been established as zone three. The zone three criteria was specified because of possible future relocation of the power plant.

H. DESCRIPTION OF RADAR INSTALLATION

The existing fire lookout tower will be moved to a nearby peak and Warren Peak lowered 20 feet to accommodate the radar installation. The debris will be used to fill about 15 feet around the north, west and south sides to create an area 300 by 480 feet on which the radar station will be built. The station will consist of three radar towers, a buried water tank, heating and emergency power building, technical supply building and an operations building with interconnecting, enclosed walkways.

I. ORIENTATION OF REACTOR SYSTEM TO RADAR BASE FACILITIES

The PM-1 plant will be located on the east slope of the peak with the finished floor of the primary system approximately 35 feet lower than the top of the peak. The secondary system will be directly upslope from the primary system on the existing roadbed to the peak. It will be located approximately 85 feet from the technical supply building, with an enclosed walkway to connect them. The finished floor of the secondary system will be approximately 30 feet lower than the altered level of the peak.

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Fig. II-5. Epicenter Location of Earthquakes with Respect to PM-1 Site

TABLE II-2

Recorded Earthquake Data
(listed in order of distance from Warren Peak)

No.	Year	Epicenter Coordinates North Latitude	Epicenter Coordinates West Longitude	Distance from PM-1 Site (Miles)	Modified Mercalli Intensity	Area Affected (Square Miles)
1	1895	43.9	103.3	60	4 to 5	--
2	1928	44.0	103.7	80	5	2,000
3	1925	44.6	107.0	110	5	3,000
4	1894	42.8	106.5	140	5	--
5	1897	42.8	106.5	140	6 to 7	--
6	1934	42.7	103.0	155	6	23,000
7	1915	43.8	101.5	160	5	Local
8	1928	43.5	108.2	185	5	3,000
9	1906	43.0	101.3	195	6	8,000
10	1954	41.5	105.5	210	5	2,000
11	1934	43.0	109.0	225	5	8,000
12	1959	44.7	111.5	325	10	550,000

NOTE: Other earthquakes have occurred near the 300-mile range, but at that distance they are considered insignificant to the Warren Peak Area.

1. Radiological Restricted Areas

The fenced area will be extended to 280 ± 30 feet beyond the reactor with no access permitted for unauthorized persons. In addition to this, a 3000-foot radius circle around the reactor radar station will be established as a restricted easement area where no permanent structures will be permitted.

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III. PM-1 NUCLEAR POWER PLANT DESCRIPTION

A. DESCRIPTION OF THE PRIMARY SYSTEM

The primary system is composed of a number of interrelated circuits and components. The arrangement of the equipment is shown in Fig. I-2, and a basic flow diagram in Fig. III-1. The modes of operation and failure of the valves are shown on the flow diagram. The modes of operation and failure of remote and self-actuated valves were determined by providing maximum safety and protection against inadvertent shutdown.

The individual circuits and components are discussed below.

1. Main Coolant Loop

The main coolant loop consists of the reactor, steam generator, primary pump and primary piping. The reactor and steam generator are described later.

The primary coolant pump is of the centrifugal type, capable of delivering a flow of 1900 gpm. The pump, designed for a pressure of 1500 psia and 600° F, will normally operate at 1300 psia and 445° F.

The primary piping is seamless, 6-inch, schedule No. 80, type 316 stainless steel. Flange joints are provided on the primary piping for quick installation or dismantling of the two packages containing the main coolant loop and its auxiliary subsystems.

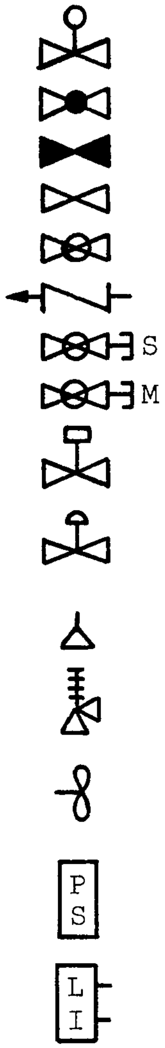
The main coolant loop contains 56 cubic feet of light water at a mean temperature of 463° F with an average operating pressure of 1300 psia.

With only the one primary coolant loop and a single coolant pump, no possibility exists for the injection of a significant quantity of cold water as a result of changes in operating conditions. Thus, the cold-water type accident cannot occur in the PM-1.

The main coolant loop drainage is carried from the bottom of the reactor vessel and other low points in the system to the waste sump tank or the shield water tank, depending on the activity of the water at the time. The main coolant loop is filled by means of a pump in the secondary system. The water enters at the bottom of the reactor. Vents at all high points in the system can be opened to release trapped gas.

The arrangement of the main coolant loop is such that natural convection will begin lowering the coolant temperature of the scrammed reactor within 10 seconds after loss of power to the primary coolant pump. Coast-down flow will protect the reactor during the 10 seconds.

LEGEND



Solenoid or Power Operated Valve; Fail Open and Normally Open

Fail Closed

Normally Closed

Manual Operated Valve, Gate

Manual Operated Valve, Globe or Needle

Check Valve

Sampling Point

Monitoring Point

Pressure Regulating Valve

Flow Regulating Valve

Spray Nozzle

Relief Valve

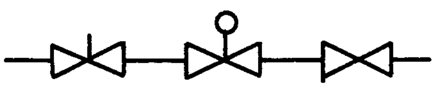
Blower

Pressure Switch

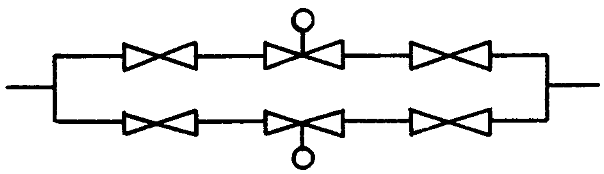
Level Indicator

RO	Remote Operated
LO	Level Operated
PO	Pressure Operated
SO	Scram Operated

Series Connection



Parallel Connection



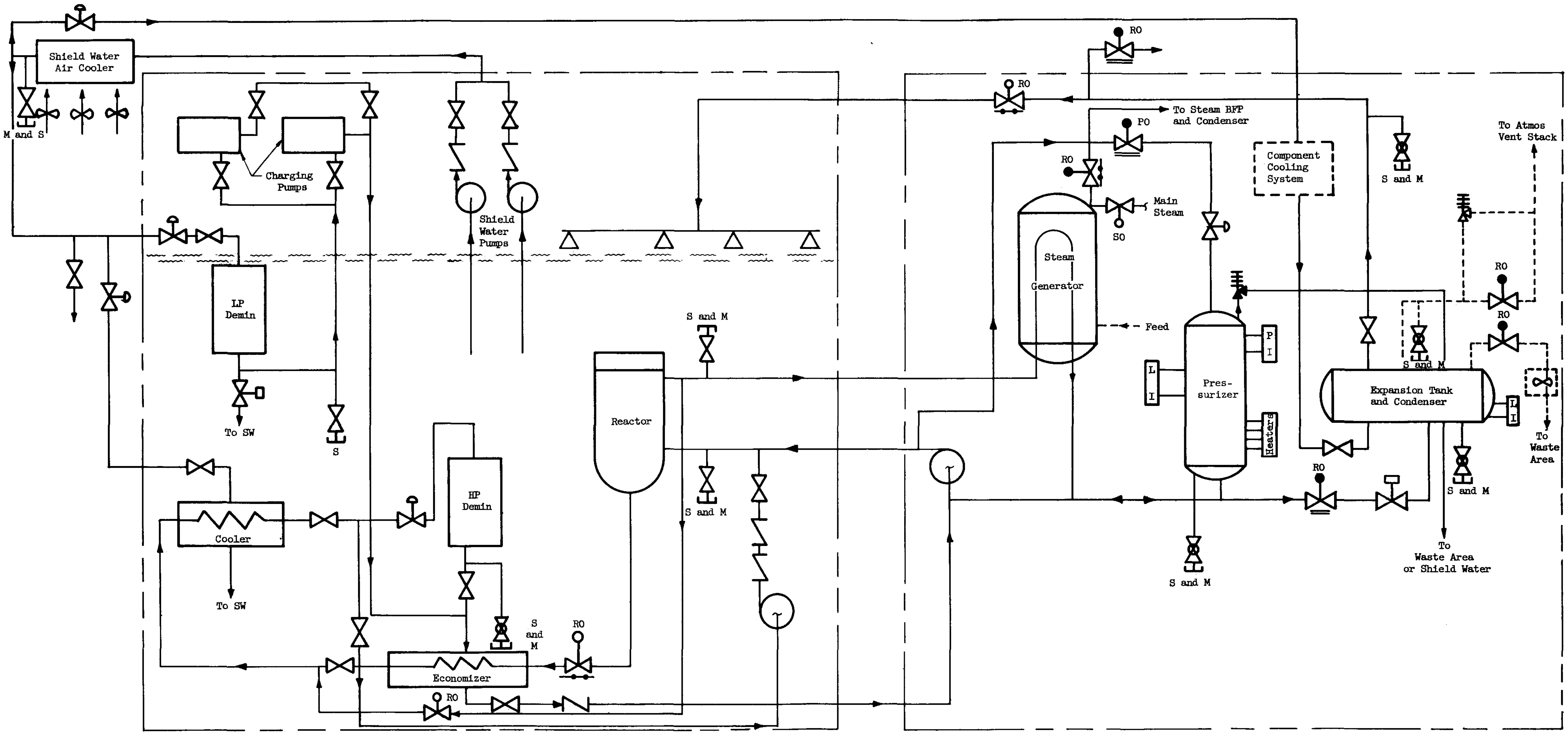


Fig. III-1. PM-1 Primary System--Basic Flow Diagram

2. Pressurizer

The pressurizer maintains primary system pressure within the designed limits above and below the 1300 psia normal operating pressure during all transients which can occur in normal operation. These limits are such that the design pressure is not exceeded during an insurge of water to the pressurizer, and the system pressure is sufficient to prevent bulk boiling any place in the core during an outsurge of water.

The transient which was determined by analog studies as producing the most severe expansion of the water in the main coolant loop occurs when:

(1) The heating load is zero.

(2) The plant gross electrical load is instantaneously decreased from 1250 to 125 kw. A gross load of 125 kw represents the plant auxiliary load only.

As a result of this transient, the primary water volume expands 1.3 cubic feet in 92 seconds and the primary system pressure increases to 1470 psia.

The transient producing the most severe contraction of main coolant loop water volume occurs when the electrical load is increased from 125 kw to 1250 kw and the heating load is zero. As a result of this transient, the primary water volume contracts 1.25 cubic feet in 37 seconds and the primary system pressure decreases to 1170 psia. The thermodynamic design power is not attained in this transient. A comparison of the power generated when the system pressure is 1170 psia shows this results in a condition that is within the design limits.

The transient producing the greatest reactor power variation is identical to the one described above with the exception that there is a full heating load. In this case the water volume in the main coolant loop contracts 0.9 cubic feet in 35 seconds and the primary system pressure decreases to 1210 psia, which is above the thermodynamic design pressure of 1200 psia.

The pressurizer volume is sufficient to maintain the above conditions without the use of the heaters or sprayers. The heaters are of the immersion type and will be used both to maintain pressure during operation and to heat the pressurizer during startup. The pressurizer is also equipped with a spray system to quench the steam in the top section of the pressurizer and thus reduce the pressure.

The assumptions used in sizing the pressurizer were that the heaters and spray system were inoperative and that the steam expansion and compression were isentropic.

The pressure limits indicated above are conservative, since no advantage has been taken of mixing on an insurge or the flashing of water on an outsurge.

3. Pressure Relief System

The pressure relief system includes pressurizer steam relief valves, an expansion tank and pressure relief condenser.

Two or more relief valves will be installed on the pressurizer to release steam at a rate which will prevent the primary system pressure from exceeding 1650 psia with all the reactor power heating the primary coolant and only the conservative end of life negative temperature coefficient controlling the reactor power. Design of the valves is in accordance with a special ruling (Case No. 1271 N) by the ASME Boiler and Pressure Vessel Committee interpreting Section VIII of the Unfired Pressure Vessel Code with respect to the safety requirements for pressurized water reactor vessels.

The pressure relief condenser and expansion tank condenses the released steam at a pressure of less than 50 psia. Condensate from the condenser and expansion tank is drained to the shield water tank or waste disposal system, depending on the activity level. If radioactive, the non-condensibles are discharged into the gas removal system. Otherwise, they are released to the environment.

4. Charging System

A positive-displacement pump is used to charge the primary system with make-up water from the low-pressure demineralizer discharge. The discharge from the pump (1 gpm) mixes with the blowdown flow (2 gpm) entering the loop between the pump section and the steam generator outlet. A standby pump is provided to insure the availability of the charging system.

5. Main Coolant Loop Blowdown System

In order to limit impurities to a tolerable concentration in the main coolant loop water (less than 2 ppm), the coolant water is continuously recirculated at a rate of 2 gpm through the high-pressure demineralizer. The fluid is passed through a regenerative heat exchanger and cooler before entering the demineralizer. The primary coolant pump produces the recirculation flow through the blowdown system.

In the case of any component failure, the blowdown system can be isolated by two remotely operated valves in series at the inlet connection to the main coolant loop and a check valve at the outlet connection.

6. Poison Injection System

The boric acid injection system is provided as a standby emergency shutdown mechanism which can maintain the reactor in a subcritical condition at any time. The amount of boric acid solution injected is sufficient to overcome the maximum reactivity in the cold, clean core condition when diluted by both the primary coolant and the shield water.

The boric acid system includes a pressurized boric acid tank and the connecting line to the reactor. The acid is injected at the reactor vessel bottom in the vicinity of the core inlet. A solenoid trip valve and a check valve are located in the connecting line at the exit of the tank. The valves may be operated by energizing a guarded electrical switch or manually. A manual charging pump is also provided, to permit manual injection of the boric acid solution.

7. Shield Water System

The shield water system accomplishes several objectives:

(1) Provides shielding during operation and maintenance.

(2) Continuously purifies the shield water to limit impurity concentration to less than 2 ppm.

(3) Provides a continuous cooling source for the pressure relief condenser.

(4) Maintains 130° F temperature of the shield water which picks up heat from the following sources:

    (1) Gamma heating in shield water

    (2) Spent core heat removal (one day after shutdown)

    (3) Demineralizer cooler

    (4) Component cooling (as in pumps).

The shield water circulating pump removes shield water from the bottom of the reactor package and pumps the water through the shield water system. This includes:

(1) Shield water air cooler

(2) Low-pressure demineralizer

(3) High-pressure demineralizer cooler

(4) Component cooling system

(5) Pressure relief condenser.

Two shield water circulating pumps are installed, one as a standby, to insure that the shield water system is continuously in operation. A failure of both pumps would necessitate a plant shutdown, since the pressure relief condenser and the component cooling system will require a continuous supply of cooling water.

The shield water air cooler employs four units which remove a total of 250 kwt from the shield water. With the failure of one of these units, sufficient capacity remains to maintain the shield water temperature at a level below 140° F which is the maximum operating temperature for the demineralizers.

The low- and high-pressure demineralizers are uninsulated and are submerged in the shield water for additional cooling.

8. Auxiliary Reactor Cooling System

The auxiliary reactor cooling system employs the demineralizer cooler and a pump to cool the reactor core after shutdown. This system is necessary when the primary coolant loop is opened at any point for maintenance. Opening the loop will disrupt natural convection and, consequently, the removal of afterheat at the steam generator.

The high-pressure demineralizer cooler, which normally cools the blowdown entering the high-pressure demineralizer, is sized to remove the core afterheat when the primary system pressure and temperature have been reduced to permit maintenance and inspection.

9. Waste Disposal System

The radioactive liquid wastes from the PM-1 power plant will be concentrated by evaporation and then stored in tanks under an earth shield.

The equipment comprising the liquid waste disposal system includes a sump tank, evapora.or, condenser and waste storage tanks.

The sump tank will accommodate the total volume of liquid and solid material required to decontaminate the main coolant loop. This tank will serve to contain radioactive wastes until they can be concentrated by the evaporator. The sump tank receives material from four sources:

(1) The main coolant loop, during the decontamination process.

(2) The secondary side of the steam generator, in the event excessive radiation is detected. This radiation might be caused by a tube failure in the steam generator.

(3) The pressure relief condenser, in the event that excessive activity is detected in the condensate.

(4) Wastes having low level activity from the decontamination room.

The evaporator will utilize batch operation. The fluid in the sump tank will be drained to the evaporator, one batch at a time; the volume of the batch being dictated by the evaporator capacity. The vapor from the evaporator will be condensed and the activity determined. This condensate will then be released to the environment provided the activity is sufficiently low. Otherwise, the condensate will be returned to the evaporator and recycled. The concentrated wastes remaining in the evaporator will be drained to the waste storage tank at the end of each evaporation cycle.

Calculations indicate that the activity of the condensate will be less than 10^{-13} curies per milliliter after one evaporative cycle, provided there has been no rupture of the fuel element cladding. Even with a large number of ruptured fuel elements, one additional evaporation cycle will lower the activity of the condensate to the level indicated above.

The waste storage tanks are designed to contain all concentrated wastes accumulated over a 20-year period. The tanks will be located under an earth shield.

The gas disposal system consists of reaction vessels, containing a silver solution and activated charcoal beds. All the radioactive gases vented from the primary system will first be passed through the silver solution to chemically remove the iodine and then discharged into the activated charcoal until the activity decays to a level at which they can be released to the environment.

B. DESCRIPTION OF THE REACTOR--MECHANICAL DESIGN

1. General Structures

The overall reactor core design, as installed within the reactor pressure vessel, is illustrated in Fig. III-2. The active core region is divided into seven fuel bundles. Six bundles, each containing a control rod, are

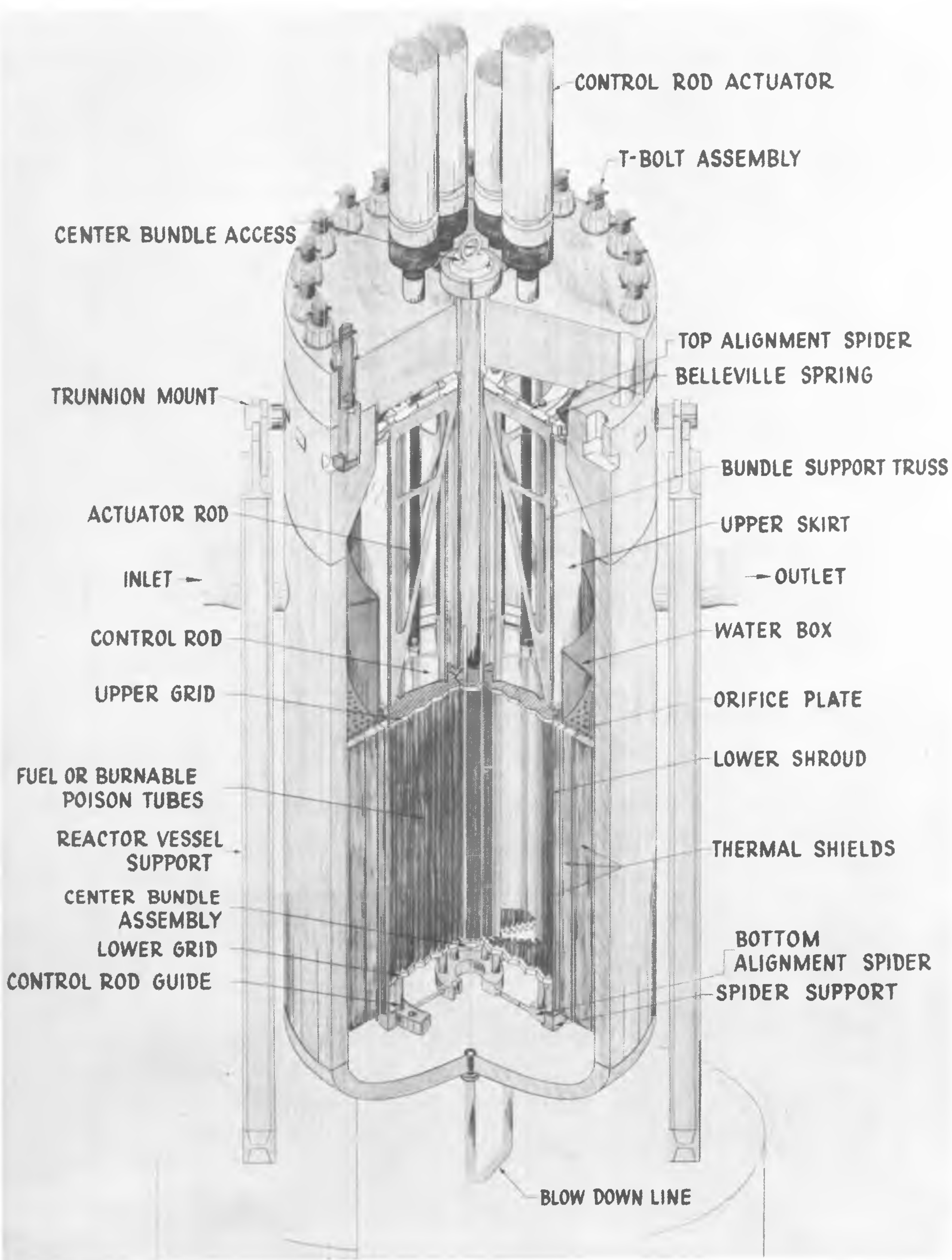


Fig. III-2. Reactor Core Design

located about the periphery of the core. The seventh bundle makes up the center of the core region. This bundle can be instrumented for experimental purposes, if desired.

Each of the peripheral fuel bundles is supported from the top by means of its support truss and guide bars. These bars, which support the active core, extend along the full height of the fuel bundle and terminate in the upper and lower alignment spiders. The upper skirt, which holds the bundle truss, also contains the exit flow orifices. The lower shroud provides a flow baffle between the core and the inlet water while also acting as the first thermal shield. A Belleville spring, compressed during installation of the reactor head, is used to provide the necessary holddown force for the complete core assembly.

The central fuel bundle is made to be removable through the pressure vessel head. It is supported from the head flange in order to eliminate any differential expansion between the bundle and instrumentation lines.

Stainless steel material will be utilized throughout the core. A majority of the structural components will be made from AISI type 304 or 347 stainless. The fuel tubes will be constructed from a modified type 347, containing low cobalt and tantalum, in order to minimize system activation. A hard material such as 17-4 PH will be used on one side of all rubbing surfaces, i.e., the control rod wear pads.

2. Fuel Bundles

Each of the fuel bundles contains its upper support structure, fuel and burnable poison tubes, upper and lower full tube support grids, plus a means for spacing the grids. In the six peripheral bundles the support grids are secured to the control rod guides. In the center bundle three fuel tubes are replaced by stainless steel tubes which hold the grid plates. All fuel and poison tubes are locked into the lower grid plate, but are allowed free thermal expansion in the upper grid. The lower grid also contains flow orifices for minimizing flow redistribution during local boiling.

The active core region approximates a right circular cylinder 23 inches in diameter and 30 inches in length. It will contain 700 to 750 fuel and 60 to 90 burnable poison elements.

The exact number and distribution will be determined during the zero power (critical) test program. The center bundle will contain 16 elements and each of the six peripheral bundles will contain a total of 141 elements.

The fuel elements are 0.50 inch in outside diameter and are spaced in a triangular array with 0.665 inch between centers. The tube wall (0.042 inch thick) is made up of a 0.030-inch fuel matrix with 0.006-inch cladding on both the inner and outer diameters. The fuel matrix is made up of a compact of stainless steel powders and approximately 25 to 26 W/O of UO_2 enriched to 93% in U-235. The total reactor loading will be approximately 26.7 kilograms of U-235.

Cylindrical burnable poison elements of 0.50-inch diameter will contain one-half to one weight percent natural boron in a stainless steel alloy. Each element will be clad with 0.012 inch of stainless steel.

3. Control Rods

Control for the complete core will be supplied by six Y-shaped control rods. One is located in each peripheral bundle, providing equal spacing around the core at an effective diameter of approximately 12-1/2 inches. Each active leg of the Y section is 3.25 inches in length by 0.25 inch thick. Wear pads are located on each leg of the control rod to provide the proper clearances and wear resistance over the life of the reactor core.

A control rod stroke of 30 inches will cover the complete core length. During detachment from the control drive mechanism, the control rod will be lowered approximately 3/8 inch to rest upon the bottom grid plate.

C. DESCRIPTION OF THE REACTOR--NUCLEAR DESIGN

1. Fuel Loading

The design conditions upon which the initial fuel loading was based were: That thermal power be 9.35 megawatts and the core life be 2 years or 18.70 megawatt years; that the cold core temperature be 68° F and the hot core temperature be 463° F; and that the moderator and coolant be water at 1300 psia.

The preliminary design established the general configuration of the core to be as follows:

Geometry, right circular cylinder	(approximately)
Diameter	23.0 in.
Height (active)	30.0 in.
Fuel element (outside diameter)	0.50 in.
Fuel element (inside diameter)	0.416 in.

	(approximately)
Fuel matrix thickness	0.030 in.
Clad thickness	0.006 in.
Number of fuel elements	725
Pitch, triangular	0.665 in.
U-235 burnup (≈1.32 gms/megawatt day)	9.01 kg

Results of the preliminary core analysis indicated that the weight percent of UO_2 in the fuel matrix required for the above core was ≈ 25%, which yielded an inventory of 26.7 kg of U-235, distributed uniformly throughout the core and fuel tubes.

2. Lumped Burnable Poison Loading

Initial studies to determine the feasibility of using lumped burnable poisons in the core showed that lumping some or all of the burnable poison would significantly reduce the peak reactivity to be controlled.

As a result of these studies, the final design of the PM-1 core will be based upon the use of lumped burnable poisons. Final distribution patterns for the lumped poisons will be determined by further analysis and experimental data gained in the critical experiments to be performed at The Martin Company.

3. Reactivity Control

Control requirements.- Preliminary studies to establish overall control requirements were performed. It was found that, by proper choice of a lumped poison system, the maximum amount of reactivity that needed to be controlled could be significantly reduced. In addition, reactivity peaking with time could either be reduced or eliminated.

Results of preliminary design studies using various lumped poison systems indicate that approximately 3 to 6% of the reactivity will have to be controlled throughout the core life at operating conditions (i.e., 463° F, equilibrium xenon, plus other fission product poisons). The reactivity, of course, decreases to zero at the end of core life (see Fig. III-3).

The difference in reactivity between the hot operating conditions and the cold, clean core condition (68° F and all of the fission products assumed to decay out of the system) was calculated for various times. This reactivity difference increased from ≈ 6.1%, initially, to ≈ 7.2% at midlife, to ≈ 8.3% at the end of life. The increase in reactivity is primarily due to the increase in fission products with time.

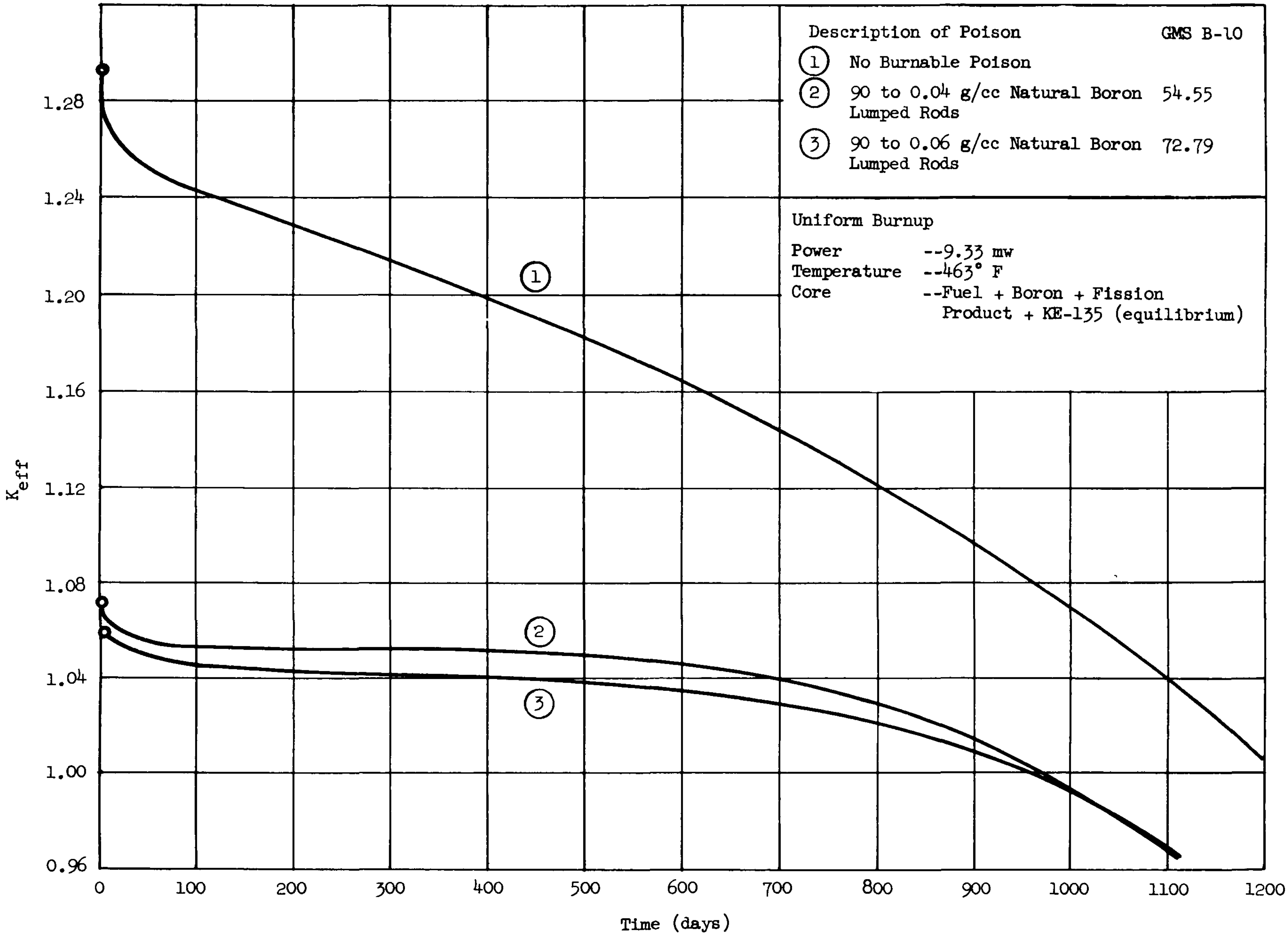


Fig. III-3. Burnable Poison Analysis--K_{eff} vs Time

The maximum amount of reactivity that must be controlled is, from Fig. III-3 (Curve 2) and the reactivity calculated above (operating condition to cold clean), ≈ 12.2% in reactivity. In using Fig. III-3 for this calculation, the K_{eff} values given are converted to reactivity.

The reactivity effect of equilibrium xenon for full power operation and maximum xenon buildup after shutdown has been calculated as a function of time. At initial startup, these values are -1.4% and an additional -0.23%, respectively.

The effects of temperature and moderator void fraction are discussed later.

The control requirements for zero days, midlife and end-of-life are 12.0%, 12.2% and 8.3%, respectively.

Control rod design and worth.- The rod poison used in the preliminary design studies was boron-stainless steel (2.5 weight percent B-10). Although boron may not be used as the final absorber material, the results of these studies are applicable to other materials.

The rod configuration and location consistent with the preliminary design core (considering actual fuel element locations, rod guides, etc.). are shown in Fig. III-2. For these conditions, the reactivity worth of the four-, five- and six-rod banks will be greater than or equal to the values given below:

Six Y rods	-23.6% Reactivity
Five Y rods	-14.8% Reactivity
Four adjacent Y rods	-9.3% Reactivity

These calculations assumed a clean core.

Control design requirements established for the PM-1 core include a stuck rod criterion, i.e., that the rods must be adequate to shut down the reactor at any time, with one rod stuck in the full out position or with any two rods stuck in the operating position.

The rod bank worth for five of the six rods was calculated to be ≈ 14.8% reactivity. Since the maximum cold, clean core reactivity is ≈ 12.2%, the shutdown reactivity margin is about -2.6%. This satisfies the condition of one rod stuck fullout.

In order to evaluate shutdown capabilities with any two rods stuck in the operating condition, consideration must be given to both the cold, clean core reactivity and the hot operating core reactivity. At present, it is planned to use a six-rod bank operation for control. An alternating three-rod bank may be used near the end of core life to permit burnup of fuel and burnable poison in the extreme upper portion of the core. This will be investigated further.

The minimum worth of the control system with two of the six rods stuck in the operating position was then determined for various times during core life as shown below:

For zero days	-12.9% Reactivity
For midlife	-12.3% Reactivity
For end-of-life	-9.3% Reactivity

Similar calculations for the increased boron loading, shown by Curve 3 of Fig. III-3, indicate that even greater shutdown margins can be obtained.

Control rod worths were calculated using the two-dimensional, three-group, IBM-704 machine code PDQ in x-y geometry. The machine-calculated rod bank worths (i.e., the difference between core reactivities with and without rods) were corrected for the effect of the step approximation required in mapping two of the three arms of the Y-shaped rods in x-y geometry. Comparison of the analytical results obtained using this technique with experimental results in previous studies indicated that the analytical results obtained are good within ± 10%. For the rod design studies described above, the calculated worth was assumed to be high by 10%.

Control rod material.- Several control rod absorber materials are at present being considered for use in the PM-1. The basic material compositions are:

(1) Boron-10-Stainless Steel (2.5 weight percent B-10)

(2) Hafnium (Hf metal)

(3) Cadmium-Indium-Silver (5, 15 and 80 weight percent, respectively)

(4) Europium Oxide--Stainless Steel (30 weight percent Eu_2O_3)

(5) Gadolinium Oxide--Samarium Oxide--Stainless Steel (15 weight percent Gd_2O_3 and 15 weight percent Sm_2O_3).

The rod absorber thickness for all cases was 0.25 inch (0.625 cm).

More detailed analyses considering epithermal absorptions, spatial burnup in the rod and iteration-type burnup will be made. The metallurgical and economical aspects of each control rod material must also be evaluated before a final selection can be made.

4. Temperature and Void Coefficient

Temperature coefficient.- The reactivity of the core was calculated for different temperatures from 68 to 473° F. Both nuclear and density temperature effects were considered. Specifically, the change in reactivity with temperature was assumed to be due to:

(1) The change in microscopic thermal cross sections with temperature.

(2) The change in reflector savings resulting in a change in the buckling with temperature.

(3) The change in density of water with temperature. Core materials other than water were assumed to have a constant density in the temperature range studied.

The reactivity as a function of temperature is shown in Fig. III-4. The average temperature coefficient from 68 to 463° F is $-1.14 \times 10^{-4} \Delta \rho/°F$. At operating temperature, the temperature coefficient is $-2.84 \times 10^{-4} \Delta \rho/°F$.

Reactivity was calculated using a three-group diffusion code, Program C-3. The thermal disadvantage factors, calculated using Program I-2, and reflector savings (both calculated as a function of temperature) were used to account for heterogeneity and reflector effects.

Void coefficient studies.- The effect of voids on reactivity was calculated for 0.5, 1.0, 1.5 and 2.0% void in the moderator. For these studies, a uniform void distribution was assumed and the void formation was represented by a change in volume of the moderator. The reactivity (relative to 68° F and no void) as a function of temperature for different void fractions is shown in Fig. III-5.

D. DESCRIPTION OF THE REACTOR PRESSURE VESSEL

The PM-1 reactor pressure vessel will be fabricated from AISI type 347 stainless steel, conforming to the requirement of SA-167, 0.08% maximum carbon.

The vessel wall thickness and top and bottom head thicknesses were determined to conform to Section VIII, ASME Boiler and Pressure Vessel Code. Radiation heating, cyclic stresses and other transient conditions were treated as outlined in the "Tentative Structural Design Basis for Reactor Pressure Vessels," issued by the U.S. Navy Bureau of Ships, 1 April 1958. The reactor vessel is trunnion-mounted to alleviate the effects of loads due to thermal expansion of connecting pipe.

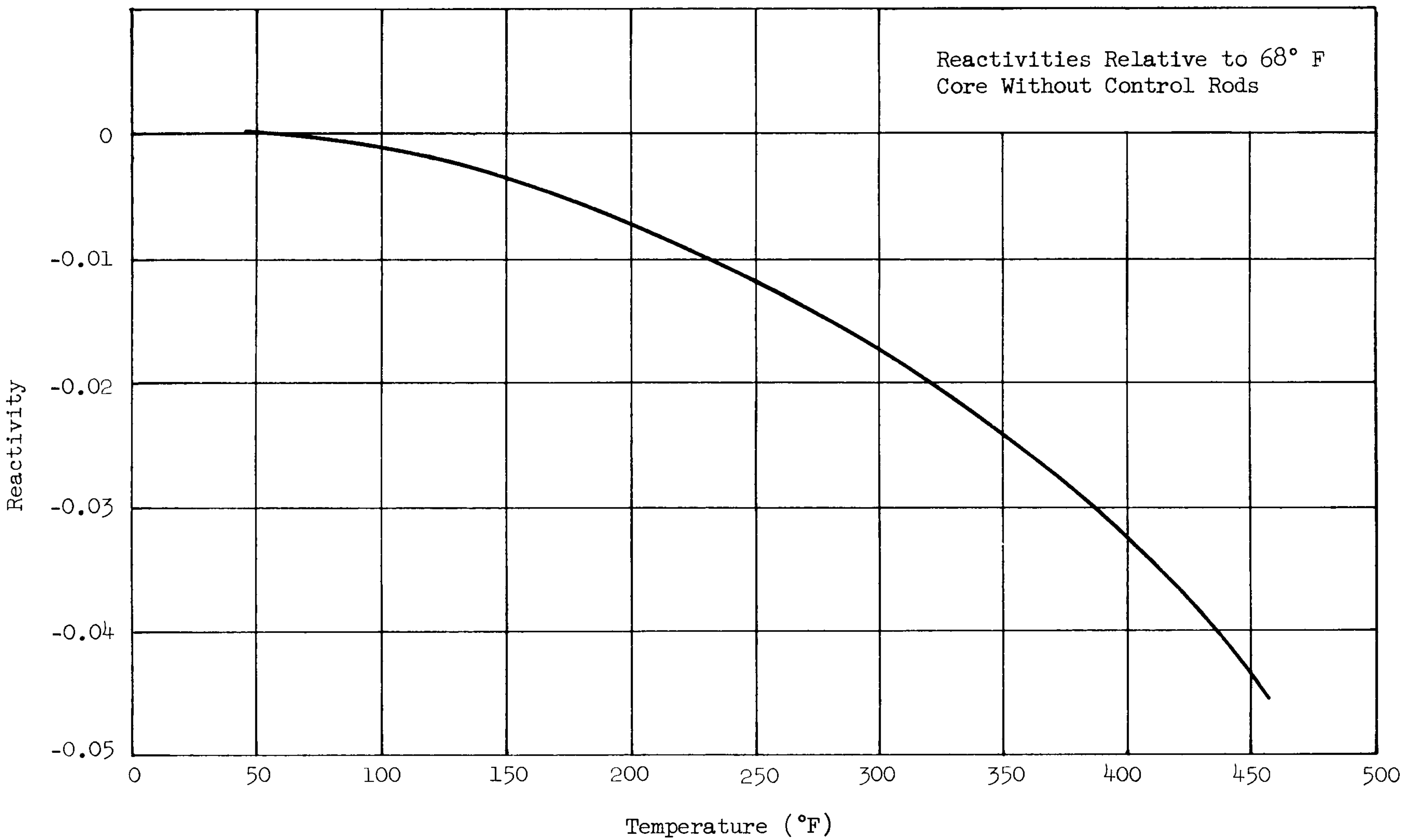


Fig. III-4. Reactivity as a Function of Temperature

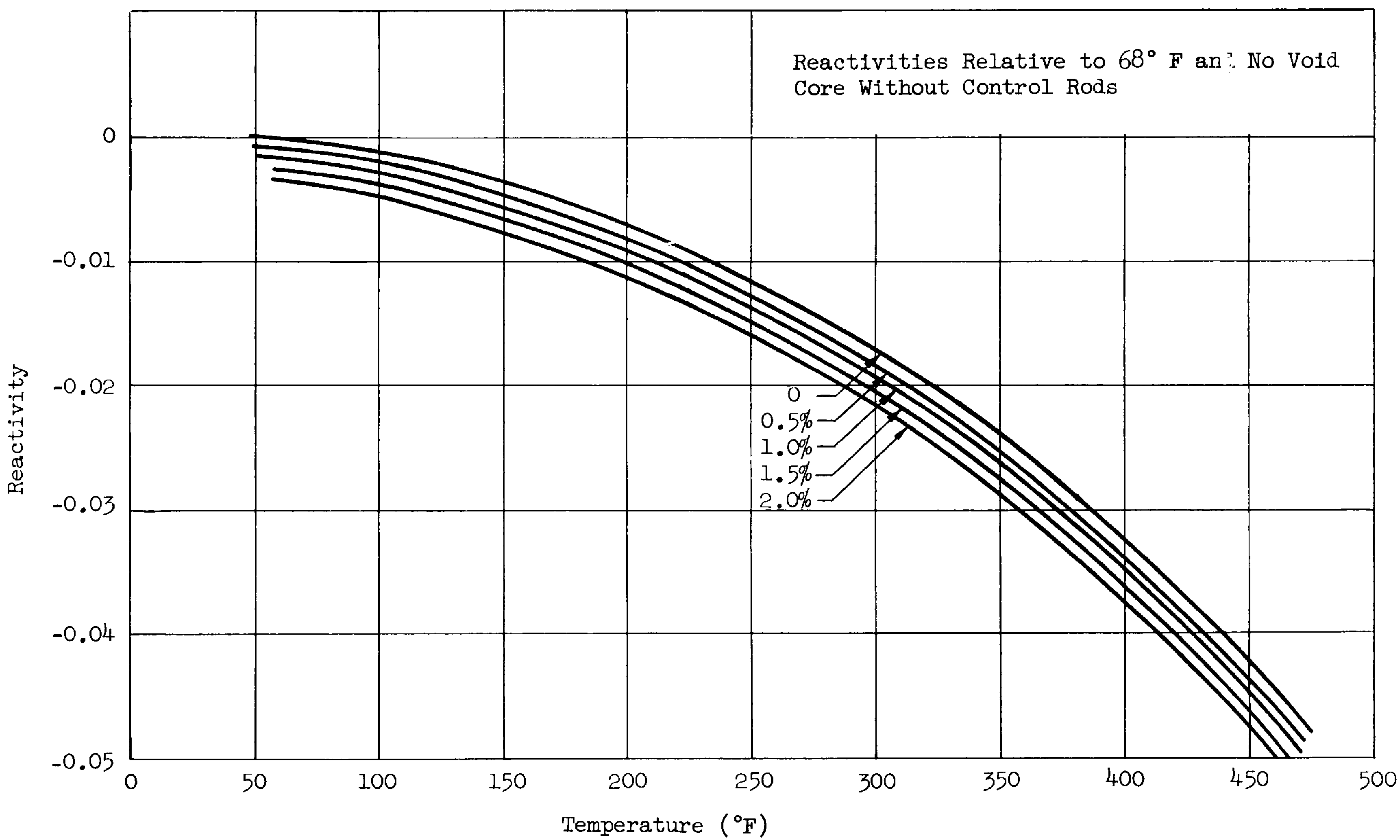


Fig. III-5. Effect of Voids on Reactivity

The reactor vessel, Fig. III-2, is basically a cylindrical shell having approximately a 40-inch outside diameter and a 2-5/16-inch wall. The vessel is designed for 1500 psi internal pressure at a temperature of 600° F.

The lower end of the vessel is closed by a two-to-one ellipsoidal head. The upper end is closed by a flat circular head having six actuator nozzles of 2-1/2-inch outside diameter at the head intersection and increasing to a 4-1/2-inch outside diameter hub located 8 inches above the top head. Each hub is drilled and tapped to accommodate the bolted flange on the control rod housing. The joint is sealed with gaskets. The actuator nozzles are equally spaced along a concentric circle 12-1/8 inches in diameter. A gasketed port 3-1/2 inches outside diameter penetrates the head on the vessel centerline. The port is provided for removal of the central fuel bundle.

The vessel is designed with an internal flange to reduce the load on the head. The flange is grooved to receive two concentric gaskets. A leak-off groove is provided for collecting any leakage past the inner gasket. Such leakage is monitored and conducted to the low-pressure demineralizer system.

The main closure utilizes twenty-two 2-inch diameter T-bolt assemblies.

Three concentric stainless steel thermal shields are located inside the pressure vessel surrounding the active core. The outside diameter of the innermost shield is 24 inches. The wall thicknesses of the shields from the innermost to the outermost are 1 inch, 1-3/4 inches and 2-1/2 inches, respectively. There is a 1/4-inch separation between neighboring shields. The overall length of the shields is 36-1/4 inches so that they extend 1-3/8 inches above and 4-7/8 inches below the active core. The shields are segmented into four parts. Each segment is replaceable by use of special extension tools operated from the refueling floor above the reactor.

The reactor core is supported from the vessel wall by a concentric orifice plate near the bottom of the water box. The thermal shields also are supported by this plate. An orificed skirt is provided in the area of the water box to direct the coolant flow. This skirt is held down by a Belleville spring between the head and the skirt.

Renewable spider-type rings are provided at the top and bottom of the vessel for proper core alignment during installation. The lower spider ring has been made removable to accommodate other types of cores.

The rings, supports and baffling will be fabricated from AISI type 304 stainless steel.

Primary coolant enters and leaves the vessel through 6-inch nozzles located diametrically opposite one another and 8 inches above the top of the active core. The annular water box is tapered from 18 inches at the inlet to 6 inches at the opposite side.

From the inlet nozzle, the coolant is distributed in the water box around the inner circumference of the reactor vessel and flows downward along and between the thermal shields. In the bottom vessel head, the coolant is re-directed upward through the core into a plenium above the core where it is baffled into the outlet nozzle by the skirt baffle. The skirt baffle is orificed to ensure proper flow distribution.

The reactor vessel is insulated thermally with 1-inch thick loose fiberglass insulation packed between the vessel wall and a double-walled stainless steel shroud.

E. DESCRIPTION OF THE CONTROL ROD ACTUATORS

The actuator system to be installed in the PM-1 Power Plant is a bank of six magnetic jack actuators powered by a solid state power supply located in the control console.

An actuator consists of the following major components which are illustrated in Fig. III-6:

(1) The pressure thimble, which encloses the jack rod bundle, scram spring, dashpot and control rod attachment. The pressure thimble is bolted to a nozzle extending from the reactor vessel head.

(2) The jack rod bundle, which moves vertically within the pressure thimble and is actuated by a movable armature and magnetic fields (also within the pressure thimble) in increments of 0.080 inch.

(3) The scram spring, situated on top of the jack rod bundle and anchored against the top of the pressure thimble.

(4) The dashpot, provided within the pressure thimble.

(5) The control rod attachment, an extension of the jack rod bundle.

(6) A hollow cylinder, containing the grip, lift and hold coils, and the position-indicating transmitters. The cylinder is situated over and around the pressure thimble and latched into position. This unit is subjected to the pressure of the shield water only and does not come into contact with the primary system fluid.

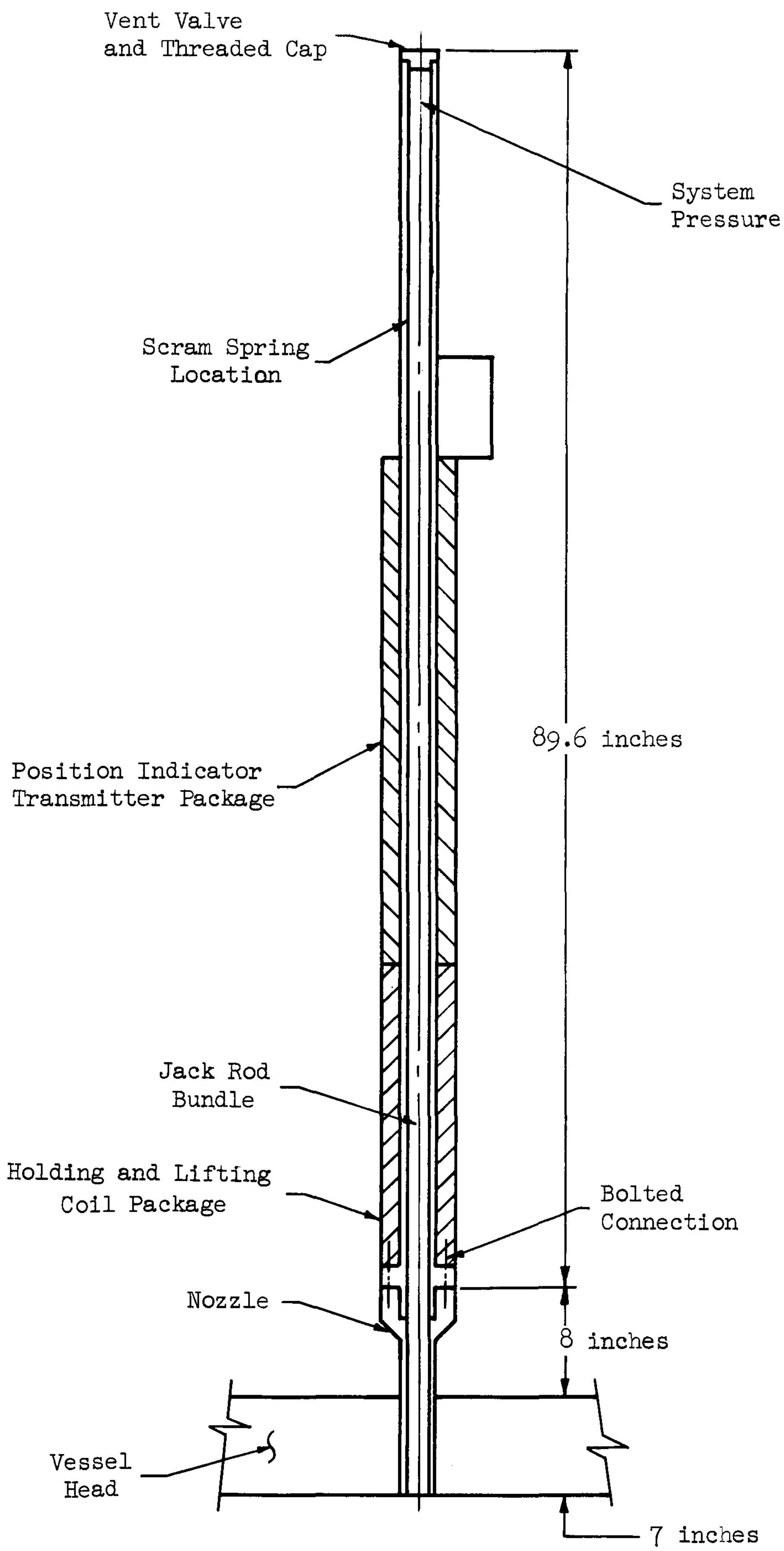


Fig. III-6 Actuator

(7) A bleed-off valve located in a plug situated in the top of the pressure thimble. This plug is screwed into position as a seat for a gasket and a retainer for the scram spring, also retaining the primary system coolant within the pressure thimble.

The components are essentially of one-piece construction and, where parts are joined, threading and welding is combined to maintain structural integrity.

The entire mechanism inside the pressure thimble is removable as a unit, through the top of the thimble, by removing the plug containing the bleed valve or by removing the pressure thimble from the reactor vessel head attachment. These are the only points of disconnect on the actuator.

F. DESCRIPTION OF THE SECONDARY SYSTEM

1. General

The secondary system performs a dual function in converting the thermal energy received in the steam generator into high-quality electrical power and into low-pressure steam for space heating. The design ratings are 1000 kwe net and 7 million Btu per hour of low-pressure steam. The major equipment in the system consists of a steam generator, a turbine-generator unit, an air-cooled steam condenser system, electrical switchgear and emergency power source. The functioning of the loop may be visualized by reference to the flow diagram (Fig. III-7) and the secondary loop layout (Fig. I-3). In general, standard commercial equipment is used. Certain components have been given special attention to insure that the PM-1 Plant requirements are met. These include the steam generator, turbine-generator unit and steam condenser system.

2. Steam Generator

The steam generator is of the vertical-U tube, integral drum type design with primary coolant in the tubes and secondary water and steam in the shell. The reference design is a cylindrical shell with hemispherical ends, 16 feet, 7 inches high and 30 inches outside diameter. The steam generator pressure vessel, as well as all pressure vessels in the system, is constructed according to ASME Code. The heat-transfer surface is composed of 420 U-shaped tubes, which are 1/2 inch OD, 18 BWG walls. The tubes yield a heat transfer area of 1080 square feet, based on the outside surface.

The steam pressure in the generator will vary almost linearly with load from 300 psia at full load to 481 psia at no load; temperatures will correspond to saturation at the given pressure.

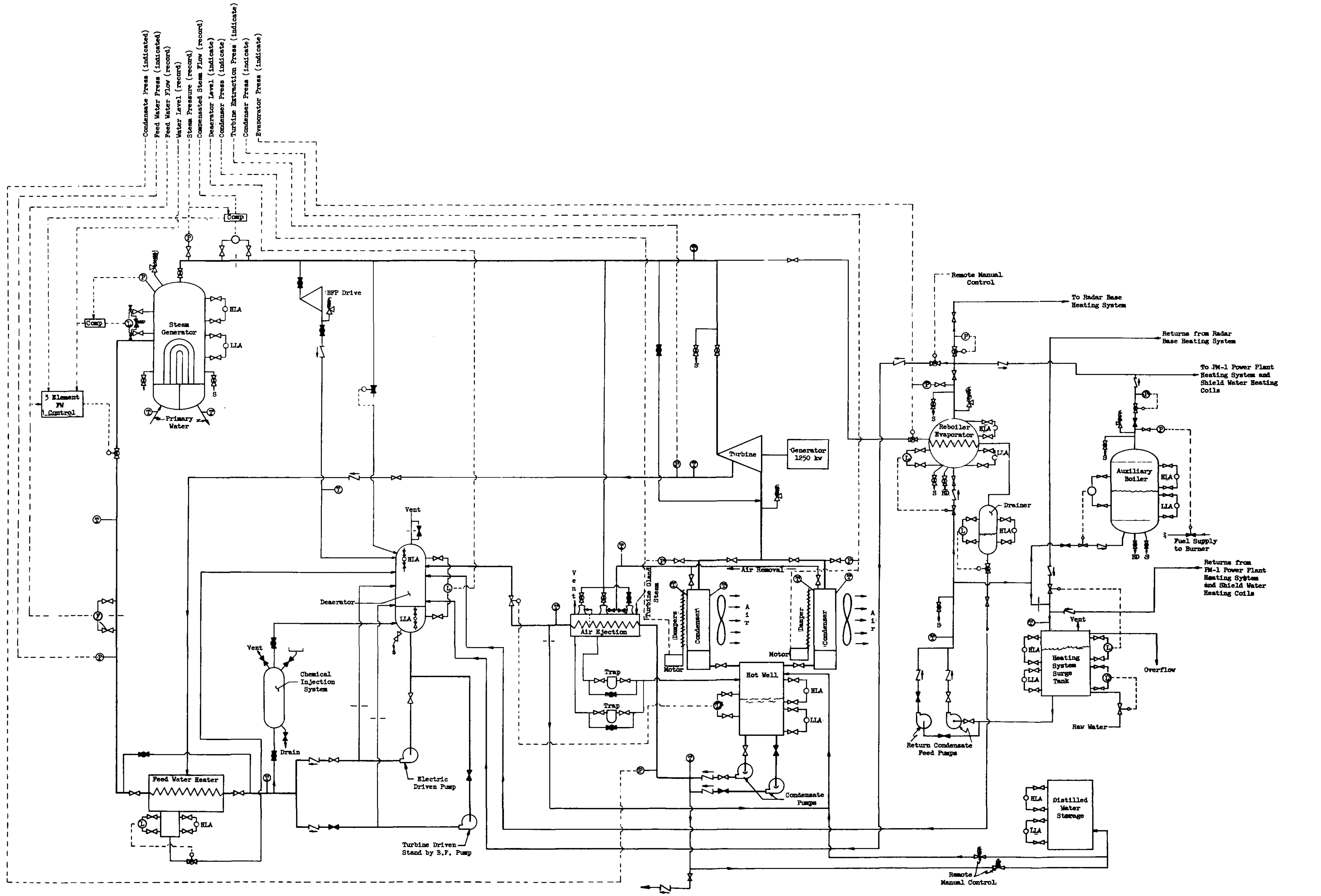


Fig. III-7. Heat Balance, Flow Diagram, Instrumentation Schematic--Secondary System--PM-1 Power Plant

3. Turbine-Generator

The turbine-generator unit is essentially a marine-type machine. It consists of a single package module containing a turbine, single reduction gear and generator, all integrally mounted on a single bed-plate. The unit is rated at 1250 kwe continuous duty.

The steam turbine is a high-speed (approximately 7500 rpm), impulse-type machine. The turbine system is equipped with protective devices, including overspeed trip, low oil-pressure trip and high back-pressure trip.

The generator is a salient-pole machine, designed to produce 1250 kwe net at 80% power factor, and 1563 kwe net at 100% power factor. Output is 2400 to 4160 volts, 60-cycle, 3-phase, 4-wire. Both ends of the windings are brought out for differential protection and, since the neutral will be solidly grounded, the windings will be braced to withstand a full line-to-ground fault. The generator is air cooled with an integral ventilation system.

A static exciter is used with a coordinated static voltage regulating system.

4. Condenser System

The air cooled condenser system consists of two identical finned-tube heat exchanger modules. Each unit is a box-shaped, all-welded aluminum structure, 8 feet, 8 inches square and approximately 30 feet long. The sides are formed from finned tubes, several rows deep. The bottom of the structure is enclosed, and the top serves as a mounting for four induced-draft fans. The total heat transfer surface of the condenser system is approximately 60,000 square feet.

Operation of the condenser is as follows: steam from the turbine exhaust enters a header located at condenser mid-length; the steam then flows in either direction inside the finned tubes, where it is condensed, and then drains by gravity to the water boxes at the ends of the unit. Air flow is induced transversely across the tubes, into the plenum space in the center of the structure and out through the exhaust fans. Air flow is controlled by automatically operated louvers which are positioned by a condenser pressure-sensing device. Fan power is monitored by manual control.

5. Switch Gear

The switch gear is of commercial design and provides control and protection to the electric generator, the lines and station power transformer. The gear includes differential relays and overcurrent relays for generator protection. Tie line and transformer protection is provided by overcurrent and ground relays. Synchronizing equipment is provided for synchronization of the main generator with the auxiliary diesel generator or the tie line.

6. General Heat Transfer Apparatus

Included in this category are the deaerator, evaporator-reboiler, feedwater heater, and jet ejector after condenser. These units are all standard commercial designs.

7. Auxiliary Power Supply

Auxiliary power for starting and shutting down the plant is provided in the form of a 200-kw, high-speed, diesel-driven generator. The unit is capable of continuous duty and satisfies the need for emergency electrical power when the nuclear plant is down.

To insure a source of uninterrupted power for instrumentation and control, a series of batteries and motor-generator units is provided. The arrangement of these is shown in the electrical one-line diagram in Fig. III-8.

8. Miscellaneous

Both condensate and boiler feed pumps have standby units. For the condensate pumps, the operating and standby units are identical. The standby boiler feed pump is steam driven to insure that the steam generator water level may be maintained during afterheat removal in the event of no electrical power.

During periods when the nuclear plant is off the line, heating for the plant is supplied by an auxiliary oil-fired boiler.

G. DESCRIPTION OF THE CONTROL AND INSTRUMENTATION SYSTEM

The control and instrumentation system for the PM-1 is subdivided into the following groups:

(1) Nuclear instrumentation

(2) Reactor safety system

(3) Primary system plant instrumentation

(4) Secondary system plant instrumentation.

Each of these groups is discussed below.

1. Nuclear Instrumentation

The nuclear instrumentation is completely transistorized and is housed in miniature plug-in assemblies.

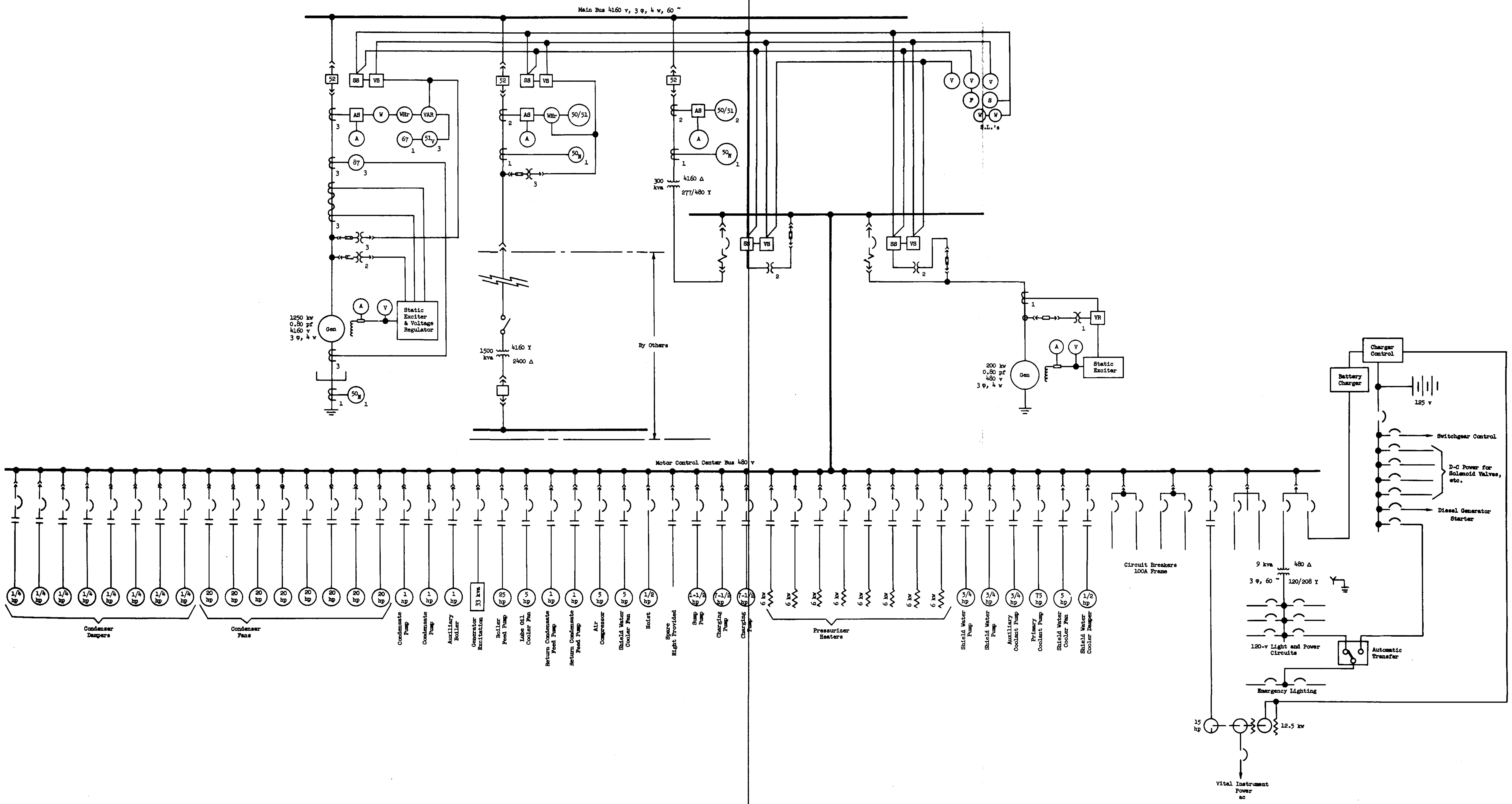


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Fig. III-8. PM-1 Nuclear Power Plant Electrical One-Line Diagram

Reactor power levels will be measured from source, through intermediate, to power range, using seven channels. The interconnection of the various channels is shown in Fig. III-9.

Channel I and II--Source Range Instrumentation. - BF_3 proportional counters are used in each of the two source-range channels. Counts per second over a range of 1 cps to 1×10^5 cps are indicated on a logarithmic scale corresponding to neutron fluxes of from 0.25 nv to approximately 2.5×10^4 nv. In addition, the source range amplifier drives a period amplifier. Reactor period is indicated over a range of -30 to ∞ to +3 seconds.

The period signal is used for high start-up rate protection in the source-range channel. If either instrumentation channel indicates the presence of a positive period of less than 15 seconds, further control rod extraction is prevented. If both channels indicate a period of less than 10 seconds, the reactor is scrammed.

Channels III and IV--Intermediate Range Instrumentation. - Gamma-compensated ion chambers are used in each of two intermediate range channels. These chambers furnish a current signal to a log N amplifier to indicate the percentages of full reactor power attained over a range of from 2.5×10^{-5}% to 10%. This corresponds to a neutron flux of 2.5×10^3 nv to 1×10^9 nv. Period indications are similar to those associated with channels I and II, including the 15-second hold and 10-second scram functions.

Channels V--VI--VII--Power Range Instrumentation. - Uncompensated ion chambers are used in each of the three power range channels. These chambers furnish a current signal to a linear amplifier to indicate reactor power over a range of from 1 to 150% of full power--which corresponds to a neutron flux of from 1×10^8 nv to 1.5×10^{10} nv. The three linear power signals are fed to individual trip circuits and to a coincidence circuit. The individual trip circuits operate annunciators when the indicated flux exceeds 120 to 130% of that associated with full power. When two of the three inputs to the coincidence circuit indicate flux levels in excess of 120 to 130% of full power, a scram is initiated.

Area Monitoring System. - An integrated area-monitoring system is provided throughout the PM-1 power plant. The radiation levels measured at each of the various stations will be recorded in the control room. Each of the monitoring channels has a preset alarm point such that the operator will be warned of any abnormal radiation levels. The table below presents the areas to be monitored and the types of contaminants to be monitored.

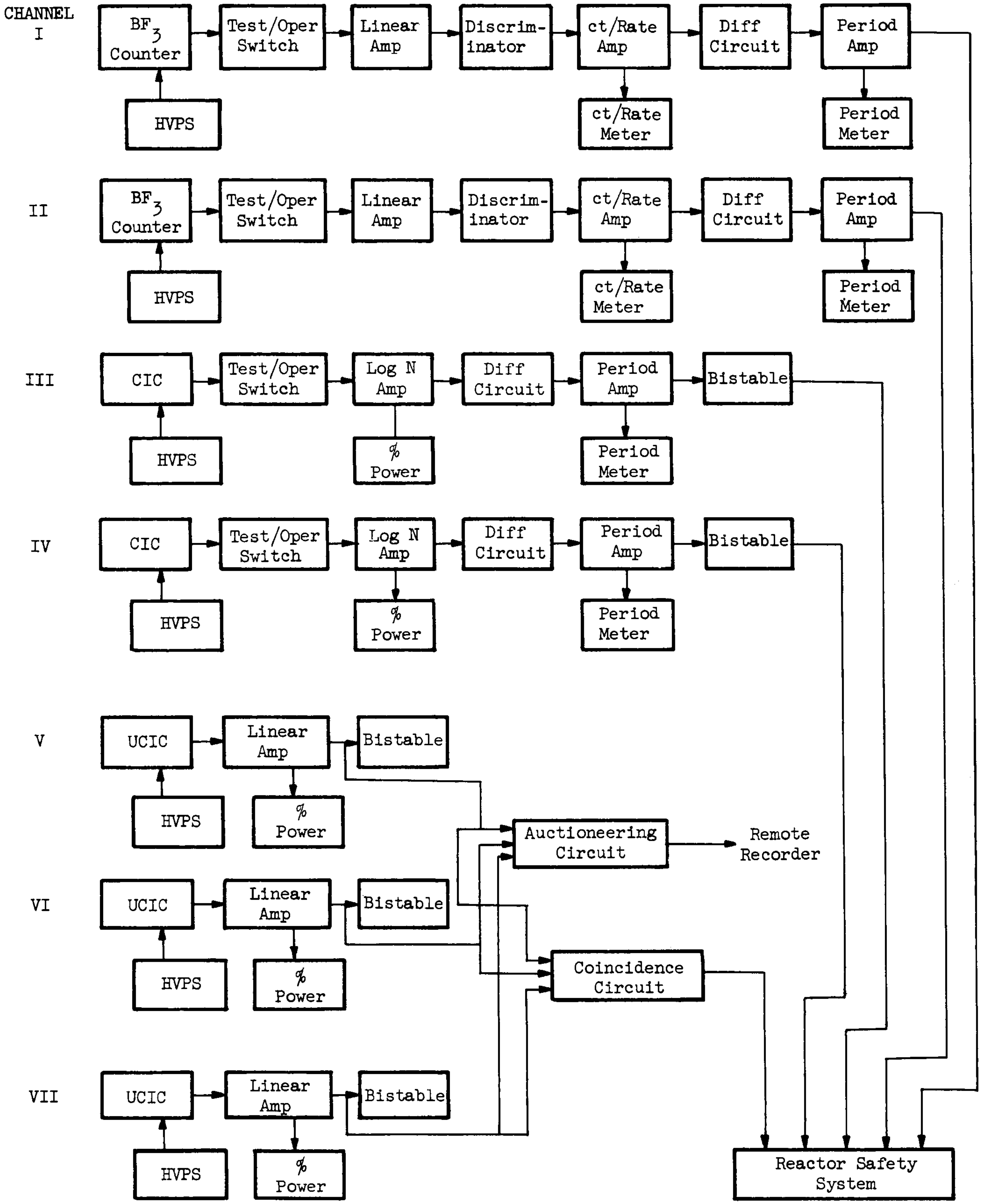

Fig. III-9. Nuclear Instrumentation Block Diagram

Location	Contaminants
Primary System Building	Gamma Radiation, Air Monitor, Independent Evacuation Alarm
Decontamination Building	Gamma Radiation, Air Monitor
Control Room	Gamma Radiation, Air Monitor
Primary System Effluents	Beta and Gamma Monitors

2. Reactor Safety System

(1) The reactor safety system utilizes signals from the nuclear and primary plant instrumentation systems to provide protection for the power plant against abnormal or unsafe operating conditions. The system employs a scram mixer which receives the following signals:

(1) High neutron flux

(2) High primary system coolant temperature

(3) Low primary system coolant pressure

(4) Low primary system coolant flow

(5) Fast reactor period during start-up.

Any one of the above signals will cause a scram by interrupting the power to the control rods. A manual scram capability is also provided at the console.

During start-up, the low-pressure scram signal is by-passed.

(2) Table III-1 lists the abnormal system conditions which result in alarms or scrams. When both an alarm and a scram are indicated, the alarm set-point precedes the scram set-point by a margin sufficient to enable the operator to take corrective measures before shut-down occurs.

3. Primary Plant Instrumentation

Basic plant parameters and operating conditions required for safe and efficient operation of the plant are measured by the primary plant instrumentation. The more important measured parameters and their functions are described:

Reactor Inlet Temperature. - A resistance temperature detector is used to measure the temperature of the primary coolant entering the reactor over a range of 50° F to 550° F. The signal is also combined with the reactor outlet temperature signal to obtain a reactor average temperature.

TABLE III-1

Operating Conditions Resulting in Reactor Scram or Alarm

Item	Operating Conditions	Alarm	Scram
1	Fast Period--Channel I	x	
2	Fast Period--Channel II	x	
3	Fast Period--Channel I and II	x	x
4	Fast Period--Channel III	x	
5	Fast Period--Channel IV	x	
6	Fast Period--Channel III and IV	x	x
7	High Neutron Flux--Channel V	x	
8	High Neutron Flux--Channel VI	x	
9	High Neutron Flux--Channel VII	x	
10	High Neutron Flux--any 2 out of 3 (7-8-9)	x	x
11	Activation of Manual Scram		x
12	Reactor outlet temperature (high)	x	x
13	Reactor Coolant Flow (low)	x	x
14	Pressurizer pressure (high)	x	
15	Pressurizer pressure (low)	x	x
16	Pressurizer level (high)	x	
17	Pressurizer level (low)	x	
18	Primary coolant pump cooling temperature (high)	x	
19	Shield water tank level (low)	x	

Reactor Outlet Temperature. - The primary coolant temperature upon leaving the reactor is measured in the same manner as that of the coolant entering the reactor. The signal from the detector indicates a range of from 50° F to 550° F. Both a high-temperature alarm and high-temperature scram are provided.

Loop Flow. - The primary coolant flow is measured by taking the differential pressure across a venturi downstream from the reactor. This signal is fed to a square-root extractor and amplifier and is used to indicate low-flow alarm and low-flow scram.

Pressurizer Pressure. - A pressure transmitter, mounted at the pressurizer, measures the system pressure over a range of 0 to 2000 psig. The amplified signal is used for pressure indication, pressurizer heater operation, spray valve actuation, low-pressure alarm, high-pressure alarm, and low-pressure scram.

Pressurizer Level. - A differential pressure amplifier is used to detect water level in the pressurizer. The amplifier output signal is used for indication, low-level alarm, high-level alarm, and heater cut-off protection.

4. Secondary System Plant Instrumentation

The secondary system controls and instrumentation system are essentially the standard system for a small steam power plant. The use of an all-electric system is being specified.

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IV. SHIELDING

A. GENERAL

A conservative approach has been taken throughout the design of the PM-1 nuclear radiation shields. In all cases, adequate shielding is provided to maintain the radiation doses received by plant personnel well below the maximum permissible doses specified in the Federal Regulation Title 10, Part 20. Special Work Permit control will be required when personnel work in certain areas, such as the primary system buildings. This control will be directly governed by the plant Health Physicist.

B. SHIELDING DURING NORMAL OPERATION

The major features of the shielding of the PM-1 are illustrated in Fig. IV-1. The shielding of the reactor and primary system is by means of an earth mound. The earth mound is composed of local aggregate from the Warren Peak area. The shielding aggregate will be selected to insure that it does not contain trace elements which will be strongly activated. The earth mound will extend to a minimum height of 22 feet above the base of the primary system packages. Shielding water within the reactor package will cover the reactor to a height of 13 feet above the reactor pressure vessel head. The vertical steam generator, in the steam generator package, provides self-shielding over the tubes containing the radiating primary coolant water, in the form of the secondary water and the steel head. Local shielding for the primary piping in the steam generator package will be provided where necessary.

Dose rates in the shelter area above the primary packages will not exceed 350 milliroentgens per hour during full-power operation. The maximum dose exists at the shelter floor level on the reactor vessel axis. A summary of total dose rates from neutrons and gammas during full-power operation at various points of interest depicted in Fig. IV-1 is given in Table IV-1. For this analysis the earth shield was assumed to have a density of 100 pounds per cubic foot. This compares very well with actual backfill material available near Warren Peak.

Significant radiation sources, during operation, are the reactor vessel, internal components, and the primary loop coolant. Primary coolant intrinsic activity was also included. Buildup factors used in computing gamma dose rates were chosen conservatively: i.e., whenever shields of two or more materials were encountered, the buildup factor for the

TABLE IV-1

Full Power Operation Dose Rates

Dose Point[a]	Coordinates[b] X (Feet)	Coordinates[b] Z (Feet)	Dose Rate (Milliroentgens per hour) Steam Generator and Primary Piping	Dose Rate (Milliroentgens per hour) Reactor Vessel	Dose Rate (Milliroentgens per hour) Total
1	-16.35	21.65	200		200
2	-16.35	24.45	140		140
3	-16.35	32.2	26		26
4	-10.1 to -18.75	32.2	26		26
5	-13.0	21.65	70		70
6	-18.75	16.4	1.8×10^3		1.8×10^3
7	-24.75	12.25	2		2
8	-10.1	21.65	130		132
9	- 1.8	23.7	150	174	324
10	0	18.35		300	300
11	2.65	16.1		1.23×10^3	1.23×10^3
12	- 4.7	32.2	30		30
13	- 6.1 to 2.55	32.2		90	90
14	0	32.2		90	90
15	-17.6	9.5	2.2×10^4		2.2×10^4
16	5.6	0		5.0×10^6	5.0×10^6
17	2.5	0		2.5×10^9	2.5×10^9
18	5.65	32.2		305	305

(a) Dose points are as shown on Fig. IV-1
(b) Coordinates of dose points are relative to the center of the core

material which, when applied over the total number of mean free paths from source to dose point, would give the highest value of the dose rate was chosen.

C. AFTER-SHUTDOWN SHIELDING

After reactor shutdown, shielding for personnel in the vicinity of the steam generator is provided by the primary shield water in the reactor package and the earth between the packages. For steam generator maintenance operations, approximately eight hours is required for primary system cooldown and drainage. Maximum dose rates in the steam generator package eight hours after shutdown will be less than 10 milliroentgens per hour. Table IV-2 lists dose rates at specific points of interest 1.67 minutes and 8 hours after reactor shutdown. Primary coolant will be drained into the waste storage tank during steam generator maintenance, thus eliminating circulating corrosion-product activity as a source of radiation to maintenance personnel. Extended operation of other pressurized water reactors has resulted in a buildup of corrosion-product activity on the surfaces exposed to the primary coolant. This buildup is expected to occur in the PM-1 primary coolant system. It is not expected that deposited activity in the steam generator and primary piping will be a major problem during maintenance operations. However, after extended periods of reactor operation, it may be desirable to decontaminate the primary system prior to performing maintenance operations.

TABLE IV-2

After Shutdown Dose Rates - Uncontained Version

Dose Point[a]	Coordinates[b] X (Feet)	Z (Feet)	Dose Rate (Milliroentgens per hour) 1.67 minutes After Shutdown	8 hours after Shutdown
19	-10	1.7	187	6
20	-10	0	3.5	0.1

(a) Dose Points are as shown in Fig. IV-1
(b) Coordinates of Dose Points are relative to the center of the core.

After shutdown, dose rates at the surface of the shield water in the reactor package will not exceed 2 milliroentgens per hour when all primary system components are in their respective normal operating positions.

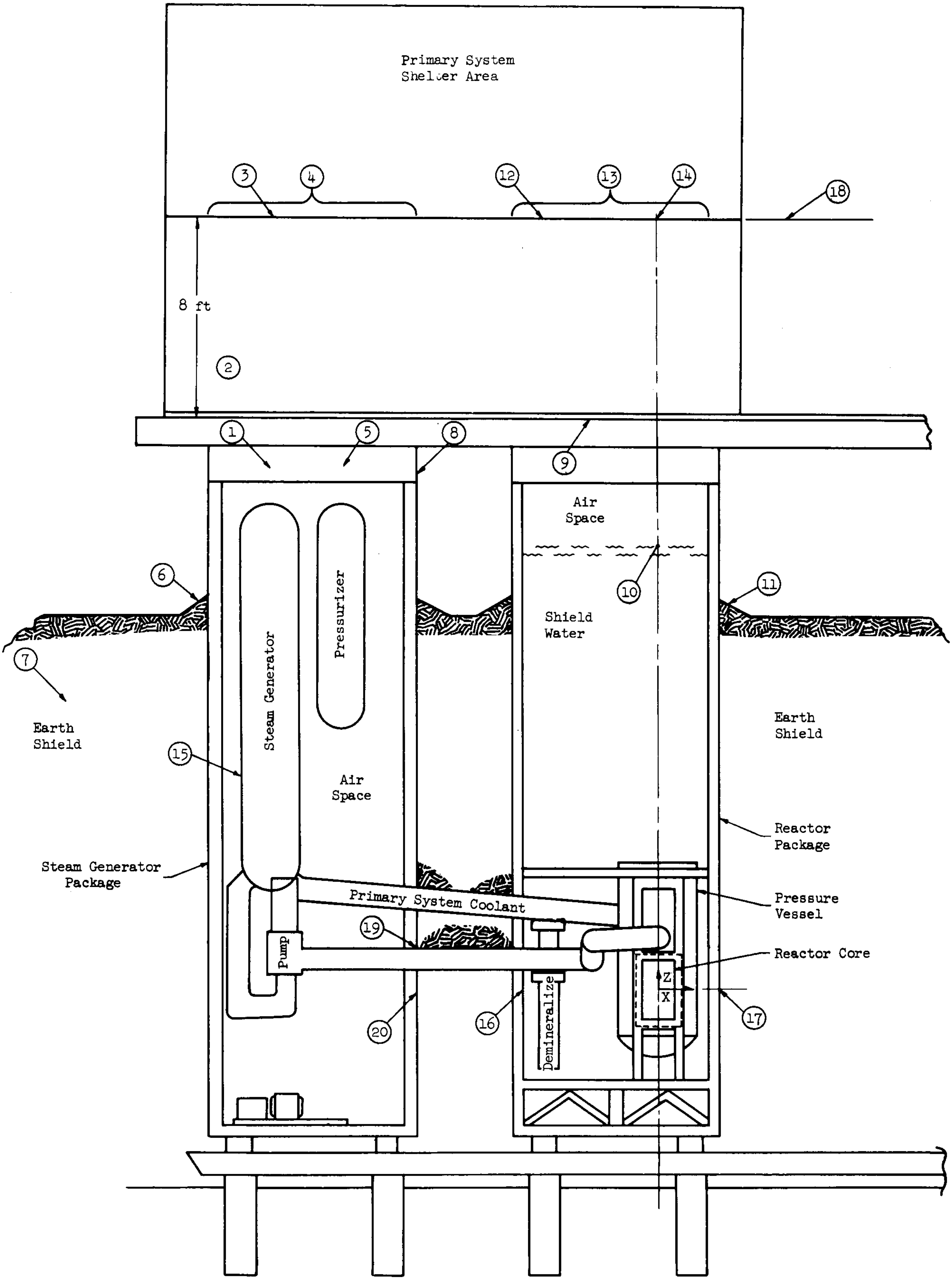

Fig. IV-1. Primary System Arrangement

Normal maintenance operations and refueling in the reactor package will be performed remotely from the surface of the shield water. Local lead shielding may be necessary for spent core and activated demineralizer removal as discussed in the following sections of this report.

D. SPENT FUEL REMOVAL, STORAGE AND SHIPMENT

Spent fuel removal procedures will involve remote handling of the reactor core within the shield water approximately eight hours after reactor shutdown. After removal of the reactor vessel head, the dose rate from fission product activity at the surface of the shield water on the core axis will be less than 2 milliroentgens per hour. Dose rates at this point represent a maximum dose rate that personnel in the operations area above the shield water would receive. Unloading the spent core and placing it into a storage cask will necessitate raising it from the reactor vessel to a height such that the top of the active portion of the core is within 65 inches of the shield water surface. For loading of core bundles into the storage casks, the core will first be raised from its operating position to a rack above the reactor vessel. The relative position of the reactor vessel, storage cask and various core and bundle positions are shown in Fig. IV-2.

Dose rates on the axis of the core and fuel bundle have been computed for the core and bundle as a function of water thickness above the source at eight hours after shutdown. From Fig. IV-3, dose rates at the surface of the shield water for core and bundle in various positions may be determined. A bundle consists of approximately one-sixth of the entire core. The core may be broken down into six equal bundles, each containing a control rod and consisting of a sixty-degree sector of the core, and a smaller central bundle of fuel tubes. In the use of Fig. IV-3 to determine dose rates at the surface of the shield water, the dose rates from the core and from a bundle removed to another position must be summed to determine total dose rate. The assumption that the core with bundle removed is still a full core will give a slightly more conservative estimate of the dose rate of the surface of the water. Other activated components of the primary system in the reactor package, such as the pressure vessel, thermal shield, primary piping and high-pressure demineralizer will not contribute significantly to the shield water surface dose rate 8 hours after shutdown. A summary of the shield water surface dose rates for various core and bundle positions is given in Table IV-3.

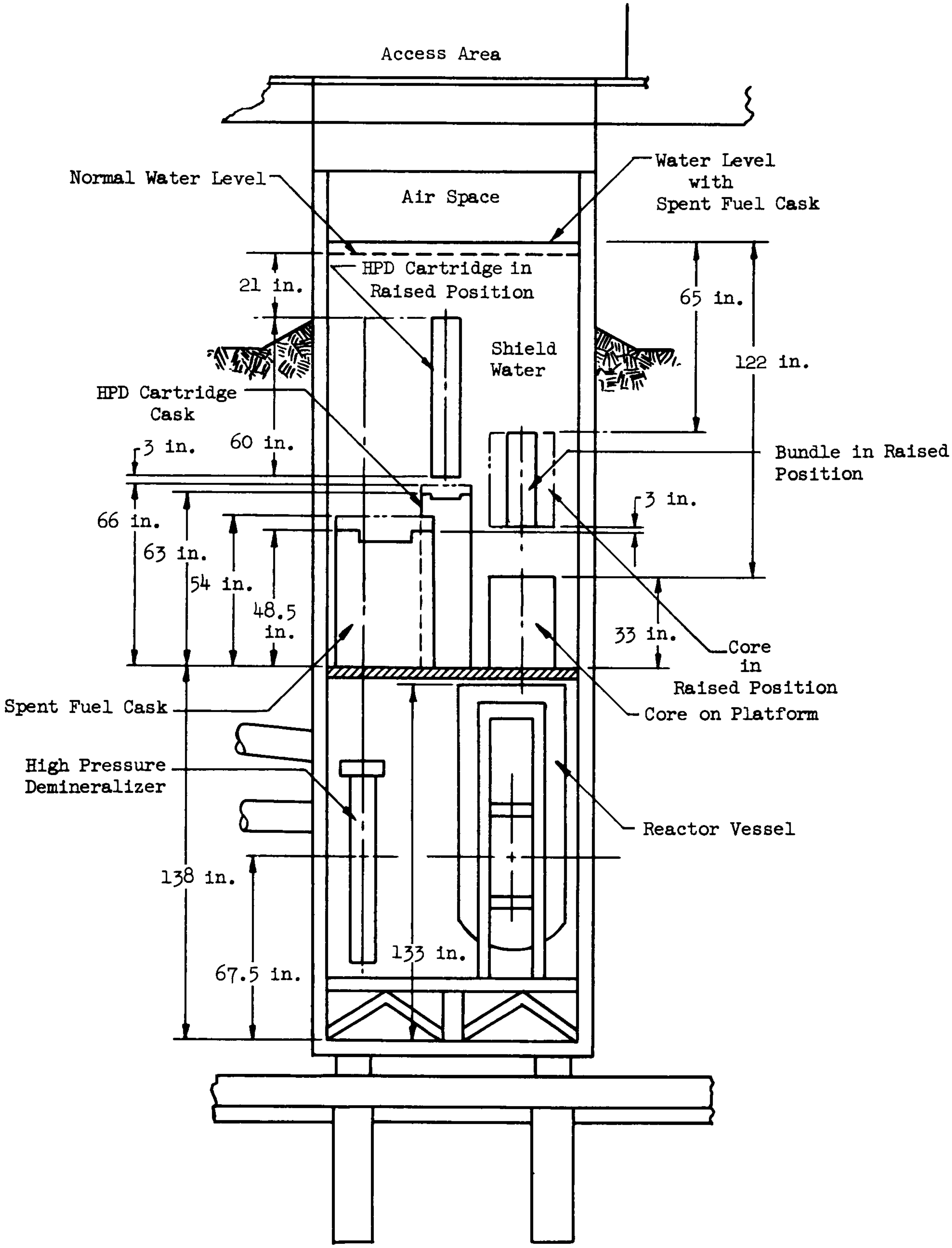

Fig. IV-2. Reactor Package Showing Spent Fuel Cask

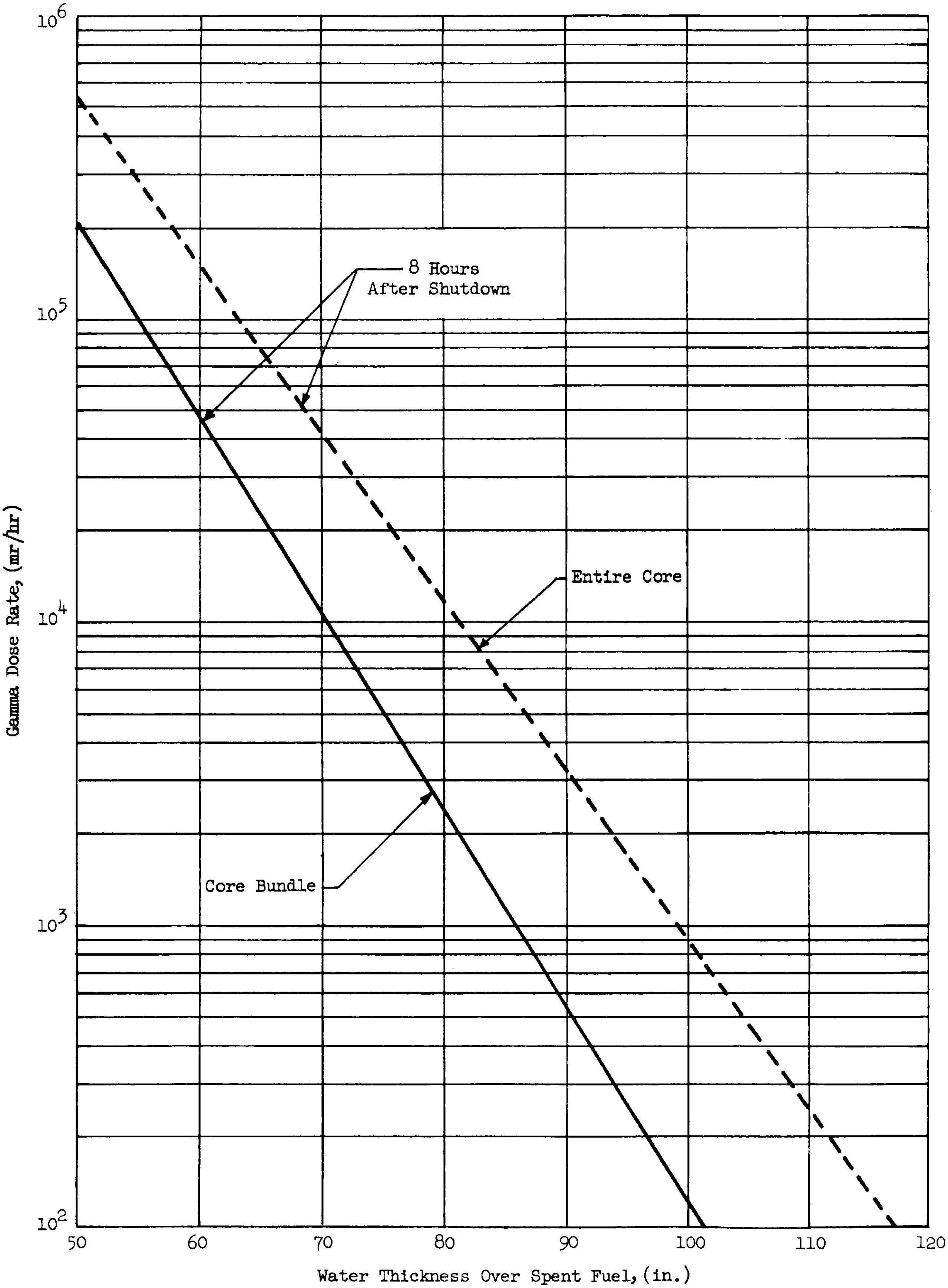

Fig. IV-3. Dose Rate at Surface of Shield Water for Various Positions of Core and Bundle

TABLE IV-3

Maximum Dose Rates at the Surface of the Shield Water Eight Hours After Shutdown

Configuration	Dose Rate (Milliroentgens/hour)
1. Core in operating position, reactor vessel head removed	$\ll 2$
2. Core on rack above reactor vessel	56
3. Entire core raised to highest position during cask loading operation	8×10^4
4. Core on rack above reactor vessel, bundle raised to highest position during cask loading	2.2×10^4
5. Core on rack, bundle in uncapped storage cask	56
6. Entire core in uncapped storage cask	200

Excessive dose rates will occur in the personnel operating area when raising the core or bundle to the height required for loading into storage casks. An estimated five inches of lead over the full core in the raised position will decrease the shield water surface dose rate from 8×10^4 to 100 milliroentgens per hour. The corresponding lead thickness for a raised bundle is four inches. An alternate approach would be to raise the shield water level during these operations.

Lead casks are designed for shipment of spent fuel after a 90-day cooling period. Two cask designs are presented, one for the entire core (cask A) and one for a fuel bundle (cask B). Both casks conform with ICC shipping regulations which require:

(1) Radiation not greater than 200 milliroentgens per hour at any place on the outside of the carrier.

(2) Radiation not greater than 10 milliroentgens per hour at one meter distance from the carrier.

(3) Radiation not greater than 11-1/2 milliroentgens for any 24-hour period at a 15-foot distance from the carrier.

TABLE IV-4

Dimensions and Dose Rates on
Spent Fuel Shipping and Storage Casks

	Overall Diameter Inches	Overall Height Inches	Time After Shutdown	Lead Thickness Inches	Dose Rates Surface Mr/hr	1 Meter Mr/hr	5 Meters Mr/hr	10 Meters Mr/hr
Cask A	A=43.2	B=54.2	8 hours	T = 9.1	2500	314	24.5	6.8
Entire Core			90 days	9.1	65	8.2	0.6	0.2
Cask B			8 hours	8.75	2350	183	12.5	3.4
Core Bundle	33.0	53.5	90 days	8.75	65	5.2	0.3	0.1

Preliminary designs of core and bundle casks are shown in Figs. IV-4 and IV-5. Maximum surface dose rates and lead thickness for these casks are listed in Table IV-4. Lead cask thicknesses were obtained from computations of dose rates in a cylindrical lead shield around the core and bundle. Dose rates through the lead from the dry core and bundle for 8 hours and 90 days after shutdown are shown in Fig. IV-6.

For computation of dose rates, fission products were assumed uniformly distributed throughout the core. In reality, a higher concentration of fission products near the center of the core is predicted by non-uniform burnout studies. Thus, computed dose rates through lead and water are somewhat conservative.

For attenuation calculations, the IBM-704 cylindrical volume source code was used. This code computes gamma flux from cylindrical or cylindrical annulus sources through concentric cylindrical shield regions by integrating over the volume of the source. A single buildup factor in exponential representation is applied along the total number of mean free paths from source to dose point. For computation of dose rates along the axis of the core in water, the dose buildup factor for a point isotopic sourcein water was used. For dose rates in lead, the corresponding lead buildup factor was used.

Gamma and beta energy release from fission products will produce significant heating. After extended core operation, approximately five per cent of the total operating power is from fission-product decay. Table IV-5 presents beta, gamma, and total energy release from fission-product decay at various times after shutdown for the PM-1 core after two years' full power operation. Because of the relatively short range of beta particles, essentially all of the beta energy will be released within the fuel elements. Gamma energy release will be in the core and surrounding absorbing media. When the fuel is placed in any of the storage and shipping casks described above, essentially all of the gamma energy will be absorbed by the fuel elements and cask. Adequate cooling of these casks will be provided through an air or liquid heat removal system.

E. HIGH PRESSURE DEMINERALIZER CARTRIDGE SHIELDING

Buildup of circulating corrosion product and intrinsic coolant activity within the high-pressure demineralizer resin over a one-year period of reactor operation at full power necessitates the use of a removal and storage cask for the high-pressure demineralizer resin cartridge. A preliminary design of a lead cask has been based on computed resin activity resulting from the removal of primary loop circulating activity.

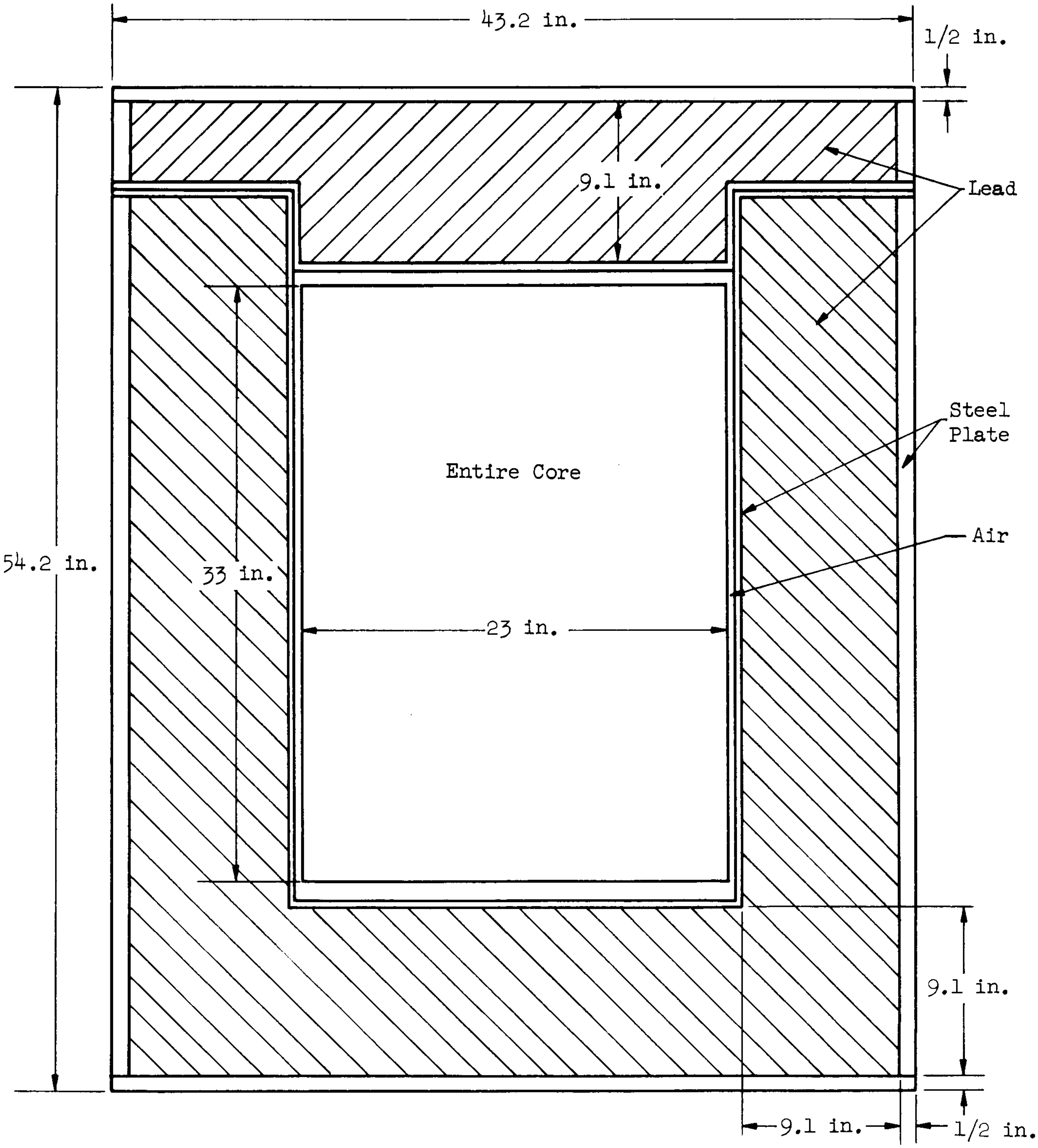

Fig. IV-4. Spent Core Shipping Cask

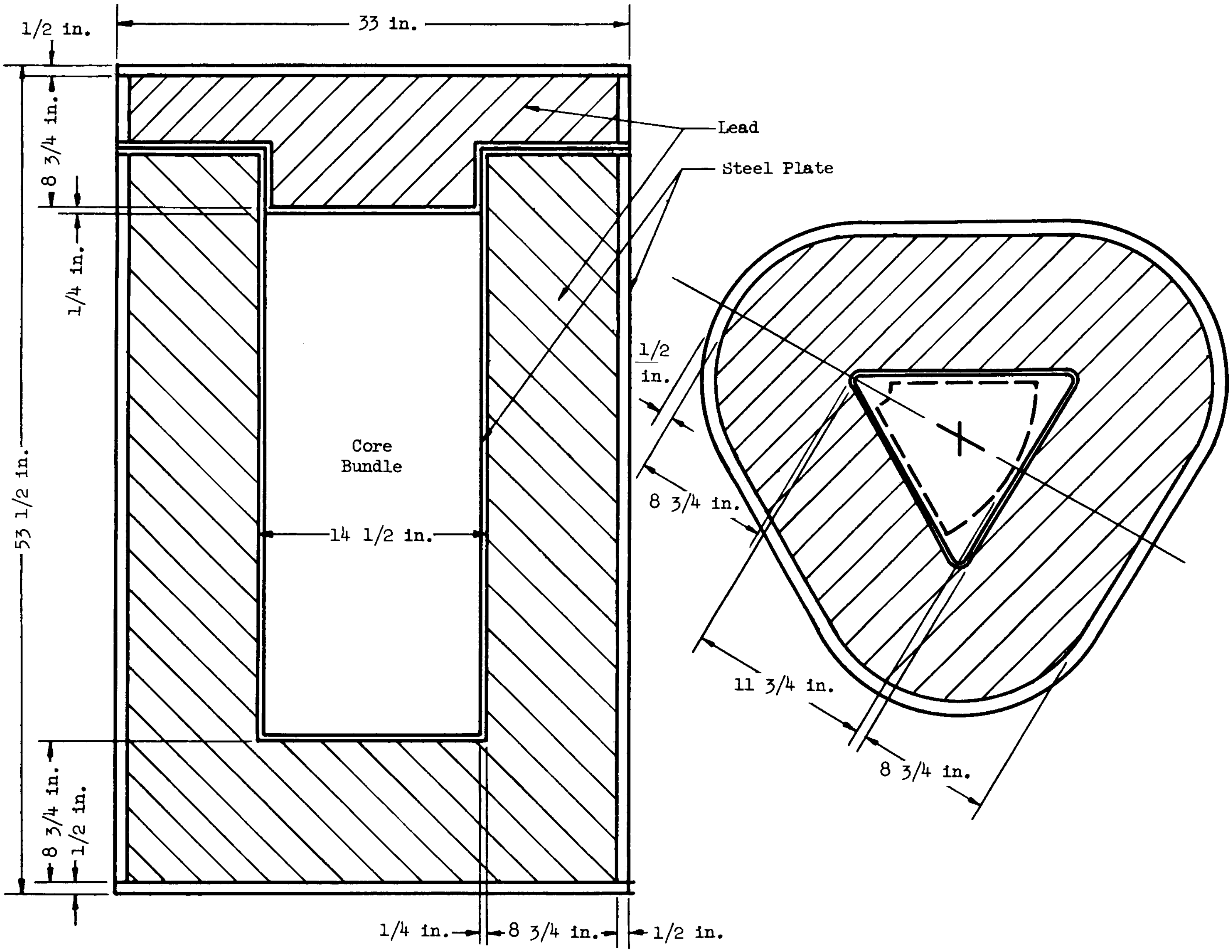

Fig. IV-5. Spent Fuel Bundle Shipping Cask

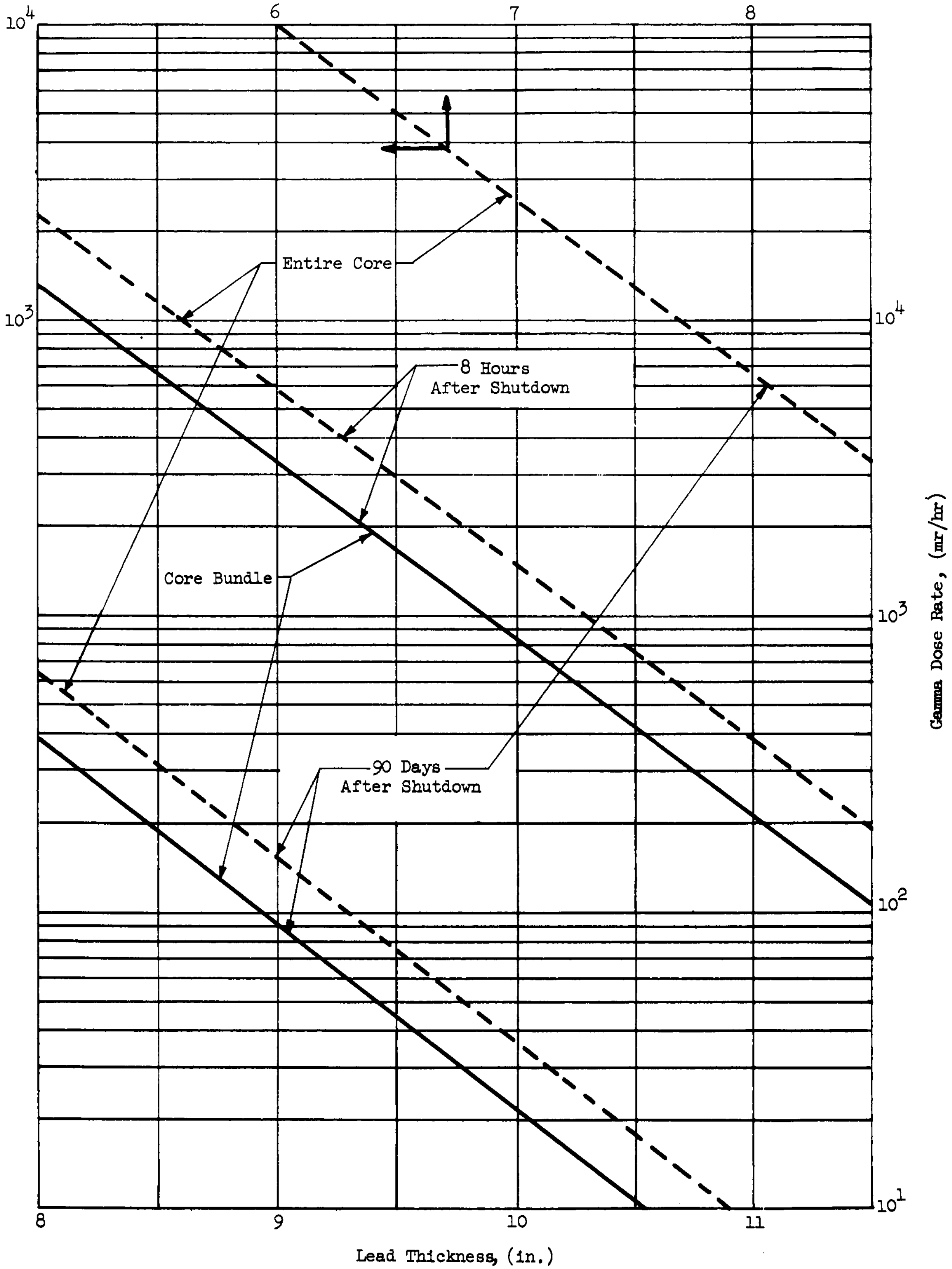

Fig. IV-6. Radial Dose Rates in Lead from Dry Spent Core and Bundle

The resin cartridge removal procedure will involve remote handling of the demineralizer cartridge under the shield water after an 8-hour cooling period. After removal of the demineralizer head, the dose rate from the cartridge activity at the surface of the shield water on its axis will be less than 2.0 milliroentgens per hour. Loading of the cartridge into a cask will necessitate raising the cartridge from the demineralizer container to a height such that the top of the cartridge is within approximately one foot of the shield water surface. The corresponding dose rate at the shield water surface with the cartridge in its highest position is 4.8×10^3 milliroentgens per hour, eight hours after shutdown. The relative positions of the reactor vessel cartridge cask, demineralizer and various cartridge positions are shown in Fig. IV-2. During the unloading operation, the above dose rates are the highest received by maintenance personnel on the operating platform. Other sources of radiation are comparatively insignificant during this operation. An estimated 3.0 inches of lead over the cartridge, in the form of a cap, will decrease the shield water surface dose rate from 4.8×10^3 to less than 100 milliroentgens per hour, or the level of the shield water can be raised to provide adequate shielding.

A lead shipping cask (Fig. IV-7) was designed to facilitate both removal, after an eight hour cooling period, and shipment, after 90 days of decay. The cask design will conform with the ICC shipping regulations. Lead cask thickness was obtained from computations of dose rates in a cylindrical lead shield around the cartridge. Dose rates through the lead from the active cartridge for saturation, eight hours, and 90 days after removal are shown in Fig. IV-8. For computation of dose rates, the activity from corrosion products and other impurities except fission products were assumed uniformly distributed in the upper eight-tenths of the demineralizer cartridge volume. The steel shell of the cask is assumed to have a half-value-thickness shielding value.

F. RADIATION LEVELS AT THE CONTROL ROOM

Radiation levels have been evaluated at the proposed control room location, which is at a horizontal distance of approximately 42 feet from the axis of the reactor vessel and at a finished floor level 5 feet above the floor level of the housing over the reactor packages. This configuration is shown in Fig. I-1. At ten megawatts (thermal) operation, the dose rate at the nearest point of the control room to the reactor package will be insignificant (less than 2×10^{-3} milliroentgens per hour).

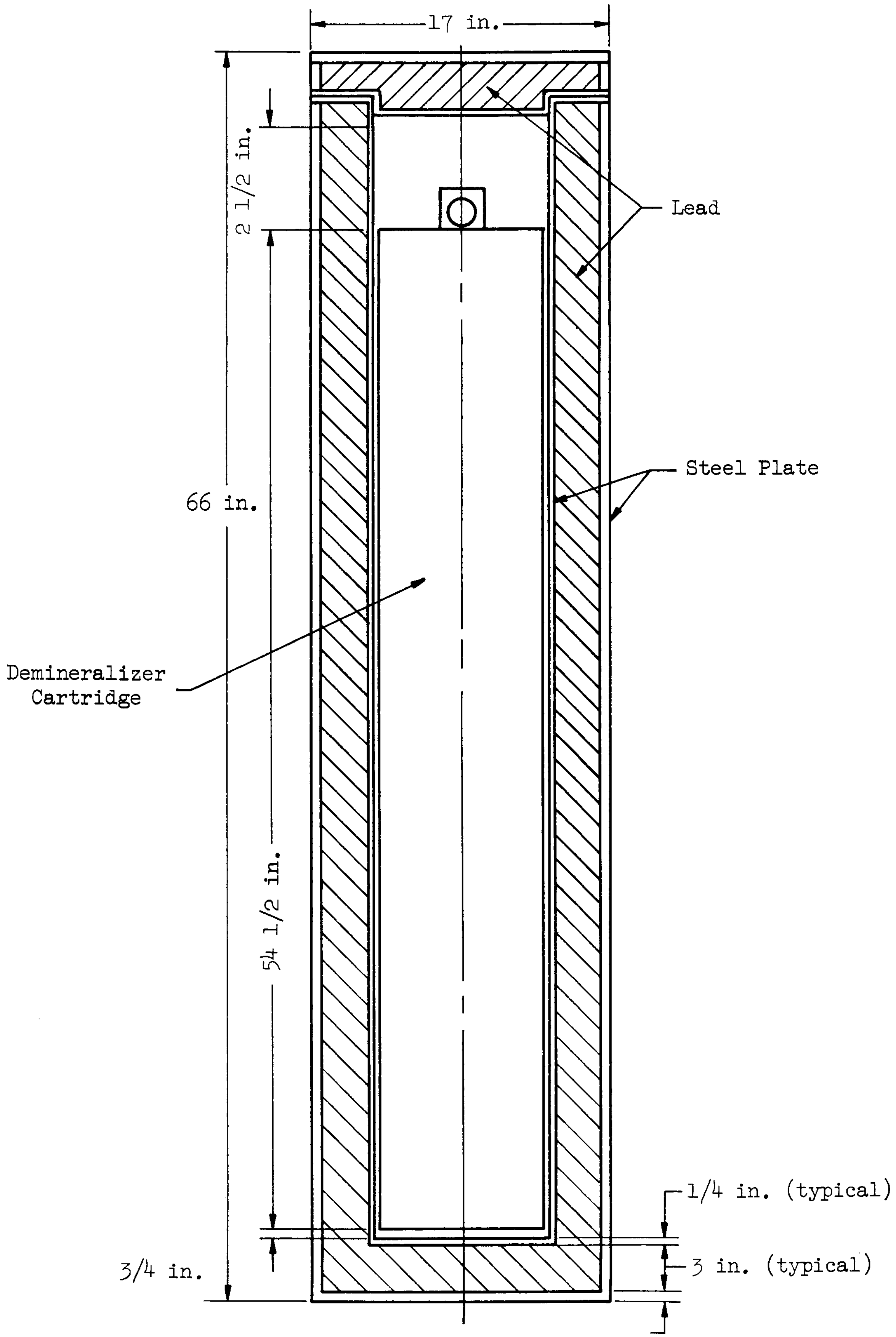

Fig. IV-7. Lead Cask for Expended Demineralizer Cartridge

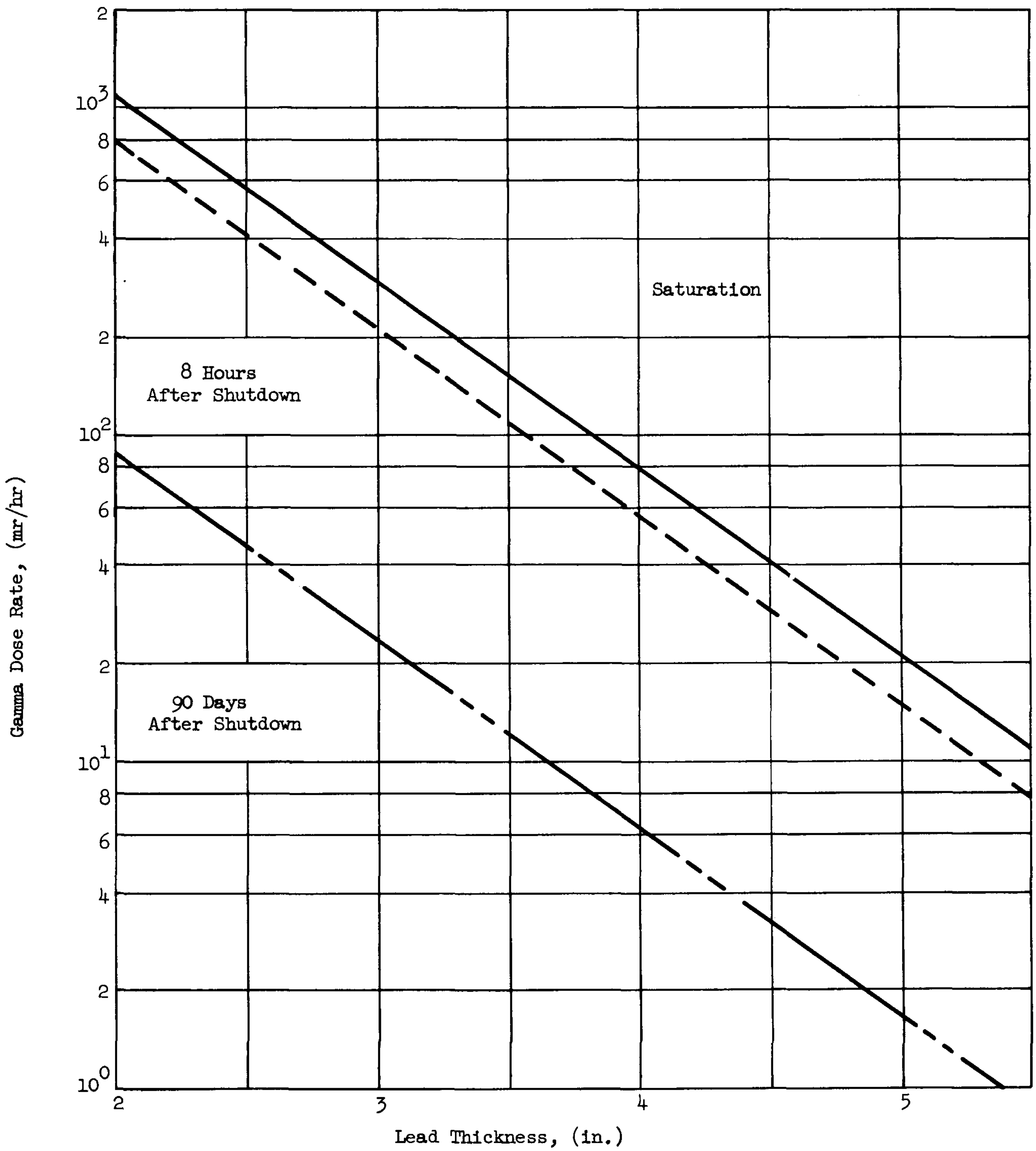

Fig. IV-8. Radial Dose Rates Through Lead from Activated Demineralizer Cartridge

V. PM-1 NUCLEAR POWER PLANT STANDARD OPERATING PROCEDURES

A. BASIC OPERATING CRITERIA

The possibility of operator error causing an accident has been minimized by establishing the following basic operating criteria:

(1) All operating, test, and emergency procedures will have been prepared in detail and reviewed in advance for correctness and plant safety.

(2) Before the PM-1 plant will be placed into normal service, an extensive test program will be conducted, including pre-start-up testing to check the installation of material and equipment, initial critical testing, low- and rated-power operation, and a six-month performance testing period to determine the adequacy of the approved operating procedures and to check the operation of systems and equipment.

(3) All operating personnel will have received an extensive training program and each individual will have to qualify for his assigned responsibility.

(4) Start-up, normal shutdown, and other repetitive operations will be performed in accordance with specific check lists.

(5) The control room will be manned at all times by a qualified operator.

(6) Operation and control of the reactor and most of the process equipment will be centralized in the control room.

(7) While most of the operating and control functions are initiated from the control room, operators may perform some functions in the equipment area at the direction of the control room supervisor or with his prior knowledge.

(8) Maintenance will be carried out in accordance with specific procedures and in accordance with established, safe practice.

(9) Check-out tests or routine maintenance of protective devices and critical operating equipment will be done in accordance with prescribed schedules.

(10) Radiation monitoring by fixed and portable instrumentation will be provided for all radiation zones.

(11) Personnel leaving a contaminated radiation zone and equipment being removed from such zones will be appropriately surveyed by a Health Physicist.

(12) Irradiated fuel is to be changed by semi-remote methods under water by means of grapples and an operator-controlled hoist.

(13) All unexpected incidents, unsafe acts, or incidents of excessive exposure to radiation will be thoroughly investigated and reported, in order to effect procedures to prevent recurrence.

(14) The PM-1 plant is so protected by automatic safety devices that no single operator error or reasonably conceivable combination of operator errors could cause a severe accident.

B. INITIAL TEST PROGRAM

The initial test program will be divided into four test phases. These are:

1. Pre-start-up Testing

It is the objective of this program to demonstrate that the plant has been built according to specifications and that it is ready for core loading and initial start-up. This program will cover all tests during the construction period prior to plant operation.

The program will include checking, adjusting, and the operation necessary to assure that initial loading and subsequent operation can be safely undertaken. Each system is to be tested functionally with special emphasis on systems important to the safety and operability of the plant. The system tests will include, as appropriate, hydrostatic and leak tests, cleanliness checks, installation checks, response and sensitivity measurements, wiring checks, equipment turnover, and vibration checks.

The following is a summary of the pre-start-up tests:

Checkout of all process systems

(1) Inspect and test all field connections.

(2) Hydrostatic tests of the primary and secondary systems.

(3) All plant containers will be checked for integrity. They will be pneumatically tested to 5 to 10 psig with a mixture of Freon and air. Freon detectors will be used for a general survey and a soap-bubble test will be used to determine the exact location of any leaks.

(4) Flush the primary and secondary systems.

(5) The various systems will be charged with their respective working fluids.

(6) The rotating equipment and plant components will be operated individually.

(7) The various fluid systems will undergo a performance test by the preliminary operation of all systems at pressure and temperature. The systems will be checked for leaks. As each fluid system is put into service, the associated alarms, controls, relief settings, and interlocks will be checked. All indicating lights and annunciator system will be checked.

(8) The various gaseous systems will be checked to assure satisfactory operating of the system and to check the performance of the individual components. Initially a pneumatic test will be made. Operation of all valves, controls, instruments and protective devices will be checked.

Checkout of nuclear instrumentation

Simulated signals will be introducted to the reactor nuclear instrumentation to determine that the system and individual components perform satisfactorily. All pulse channels will be checked with a pulse generator after installation. The ionization chamber channels will be checked with a current source. Alarm and trip signal channels will be set at designated levels. The detectors will be connected and the overall system will be checked for operation with a portable source.

The radiation-monitoring system will be checked for satisfactory operation of all channels, including the operation of all alarms and trip signals. The permanently-installed channels in the plant will be checked after installation with a portable source.

To prove the satisfactory operation of the reactor shutdown system before operation with a core, all the input signals causing a scram will be individually initiated or simulated and the system will be checked for proper operation.

To insure that the primary system is in a condition of cleanliness suitable for critical operation, the primary system filter will be removed, and any foreign material collected will be extracted.

2. Initial Critical Testing

The initial critical tests will provide information required for proper operation of the reactor core and control systems. During the initial critical test program, the reactor power developed will be restricted to low power not requiring the secondary plant for heat dissipation.

The initial criticality tests will consist of operational tests of the control rod mechanisms, initial calibration of nuclear instrumentation, complete check-out of the reactor control console, approach to criticality, calibration of control rod worth determination of reactivity coefficients, and repeated approaches to criticality and manual low-power operation. The results of these tests will be compared with the predicted results. The objective of these tests is to insure safe operation of the reactor core and to provide data necessary for initial evaluation of the core performance.

The program will be sequenced as follows:

(1) After the fuel is loaded in the pressure vessel, the charging pump will be used to fill the reactor slowly. The neutron level will be continually observed during the operation to avoid any possible nuclear hazard.

(2) Operational tests will be made of the control rod mechanisms. The scram circuits will be tested and the scram times determined to assure proper operation of the control rod drive mechanisms, scram reliability and proper latching of the mechanisms after installation and prior to initial plant start-up.

(3) With the reactor filled with water at atmospheric pressure and ambient temperature, individual control rods shall be withdrawn slowly and cautiously as a 1/m curve is plotted as a function of control rod position. Criticality will be accomplished safely, and the control rod configuration for cold criticality will be determined. A calibration of the control rod worth will be made as well as the determination of the various reactivity coefficients.

Detailed procedures will be written prior to the start of any testing.

3. Low- and Rated-Power Operation

The plant will be operated initially at low power integrating in the secondary system. The reactor power will be increased in steps as the turbine generator is loaded down to full power. The intent of this sequence of tests is to demonstrate the ability of the overall plant to meet the operational design specifications. In addition, an inter-calibration of the nuclear instrumentation will be made to check the overlap of the source, intermediate and power range of the instruments. Several radiation surveys will be made to assure the adequacy of the shielding design.

4. Six-Month Performance Test

A six-month testing program will be undertaken at the completion of the low and rated power tests. This period will also serve as a final training and check-out of customer operating personnel. These tests will investigate both steady-state and normal powerplant transient operation to provide performance data. Final operating procedures and maintenance programs will be formulated. All equipment and systems will be tested to assure that they meet the intended design requirements. The following are typical of the type tests that will be conducted.

(1) Plant start-up tests to determine operating characteristics under varying start-up conditions to compare with specified design requirements.

(2) Transient tests to determine the ability of the plant to handle various rate of change of load demand.

(3) Steady-state tests to determine that the plan can meet full design power requirements.

(4) Five hundred-hour continuous operation to demonstrate the reliability of continuous plant operation.

(5) Shutdown tests to determine operating characteristics under varying shutdown conditions to compare with design requirements.

(6) Performance tests on the various plant components to assure satisfactory operation, to determine heat rate and heat balance, to determine the ability to maintain design conditions.

(7) Periodic control rod calibrations and control rod position for criticality to determine the rod configuration for criticality at cold and hot operating conditions over the lifetime of the core.

(8) Pressure, temperature and flow coefficient measurements of reactivity.

(9) Xenon transient tests to determine the reactivity change as a function of time due to the build-up and decay of xenon after shutdown following extended operation at full power. Also, determine the ability of the reactor to override the xenon transient at various stages of core burn up.

C. NORMAL PLANT START-UP AND POWER OPERATION

The normal plant start-up and power operational procedures will be finalized by the experience attended during the six month performance testing. By the completion of the six month test program the operating characteristics of the plant, and the plant capabilities and limitations will be thoroughly understood.

The following are the basic steps of plant start-up. Detailed start-up and operating procedures will be established before the initial start-up.

(1) To assure the plant safeguard features are functioning properly, and that all equipment is in proper condition for operation.

  (1) Complete the instrumentation check list to ascertain that all safety circuitry and interlocks are operable and calibrated.

  (2) Complete the valve check-off list to assure the various systems are aligned properly.

  (3) Check that adequate make-up and reserve feed water is available.

  (4) Clear the reactor area of all personnel.

(2) Start the primary system pump.

(3) Establish system over-pressure with the pressurizer.

(4) With greater than 2 counts/sec. on the start-up counters start withdrawal of the control rods until criticality is established. A continuous check is to be made of the multiplication while the reactor is subcritical and of the reactor period after criticality is established.

(5) Increasing reactor power will not be permissible if there is not at least one decade overlap between the start-up counters and the LogN indication.

(6) To avoid developing thermal stresses the primary coolant temperature rise rate will be restricted.

(7) Maintain a reactor power level which will permit a controlled warmup of the primary system.

(8) Maintain a sufficient system overpressure with the pressurizer to prevent boiling at all stages of warmup.

(9) The reactor will be carefully brought to 10 - 15% rated power and operating steam pressure will be established in the secondary steam drum. The secondary steam lines will then be warmed for at least 30 minutes.

(10) The turbine-generator will be brought up to speed at no load and the turbine over speed trip checked.

(11) The turbine generator will be synchronized and loaded as the reactor is increased in power simultaneously.

(12) When the system electrical requirements are met steady state conditions will be established.

The main functions of the operating personnel during normal power operations are:

(1) Surveillance of plant equipment for proper functioning and making the necessary adjustments and repairs. These include:

  (1) Routine preventive and overhaul maintenance.

  (2) Process steam and water sampling.

  (3) Observing and recording information from the plant instrumentation.

  (4) Routine checks of control rod functioning.

  (5) Periodic tests of automatic, remote-controlled and safety valves.

(2) Manual control of the reactor power by manipulation of the control rods.

(3) Evaluation of abnormal conditions and taking of emergency action as required.

(4) Confining any radioactive contamination to the smallest possible area; and preventing contamination of personnel and environs above maximum permissible levels.

(5) Preparation and handling of irradiated fuel for off-site shipment.

D. PLANT SHUTDOWN

There are two types of plant shutdown, normal shutdown and emergency shutdown.

1. Normal Shutdown

A normal shutdown is considered one that is planned in advance and does not result in a thermal shock of the primary plant equipment. The normal plant shutdown is accomplished by the following steps.

(1) The turbine and reactor power are reduced and the generator is separated from the distribution system.

(2) The reactor coolant temperature is reduced at a controlled rate so as not to thermally stress primary system equipment.

(3) The final position of the control rods is "all in."

2. Emergency Shutdown

An emergency shutdown is usually considered to be a scram-initiated shutdown, whether the scram is caused by one of the safety signals or by manually pushing the scram button.

In all instances of emergency shutdown, a check is made to make certain that the expected sequence of events actually has occurred and that the plant has been safely shut down. Then action is directed toward locating and correcting the conditions that required the shutdown.

E. REFUELING

The basic principles by which the PM-1 reactor will be refueled are as follows:

(1) Refueling will be conducted through the top of the reactor after pressure vessel head removal.

(2) The insertion and removal of fuel will be done under water by means of grapples and hoist controlled by operators. The water provides both shielding and coolant.

(3) The refueling operation will be conducted with the control rods disconnected and in the core.

(4) The radiation exposure of the refueling operators will be controlled by handling the irradiated fuel under water, and by monitoring the movement of all irradiated fuel by portable instrumentation.

(5) Irradiated fuel will be stored in appropriate casks until sufficient cool down will allow shipment.

(6) Unirradiated fuel will be stored in air in the fuel vault.

(7) A detailed refueling procedure will be written and approved before refueling operations are initiated.

F. WASTE DISPOSAL OF SOLIDS, LIQUIDS AND GASES

The basic objective of the waste disposal system is the routine disposal of radioactive wastes that must not result in the exposure of personnel on or off the PM-1 plant premises in excess of the permissible limits. Sufficient conservatism is included in the design and operational procedures to assure that the average off-site exposures will be a significant factor below the permissible limits.

G. MAINTENANCE

Preventive maintenance will be conducted in the same way as is practiced in conventional plants with the exception of the radiation - contamination problem. The basic principles by which maintenance will be performed are:

(1) To minimize the possibility of error or damage to equipment, all maintanance will be done in accordance with a specific check list.

(2) Equipment and system maintenance records will be kept to facilitate scheduling and completion of all necessary maintenance. These records will include equipment history and overhaul reports.

(3) Radiation monitoring will be provided during the initial approach to the equipment and periodically during disassembly.

(4) The equipment will be decontaminated as necessary prior to the performance of maintenance.

(5) Personnel radiation exposure records and limitations will be adhered to in assigning time limitations on jobs in radiation fields. Checks on work area exposure rates will be made by portable and personnel monitoring devices. All such work will be done with the advanced cognizance and approval of both the health physics representative and the shift supervisor.

(6) Protective clothing will be worn and other measures observed as necessary to avoid spread of radioactive materials.

(7) When the possibility of radioactive airborne particles exists, fresh air breathing equipment will be used.

(8) All maintenance or repair to the internal parts of the reactor will be under the direct supervision of a qualified individual.

(9) No maintenance personnel will be allowed near the reactor during critical experiments or power increases.

(10) Maintenance work is permitted during normal steady state operation.

(11) No maintenance is to be performed without prior approval of the shift supervisor.

(12) Major overhaul periods requiring complete plant shutdown will be scheduled as required, commensurate with safe plant operation.

H. INVESTIGATION OF UNUSUAL OR UNEXPECTED INCIDENTS

Unusual or unexpected incidents that result in occurrence beyond the normal experience of operation involving either excessive radiation exposure of personnel, incidents resulting in serious injury, so called "close calls," or unsafe working practices will be promptly reported to supervision.

In order to prevent more adverse consequences following such incidents it will be standard procedure to:

(1) Make a thorough investigation of all such incidents.

(2) Report all such incidents in written detail.

(3) If the safety of any operation is questionable the operation will be suspended until the questionable portion is resolved.

(4) Establish and enforce rules or procedures necessary to prevent recurrence of the incident.

I. PROTECTION AGAINST OPERATIONAL ERRORS

Thorough training of the operating staff, work conditions such as, air conditioning, color combinations, and lighting conducive to attention to duties, and systematically planned operating and maintenance procedures will minimize the possibility of operator error. Safety features incorporated in the plant are designed to protect against accidents in the event of operator error that might occur.

The following measures will be taken to protect against operator error.

(1) Each operating position will have its duties and responsibilities clearly established.

(2) All personnel must qualify for their position (oral and written examinations).

(3) Definition will be made as to the decision making authority for the various position levels.

(4) Performance standards, both operational and maintenance will be established.

(5) All repetitive operations such as start-up, shutdown, and maintenance will be carried out according to check lists.

(6) The operating personnel will be trained in the appropriate corrective action to all of the significant error situations that can be predetermined.

(7) Most situations requiring fast action to prevent serious and adverse conditions are designed with automatic trip or scram protection.

(8) The control panel is human engineered to:

 (1) Minimize constant attention to controls to reduce operator fatigue.

(2) Clearly identify controls and instruments.

(3) Carefully group controls and instruments to minimize the probability of operating the wrong control.

(9) The more critical sequential operations have been interlocked and/or annuciated.

J. EMERGENCY PROCEDURES

The action planned for an emergency that may endanger the safety of the plant or personnel by continued operation is as follows:

(1) Shutdown the reactor as quickly as the situation demands.

(2) Take action as may be appropriate to protect personnel, equipment or environs.

(3) Emergency procedures will be established for the more plausible situations such as fire, loss of power, etc.

VI. HAZARDS EVALUATION

A. GENERAL

In evaluating the potential hazards associated with the PM-1, a hypothetical accident was studied which embodied failures believed to be beyond the realm of credibility. An analysis was made of the results of this hypothetical accident in terms of the potential danger to the population in the area of Warren Peak. The particular population group considered is the town of Sundance, Wyoming. All personnel at the radar station will be under direct control of the military commander and, therefore, their involvement in any accident can be governed by emergency procedures.

The accident model assumes that the primary coolant system is ruptured as a result of a nuclear excursion while operating at full power. The nuclear excursion was assumed to be the result of the introduction of a 2% step increase in reactivity. The analysis showed that this excursion released a greater energy than a similar accident at cold conditions. Further, the accident model assumes that the primary system ruptures in the steam generator package, where, unlike the reactor package, no shield water is available to quench the flashing steam.

The sequence of events following the rupture of the primary system assumed that the reactor core melted releasing 22% of the contained fission products resulting from two years of full power operation. This is again extremely pessimistic. The excursion analysis has shown that the center of the fuel tube wall does not reach the melting temperature of stainless steel (1400° C). The cladding temperature would, of course, be considerably lower than the center fuel temperature. Thus, the release of any of the contained fission products is very questionable.

It was assumed that the steam cloud formed upon rupture, filled the primary system shelter area causing the building to fail. This releases the cloud to the atmosphere.

The released cloud was assumed not to rise but rather to move downwind in a horizontal line from the point of release. This is again a very pessimistic assumption. The steam cloud at the point of release will be at a temperature of approximately 200° F (T_S at 6600 feet). This cloud under lapse conditions will rise significantly.

It is therefore believed that the hypothetical accident analyzed is extremely pessimistic and could not be achieved except possibly through planned sabotage.

B. NUCLEAR EXCURSION

The excursion analysis, performed on an IBM-704 based upon a program prepared by The Martin Company, provided the results indicated on Table VI-1 below. The program utilized, assumes that all thermal effects which act to shut down the excursion can be lumped into two effects:

(1) Fuel tube temperature rise

(2) Steam formation within the moderator (i.e., void formation).

TABLE VI-1

PM-1 Excursion Data

Step - Operating Power (9.35 MW)

Reactivity Input	Asymptotic Period (MS)	Energy (MW-Sec)	Fuel Temperature
0.010 Step	14	38	532° C
0.015 Step	1.8	97	987
0.020 Step	1.0	148	1296
	Step--Zero Power (10^{-6} mw)		
0.010 Step	19	39	185
0.015 Step	1.8	100	796
0.020 Step	1.1	155	1045
	Ramp--Operating Power		
0.0050 Kex/sec	1030 Max.	23	341
0.0075 Kex/sec	598 Max.	23	342
0.0100 Kex/sec	385 Max.	23	343
	Zero Power		
0.0050 Kex/sec	431 Max.	27	136
0.0075 Kex/sec	317 Max.	30	166
0.0100 Kex/sec	111 Max.	33	191

In addition, the following assumptions are made:

(1) Heat transfer through both fuel and moderator is conductive.

(2) The thermal diffusivities of the fuel element wall and moderator are constant.

(3) Steam formation begins when the fuel tube surface temperature reaches the moderator saturation temperature.

(4) The steam formation rate is proportional to the heat current entering the moderator.

The computer program consists of a transient thermal subroutine embodying the above assumptions coupled to a nuclear kinetic routine. The nuclear kinetic routine provides the heat release rates for the thermal subroutine which in turn feeds back the reactivity losses.

The following parameters were used in the excursion analysis:

Operating power--9.35 Mw

Initial coolant temperature = 225.2° C

Saturation temperature = 288.9° C

Zero power--1 watt

Initial coolant temperature = 20° C

Saturation temperature = 100° C

Neutron lifetime--1.4 x 10^{-5} sec

Delayed neutron fraction (ß)--0.0075

Temperature coefficient of reactivity--3.95 x 10^{-5}/° C

Void coefficient of reactivity--0.37 Kex/% core void fraction

C. RUPTURE OF THE PRIMARY COOLANT SYSTEM

In the accident analyzed, it was assumed the primary coolant system was ruptured in the steam generator package. The rupture of this system produces a steam cloud in the Primary System Shelter. The amount of steam released becomes unimportant when the assumption is made that the shelter fails releasing the cloud. Any steam overpressure in the shelter in excess of that pressure required to fail the building will tend to produce a larger cloud volume and thus reduce the specific activity of the released cloud. In the accident analysis. It was assumed that the cloud volume was exactly equivalent to the shelter volume (136 cubic meters).

D. RADIONUCLIDE INVENTORY

The fission product inventory of the PM-1 after two years of operation at full power is 29.7×10^6 curies. This is made up of the following:

Halogens	2.7×10^6 curies
Noble Gases	2.2×10^6 curies
Solids	24.8×10^6 curies
Total	29.7×10^6 curies

The fission products resulting from the nuclear excursion can be neglected as they represent only about 4% of the stored fission products. Likewise, the primary coolant activation products can be neglected because of their low concentrations and in the case of the radionitrogens the short halflife.

Even though the excursion analysis indicated that the fuel tubes would not melt, the following fission product release was assumed in the accident evaluation. The percentages being taken from reference 2.

Halogens	(70% of those contained)	1.89×10^6 curies
Noble Gases	(100% of those contained)	2.20×10^6 curies
Solids	(10% of those contained)	2.48×10^6 curies
	Total	6.57×10^6 curies

The 6.57 megacuries represent 22% of the total contained fission products.

E. METEOROLOGICAL PARAMETERS

For the accident evaluation it was assumed that the radioactive cloud did not rise but rather that the cloud center moved horizontally from the point of release. Calculations indicate that the cloud would rise approximately 1700 feet under average conditions.

In order to determine wind velocities for the hazards evaluation it was necessary to rely on the judgement of local observers because of the lack of recorded data. These observers estimated a mean wind speed of 15 miles per hour for the summer months and 18 miles per hour for the winter months. The minimum recorded wind velocity known for Warren Peak over a period of two summers was 5 miles per hour. For the hazards evaluation the following wind conditions were employed:

Parameter	Average Condition	Stable Conditions
Wind Velocity	7 meters/sec (15 mph)	2.3 meters/sec (5 mph)
Sutton's Diffusion Coefficient	0.30 meters	0.05 meters
Stability Parameter	0.25	0.50

F. TOPOGRAPHICAL FACTORS

The topographic features of the site area have a significant influence upon the radiation doses received by a downwind observer. Figure VI-1 shows a topographic cross-section through the PM-1 site and Sundance with superimposition of cloud configurations for both average and stable conditions. The distances from the center of the cloud to ground are based upon this cross-section. The significance of the topography of the site area is that under stable conditions the cloud will not touch the ground at any point. Under average conditions the cloud will touch the ground throughout its passage downwind.

G. HAZARDS EVALUATION

A hazards evaluation based upon the foregoing model is presented. The evaluation is made both for stable and average meteorological conditions. It includes the direct external radiation doses from the cloud, and internal radiation doses from inhalation of significant radionuclides in the cloud. Figures VI-2 and VI-3 show the surface path of a typical cloud and Fig. VI-1 shows a cross-section through the cloud path combined with the topographic cross-section. Figures VI-2 and VI-3 also present isodose curves as a function of radius.

1. Average Meteorological Conditions

Table VI-1 is a summary of the radiobiological consequences for the postulated accident during average conditions. In all cases, the individual exposed at various distances from the origin is in the same vertical plane as that containing the line representing the movement of the cloud center above him.

Direct External Radiation Doses. - The external gamma doses resulting from direct radiation from the cloud as it passes overhead vary from 0.44 roentgen at 500 meters to 0.27 milliroentgen at 9650 meters (Sundance). In addition to this direct radiation from the cloud, a person would receive a significant gamma dose from the fallout left behind by the cloud's passage. The fallout doses based upon a two day exposure prior to cleanup or evacuation of the area vary from 177 roentgen at 500 meters to 670 milliroentgen at 9650 meters.

The external beta doses (TID) vary from 6.5 roentgen at 500 meters to 15 milliroentgen at 9600 meters based upon direct radiation from the cloud during its passage. Doses from beta radiation from fallout are not treated because of the relative ease of protecting from them. A person's clothing provides significant shielding while being in a building or motor vehicle will provide even greater protection.

Internal Radiation Doses from Inhalation. - The internal exposure to the thyroid from radioiodine (TID) varies from 3.95 rem at 500 meters to 20.66 millirem at 9650 meters. The internal exposure to the bone from radiostrontium varies from 1.85 rem at 500 meters to 9.7 millirem at 9650 meters. The internal exposure from Cerium 144 and Praseodymium 144 to the bone varies from 1.86 rem at 500 meters to 9.7 millirem at 9650 meters. The Cesium-137 and Barium-137 exposure to the muscle varies from 48 millirem at 500 meters to 0.25 millirem at 9650 meters.

2. Stable Meteorological Conditions

Table VI-3 is a summary of the radiobiological consequences for the accident postulated during stable meteorological conditions.

Direct External Radiation Doses. - The direct gamma radiation doses received by individuals at various distances from the origin of

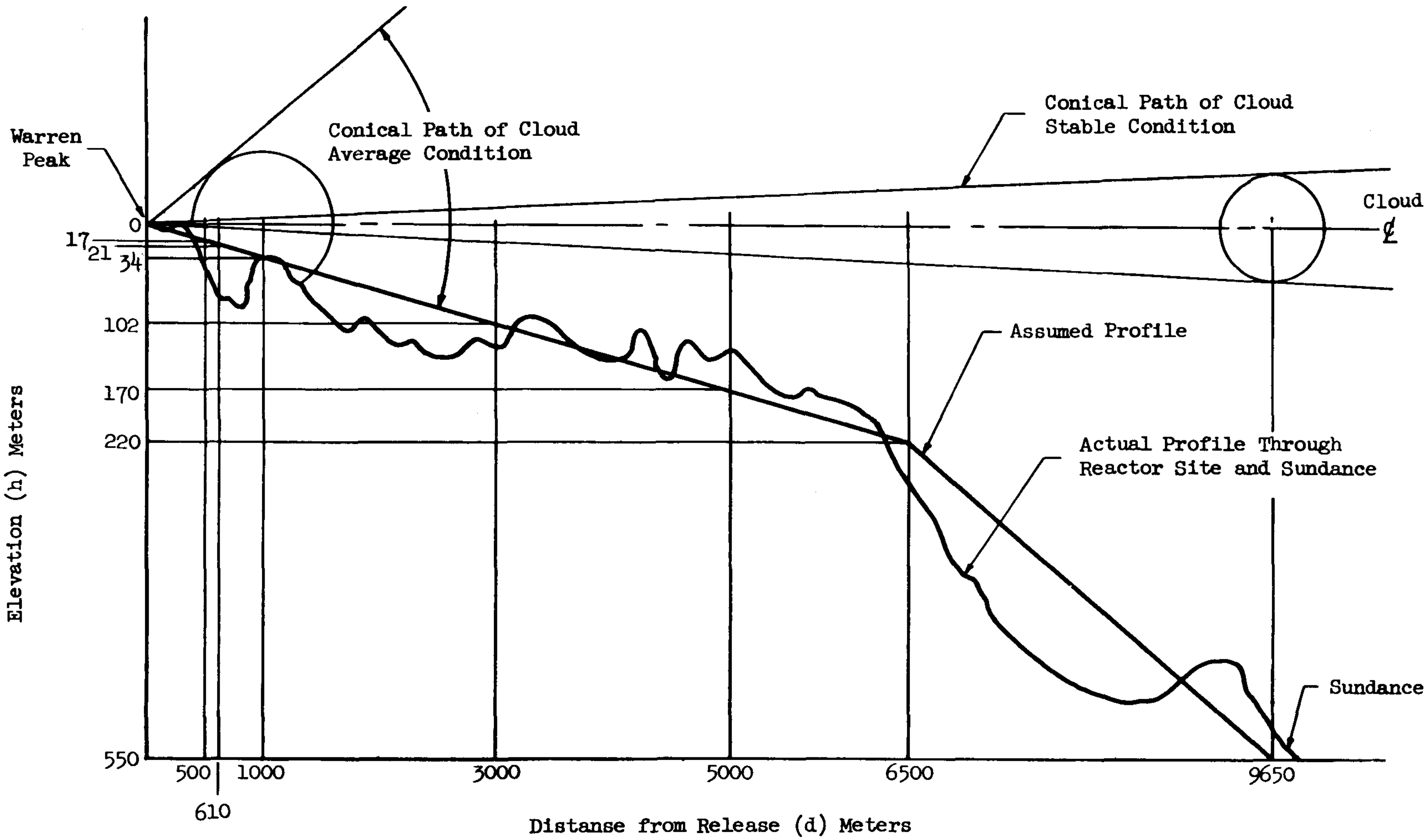

Fig. VI-1. Cross-Section of the Reactor Site, Showing the Conical Path of the Radioactive Cloud for the Stable and Average Meteorological Conditions

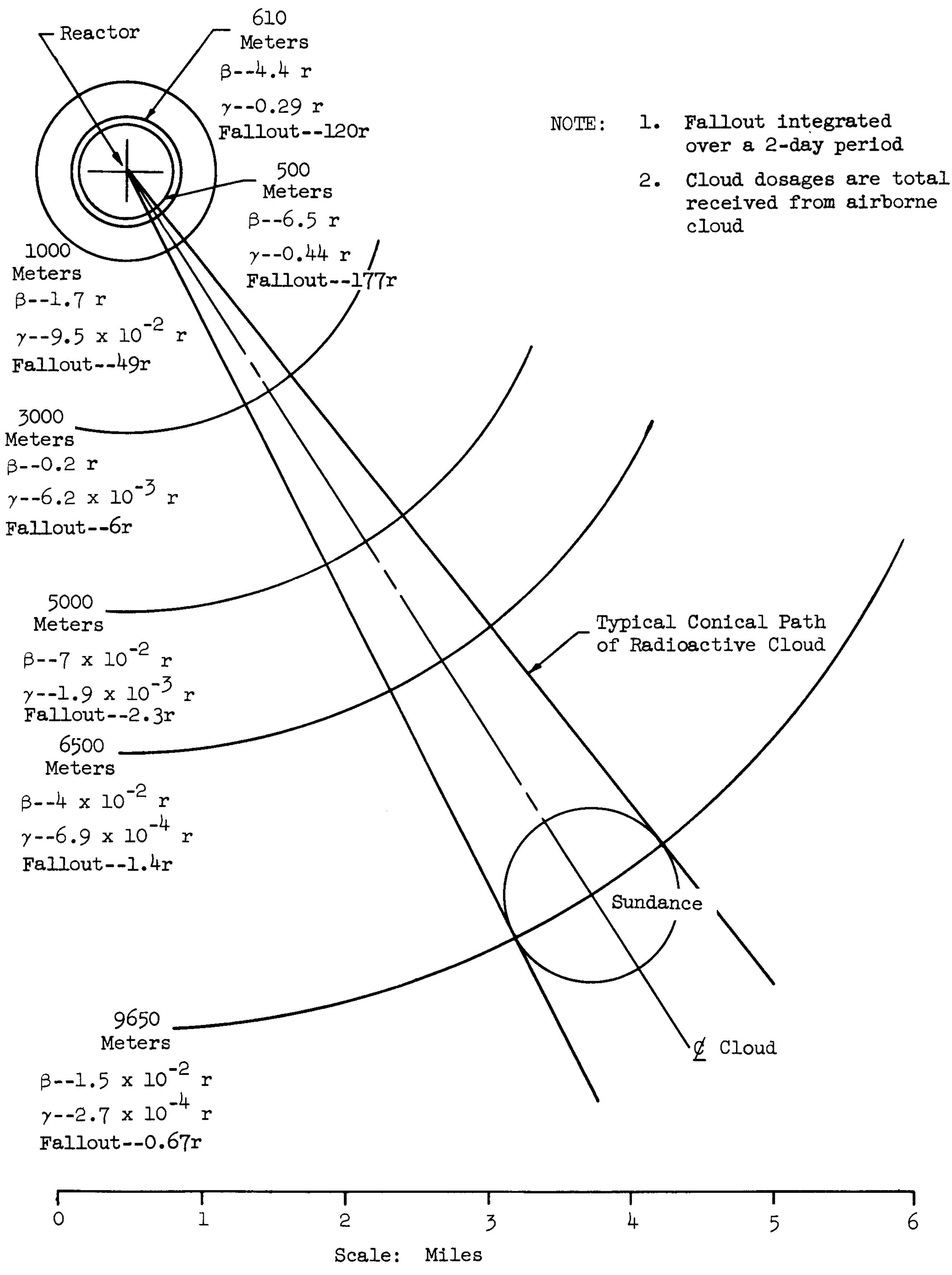

Fig. VI-2. External Dosages for Postulated Accident During Average Meteorological Conditions

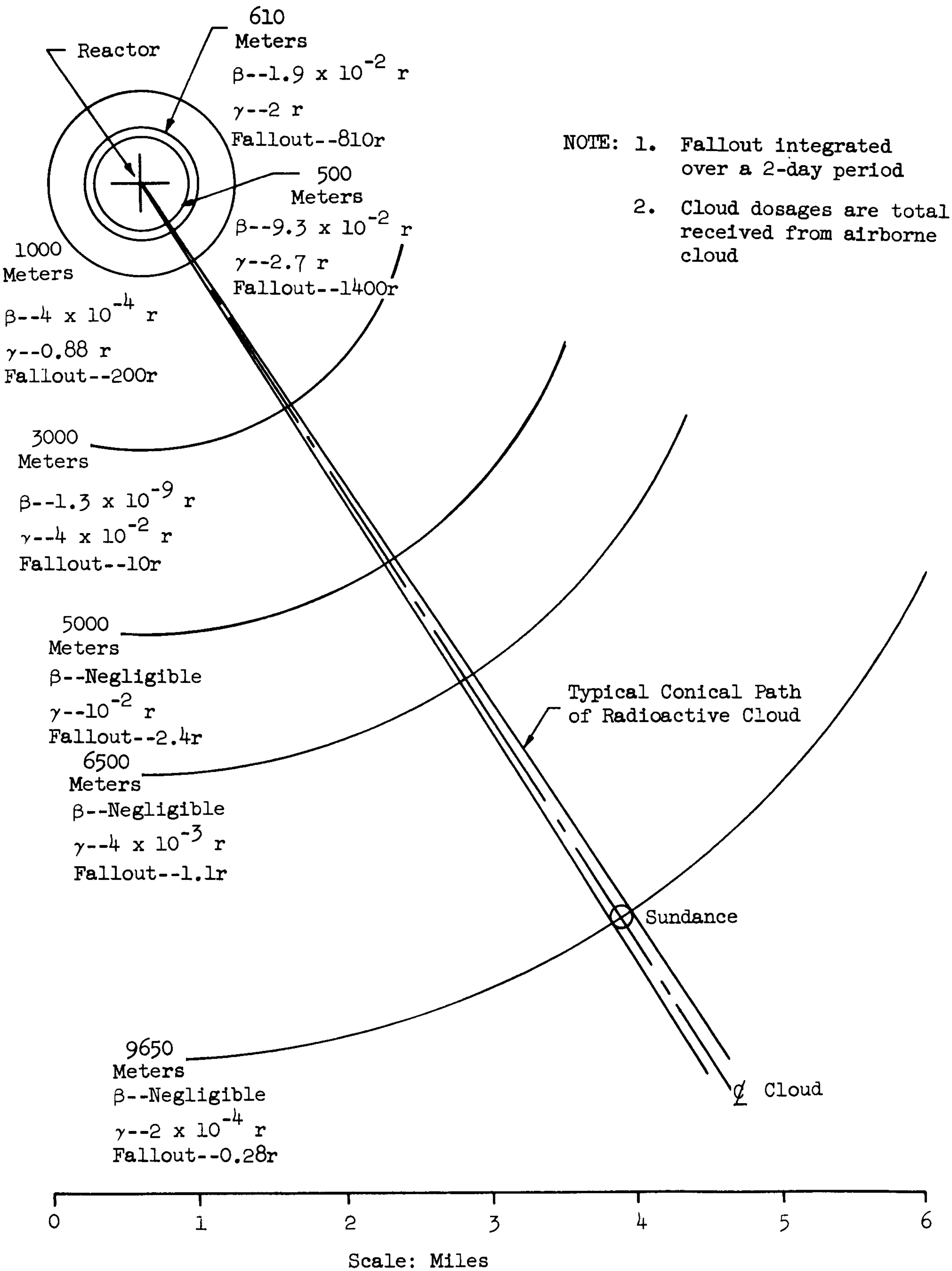

Fig. VI-3. External Dosages for Postulated Accident During Stable Meteorological Conditions the accident from the incident cloud were determined. The external gamma doses (TID), vary from 2.7 roentgen at 500 meters to 0.2 milliroentgen at 9650 meters. The gamma radiation doses resulting from fallout vary from 1400 roentgen at 500 meters to 280 milliroentgen at 9650 meters based upon two day's of integrated exposures. The two day period would be adequate for decontamination in the outlying areas. At the PM-1 site the two day period is extremely pessimistic. Decontamination and/or evacuation of the military installation within the 500 meter circle could be accomplished in a matter of a few hours while closely monitoring the radiation doses being received.

The external beta doses vary from 93 milliroentgen at 500 meters to an infinitesimal dose at 9650 meters due to direct cloud radiation. Fallout beta doses were not considered because of the relative ease of protecting persons from them.

Internal Radiation Doses From Inhalation. - There are no internal radiation doses from inhalation of incident radionuclides from the cloud in that the cloud does not touch the ground at any point. Figure VI-1 shows this condition graphically.

3. Summary of Hazards Evaluation

The external radiation doses, both beta and gamma, present only minimal hazards to the general public near the Warren Peak site. The total external gamma doses at Sundance vary from approximately 1000 milliroentgen with average meteorological conditions to approximately 300 milliroentgen with stable meteorological conditions based upon a two day total integrated dose. In either case, the total doses are sufficiently small so as to present no serious consequences. The internal radiation doses at Sundance are in the millirem range and again do not present any serious consequences.

Within the 500-meter circle around the PM-1 powerplant, higher radiation doses would be received. However, the personnel within this area will be under direct control and would be promptly notified in an emergency. The training of the reactor operating personnel would allow them to make a rapid evaluation of the hazards present and take any necessary action to safeguard the radar site. Emergency equipment such as radiation monitors, will be available for immediate use.

The doses determined in this evaluation are all based upon the most pessimistic assumptions. In other words, these doses represent the "worst case", and therefore, in any lesser accident, or more favorable meteorological conditions the doses received would be greatly reduced or completely eliminated.

TABLE VI-2

Summary of Radiological Consequences Postulated Accident During Average Meteorological Conditions

1 (3) Distance (d) from release		2 (4) Radioactive Cloud Volume (V)	3 Cloud radius	4 (5) Height of Cloud centerline above ground (h)	5 Time for Cloud centerline to reach distance (d)	6 (6) Dry Deposition	7 (7) Total Curies in Cloud (Dry Deposition)	8 Concentration In Cloud (Dry Deposition)	9 Time for Cloud to pass over point on ground	10 (8) External β Dose on ground from airborne cloud	11 (9) External γ Dose on ground from airborne Cloud	12 (10) External γ Dose Due to Surface Deposition
meter	feet	$meter^3$	meter	meter	second	$curies/meter^2$	curies	microcuries/cc	second	Roentgens	Roentgens	T.I.D. - 2 days Roentgens
0	0	136	3.2	0	0	-	6.57×10^6	4.83×10^4	0.5	-	-	-
500	1.64×10^3	2×10^6	78	17	71	3.4	2.71×10^6	1.36	22	6.5	4.4×10^{-1}	1.77×10^2
610	2×10^3	3×10^6	90	21	87	2.3	2.59×10^6	8.6×10^{-1}	25	4.4	2.9×10^{-1}	1.2×10^2
1000	3.28×10^3	9×10^6	130	34	143	9.4×10^{-1}	2.3×10^6	2.6×10^{-1}	36	1.7	9.5×10^{-2}	49
3000	9.84×10^3	1.7×10^8	345	102	429	1.2×10^{-1}	1.8×10^6	1.1×10^{-2}	94	2×10^{-1}	6.2×10^{-3}	6
5000	1.64×10^4	7×10^8	551	170	714	4.4×10^{-2}	1.6×10^6	2.3×10^{-3}	150	7×10^{-2}	1.9×10^{-3}	2.3
6500	2.13×10^4	1.6×10^9	726	220	929	2.7×10^{-2}	1.5×10^6	9.4×10^{-4}	200	4×10^{-2}	6.9×10^{-4}	1.4
9650	3.17×10^4	4.2×10^9	1000	550	1380	1.3×10^{-2}	1.4×10^6	3.3×10^{-4}	240	1.5×10^{-2}	2.7×10^{-4}	0.67

1 (3) Distance (d) from release	13 (11) Internal Exposure of Critical Organs from Various Fission Products Inhaled during Passage of Cloud Total REM								
meter	I^{131} Thyroid	I^{132} Thyroid	I^{133} Thyroid	I^{134} Thyroid	I^{135} Thyroid	Sr^{89} Bone	Sr^{90} Bone	Ce^{144} - Pr^{144} Bone	Cs^{137} - Ba^{137} Muscle
0	7.5×10^2	42	4×10^2	38	10^2	85	5.5×10^2	6.3×10^2	1.6×10^2
500	2.2	1.3×10^{-1}	1.2	1.1×10^{-1}	3.1×10^{-1}	2.5×10^{-1}	1.6	1.86	4.8×10^{-2}
610	1.7	10^{-1}	9.2×10^{-1}	8.8×10^{-2}	2.4×10^{-1}	1.9×10^{-1}	1.3	1.5	3.8×10^{-2}
1000	8.3×10^{-1}	4.8×10^{-2}	4.4×10^{-1}	4.2×10^{-2}	1.1×10^{-1}	9.3×10^{-2}	6.1×10^{-1}	7×10^{-1}	1.8×10^{-2}
3000	1.16×10^{-1}	6.7×10^{-3}	6×10^{-2}	5.9×10^{-3}	1.6×10^{-2}	1.3×10^{-2}	8.4×10^{-2}	9.7×10^{-2}	2.5×10^{-3}
5000	7.3×10^{-2}	2.5×10^{-3}	2.3×10^{-2}	2.2×10^{-3}	6×10^{-3}	4.9×10^{-3}	3.2×10^{-2}	3.6×10^{-2}	9.5×10^{-4}
6500	2.6×10^{-2}	1.5×10^{-3}	1.4×10^{-2}	1.3×10^{-3}	3.6×10^{-3}	2.9×10^{-3}	4.9×10^{-2}	2.2×10^{-2}	5.7×10^{-4}
9650	1.16×10^{-2}	6.7×10^{-4}	6.2×10^{-3}	5.9×10^{-4}	1.6×10^{-3}	1.3×10^{-3}	8.4×10^{-3}	9.7×10^{-3}	2.5×10^{-4}

Notes and Parameters

1. Average meteorlogical condition was defined as a condition with the following parameters:

   (a) wind speed ($\bar{u}$) = 7 meter/sec
   (b) diffusion coefficient (C) - 0.3 meter
   (c) stability parameter (n) - 0.25
   (d) isotropic turbulence was assumed.

2. The centerline of the cloud was assumed to move normal to the release above the ground with no initial rise.

3. The distances taken in column 1 are arbitrary, the 1000 meter distance being Government controlled property and the 9650 meter distance is the Sundance area.

4. Radioactive Cloud Volume, Column 2, was calculated from the following equation. The term d_c is a corrected distance equal to $d + d_o$. The distance d_o is a point up wind where the cloud would theoretically be a point source. In the average condition, d_o = 18 meters.

   $$V = (\pi^{1/2} C d_c^{(2-n)/2})^3$$

5. Height of Cloud centerline above ground, column 4, was taken from an elevation cross-section of the area through the reactor site and Sundance. Sundance is 1800 feet (550 meters) below the reactor site.

6. Ground deposition from dry fallout, column 6, was calculated from the following equation:

   $$W_{dry-man} = \frac{nQ}{2e\pi^{1/2} C_d 2-n/2}$$

   Where Q was considered to be the number of curies in the original cloud that would fall out, this Q eliminates the source strength of the noble gasses which would not deposit out.

7. Column 7 is the total curie count in the cloud corrected for fallout and decay. The original inventory of fission products in the core after two years of operation at 10 megawatt power was taken as 2.97×10^7 curies, 2.2×10^6 curies of noble gasses, 2.7×10^6 curies of halogens, and 2.48×10^7 curies of solid particles. The release assumed for this analysis was 22.12% of the total inventory with the following breakdown.

Noble gases	100%	- 2.2×10^6 curies
Halogens	70%	- 1.89×10^6 curies
Solid particles	10%	- 2.48×10^6 curies
Total released		6.57×10^6 curies

8. External β dosage, column 10, describes the total dosage during the passage of the cloud. It was calculated from the following equation.

   $$D_\beta = 9.41 \times 10^{-12} \frac{Q_\beta \, e^{-\left(\frac{h^2}{c^2 d^{2-n}}\right)}}{\pi \bar{u} C^2 d^{2-n}}$$

   where Q_B was taken as 8.00×10^{17} mev/sec, the total beta source strength, and h was distance of the ground below the centerline of the cloud, Q_B was corrected for decay.

9. The external γ dosage, Column 11, was taken from Holland's Nomograms of cloud gamma dosage from power excursion products, Meteorology and Atomic Energy, July 1955.

10. External γ Dose due to surface deposition, Column 12, was calculated from equation:

    $$D_d = 3 \times 10^{-3} X_0 \int_{t_1}^{t^2} \epsilon(t)\, dt$$

    The fission products were assumed to have been released from a reactor operating at a steady power of 10 megawatts for 2 years.

    $(\epsilon t) = t^{-0.2}$ with $t_1 = 0$ and t_2 = 2 days

    $D_d = 1.15 \times 10^3 X_0$

    X_0 being the γ deposition factor.

11. Total internal exposures of critical organs from fission products inhaled during passage of cloud, Column 13, was calculated from numbers published by T. J. Burnett, in an article entitled "Reactors, Hazard vs Power Level", Nuclear Science and Engineering, Vol. 2, No. 3, May 1957.

    Breathing rate was assumed to be 8.3 cc/sec. The inhalation uptake factor for the various isotopes was assumed to be as indicated below:

I	- 0.15
Sr	- 0.22
Ce-Pr	- 0.1
CS-Ba	- 0.36

TABLE VI-3

Summary of Radiological Consequences for Postulated Accident During Stable Meteorological Conditions

1 (3) Distance (d) from release		2 (4) Radioactive Cloud Volume (V)	3 Cloud Radius	4 (5) Height of Cloud Centerline above ground (h)	5 Time for Cloud Centerline to reach distance (d)	6 (6) Dry Deposition	7 (7) Total Curies in Cloud (Dry Deposition)	8 Concentration in Cloud (Dry Deposition)	9 Time for Cloud to pass over point on ground	10 (8) External β Dose on ground from Airborne Cloud	11 (8) External γ Dose on ground from Airborne Cloud	12 (9) Time for Complete Deposition	13 (9) External γ Dose Due to Surface Deposition	14 Internal Exposure from inhaled fission products in the cloud
meter	feet	$meter^3$	meter	meter	second	$curies/meter^2$	curies	microcuries/cc	second	Roentgens	Roentgens	Seconds	TID - 2 days Roentgens	REM
0	0	136	3.2	0	0	--	6.57×10^6	4.8×10^4	2	--	--	--	--	
500	1.64×10^3	2.3×10^3	8.2	17	217	86	2.11×10^6	9.2×10^2	7	9.3×10^{-2}	2.7	3.7×10^4	1.4×10^3	See
610	2×10^3	3×10^3	8.95	21	265	61	2×10^6	6.7×10^2	8	1.9×10^{-2}	2	4.6×10^4	8.1×10^2	Note 10
1000	3.28×10^3	7.5×10^3	12.2	34	435	26	1.75×10^6	2.3×10^2	11	4×10^{-4}	8.8×10^{-1}	7.4×10^4	2×10^2	Below
3000	9.84×10^3	5.2×10^4	22.2	102	1304	3.8	1.37×10^6	26.3	19	1.3×10^{-9}	4×10^{-2}	2.2×10^5	10	
5000	1.64×10^4	1.5×10^5	33	170	2174	1.5	1.22×10^6	8.1	29	Negligible	1×10^{-2}	3.7×10^5	2.4	
6500	2.13×10^4	3×10^5	41.5	220	2826	1	1.12×10^6	3.7	36	Negligible	4×10^{-3}	4.8×10^5	1.1	
9650	3.17×10^4	6×10^5	52.3	550	4196	4.9×10^{-1}	1.02×10^6	1.7	46	Negligible	2×10^{-4}	1.2×10^6	.28	

Notes and Parameters

1. Stable meteorological condition was defined as a condition with the following parameters:
   (a) Wind speed (u) = 2.3 meter/sec
   (b) Diffusion coefficient (c) -0.05 $meter^{n/2}$
   (c) Stability parameter (n) -0.50
   (d) Isotropic turbulence was assumed.

2. The centerline of the cloud was assumed to move normal to the release above the ground with no initial rise. In this stable condition, the cloud never reaches the ground after it leaves the release point. This can be seen by comparing the cloud radius, column 3 and the ground level below the release point, column 4, and Fig. VI-1.

3. The distances taken in column 1 are arbitrary, the 1000 meter distance being Government controlled property and the 9650 meter distance is the Sundance area.

4. The radioactive Cloud Volume, Column 2, was calculated from the following equation. The term d_c is a corrected distance equal to $d + d_o$. The distance d_o is a point upwind where the cloud would theoretically be a point source. In the stable condition, d_o = 300 meters.

$$V = (\pi^{\frac{1}{2}} C d_c^{(2-n)/2})^3$$

5. Height of the cloud centerline above ground, column 4, was taken from an elevation cross-section of the area thru the reactor site and Sundance. Sundance is 1800 feet (550 meters) below the reactor site.

6. Ground deposition from dry fallout, column 6, was calculated from the following equation.

$$W_{dry\text{-}max} = \frac{n\,Q}{2_{e\pi^{\frac{1}{2}}}\,cd^{2-n/2}}$$

Where Q was considered to be the number of curies in the original cloud that would fallout. That is to say, Q eliminates the source strength of the Noble gasses which would not deposit out.

7. Column 7 is the total curie count in the cloud corrected for fallout and decay. The original inventory of fission products in the core after two years of operation at 10 megawatt power was taken as 2.97×10^7 curies, 2.2×10^6 curies of noble gasses, 2.7×10^6 curies of halogens, and 2.48×10^7 curies of solid particles. The release assumed for this analysis was 22.12% of the total inventory, with the following breakdown.

Noble gases	- 100%	-	2.2×10^6 curies
Halogens	- 70%	-	1.89×10^6 curies
Solid particles	- 10%	-	2.48×10^6 curies
Total release			6.57×10^6 curies

8. External β and γ dosage from the airborne cloud, columns 10 and 11 describe the total dosage during the passage of the cloud. These numbers, especially the β dosage, are very small; for the entire cloud remains meters above the ground as it moves downwind. The β dosage was calculated from the equation given below and the γ dosage was taken from Holland's Nomograms of cloud gamma dosage from power excursion products, Meteorology and Atomic Energy, July 1955.

$$D_\beta = 9.41 \times 10^{-12} \frac{Q_\beta e^{-\frac{h^2}{c^2 d^{2-n}}}}{\pi \bar{u} c^2 d^{2-n}}$$

9. External γ dose due to surface deposition column 13, was calculated from equation

$$D_d = 3 \times 10^{-3} X_o \int_{t_1}^{t_2} f(t)\, dt$$

The fission products were assumed to have been released from a reactor operating at a steady power of 10 megawatts for 2 years

$$f(t) = t^{-0.2}$$

The numbers shown in the table have been corrected for fallout time. Because the fallout time for complete deposition, column 12 is so large, it was assumed that only a portion of the products falling from the cloud would be deposited on the ground. The assumed percent of the total fallout in each area was prorated to the distance (h) to which it had to fall. These percentages are as follows:

Distance from Release	Height	% Fission Products Deposited
500 meter	17 meter	30%
610 meter	21 meter	25%
1000 meter	34 meter	15%
3000 meter	102 meter	5%
5000 meter	170 meter	3%
6500 meter	220 meter	2%
9650 meter	550 meter	1%

10. Internal exposures from inhaled fission products in the cloud would be zero for this stable condition. With the cloud moving normal to the release point, it is virtually impossible for anyone to come in direct contact with it, as shown in Fig. VI-1.

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