ANL-7442
Reactor Technology

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

FINAL SAFETY ANALYSIS REPORT ON THE USE OF PLUTONIUM IN ZPR-6 AND -9

by

W. Y. Kato, G. K. Rusch,
L. R. Dates, C. E. Till,
A. Ancarani,* J. van Doorninck,
C. L. Cheever, and E. M. Bohn

Applied Physics Division

February 1970

*Guest Scientist from Euratom.

DISCLAIMER

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

FINAL SAFETY ANALYSIS REPORT ON THE USE OF PLUTONIUM IN ZPR-6 AND -9

by

W. Y. Kato, G. K. Rusch,
L. R. Dates, C. E. Till,
A. Ancarani, J. van Doorninck,
C. L. Cheever, and E. M. Bohn

ABSTRACT

The Final Safety Analysis Report describes the Argonne split-table type critical facilities, ZPR-6 and -9 located at the Illinois Site, the modifications which were made to the reactor cells and facilities, and analyzes the hazards associated with the use of up to 1400 kg of plutonium fuel in each of the facilities.

Modifications to the cells to improve their containment characteristics were necessary to limit exposures to on-site personnel to less than 0.1 maximum permissible body burdens for plutonium (ICRP values) under design basis accident conditions. These modifications included the construction of: (1) a 3/16-in. thick steel confinement shell over the two blast-resistant highly reinforced concrete cells; (2) an emergency sand filter and a double bank HEPA filter exhaust system with a total particulate attenuation of about 10^5 connected to a 150-ft stack; (3) special airlocks to maintain negative pressures in rooms adjacent to the cells which could not be covered with the steel shell; and (4) a steam turbine generator as a source for emergency power. Modifications to the facilities included: (1) new table drive motors for the reactors to decrease the speed for mechanical reactivity addition; (2) a power feed rail system for the drive motors to insure proper table speeds; and (3) an air-cooling system to remove heat from the core due to alpha activity.

In addition, for personnel protection the following were installed: (1) high sensitivity impactor-type plutonium air-monitoring equipment; (2) special processing hoods for loading and unloading of drawers; and (3) an emergency surveillance station.

Reactor operating procedures and the principles of reactor management are discussed in detail. The report concludes with a discussion of a maximum credible accident and a design basis accident. The analysis of the hypothetical design basis accident in which reactivity is continuously added, with no shutdown control mechanisms operable, and shutdown occurring only by melting of fuel shows that the modifications more than adequately limit the amount of radioactive particulate released to the atmosphere.

Chapter I

INTRODUCTION AND SUMMARY

The Argonne National Laboratory has in operation at the Illinois Site two split table type critical facilities, ZPR-6 (Fig. I-1) and ZPR-9, investigating the physics of fast reactor systems since July 1963 and January 1964, respectively. Authorization for the use of ^{235}U as fuel in cores having volumes up to about 3700 liters and containing up to about 1644 kg ^{235}U is based on the following documents:

Safety Analysis Report, Argonne Fast Critical Facility (ZPR-6) ANL-6271 (Ref. 1); Safety Analysis for the ZPR-9 Assembly, An Addendum to ANL-6271 (Ref. 2); letters, K. Dunbar to A. V. Crewe dated April 14, 1966 (Ref. 3), K. Dunbar to R. B. Duffield dated December 13, 1967 and G. K. Rusch and R. A. Karam (Refs. 4 and 5).

Since the beginning of operation of ZPR-6 and -9, a number of ^{235}U fueled assemblies (as reported in Refs. 6-13) have been investigated in these facilities. The assemblies studied have ranged from relatively small (100-500 liters) ^{238}U or tungsten based cores, to about 3700 liter uranium oxide assemblies. Systems with a depleted uranium blanket and a number of aluminum, aluminum oxide, and beryllium oxide reflected assemblies have been studied. Although the basic fuel is of metal composition, oxide and carbide core compositions have been studied by using uranium oxide, iron oxide and graphite plates to introduce oxygen or carbon into the system.

There is need for integral experimental physics data of large dilute plutonium fueled fast reactors because of the national LMFBR program. Modifications have been made to the ZPR-6 and -9 facilities in order that plutonium may be used as fuel and thus fulfill the national requirements for additional large plutonium fueled flexible critical facilities. The modifications include: (1) the construction of a confinement shell over the two reactor cells; (2) the addition of an emergency sand filter exhaust system; (3) installation of special processing hoods for loading and unloading of fuel drawers; (4) installation of special monitoring equipment; (5) new motors and special power feed rails for the tables; and (6) construction of a new vault for storage of uranium and plutonium fuels. The construction and installation of these modifications were authorized on the basis of a Preliminary Safety Analysis Report (Ref. 14) and the approval letter, K. A. Dunbar to W. M. Manning (Ref. 15). This report discusses the details of the modifications as well as the hazards associated with the use of plutonium fuels.

The general scope of the experimental program will be basically the same as that outlined in the SAR, ANL-6271, but altered to include the use of plutonium fuel. It is proposed to investigate the physics parameters of

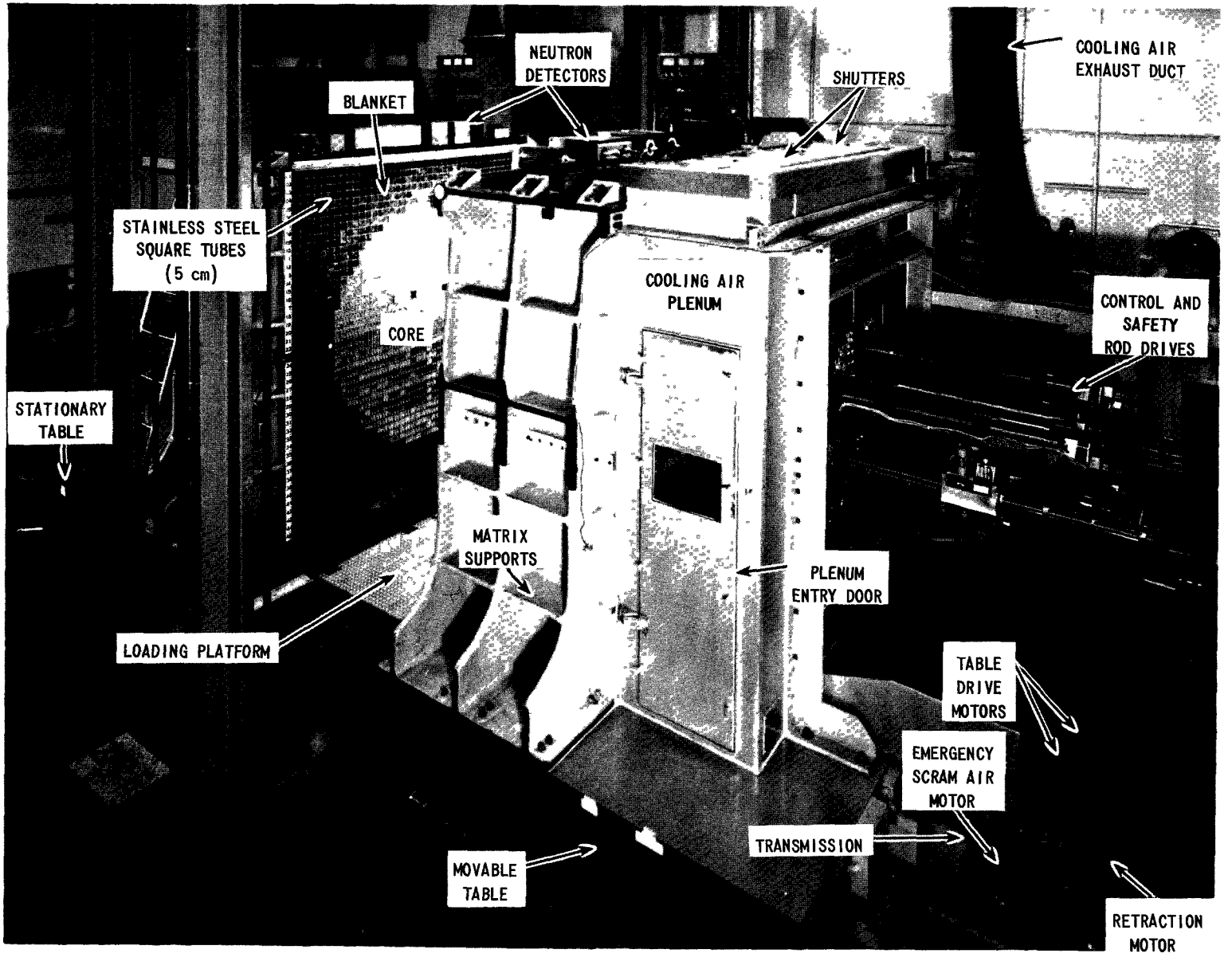


Fig. I-1. ZPR-6. ANL Neg. No. 113-2963A.

both uniformly and zone loaded systems using plutonium and ^{235}U fuels. Measurement of physics parameters such as critical mass, reactivity perturbation measurements of various materials, spectra, flux and power distributions, and Doppler effect will be carried out.

Except for the radiological hazards, the reactor safety characteristics of plutonium-fueled assemblies are not very different from ^{235}U fueled assemblies of the same general type. The only significant difference inherent with the use of plutonium is the smaller value of the delayed neutron fraction, β. The operating kinetics of the system will be kept the same by reducing the maximum rate at which reactivity can be inserted by control rod and table motion for the intermediate and slow closure speeds by approximately a factor of 3 (the ratio of the delayed neutron fractions for ^{235}U and plutonium).

The radiological hazards however, are quite different for the two fuels. The high alpha activity of plutonium, its long biological lifetime and its bone seeking character combine to set very low maximum body burdens for all plutonium nuclides. This is reflected in two major safety areas--more elaborate procedures for handling the fuel and more stringent containment requirements.

Based on the accumulated experience in handling plutonium at the Illinois and Idaho Sites of ANL, handling procedures have been specified that will present no unacceptable hazards to personnel.

Significant modifications to the existing facility that will insure that there is no increased hazard to the surrounding area or to the Laboratory Site itself are proposed. The containment concept is based on 5 main elements:

(1) Plutonium fuel pieces individually contained in 15-mil stainless steel jackets.

(2) Primary containment afforded by the existing blast-resistant reinforced concrete cells.

(3) A confinement shell that encloses the concrete cells in a subatmospheric pressure environment, collecting any gaseous or particulate leakage from the cell.

(4) Installation of an emergency venting system for the cells consisting of a sand filter, two banks of HEPA filters and a 46 m stack. The use of the emergency venting system will maintain cell integrity even if the cell pressure were to rise under extraordinary conditions.

(5) The effluent from the confinement shell volume, the emergency venting and the normal air conditioning in the cell, control room, vault and

vault workrooms is discharged through an exhaust arrangement consisting of banks of high efficiency absolute (HEPA) type filters in series with a discharge stack 46 m in height.

In essence, the cell and shell provide protection against external consequences of a nuclear incident and the sand filter system protects against the compounding effects of a severe metal fire, or the effects of a fire alone.

Various accidents are considered and are discussed in this report. The maximum credible accident is considered to result from criticality occurring with the halves closing at intermediate speed, with all trip circuits inoperable except for a single safety or high level trip, and this trip being set so high that the excursion is only sensed when the reactor reaches prompt critical. This is directly analogous to the maximum credible accident considered in the original safety analysis report.[1] To provide a direct comparison with the analysis in the original report, which analyzed this accident on a 50 liter core, an analogous 50 liter core is also analyzed in this report. The results show that the change in system characteristics caused by the change of fuel to plutonium do not significantly affect the conclusions of the original analysis. The analysis is then repeated for a nominal 3500 liter carbide mockup, which is considered to be much more typical of the cores likely to be built on these facilities, and which was therefore chosen as the reference core for these and subsequent calculations. The conclusion in each case was that the maximum credible accident can result in a substantial increase in the fuel temperature (200°C for the fuel temperature and 2.3×10^{17} fissions for the total energy release for the 50 liter case and 58°C for the fuel temperature and 4.6×10^{17} fissions for the energy release for the 3500 liter core), but the maximum fuel temperature does not approach the ignition point of the fuel and there is little likelihood of release of radioactivity.

The failure of all interlock and trip circuits coupled with the complete breakdown of administrative controls and operation with total disregard for the state of the reactor is not considered credible nor in the general realm of possibility. In order, however, to demonstrate how the reactor cells and ancillary equipment can adequately handle a severe nuclear accident, an accidental nuclear excursion of a very large magnitude designated a design basis accident (DBA) is hypothesized and analyzed. This accident represents an exceedingly severe one but is not to be considered the maximum conceivable accident. The design basis accident which is entirely hypothetical is analyzed essentially as an aid in establishing the adequacy of the containment system design.

In order to hypothesize a severe nuclear accident which taxes the containment capability of the cell, the MCA (intermediate speed criticality accident) with the nominal 3500 liter core is considered. In addition to

postulating that the reactor is overloaded so that it becomes critical at intermediate speed, the low level trip and period trip circuits inoperative, it is further postulated that all high level trip circuits are inoperable and the reactor operators continuously add reactivity disregarding the state of the reactor. This accident is analyzed using a point reactor kinetics code which has been coupled to several subroutines which calculate the neutronic and excursion temperature distribution using appropriate constants for heat conductivity, metal burning rates and the cell pressure as a function of time.

The hypothetical DBA analysis results in the melting and burning of about 60 kg of plutonium and the vaporization of about 22 kg of sodium and the production of 2.7×10^{20} fissions. The resulting cell pressure reaches a maximum of about 35 psia. The emergency venting system prevents further buildup in cell pressure. The total amount of plutonium which reaches the atmosphere is no greater than 60 mg. This results in inhalation hazards of 7.4×10^{-4} mpbb at 225 m under the average worst meteorological conditions. This is about a factor 150 less than the AEC criterion of no more than 0.1 mpbb to an observer at the nearest facility under DBA conditions. The doses to an observer from the noble gases and radioiodine fission products are 44 Rem at 225 m and 20 Rem at 1300 m to the nearest site boundaries assuming (1) low wind conditions, and (2) total instantaneous release (no holdup in reactor cell) up the stack of all noble gas fission products from the 2.7×10^{20} total fissions.

It is concluded that the ZPR-6 and -9 facilities may be operated using plutonium fuel at the Argonne, Illinois Site of Argonne National Laboratory without undue risk to the health and welfare of the general public in the vicinity of the Laboratory or to Laboratory personnel.

REFERENCES

1. W. Y. Kato, G. J. Fischer, and L. R. Dates, "Safety Analysis Report, Argonne Fast Critical Facility (ZPR-VI)," ANL-6271 (December 1963).
2. C. E. Cohn, G. S. Rosenberg, L. R. Dates, W. B. Loewenstein, and G. H. Golden, "Safety Analysis for the ZPR-9 Facility [An Addendum to ANL-6271, Safety Analysis Report, Argonne Fast Critical Facility (ZPR-6)]," ANL-7166, (February 1966).
3. K. Dunbar, Letter to A. V. Crewe (April 1966), and G. K. Rusch and R. A. Karam, "Analysis for Large Fast Critical Assemblies (ZPR-6 and ZPR-9)," ANL-6271 Addendum (July 1966).
4. G. K. Rusch and R. A. Karam, "Safety Analysis of a Nominal 3700-liter UO_2 Core in the ZPR-6 and -9 Facility," ANL-6271 Addendum 2 (July 1968).

5. K. A. Dunbar, Letter to R. B. Duffield (December 13, 1967).

6. G. K. Rusch et al., "Investigations of a 600-liter Uranium Carbide Core (ZPR-6 Assembly No. 2)," Reactor Physics Division Annual Report, July 1, 1963 to June 30, 1964, ANL-7010, p. 91 (January 1965).

7. R. C. Doerner et al., "Physics Measurements in Tungsten Based Aluminum Reflected Fast Reactors," Reactor Physics Division Annual Report, July 1, 1963 to June 30, 1964, ANL-7010, p. 116 (January 1965).

8. F. H. Helm et al., "Physics Parameters and Sodium Void Coefficients in a 950-liter Pancake Core," Reactor Physics Division Annual Report, July 1, 1964 to June 30, 1965, ANL-7110 (1966).

9. R. C. Doerner et al., "Experimental Physics Studies in Fast Reactor Nuclear Rocket Engine Cores," Reactor Physics Division Annual Report, July 1, 1964 to June 30, 1965, ANL-7110 (1966).

10. R. A. Karam et al., "Measured Physics Parameters in a Zonal Fast Uranium Carbide Core (ZPR-6, Assembly 4Z)," Reactor Physics Division Annual Report, July 1, 1965 to June 30, 1966, ANL-7210, p. 137 (December 1966).

11. C. E. Till et al., "Doppler Effect Measurements on a Dilute Carbide Fast Assembly (ZPR-6, Assembly 4Z)," Reactor Physics Division Annual Report, July 1, 1965 to June 30, 1966, ANL-7210, p. 129 (December 1966).

12. R. C. Doerner et al., "Physics Measurements in Fast Tungsten Rocket Reactor Critical Experiments," Reactor Physics Division Annual Report, July 1, 1965 to June 30, 1966, ANL-7210, p. 189 (December 1966).

13. R. A. Karam et al., "Measured Physics Parameters in a Large Uranium Carbide Fast Core Assembly 5, ZPR-6," Reactor Physics Division Annual Report, July 1, 1966 to June 30, 1967, ANL-7310, pp. 167-189 (1967).

14. W. Y. Kato, C. E. Till, G. K. Rusch, A. Ancarani, and J. Haugsnes, "Preliminary Safety Analysis Report on the Use of Plutonium and U^{233} in ZPR-6 and -9 [Addendum No. 3 to Safety Analysis Report, Argonne Fast Critical Facility (ZPR-6), ANL-6271]," ANL-7211 (to be published).

15. K. A. Dunbar, Letter to W. M. Manning dated October 25, 1967.

Chapter II

PROPOSED EXPERIMENTS

A. Introduction

Critical experiment studies of fast reactors are conducted to provide accurate integral information which, in conjunction with theoretical studies, supply physics information required in the design of economic large fast breeder reactors. Although in principle it is possible to design large reactors using microscopic nuclear data alone, in practice the accuracy of the data required is such that design needs must be largely supplied by carefully selected integral experiments. Such experiments are designed in close conjunction with theoretical and analytical studies of large fast reactors to insure that the data correspond to the programmatic needs.

The uses for a critical facility are twofold: First, experiments are carried out to investigate the ability to predict the performance of reactors from calculations based on fundamental or adjusted parameters. Experiments are designed specifically to establish a basis for judging calculational methods and adjusting parameters. Some examples of this function are:

(a) Determination of critical size or critical enrichment for power reactor composition. Integral studies are performed on large uranium and plutonium fueled systems in order to obtain the basic static characteristics of such systems. Diagnostic sequences of experiments are made to provide data on clean plutonium fueled systems. Zoned studies are performed to validate the method and to obtain some of the characteristics of large systems with much reduced material inventories. Flux and power distributions are measured.

(b) Measurement of reactivity effects such as temperature coefficients, control rod effectiveness, blanket compositions and effectiveness.

(c) Investigation of heterogeneity effects to assure that such effects in the critical facilities, or in the associated reactors, are understood and that any necessary corrections are made.

(d) Measurements related to the determination of the neutron spectrum and the adjoint function (importance).

(e) Determination of parameters that are relevant to the prediction and understanding of the long term behavior of the operating reactor. An example of such a quantity is the capture-to-fission ratio of the fissile isotopes. Studies related to the effects of the higher plutonium isotopes are made experimentally by the use of zones containing such material.

Second, the facilities are used as a proof-test of proposed reactor designs. In these experiments, a reactor design is studied to verify that it performs as expected, and empirical determinations of the necessary corrections for such parameters as composition, control rod constitution and power distribution are experimentally made.

The principal objective of the critical experiment program is essentially to improve the utility and efficiency of integral fast reactor measurements by a process of comparison of experimental and calculated data followed by attempted identification of the causes of any discrepancies. In general, the course followed is: (a) a critical examination of the accuracy of the integral parameter being investigated; (b) comparison of the measured values with those calculated using one or more standard cross section sets and identification of discrepancies outside the range of experimental error; (c) examination of the basic nuclear data to assess the accuracy of the multigroup cross section sets; (d) proposals for modification of the integral measurements to improve their validity; and (e) if necessary, the suggested development of improved or new experimental techniques to aid in elucidating the causes of discrepancies.

B. Authorization Request

This report requests the additional authorization to investigate plutonium fueled systems mocking up metal, oxide and carbide assemblies having core volumes as large as about 3500 liters and containing up to about 1400 kg of fissile plutonium in each of the two facilities, ZPR-6 and -9. (Both facilities are currently authorized to investigate ^{235}U fueled systems having core volumes up to about 3700 liters.) It is anticipated that both uniform and zoned cores will be built. In uniform systems either plutonium or ^{235}U fuel in the form described in Chapter V may be used.

For zoned systems plutonium, which is undiluted by ^{238}U, enriched ^{235}U or the ^{238}U-diluted plutonium alloy may be used. The use of undiluted plutonium fuel will be restricted in amount so that the reactivity worth of such fuel will constitute no more than 25% of the total reactivity of the core, defined by

$$\left[\frac{\int_{\substack{\text{undiluted}\\ \text{fuel region}}} \int_E \Phi^*(\vec{r})\Phi(\vec{r},E)\nu(E)\Sigma_f(\vec{r},E)\, dE d\vec{r}}{\int_{\substack{\text{entire}\\ \text{core}}} \int_E \Phi^*(\vec{r})\Phi(\vec{r},E)\nu(E)\Sigma_f(\vec{r},E)\, dE d\vec{r}}\right] \leq 0.25$$

where

$$\Phi(\vec{r},E),\ \Phi^*(\vec{r}),\ \nu(E) \text{ and } \Sigma_f(\vec{r},E)$$

are the neutron flux, average fission neutron importance, the neutrons per fission and the macroscopic fission cross section at point $\vec{r}$ and energy E.

The ZPR-6 and -9 facilities are currently authorized to use up to 1 kg of ^{233}U in small "danger" coefficient or Doppler effect samples. The hazards associated with the use of up to 1 kg of ^{233}U has been discussed in detail in ANL-6271. It is proposed to maintain this limitation of 1 kg for ^{233}U in the studies proposed in this safety analysis report.

C. Characteristics of Typical Cores

Large flexible critical facilities such as ZPPR, ZPR-6 and -9 may be used to study many different types of core compositions and geometries. Since it is impossible to foresee and describe all of the possible cores to be studied on ZPR-6 and -9, this section will consider only typical representative cores which may be studied on these facilities.

Parameters which will be measured will vary from assembly to assembly. In general, they may include those listed below.

(a) Critical mass.

(b) Control rod calibration.

(c) Neutron flux and power distribution using foils or counters.

(d) Neutron spectra using in-core detectors or time-of-flight techniques.

(e) Spectral indices using threshold fission detectors, nuclear emulsions and radiation damage detectors.

(f) Material worth measurements using simple material replacements methods, oscillator, and remote controlled sample changers.

(g) β/ℓ using noise analysis and pulsed neutron techniques.

(h) Doppler effect measurements of various materials using static or oscillation techniques.

1. Core Parameters

The characteristics of typical cylindrical cores with the L/D ~ 1 in the range of interest for the experimental program are listed in Table II-1. The cores are reflected with 30 cm of blanket, axially and radially, with the compositions given in Table II-2. Except as noted, the calculations were done with the one-dimensional multigroup diffusion code Mach 1 (Ref. 1) using the ANL 22-group cross section set 224. The physics characteristics and reactivity coefficients relevant to safety are sufficiently similar for

oxide and carbide assemblies, that only carbide and metal core characteristics are given. The primary difference between the carbide, oxide or metal cores will be a small difference in the overall Doppler coefficients because of differences in neutron spectra. The carbide values are approximately representative of ceramic cores, whether oxide or carbide. These cores provide the reference data for the analyses carried out in the subsequent section of this report.

Table II-1

CHARACTERISTICS OF TYPICAL Pu REACTORS

Type of Core	Pu Metal		Pu Carbide		
Nominal Volume, liters	500	3500	50	500	3500
Actual Volume, liters	591	3469	50 3	590	3466
Radius, cm	46 8	84 0	20 0	46 7	84 0
Height, cm	86 0	156 3	40 0	86 0	156 3
L/D Ratio	0 92	0 93	1 0	0 92	0 93
Critical Mass (kg fissile isotope)	390	1506	75 4	370	1354
Volume Fractions, %					
^{239}Pu	3 46	2 28	7 87	3.30	2 05
^{240}Pu	0 483	0 318	1 10	0 460	0 286
^{241}Pu	0 0806	0 0530	0 183	0 0766	0 0477
^{235}U	0 0278	0 0388	0 0029	0 0074	0 0190
^{238}U	25.54	27 05	28 0	14 80	16 40
Mo	0 403	0 265	0 92	0 383	0 239
C			21.97	10 97	10 95
Na	45 0	45 0	15 0	45 0	45 0
SS	20 0	20 0	19 9	20 0	20 0
$y = \frac{\text{N fertile}}{\text{N fissile}}$	7.37	11.9	3.56	4.50	8 0
$\beta_{eff} \times 10^{-3}$	3 57	3 62	3 37	3 20	3 27
$\ell_p \times 10^{-7}$, sec	1.38	1 97	0 744	2 28	3 70
ρ, Ih/% Δk	978 9	985 8	988 2	1027	1033

Table II-2

BLANKET VOLUME FRACTIONS, %

	System			System	
Isotope	Pu Metal	Pu Carbide	Isotope	Pu Metal	Pu Carbide
^{238}U	61 876	39 341	Fe	9 94	9 94
^{235}U	0 124	0 079	Cr	2 66	2.66
C		22 580	Ni	1.40	1.40
Na	20 00	20 00			

2. Reactivity Coefficients

a. Central Drawer Material Worths

The effect on reactivity of the removal of the central drawer from both halves of the assembly and the effect of substituting a full drawer of fuel plates for this drawer is given in Table II-3. The effect of substituting a drawer of plutonium fuel plates is less than substituting ^{235}U plates because of the ^{238}U diluent in the plutonium fuel.

Table II-3

CENTRAL DRAWER WORTHS (BOTH HALVES)

Reactivity Change % for:	System			
	Pu Metal		Pu Carbide	
	500 ℓ	3500 ℓ	500 ℓ	3500 ℓ
1. Central Drawer Removal	-0.27	-0.052	-0.31	-0.061
2. Substitution of a Full Drawer of Pu-U Alloy Fuel Plates[a] for a Central Drawer	+1.72	+1.11	+1.75	+1.16

[a] 28 wt % Pu-69.5 wt % U-2.5 wt % Mo alloy fuel plates.

The effect of removing the contents of the central drawer and substituting other materials is shown in Table II-4. Each substitute material occupied the full length of 5.1 x 5.1 sq cm cross section drawer. The length of the drawer corresponding to the length of the core under consideration.

Table II-4

WORTHS OF VARIOUS MATERIALS IN THE CENTRAL DRAWER (BOTH HALVES)

Reactivity Change % for Removal of Central Drawer and Substitution of		System			
		Pu Metal		Pu Carbide	
		500 ℓ	3500 ℓ	500 ℓ	3500 ℓ
Material	Density-g/cc				
U^{235}	18.9	+6.65	+4.78	+6.78	+5.03
U^{238}	19.7	-0.63	-0.30	-0.70	-0.37
Th^{232}	11.6	--	--	--	--
Paraffin	0.89	-1.00	-0.63	-0.60	-0.36
Be	1.98	-0.64	-0.39	-0.49	-0.33
C	1.95	-0.47	-0.23	-0.43	-0.20
H_2O	1.0	-1.19	-0.73	-0.81	-0.49

b. Edge Drawer Worths

The effect on reactivity of the removal of a drawer of core materials at the core-blanket interface and the effect of substituting a full drawer of fuel plates for the same drawers are given in Table II-5.

Table II-5

EDGE DRAWER WORTHS (BOTH HALVES)

Reactivity Change % for:	System			
	Pu Metal		Pu Carbide	
	500 ℓ	3500 ℓ	500 ℓ	3500 ℓ
1. Edge Drawer Removal	-0.055	-0.008	-0.063	-0.009
2. Substitution of a Full Drawer of Pu-U Alloy Fuel Plates[a] for an Edge Drawer	+0.179	+0.045	+0.214	+0.093

[a]28 wt % Pu-69.5 wt % U-2.5 wt % Mo alloy fuel plates.

Substitution of other material in place of the contents of the edge drawer gives reactivity changes as shown in Table II-6.

Table II-6

WORTHS OF VARIOUS MATERIALS IN AN EDGE DRAWER (BOTH HALVES)

Reactivity Change % for Removal of Edge Drawer and Substitution of:		System			
		Pu Metal		Pu Carbide	
Material	Density	500 ℓ	3500 ℓ	500 ℓ	3500 ℓ
^{235}U	18.9	+0.668	+0.174	+0.818	+0.233
^{238}U	19.7	-0.045	-0.008	-0.054	-0.012
^{232}Th	11.6	-	-	-	-

c. Dual-purpose and Safety Rod Worths

Typical worths of the 10 fuel-bearing dual-purpose and the 12 boron-10 safety rods for some of the reference cores are shown in Table II-7. The dual-purpose rods were assumed to contain core composition and the positions in the core of both dual-purpose and safety rods were

rather arbitrarily selected so that the calculations are only illustrative of typical worths in these systems. Where a deficiency in available shutdown is shown, the contents of the dual-purpose drawers or the positions of either of the rod types would in practice be adjusted to meet the shutdown requirements. The calculations were done by altering the material densities in annuli concentric with the core center to account for the absence of the dual-purpose rods or the presence of boron-10 safety rods.

Table II-7

CONTROL AND SAFETY ROD WORTHS

REACTIVITY %

Rod Configuration	System	
	500 ℓ Systems	
	Pu Metal	Pu Carbide
10 dual-purpose rods between 13 and 19 cm radius	-1.25	-1.42
12 B^{10} safety rods between 21 and 23 cm radius	-2.58	-3.60
Total Shutdown	-3.83	-5.02

3500 ℓ Systems

	Pu Metal	Pu Carbide
2 dual-purpose rods between 14 and 20 cm radius and 8 dual-purpose rods between 20 and 26 cm radius	-0.25	-0.29
12 B^{10} safety rods between 32 and 34 cm radius	-1.58	-2.61
Total Shutdown	-1.83	-2.90

d. Fatman Effects

The human body is a good moderator and in large assemblies where the k_{eff} of a single half is high, the introduction of moderating material between the halves should be controlled. It has been the practice to use Boral neutron shields placed against the unreflected faces of the separated halves of the reactor whenever loading changes are made on the larger assemblies. The shields have been used until Fatman effect measurements have shown the reactor to be safe. Boral shields are available in 0.635- and 1.27-cm thicknesses. Two rotating sections of the 1.270-cm thick Boral shield, one offset with respect to the axis of rotation of the other, allow access to any portion of the core while exposing only a small fraction of the core face. In addition, a solid 0.635-cm thick Boral shield mounted in a similar framework is available to cover the entire core face of the other half of the reactor during loading operations.

Calculations relevant to this problem were made using Mach 1 and ANL cross section set 201 (which extends down to thermal energies). The Fatman effect was conservatively simulated by assuming the entire normally unreflected face of one of the separated halves of the reactor was covered with a slab of Benelex. (Benelex has the following composition in atoms/cm^3: H, 5.5×10^{22}; C, 3.4×10^{22}; O, 2.4×10^{22}.) The k_{eff} of one-half of the core was first calculated. Then calculations were made with Benelex thicknesses of 20 and 50 cm against one-half and repeated with a 0.635-cm thick sheet of Boral between the core and Benelex. (Boral has the following composition in atoms cm^{-3}: Al, 4.6×10^{22}; B, 2.59×10^{22}; C, 0.65×10^{22}.) The results are shown in Table II-8. It will

Table II-8

FATMAN EFFECTS

($k_{effective}$)

Configuration	System				
	Pu Metal		Pu Carbide		
	500 ℓ	3500 ℓ	5000 ℓ	3500 ℓ	3500 ℓ (L/D = 1/3)
Bare Half	0.789	0.862	0.763	0.827	0.608
Bare Half + 20 cm Benelex	0.886	0.905	0.880	0.882	0.778
Bare Half + 50 cm Benelex	0.886	0.905	0.880	0.882	0.778
Bare Half + 1/4 in. Boral + 20 cm Benelex	0.807	0.866	0.785	0.833	0.632

be noted that 20 cm of Benelex constitutes an effectively infinite reflector, and that the insertion of the 0.635 cm of Boral between the core face and the Benelex reduced the multiplication to values very little above that for the bare half alone.

It is therefore concluded that while the Fatman effect must be carefully investigated and that adequate safeguards must be employed to guard against the unexpected, there is no indication that the Fatman effect presents an operational problem or a hazard.

e. Worth of Gap between Halves

Reactivity changes caused by separation of the halves were calculated by the method outlined by Tamor.[2] The reactivity loss as a function of gap width for the various reference core sizes is shown in Fig. II-1.

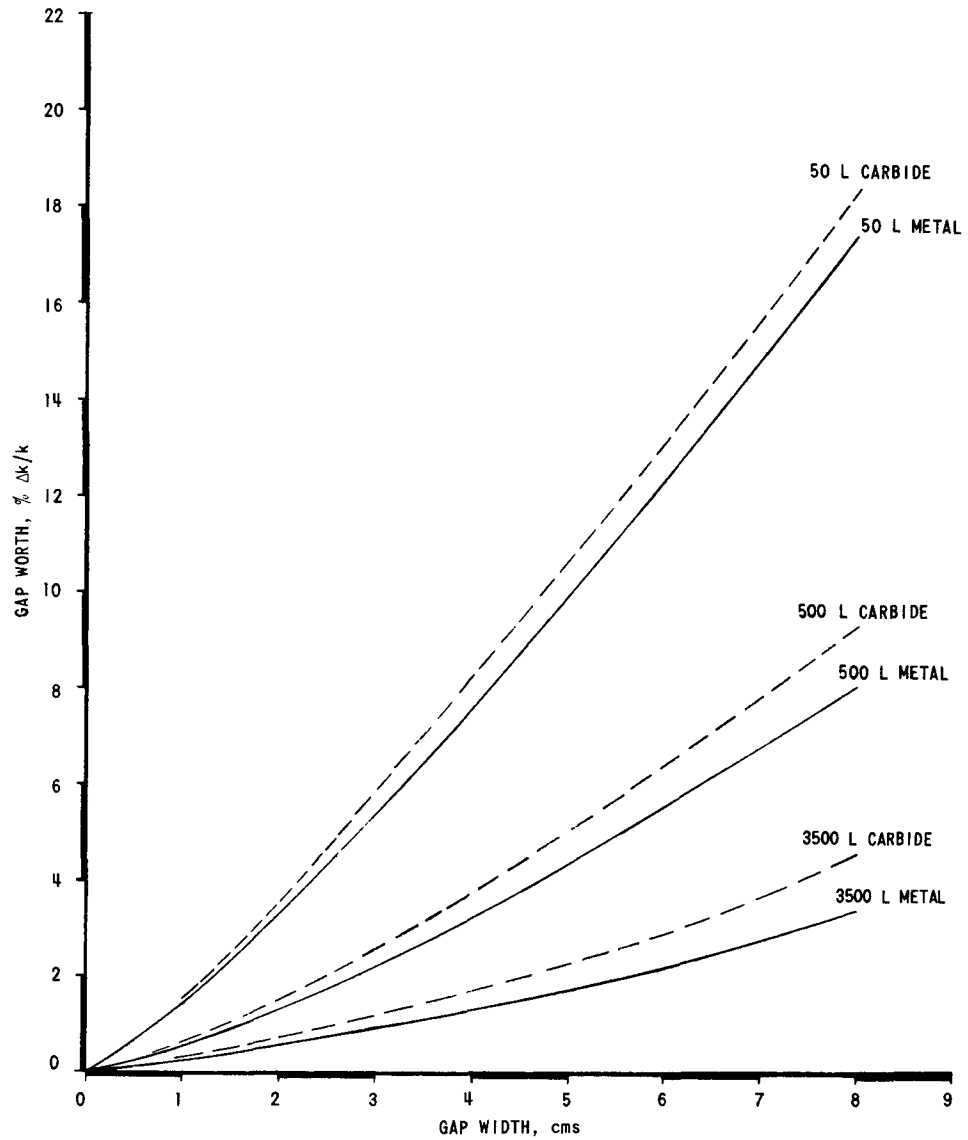


Fig. II-1. Reactivity Loss vs Gap Width

Some check on the accuracy of the gap worth calculations is possible from the gap worth measurements made on ZPR-6 Assemblies 1 and 2 as reported in Refs. 5 and 6. The measurements were made on small gaps (less than a centimeter) and serve as a partial check on the gap worth calculation. Assembly 1 was a 125 liter metal core and Assembly 2 was a 650 liter carbide core. Figure II-2 is a plot of the variation of gap worth with core size for both carbide and metal cores as calculated by Tamor's method for a separation of 0.5 cm. The measured gap worths for the ZPR-6 Assemblies 1 and 2 are shown to agree well both in magnitude and in the predicted variation with core size.

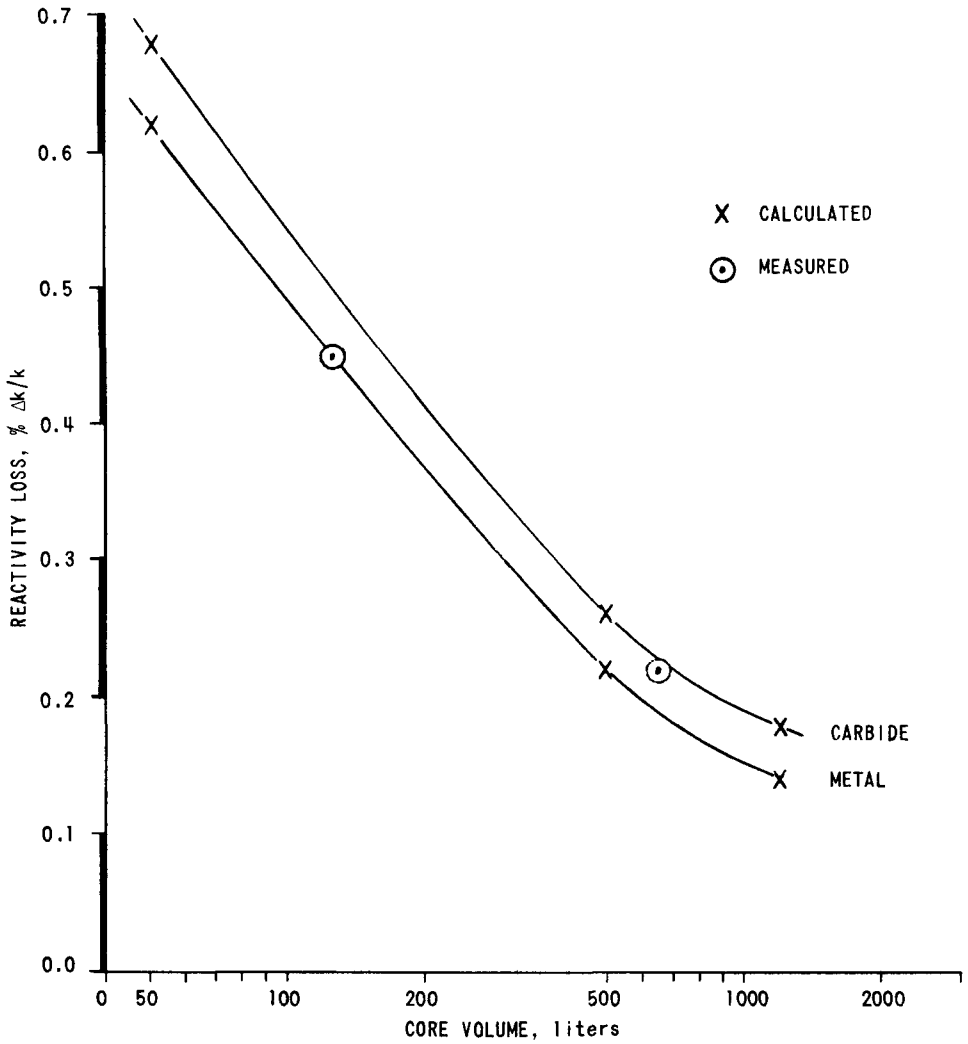


Fig. II-2. 5 mm Gap Worth vs Core Size

f. Expansion Coefficient

Expansion of the fuel into a void space provided at the rear of each of the core drawers gives a prompt negative temperature coefficient that would tend to oppose any excursion. The void space provided in the

nominal 38-cm (15-in.) drawers is 0.48 cm, in the 61-cm (24-in.) drawers is 0.79 cm and in the 81-cm (32-in.) drawers is 1.07 cm. A spring inserted in the void insures that fuel expansion is away from the core center and toward regions of lower importance, reducing reactivity.

The fractional reactivity loss per fractional increase in length ($\Delta k/k \big/ \Delta L/L$) was calculated for two cases. The first case assumed uniform expansion of the fuel plates only, which upon expansion displaced axial blanket material. The second case assumed uniform expansion of all the materials in the core drawer. Both calculations were done in slab geometry. The results for the various reference cores are given in Table II-9.

Table II-9

REACTIVITY COEFFICIENTS FOR UNIFORM EXPANSION

System		($\Delta k/k \big/ \Delta L/L$) for expansion of	
Type	Volume	Fuel Plates Only	All Core Material
Pu Metal	500	-0.38	-0.43
	3500	-0.41	-0.22
Pu Carbide	50	-0.48	-0.63
	500	-0.36	-0.39
	3500	-0.39	-0.25

g. Doppler Effect

The 3:1 ^{238}U to plutonium fuel plate composition insures that the reactivity coefficient of the fuel due to the Doppler effect will always be negative. Further, for the large cores it is of sufficient magnitude that it becomes an important contributor to the overall safety of the system.

The two Doppler coefficients which exist are: the prompt coefficient associated with the fuel plates alone and the slower-acting coefficient associated with the ^{238}U plates making up the balance of the total ^{238}U content of the particular core under study. The coefficient for the fuel plates alone and the total coefficient are both listed in Table II-10. For rapid excursions only the fuel plate component is assumed to be present.

However, after some few seconds, the Doppler component of the depleted fuel plates comes in strongly as a compensating influence on an excursion.

Table II-10

DOPPLER EFFECT T(dk/dT)

System	Fuel Plates* Only Heated	All Materials** Heated
500 ℓ Pu Metal	-0.00029	-0.00091
3500 ℓ Pu Metal	-0.00042	-0.0023
50 ℓ Pu Carbide	-0.00014	-0.00021
500 ℓ Pu Carbide	-0.0015	-0.0027
3500 ℓ Pu Carbide	-0.0025	-0.0086

* Only the U-238 and Pu in the fuel plates were considered for this Doppler effect calculation.

** All materials in the core were assumed to heat uniformly. The Doppler effect of the U-238 in the core was included in the calculation.

The Doppler coefficient calculations were adjusted to give agreement with recent Doppler effect experiments on ZPR-6 (Ref. 3) and ZPR-3 (Ref. 4). The experiments on Assembly 4Z on ZPR-6 (Ref. 3) were used to provide the normalization for the ^{238}U Doppler effect values. Assembly 4Z was a mockup of a 2600-liter carbide, ^{235}U-fueled core. It had a spectrum comparable to the 3500-liter plutonium-fueled carbide core under study here, and had a similar atom density of ^{238}U.

Experiments on plutonium carbide mockups in ZPR-3 (Ref. 4) have shown that the Doppler effect of the plutonium constituent is very small compared to ^{238}U on a per-atom basis. In fact, there is some doubt whether it is even positive, but it is certainly small. Taking into account in the most pessimistic manner the limits of error on these measurements, the maximum conceivable positive plutonium Doppler effect was 10% of the negative ^{238}U Doppler effect in 7:1 ^{238}U to plutonium samples. Therefore, the plutonium Doppler effect was assumed for purposes of these calculations to be positive and to have an absolute magnitude of 70% of the ^{238}U Doppler effect on a per-atom basis.

There is additional evidence that the positive nature of the Doppler effect for other fissile elements is also substantially overestimated by theory. Recent measurements[3] on ZPR-6 indicate that the ^{235}U Doppler effect, while positive, is only about 1/3 to 1/5 magnitude of the negative ^{238}U Doppler effect, on a per-atom basis. Existing cross section sets predict ^{235}U Doppler effects at least double this, so that this lends additional indirect evidence that the calculated Doppler effect for the fissile element plutonium could be substantially overestimated.

REFERENCES

1. D. A. Meneley, L. C. Kvitek, and D. M. O'Shea, "MACH1, A One-dimensional Diffusion-theory Package," ANL-7223 (June 1966).
2. S. Tamor, "The Effect of Gaps on Pile Reactivity," ORNL-1320 (1952).
3. C. E. Till, R. A. Lewis, and E. F. Groh, "Doppler Effect Measurements on a Dilute Fast-Carbide Assembly," Trans. Am. Nucl. Soc. 8(2), 454 (1965).
4. G. J. Fischer et al., "Doppler Effect Measurements in Plutonium-fueled Fast Power Breeder Spectrum," Reactor Physics Division Annual Report, July 1, 1964 to June 30, 1965, ANL-7110, p. 120 (December 1965).
5. G. K. Rusch et al., "Investigations of a 600-liter Uranium Carbide Core (ZPR-6 Assembly No. 2)," Reactor Physics Division Annual Report, July 1, 1963 to June 30, 1964, ANL-7010, p. 91 (January 1965).
6. R. C. Doerner et al., "Physics Measurements in Tungsten Based Aluminum Reflected Fast Reactors," Reactor Physics Division Annual Report, July 1, 1963 to June 30, 1964, ANL-7010, p. 116 (January 1965).

Chapter III

SITE

A. Location

The ZPR-6 and -9 facilities are located in the Reactor Physics Laboratory, Building 316-W near the center of the Argonne National Laboratory Site. The Argonne National Laboratory is located near Lemont in DuPage County Illinois, about twenty-five miles southwest of the Chicago Loop as seen in Fig. III-1. The principal population centers near the Laboratory are communities such as Downers Grove, Westmont, Hinsdale and Lisle north of the Laboratory and Lemont 4.0 km southwest of the Laboratory. The areas south and west are primary rural. The more heavily populated Chicago suburbs border closely on the north and east sections of the Laboratory. The DesPlaines river and the Chicago Sanitary and Ship Canal are located beyond the southern border boundary of the Laboratory Site thus, extending the uninhabited area about 1 km beyond this border. The north, west and east boundaries are approximately 2 km from the Reactor Physics Laboratory. The Reactor Physics Laboratory is located in the 300 area of the Argonne National Laboratory Site as shown in Fig. III-2. Other major reactors such as CP-5, EBWR, and Juggernaut are also located in the same area.

B. Population Distribution

1. ANL Site

At the present time, there are approximately 5200 employees on the 3700 acre Argonne Site. It is estimated that by about 1980, there may be as many as 6400 persons on the Argonne Site. Table III-1 gives a list of on site facilities within 1000 meters of the ZPR-6 and -9 facilities with an estimate of the number of employees in each building during a normal working day.

2. Off-site

a. General

Within a 10 km radius of the reactor site are the communities of Lemont (pop. 3,397), 4 km southwest; Downers Grove (pop. 21,154), Westmont (pop. 5,997), and Clarendon Hills (pop. 5,885), 10 km to the north; Romeoville (pop. 3,574), 9 km to the southwest; and Willow Springs (pop. 2,348), 8 km to the east. The 1960 populations of communities near Argonne are shown in Fig. III-3. Of the 874,366 people shown as living within 24 km (15 miles) of Argonne, 11,912 were within 4.8 km (3 miles), 27,381 were within 8 km (5 miles), and 213,919 were within 16 km (10 miles).

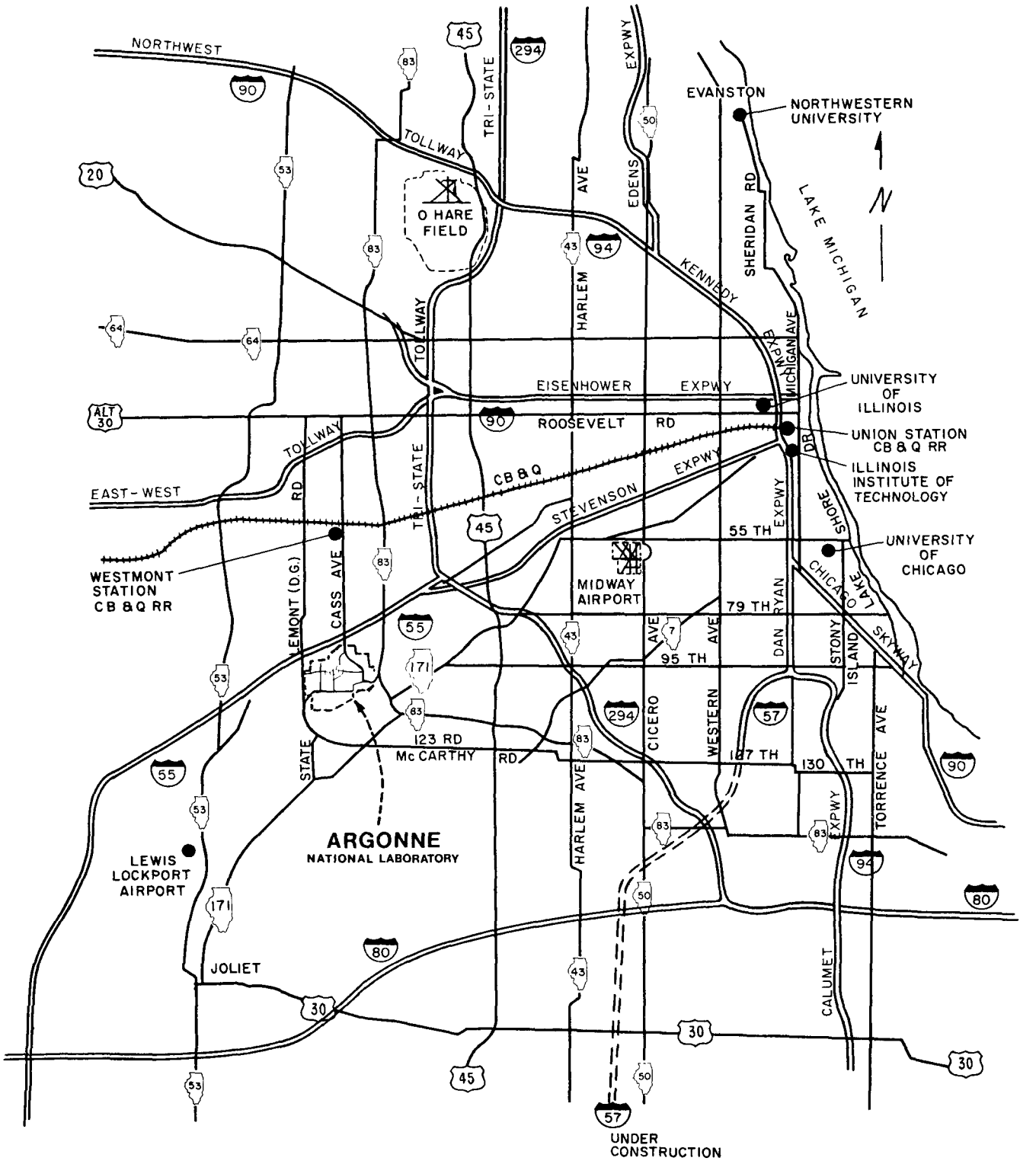


Fig. III-1. Regional Map

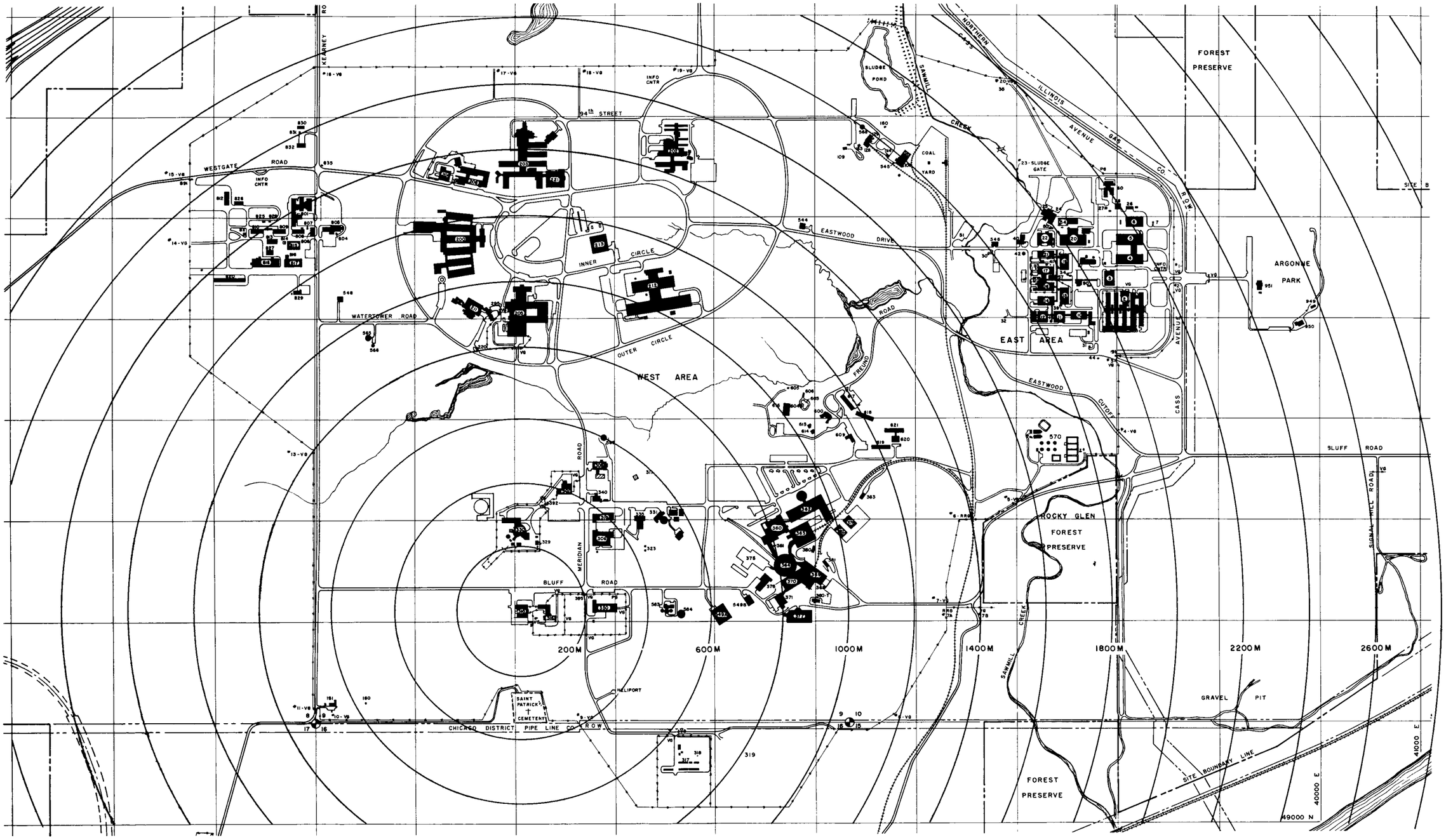
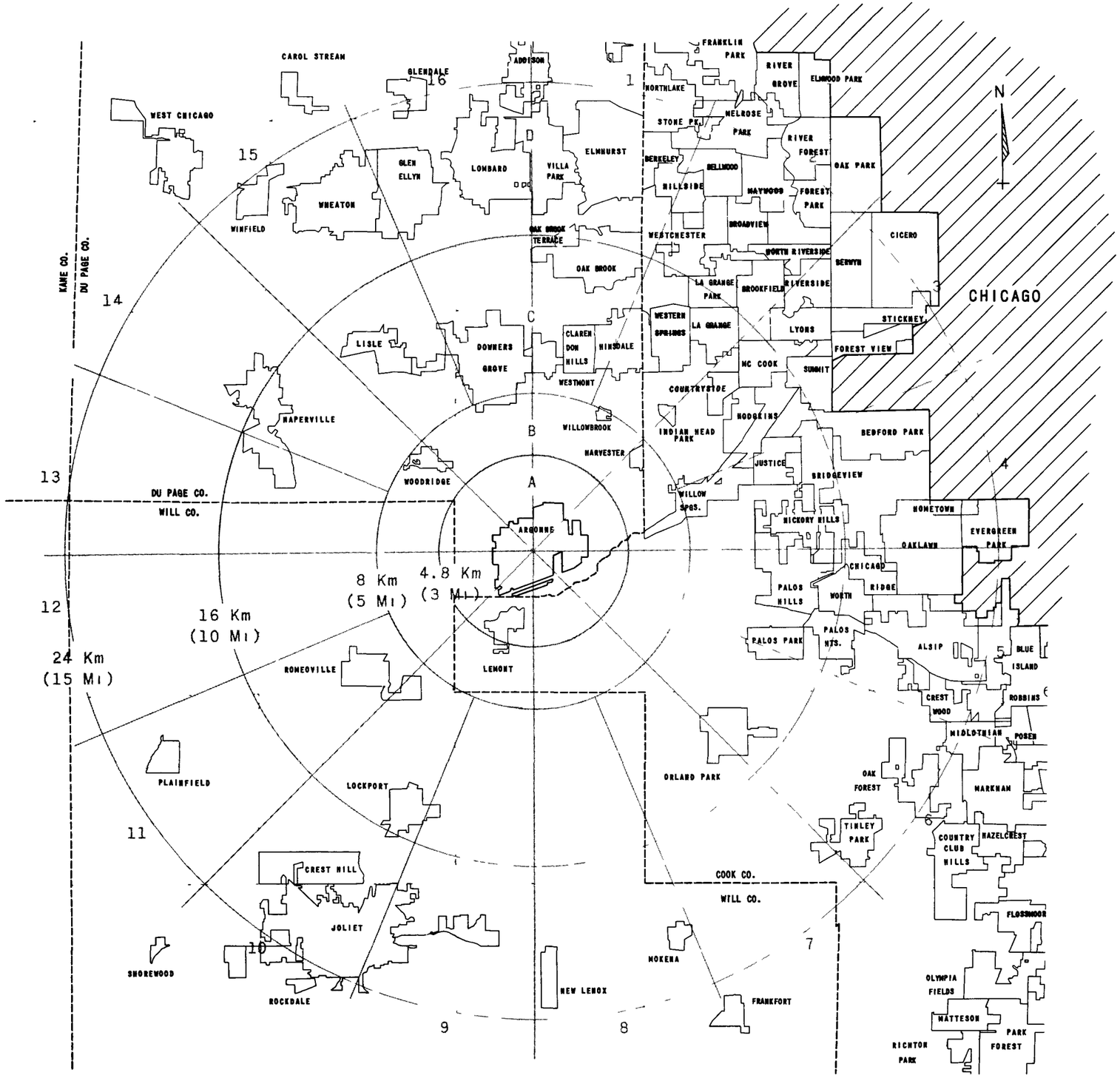


Fig. III-2. Argonne National Laboratory, Illinois Site

TABLE III-1

POPULATION OF NEIGHBORING ON-SITE FACILITIES

Building	Name of Facility	Distance from Reactor Cells	Pop.	Sub-totals
316	Reactor Physics Laboratory (critical facilities)	Within 200 m	70	70
330	Cp-5 Research Reactor	Between 200-400 m	60	
306	Reclamation	Between 200-400 m	40	
310	Chemical Engineering	Between 200-400 m	40	
350	Plutonium Fabrication Facility	Between 200-400 m	30	170
301	Hot Laboratory	Between 400-600 m	25	
335	Juggernaut Research Reactor	Between 400-600 m	20	
331	EBWR Research Reactor	Between 400-600 m	20	
340	Experimental Animal Quarters	Between 400-600 m	10	
308	Sodium Loop Facility	Between 400-600 m	25	100
362	High Energy Physics	Between 600-1000 m	240	
ZGS	ZGS Complex	Between 600-1000 m	485	
205	Chemical Engineering	Between 600-1000 m	290	
212	Fuels Technology Center	Between 600-1000 m	330	
211	Cyclotron Building	Between 600-1000 m	20	1365
Total			1705	



COMMUNITY	POP.	COMMUNITY	POP.
ADDISON	6741	MARKHAM	11704
ALSIP	3770	MATTESON	3225
BEDFORD PARK	737	MAYWOOD	27330
BELLWOOD	20729	McCOOK	441
BERKELEY	5792	MELROSE PARK	22291
BERWYN	54224	MIDLOTHIAN	6605
BLUE ISLAND	19618	MOKENA	1332
BRIDGEVIEW	7334	NAPERVILLE	12933
BROOKFIELD	20429	NEW LENOX	1750
CAROL STREAM	836	NORTHLAKE	12318
CHICAGO	3550404	NORTH RIVERSIDE	7989
CHICAGO RIDGE	5748	OAK BROOK	324
CICERO	69130	OAK BROOK TERRACE	1121
CLARENDON HILLS	5885	OAK FOREST	3724
COUNTRY CLUB HILLS	3421	OAKLAWN	27471
COUNTRYSIDE	2393	OAK PARK	61093
CREST HILL	5887	OLYMPIA FIELDS	1503
CRESTWOOD	1213	ORLAND PARK	2592
DOWNERS GROVE	21154	PALOS HEIGHTS	3775
ELMHURST	36991	PALOS HILLS	3766
ELMWOOD PARK	23866	PALOS PARK	2169
EVERGREEN PARK	24178	PARK FOREST	29993
FLOSSMOOR	4624	PLAINFIELD	2183
FOREST PARK	14452	POSEN	4517
FOREST VIEW	1042	RICHTON PARK	933
FRANKFORT	1135	RIVER GROVE	8464
FRANKLIN PARK	18322	RIVERSIDE	9750
GLENDALE	173	ROBBINS	7511
GLEN ELLYN	15972	ROCKDALE	1272
HARVESTER	299	ROMEOVILLE	3574
HAZELCREST	6205	SHOREWOOD	358
HICKORY HILLS	2707	STICKNEY	6239
HILLSIDE	7794	STONE PARK	3038
HINSDALE	12859	SUMMIT	10374
HODGKINS	1126	TINLEY PARK	6392
HOMETOWN	7479	VILLA PARK	20391
INDIAN HEAD PARK	385	WESTCHESTER	18092
JOLIET	66780	WEST CHICAGO	6854
JUSTICE	2803	WESTERN SPRINGS	10838
LA GRANGE	15285	WESTMONT	5997
LA GRANGE PARK	13793	WHEATON	24312
LEMONT	3397	WILLOW BROOK	157
LISLE	4219	WILLOW SPRINGS	2348
LOCKPORT	7560	WINFIELD	1575
LOMBARD	22561	WOODRIDGE	542
LYONS	9936	WORTH	8196

Fig. III-3

Population of Communities near
Argonne National Laboratory
(Based on 1960 Census)

Although the nearest ANL Site boundary is about 1.5 km from the reactor, the area between this boundary and the 2-km-radius circle is relatively uninhabited. To the south the DesPlaines River, the Chicago Sanitary and Ship Canal, and The Atchison, Topeka and Santa Fe railroad pass through this area.

b. Present Population

A map of the metropolitan area west of Chicago showing the population of various villages and cities with a 24 km radius is shown in Fig. III-3. A "Township, Area and Population Breakdown," Table III-2,

Table III-2

TOWNSHIP, AREA, AND POPULATION BREAKDOWN

Township	Population			Area (sq km)			Persons per sq km (Rural)
	Total	City	Rural	Total	City	Rural	
Addison	41,808	27,514	14,294	93	26	67	212
Berwyn	54,224	54,224	-	9.6	9.6	-	-
Bloom	70,530	64,007	6,523	93	31	62	105
Bloomingdale	14,924	5,852	9,072	93	11.4	81.6	111
Bremen	55,392	44,626	10,766	93	45.8	47.2	227
Calumet	19,299	19,299	-	10	10	-	-
Cicero	69,130	69,130	-	14.8	14.8	-	-
Downers Grove	66,664	44,298	22,366	124	32	92	244
Du Page	4,725	3,470	1,255	93	5	88	14
Frankfort	5,784	2,467	3,317	93	4 .	89	37
Homer	4,078	-	4,078	93	-	93	44
Joliet	94,116	67,755	26,361	93	31	62	424
Lemont	6,732	3,397	3,335	54.4	3.1	51.3	65
Lisle	20,982	12,847	8,135	93	17	76	106
Lockport	26,882	13,551	13,331	93	14	79	170
Lyons	82,214	65,862	16,352	95	69	26	619
Milton	51,361	40,483	10,878	93	28	65	168
Naperville	8,218	5,668	2,550	93	5	89	29
New Lenox	6,232	1,750	4,482	93	7	86	52
Orland	7,444	3,227	4,217	93	9	84	50
Palos	17,728	13,886	3,842	91	29	62	62
Plainfield	6,655	2,183	4,472	93	3	90	49
Proviso	160,275	157,891	2,384	77	68	9	256
Rich	35,258	32,389	2,869	93	31	62	46
Riverside	17,875	17,732	143	10	9	1	27
Stickney	31,404	7,327	24,077	47	40	7	3321
Thornton	138,444	135,613	2,831	93	72	21	137
Troy	2,679	655	2,024	93	3	90	22
Wheatland	1,023	-	1,023	93	-	93	11
Winfield	16,437	11,531	4,906	93	10	83	59
Worth	107,761	91,988	15,773	96	83.4	12.4	1269
York	89,988	76,218	13,770	93	54	39	354

is also included, for more comprehensive information. In the table, townships which fall entirely or partly within the 24 km radius are listed alphabetically. For each township the following figures are shown: total, city, and rural population; total, city, and rural area; and persons per sq km in the rural area. The total population was taken from the 1960 Census report. The city population is a total of all city populations, whole or part, listed for that township in the census report. The rural population is the difference between city population and total population. The area figures were obtained by measuring with the use of the map scale of miles and a planimeter. The number of persons per sq km was computed by dividing rural population by rural area.

The area within the 24 km radius was divided into segments by drawing in circles and radii, with Building 316-W as a center point. The radii of the circles indicate the distance in miles from the center point. These distance zones are identified by the letters A, B, C, and D, starting at the center. The radius of the A zone is 4.8 km, the B zone 8 km, the C zone 16 km, and the D zone 24 km. The A and B zones are divided by the radii into 8 equal segments, respectively, and the C and D zones into 16 equal segments, respectively. The area of each segment in the A zone is 9.1 sq km, in the B zone 16.3 sq km, in the C zone 38.1 sq km, and in the D zone 63.4 sq km. The segments are numbered 1 through 16, starting at the top or north, and reading clockwise around the circumference. Each segment is identified by using the zone letter and the number of the segment. For example, the first segment in the inner circle of A zone reading clockwise from the north would be A-2, the first segment between the inner and the second circle would be B-2, the first segment between the second and third circles C-1, and the first segment between the third and fourth circles, D-1. The population of part of a city, in those cases where a city is divided into 2 or more segments, was determined by estimating the percentage of the population of the city for each segment and using the same percentage of the population of the city for each segment into which it is divided, except for the City of Chicago. The population figures for that part of Chicago which falls within the 24 km radius were taken from an Official One Mile Intersection Map prepared by the Chicago Civil Defense Corps, which lists population per square mile as received from the U.S. Bureau of the Census.

The "Segment Area and Population Breakdown" as shown in Table III-3 was prepared as follows: The segments are listed numerically starting with the A zone, followed by the B, C, and D zones. For each segment, rural area and population, city area and population, and total area and population are listed. The rural area of each segment was measured with a planimeter and subtracted from the segment total to obtain the city area in those segments having both rural and city area. The rural population was computed by multiplying the rural area by the persons per sq km in the rural area. The city population per segment is the sum of the population figures of those cities within the segment.

Table III-3

SEGMENT AREA AND POPULATION BREAKDOWN

Segment No.	Rural		City		Total	
	Area	Population	Area	Population	Area	Population
A-2	5.2	1,264	3.9*	-	9.1	1,264***
A-4	5.7	1,297	3.4*	-	9.1	1,297***
A-6	7.2	842	1.8*	-	9.1	842***
A-8	8.0	567	1.0*	-	9.1	567***
A-10	5.2	336	3.9**	3,125	9.1	3,461***
A-12	8.0	1,165	1.0*	-	9.1	1,165***
A-14	7.8	1,420	1.3*	-	9.1	1,420***
A-16	7.8	1,896	1.3*	-	9.1	1,896***
B-2	15.8	3,855	0.5	124	16.3	3,979
B-4	12.9	3,187	3.4	676	16.3	3,863
B-6	16.3	1,032	-	-	16.3	1,032
B-8	16.3	987	-	-	16.3	987
B-10	14.5	681	1.8	272	16.3	953
B-12	16.3	233	-	-	16.3	233
B-14	14.5	683	1.8	336	16.3	1,019
B-16	15.5	3,197	0.8	206	16.3	3,403
C-1	18.9	5,557	19.2	16,567	38.1	22,124
C-2	12.9	6,073	25.1	36,644	38.1	42,717
C-3	18.1	11,221	18.1	6,157	38.1	17,378
C-4	13.2	2,980	24.9	9,707	38.1	12,687
C-5	21.2	1,312	16.8	10,623	38.1	11,935
C-6	31.8	1,824	6.2	2,381	38.1	4,205
C-7	34.2	1,655	3.9	363	38.1	2,018
C-8	38.1	1,661	-	-	38.1	1,661
C-9	37.5	2,453	0.5	1,083	38.1	3,536
C-10	32.6	5,313	5.4	7,156	38.1	12,469
C-11	31.8	2,988	6.2	2,895	38.1	5,883
C-12	38.1	510	-	-	38.1	510
C-13	34.7	1,067	3.4	3,621	38.1	4,688
C-14	31.8	3,322	6.2	5,820	38.1	9,142
C-15	32.3	3,936	5.7	5,700	38.1	9,636
C-16	23.3	6,822	14.8	19,127	38.1	25,949
D-1	17.9	4,660	45.6	56,174	63.4	60,834
D-2	7.2	1,853	56.2	125,484	63.4	127,337
D-3	1.3	2,201	62.1	92,465	63.4	94,666
D-4	16.1	42,687	47.3	67,678	63.4	110,365
D-5	16.1	17,050	47.3	42,258	63.4	59,308
D-6	46.6	7,149	16.8	10,313	63.4	17,462
D-7	60.6	2,628	2.8	650	63.4	3,278
D-8	61.6	3,556	1.8	2,816	63.4	6,372
D-9	52.6	10,688	10.9	21,566	63.4	32,254
D-10	33.1	8,704	30.3	47,958	63.4	56,662
D-11	60.8	3,661	2.6	2,183	63.4	5,844
D-12	63.4	908	-	-	63.4	908
D-13	63.4	1,423	-	-	63.4	1,423
D-14	59.5	2,639	3.9	3,492	63.4	6,131
D-15	44.8	6,551	18.6	26,250	63.4	32,801
D-16	31.6	7,302	31.8	37,500	63.4	44,802
				TOTAL	1827.8	874,366
				ANL		4,000
				GRAND TOTAL		878,366

*ANL Exclusion Area

**ANL = 0.5; Lemont = 1.0

***Does not include exclusion area population

c. Future Population

The projected population is based upon information received from, and consultation with the Northeastern Illinois Metropolitan Area Planning Commission, utility companies and local governmental agencies. Figure III-4 entitled Argonne National Laboratory and the Immediate Surrounding Area, has been divided into four quadrants and at radii of 2.5 and 7 km. The present, and estimated populations in 1970 and 1980 for each of the twelve areas are as follows:

	Off-site Population		
Area Number	1960	1970	1980
0-2 km radius			
A-1 (Northeast)	-	-	-
A-2 (Southeast)	-	-	-
A-3 (Southwest)	140	160	170
A-4 (Northwest)	495	620	795
Total	635	780	965
2-5 km radius			
B-1 (Northeast)	2,561	3,329	4,328
B-2 (Southeast)	1,409	1,691	2,029
B-3 (Southwest)	4,626	5,551	6,661
B-4 (Northwest)	3,316	4,642	6,499
Total	11,912	15,213	19,517
5-7 km radius			
C-1 (Northeast)	7,520	13,723	24,315
C-2 (Southeast)	2,019	3,029	4,543
C-3 (Southwest)	1,186	1,660	2,324
C-4 (Northwest)	4,150	7,075	11,320
Total	14,875	25,487	42,502

C. Meteorology

1. General

A summary of meteorological data collected at Argonne is as follows:

(a) At 19 ft above ground, the daily average wind speed varies from 5 mph in August to 10 mph in March and gusts exceed 50 mph about once a year. At 150 ft above ground, this speed is exceeded about 15 times a year. No air measurements above 150 ft, without the aid of balloons, are available at Argonne.

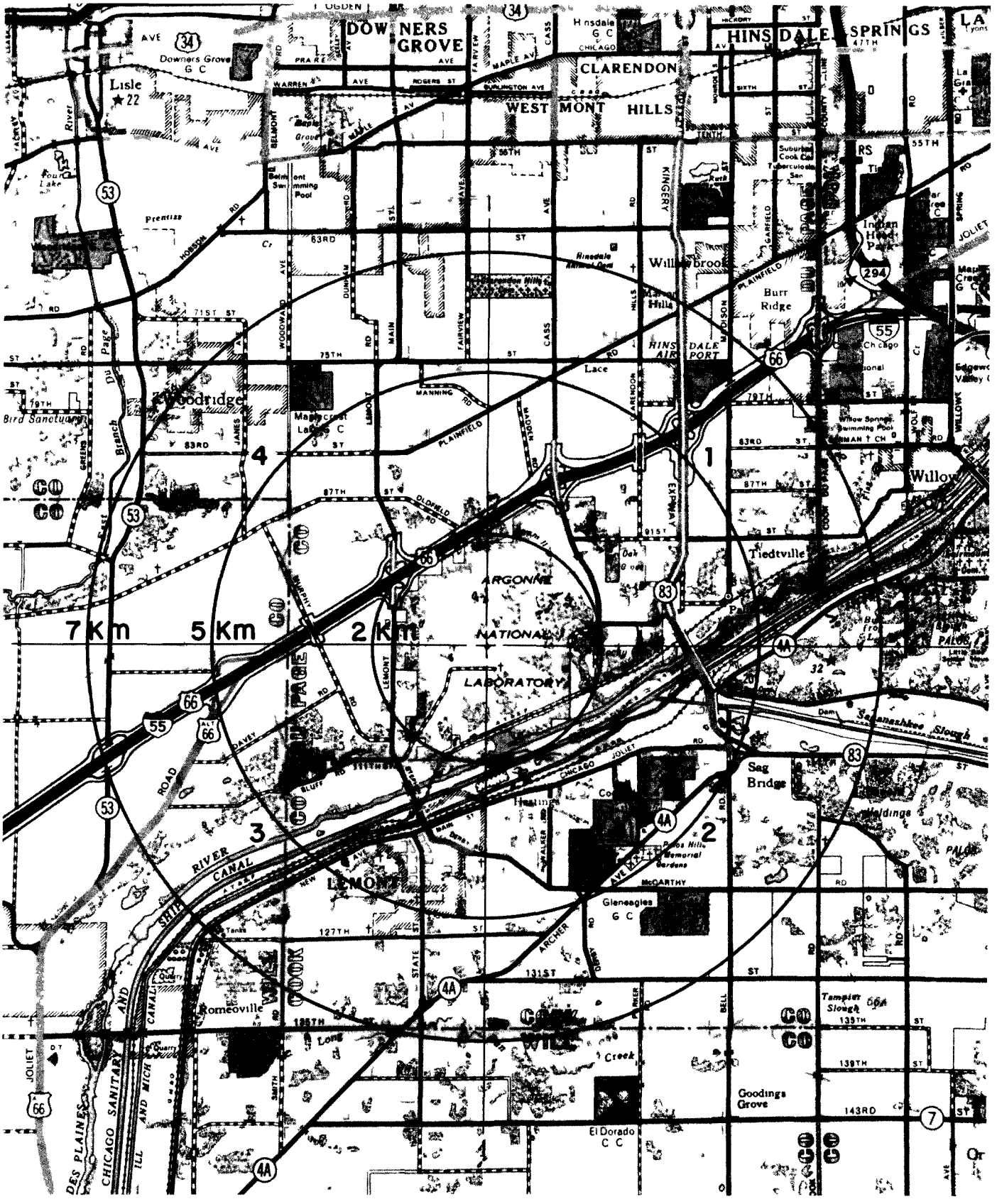


Fig. III-4. Argonne National Laboratory and the Immediate Surrounding Area

(b) Daily temperatures average from the low 20's in the winter to the high 70's in the summer. There are about 6850 degree-days per year. Rarely are there maximum temperatures above 100°F, and the minimum temperature recorded is -19°F.

(c) Precipitation averages about 33 in. per year. Mid-day humidity values average about 50% in October to 75% in January; however, night readings are about 80 to 90% all year.

(d) Solar radiation, in Langleys on a horizontal surface, varies from about 150 in December to 600 per day in June.

(e) Atmospheric pressure varies between a high of 30.00 in. Hg and a low of 28.30 in. Hg with a mean of about 29.20 in. of mercury.

(f) Soil temperatures three or four feet below grade vary only from 50°F in July to 52°F in February. Temperature at depths of four inches vary from the 80's in the summer to just below freezing in mid-winter.

Annual climatological summaries based on measurements taken at the DuPage Site of the Argonne National Laboratory have been prepared and published as reports designated as ANL-4538, ANL-4793, ANL-4928, ANL-5256, and ANL-5592. These reports cover the period from July 1949 to June 1954. The normals, means, and extremes for the Chicago Area, as furnished by the United States Weather Bureau, are shown in Table III-4.

D. Geology and Hydrology

1. General Topography

The topography of the vicinity in general is flat to slightly hilly with an elevation of between 700 and 750 ft above sea level for the major portion of the Argonne Site. The land for several miles north and east has an elevation in this same range but drops slowly toward the southwest. An intermittent stream, Sawmill Creek, crosses Argonne and empties into the DesPlaines River which borders the southern edge of the site. The normal level of the DesPlaines River adjacent to Argonne is 583 ft above sea level, and its flood plain extends approximately 1/8 mile onto Argonne property. The land rises sharply from the edge of this flood plain, where outcroppings of the Niagara dolomite may be seen, to the top of the bluffs overlooking the DesPlaines valley. These bluffs are approximately 650 ft above sea level. From here the land slopes gradually upward for nearly a mile to the immediate area of the Reactor Physics Laboratory where ZPR-6 and -9 are located which is at elevation 745 ft above sea level.

Table III-4

METEOROLOGY

1a

TEMPERATURES FOR CHICAGO AREA

Month	Temperature: Normal: Daily Maximum, °C	Daily Minimum, °C	Monthly, °C	Extremes: Record Highest, °C	Year	Record Lowest, °C	Year
(a)	(b)	(b)	(b)	17		17	
J	0.39	- 8.28	- 3.94	19.4	1950	-26.1	1951
F	1.67	- 6.78	- 2.56	20.6	1954	-26.1	1951
M	7.22	- 1.67	2.78	27.8	1945	-21.7	1943
A	14.2	3.67	8.94	28.9	1958	- 7.22	1954
M	20.9	9.28	15.1	34.4	1949	- 1.11	1947
J	26.7	14.9	20.8	40.0	1953	1.67	1945
J	29.6	17.7	23.7	39.4	1956	9.44	1947
A	28.3	16.7	22.6	38.3	1947	7.78	1950
S	24.4	12.9	18.7	38.3	1953	2.22	1943
O	18.0	6.61	12.3	32.8	1954	- 6.66	1948
N	8.67	- .389	4.17	27.2	1950	-18.9	1950
D	1.83	- 6.34	- 2.22	18.3	1951	-24.4	1951
Yr	15.2	4.89	10.1	40.0	June 1953	-26.1	Feb 1951

1b

PRECIPITATION FOR CHICAGO AREA

Month	Precipitation: Normal Total, cm	Maximum Monthly, cm	Year	Minimum Monthly cm	Year	Maximum in 24 hr, cm	Year
	(b)	17		17		17	
J	4.67	9.58	1950	0.965	1956	3.51	1950
F	3.57	8.51	1950	0.838	1958	3.18	1949
M	7.24	12.7	1954	0.838	1958	6.35	1948
A	7.16	21.2	1947	3.25	1946	10.4	1947
M	9.30	19.3	1945	1.98	1950	7.44	1951
J	10.5	16.3	1958	1.98	1956	11.6	1959
J	6.94	22.8	1957	3.38	1945	15.8	1957
A	8.10	15.0	1954	2.54	1953	7.90	1955
S	8.20	15.3	1945	1.17	1956	6.48	1951
O	6.50	30.6	1954	0.762	1956	14.3	1954
N	5.92	9.50	1951	2.26	1949	4.67	1959
D	4.95	16.9	1949	0.864	1943	6.04	1949
Yr	83.1	30.6	Oct 1934	0.762	Oct 1956	15.8	July 1957

1c

PRECIPITATION FOR CHICAGO AREA

Month	Snow, Sleet: Mean Total, cm	Maximum Monthly, cm	Year	Maximum in 24 hr, cm	Year
	17	17		17	
J	19.6	52.8	1943	16.8	1957
F	16.5	42.2	1950	21.1	1955
M	15.2	50.6	1954	30.0	1954
A	1.27	6.10	1951	6.10	1951
M	T	0.51	1954	0.51	1954
J	0	0		0	
J	0	0		0	
A	0	0		0	
S	0	0		0	
O	76	7.62	1952	7.62	1952
N	9.14	36.3	1951	20.3	1951
D	25.9	84.6	1951	25.4	1951
Yr	88.4	84.6	Dec 1951	30.0	Mar 1954

1d

WIND FOR CHICAGO AREA

Month	Wind: Mean Hourly Speed (m/sec)	Prevailing Direction	Fastest Speed: Speed (m/sec)	Direction	Year
	17	15	17		
J	4.87	W	22.4	W	1950
F	5.00	W	20.1	W	1951
M	5.14	W	24.1	NW	1955
A	5.00	ENE	22.4	NW	1951
M	4.47	NE	24.1	S	1950
J	3.93	SSW	22.4	W	1953
J	3.44	SSW	20.6	NW	1959
A	3.35	SW	24.1	NW	1949
S	3.84	S	21.5	SW	1959
O	4.16	S	20.1	S	1949
N	5.05	W	26.8	SW	1952
D	4.83	W	22.4	SW	1948
Yr	4.43	SSW	26.8	SW	Nov 1952

(a) Length of record, years.

(b) Normal values are based on the period 1921-1950 and are means adjusted to represent observations taken at the present standard location.

2. Bedrock

The average elevation of the bedrock directly beneath the ZPR-6 and -9 Site is approximately 610 ft above sea level. The Argonne deep well No. 1, drilled to a depth of 1595 ft, reveals stone formations and their thicknesses as depicted in Fig. III-5. Preglacial erosion has worn off bedrock of more recent origin and has cut deeply into the dolomitic top surface of the presently existing bedrock in the DesPlaines River valley.

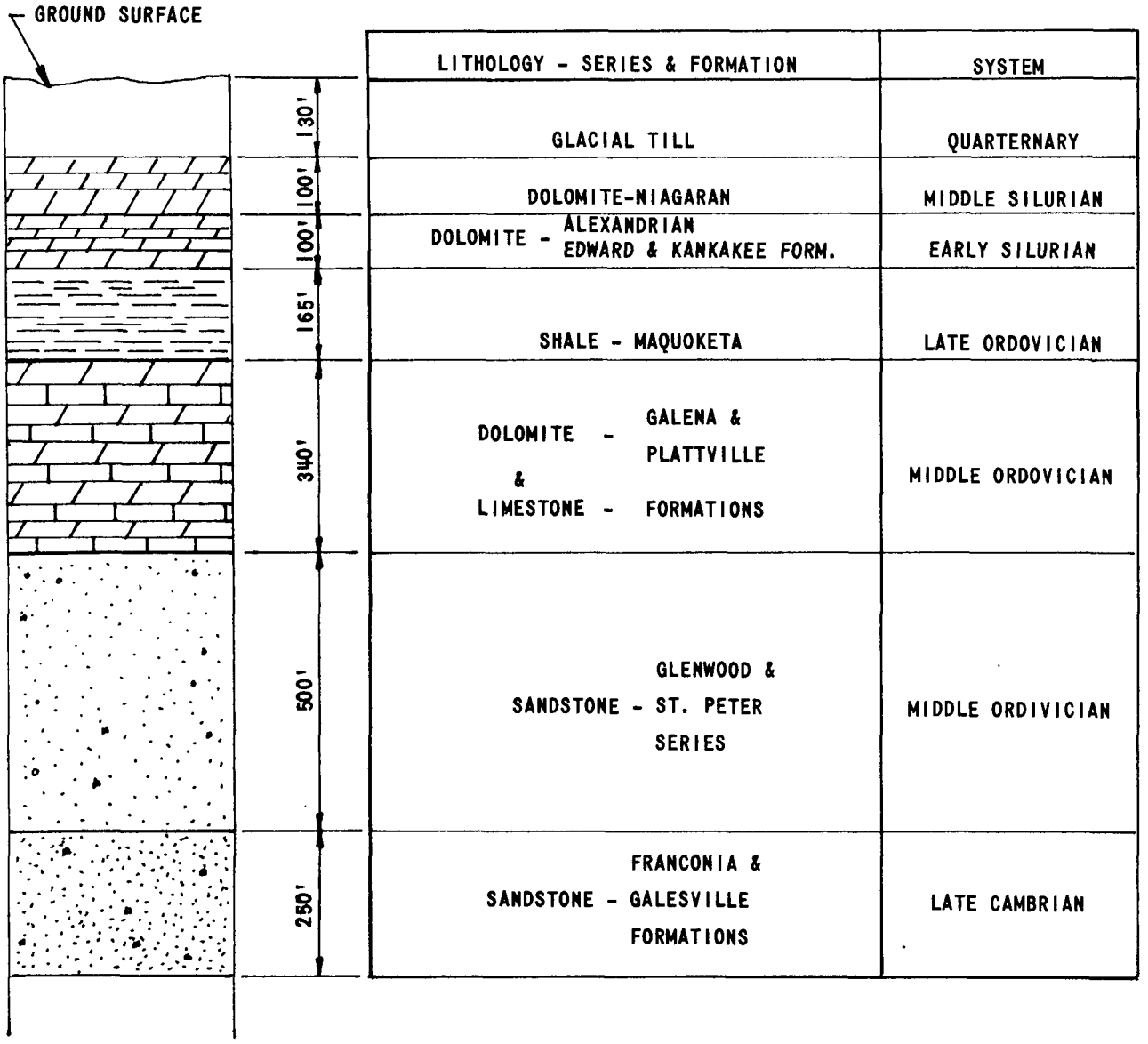


Fig. III-5. Bedrock near ZPR-6 and -9 Site

Glacial till of the Wisconsin ice sheet deposited by the Valparaiso moraine overlies this bedrock and partially fills the DesPlaines River valley.

3. Subsurface Soils

The soil residing above the bedrock is composed of dense and very tough silty clays and clay-like silts. There are some traces of sand. These clays were laid down, as mentioned above, as the glacial drift of the Valparaiso moraine.

4. Hydrology

Due to the tight nature of the soil, percolation of water through it is very slow. Therefore, drainage from the ZPR-6 and -9 area will be mostly by surface runoff. Potable water for the Laboratory is obtained from two principal aquifiers, the Niagara dolomite and the deeply buried Ordivician and Cambrian sandstones. The Laboratory has five shallow supply wells that obtain water from the Niagara dolomite, and one deep well, cased and sealed through the Niagara dolomite, that obtains its water from the deep sandstones. Only the Niagara dolomite need be considered in connection with possible contamination from an accidental spill of radioactive liquid, as the deep aquifiers are separated from the surface by hundreds of feet of rock formations (see Fig. III-5), of which the Maquoketa shale is particularly impermeable. In the event of such a spill circumventing protective measures provided, the contaminated liquid would move very slowly through the existing glacial till and would be diluted by the normal ground water descending with it. As the diluted liquid approached the level of the bedrock it would be further diluted by water that is moving across and down the fissured dolomite. From this point the hydraulic gradient of the Niagara dolomite, which is generally to the south-eastward, would be followed (see Fig. III-6).

Tests of water movement through the glacial till at the Laboratory were made in 1952 by W. J. Drescher of the U.S. Geological Survey.[1] These studies determined that the downward rate of movement of water through the till is 0.1 in. per day. At that rate it would take approximately 36 years for any contaminated water entering the surface soil at the Reactor Physics Laboratory to reach the bedrock layer of Niagara dolomite directly beneath the site.

At the present rate of pumping it is very likely that contaminated water introduced into the glacial till at the Reactor Physics Laboratory would decay and partial soil retention through leaching and/or ion exchange by the natural soil would certainly take place in the long interval of time between insertion at the ZPR-6 and -9 Site and ultimate discharge into the DesPlaines River.

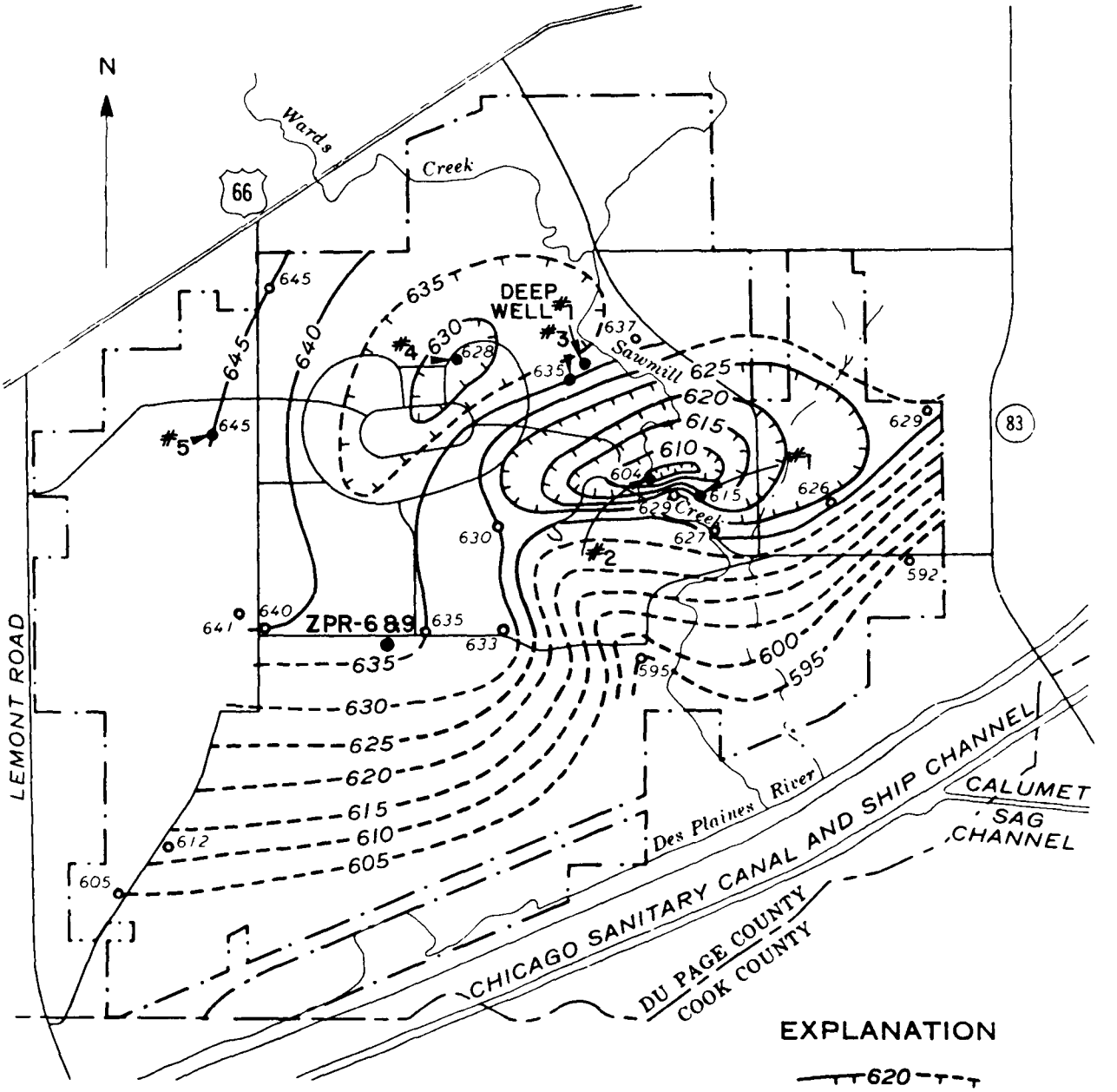


Fig. III-6. Piezometric Surface of Water in the Niagara Dolomite at Argonne National Laboratory, Illinois, June 2-3, 1960

The DesPlaines River, the Chicago Sanitary and Ship Canal, the Illinois and Michigan Canal, and the Sag Channel have all been interconnected by the time they reach Joliet, which lies about 22 km southwest of Argonne National Laboratory. The water flow in the DesPlaines River just prior to Joliet varies between 12 and 340 m^3/sec, depending upon the flood stage and season. When the river is in flood stage, excess water is diverted to the higher capacity Sanitary Canal at a point just south of Lemont, before reaching Joliet. The flow rate of the Sanitary Canal for 1944 and 1945 averages about 96 m^3/sec; of this about 45 m^3/sec is water which has been diverted from Lake Michigan, and the balance is made up of domestic and industrial discharge from this district. Both the river and the canal flow a distance of approximately 32 km/day. Therefore, any surface drainage from the ZPR-6 and -9 Site would be diluted first by the river bordering the Laboratory, and then further diluted by the canal, normally within 17 hours. If such drainage occurred during a wet period, it would be diluted even more extensively and sooner.

Canal water is used prior to and in Joliet for some industrial uses, such as hydroelectric generators and condensers, as well as for irrigation at the state prison. After leaving Joliet, the DesPlaines River flows southwest to a point approximately 23 km further, where the river is joined by the Kankakee River, forming the Illinois River. The first use made of this water for drinking purposes is at Alton, Illinois, which is about 442 km from Argonne National Laboratory and 16 km downstream from the point where the Illinois River merges with the Mississippi River.

E. Seismology

Realistic predictions regarding earthquakes, or earth shocks, as well as their frequency and severity, can only be based upon the seismic history of the area concerned and the statistical representation which history provides. Significant geological features, such as known slip-planes or faults, pertinent to an area play an important role in the seismic history of that area and are, therefore, to be considered as factors in the prediction of an earthquake.

There is one known fault zone in the area. This fault, known as the "Sandwich" fault (passing through Sandwich, Illinois), strikes approximately S 60° E and is located approximately 33 km (20 miles) from the site as measured along a line normal to the fault and bearing from the site approximately S 30° W. It is uncertain what the dip of this fault may be. It is estimated that the major motion along this fault occurred during the Palezoic era and sometime before the Pennsylvanian period ended and after the Silurian period. (Some estimate the fault was formed due to the uplift of the LaSalle anticline in northern Illinois after the Mississippian period.) There may be other faults paralleling and associated with the Sandwich fault.[2] These fault indications, which lie about 3.3 km (2 miles) north of the known fault, would broaden the zone of faulting.

A small area (of about 25 square miles) of severe faulting is located approximately 36 km (22 miles) north of the site and has been mapped as the "DesPlaines Disturbance."[3] Because of the number, severity and local nature of these faults (tilted blocks and broken rock to depths of 1600 feet) conclusions made from detailed studies of this area are that this chaotic disturbance was the result of the focusing of regional forces or the result of the impact of a large meteorite.[4]

A chronological list of the earthquakes with epicenters within a 600 km radius of the ZPR-6 and -9 Site and with a probable intensity of III (as evaluated in terms of the Modified Mercalli Intensity Scale of 1931) or greater, felt at the ZPR-6 and -9 Site is as follows:[5]

Map Location Number	Date	Epicenter	Extent of Tremor in Square Miles	Maximum Intensity at Center	Estimated Probable Intensity at Site
1	8/20/1804	Chicago, Illinois	30,000	-	-
2	12/16/1811	New Madrid, Missouri	2,000,000	X	IV
2	1/23/1812	New Madrid, Missouri	2,000,000	X	IV
2	2/7/1812	New Madrid, Missouri	2,000,000	X	IV
3	10/31/1895	Charleston, Missouri	1,000,000	VIII	IV
4	5/26/1909	Aurora, Illinois	500,000	VIII	VII
5	7/18/1909	Central Illinois	40,000	VII	III
6	1/2/1912	Between Aurora and Morris, Illinois	40,000	VI*	VI
7	4/9/1917	DeSoto, Missouri	200,000	VI	IV
8	3/8/1937	Lima, Ohio	150,000	VII-VIII	III
9	11/23/1939	Waterloo, Illinois	150,000	V	II-III
10	8/9/1947	Battle Creek, Michigan	50,000	VI	III

Other strong tremors felt in the area, but with epicenters greater than 600 km away, are as follows:

Date	Epicenter	Extent of Tremor in Square Miles	Maximum Intensity at Center	Estimated Probable Intensity at Site
8/31/1886	Charleston, South Carolina	2,000,000	X	III
2/28/1925	Quebec, Canada	2,000,000	VIII	III

Figure III-7 shows the relative epicenter locations of the various earthquakes that have been experienced at the ZPR-6 and -9 Site (Intensity III or more). None of the earthquakes shown can be considered as having a damaging effect on structures of adequate design and located at the site.

*0.07 g maximum for a period ranging between 0.3 to 0.6 seconds.[6]

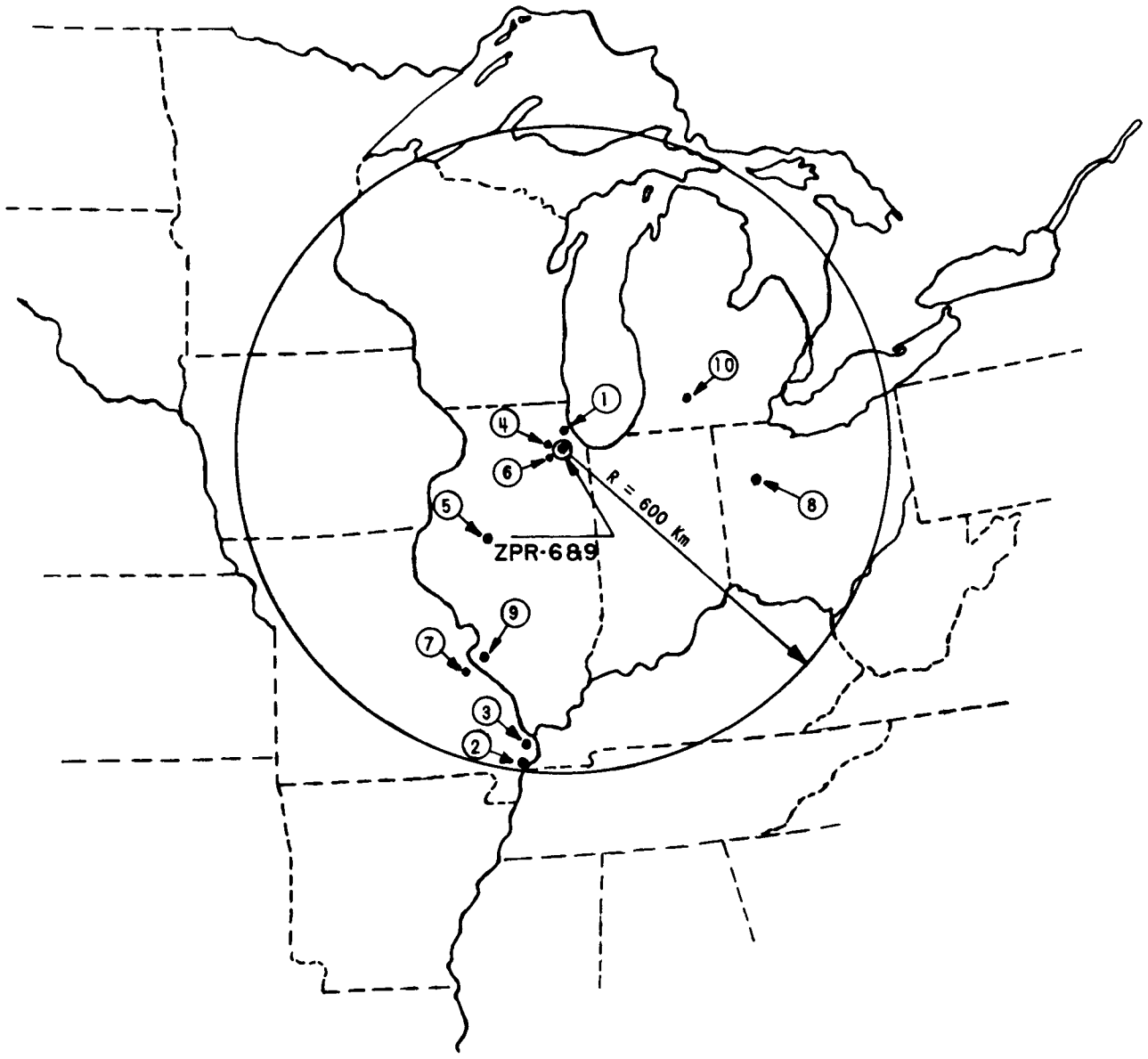


Fig. III-7. ZPR-6 and -9 Proximal Earthquakes

REFERENCES

1. Knowles, Drescher, LeRoux, "Ground-Water Conditions at Argonne National Laboratory," Illinois 1948-1960 Geological Survey Water-Supply Paper 1699-0.

2. Buschbach, T. C., "Cambrian and Ordovician Strata of Northeastern Illinois," Illinois State Geological Survey Report of Investigations 218.

3. Longwell, C. R., et al., "Tectonic Map of the United States," American Association of Petroleum Geologists.

4. Emrich, G. H., and Bergstrom, R. E., "DesPlaines Disturbance, Northeastern Illinois," Geological Society of America Bulletin V.73.

5. "Earthquake History of the United States--Part I," U.S. Dept. of Commerce, Coast and Geodetic Survey, Washington.

6. Seismology Division, Coast and Geodetic Survey, U.S. Dept. of Commerce, Environmental Science Services Administration, "Report of Seismicity of the Grundy County, Morris, Illinois Area" (Sept. 17, 1965). (Report compiled for construction permit and operating license for Dresden Nuclear Power Station Unit #2.)

Chapter IV

DESCRIPTION OF FACILITY

A. General Description

The ZPR-6 and -9 facilities[1-5] are split-table type critical assemblies consisting of one stationary and one movable table as shown in Fig. IV-1. A schematic diagram of the facility showing details is given in Fig. IV-2. The two tables are 3.7 meters (12 ft) wide and 2.4 meters (8 ft) long which rests on a cast steel bed. On each table are stacked horizontally a 45 row by 45 column array of 5.5 cm (2 in.) square stainless steel tubing of 1.2 meters (4 ft) forming half of a 2.4 meters (8 ft) cube reactor. The tubes are constrained in position by massive supports on both sides. Fissile material such as enriched uranium, plutonium, and materials such as depleted uranium, stainless steel, aluminum, zirconium, graphite and oxides of heavy metals are available to simulate various reactor compositions. These materials are loaded into drawers 5 x 5 cm of various lengths such as 38 cm (15 in.), 61 cm (24 in.) or 81 cm (32 in.) for insertion from the interface into the matrix tubes. Each half of the reactor has five fuel-bearing dual-purpose control/safety rods. A dual-purpose safety rod is in the core during normal operation and as the name implies is ejected (by spring action) from the core when scrammed. A dual-purpose control rod is also ejected (by spring action) from the core when scrammed; however, the dual-purpose control rod may be in, partially in, or out of the core whenever the tables are together.

In addition, up to twelve ^{10}B safety blades (six per half) are available for additional shutdown if needed. These blades contain approximately 268 gms of ^{10}B and are inserted into the reactor by spring action when a scram occurs. Four of the ^{10}B blades (two per half) may also be used as control rods when necessary. The basic differentiation between whether a dual purpose rod or ^{10}B safety blade is classified as a control or safety rod is the fact that a rod designated as a safety rod is moved to its position of maximum reactivity worth (DP rods into the core, ^{10}B rods out of the core) before the movable table is started to closure and are thus available to remove reactivity. Rods designated as control rods are moved to positions of their maximum reactivity worth after the tables are together. Regardless whether a rod is designated as a control or safety rod, the rod will scram upon a scram signal by spring operation. After the tables are together any rod may be designated as a control or safety rod at the option of the operator.

Table movement is normally effected by one of four electric motors. The speed of the movable table from full separation of 152 cm to 45 cm is 25 cm/min (10 in./min) from 45 cm to 8 cm is 5.1 cm/min (2 in./min) and from 7.5 cm to 0 separation is 0.42 cm/min (1/16 in./min). Each of the

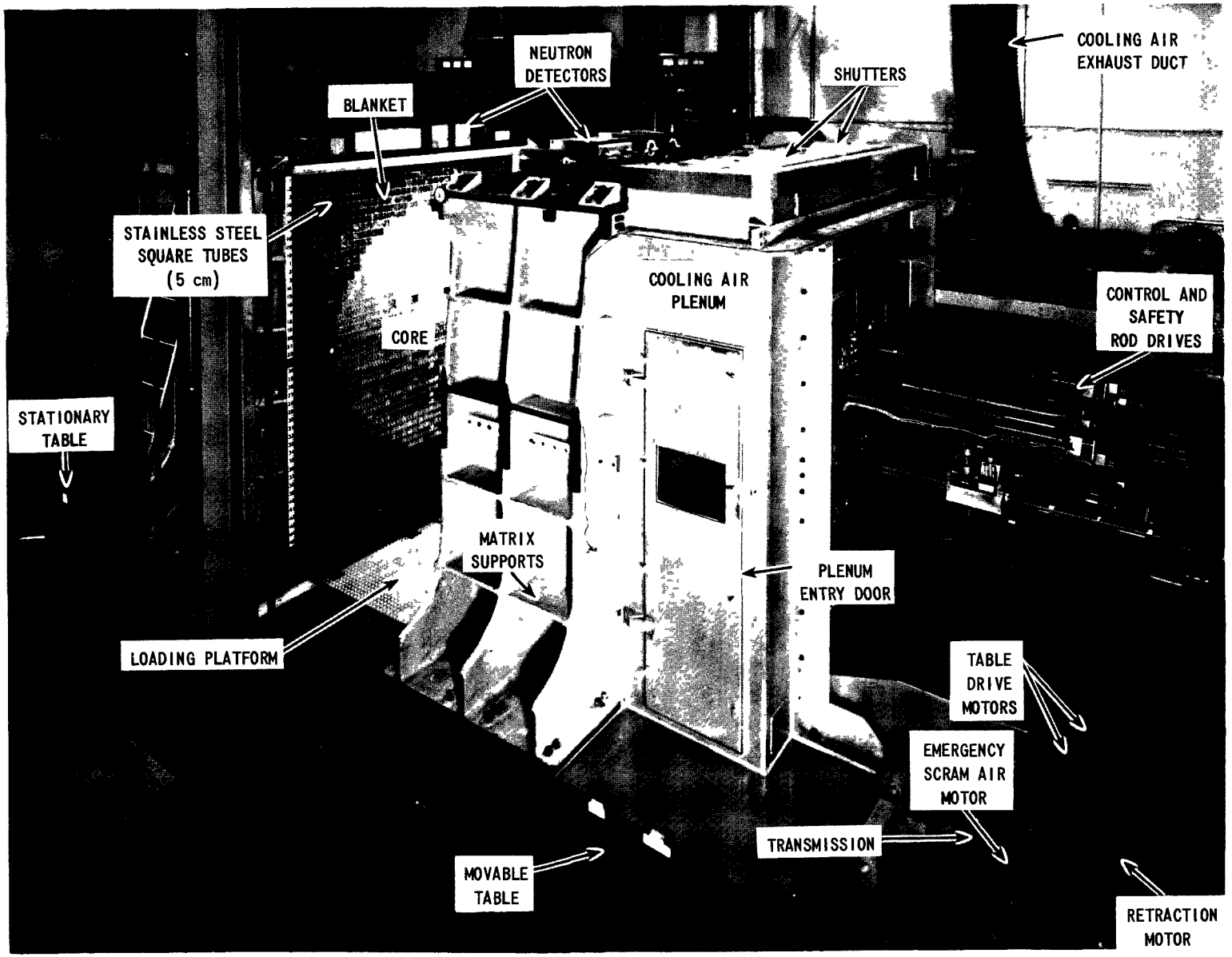


Fig. IV-1. ZPR-6. ANL Neg. No. 113-2963A.

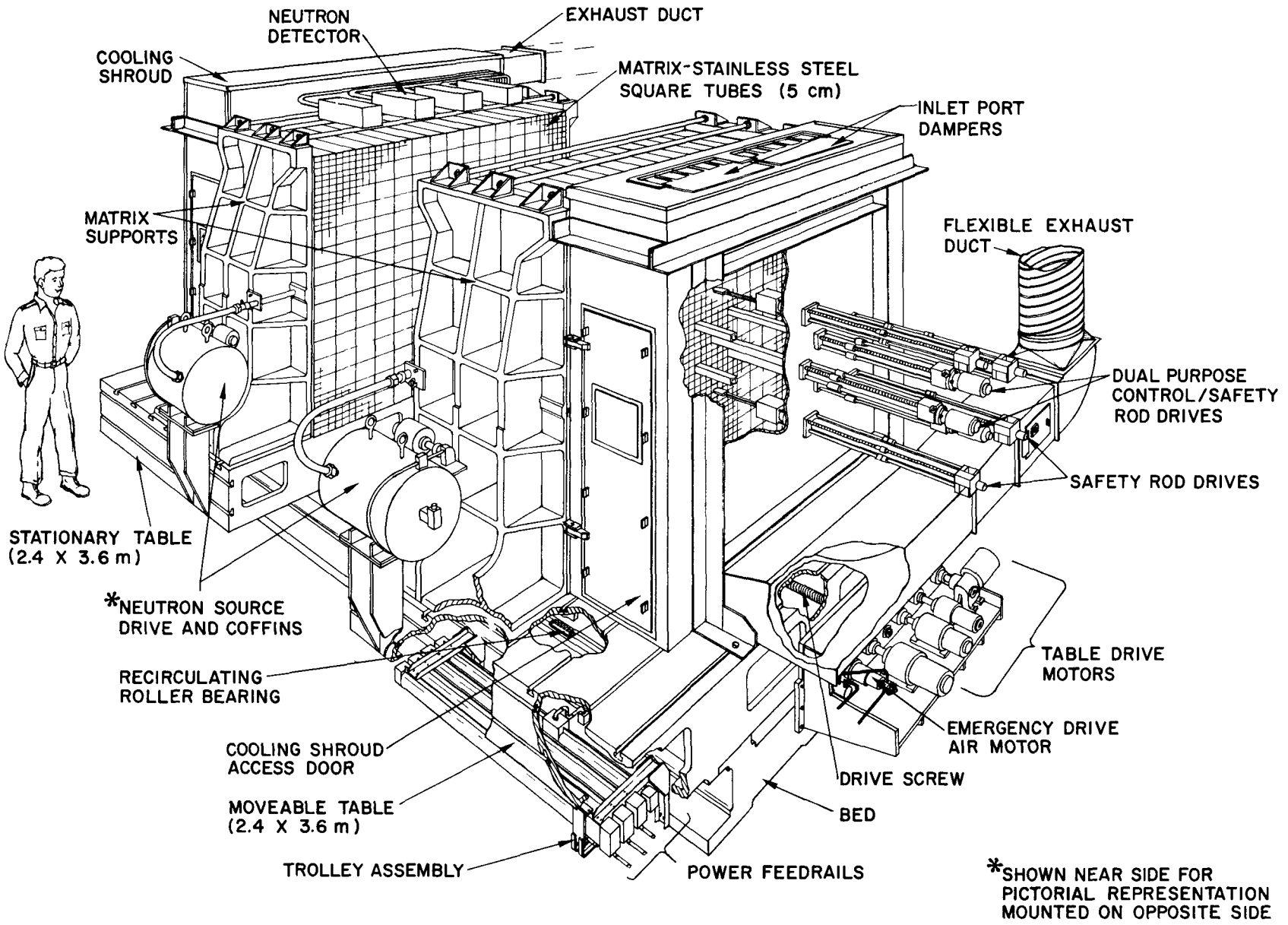


Fig. IV-2. Fast Critical Facility--ZPR-6 and -9

three motors can drive the table in or out over its respective operational range. The fourth motor is automatically energized when a scram condition occurs and separates the reactor halves. This electric scram motor is backed up by an air motor which is powered by compressed-bottled nitrogen stored in the reactor cell.

Each half of the assembly contains an americium-beryllium startup neutron source whose intensity is about 10^7 neutrons per second which may be driven in or out of the assembly by remote control. Control console and nuclear instrumentation are located in the control room to remotely control the movement of the table and rod drive mechanisms and to monitor the neutron population. A more detailed description of the ZPR-6 and -9 facilities can be found in Refs. 1 and 4. ZPR-9 is an identical facility to ZPR-6 except that it currently has a smaller steel matrix assembly (1.6 x 1.6 x 2.4 m). Within the next year, sufficient stainless steel matrix tubes will become available so that the ZPR-9 will have a matrix assembly (2.4 x 2.4 x 2.4 m) the same size as ZPR-6.

Although the matrix supports and control rod mounting plate have been sized to match a 2.4 m^3 (8 ft cube) matrix assembly, only minor modifications are necessary to make a 3 x 3 x 2.4 m (10 x 10 x 8 ft) matrix assembly. In the course of the present experimental program it may be necessary to make a larger cross sectional area matrix. Each of the tables is designed to carry approximately 82 metric tons easily which is adequate to handle the larger cores. A summary of the physical characteristics of the reactor facilities in the ZPR-6 and -9 cells is given in Table IV-1.

Table IV-1

SUMMARY OF PHYSICAL CHARACTERISTICS OF ZPR-6 AND -9

Bed and Table Assembly	
Bed	
Length	6.5 m (256 in.)
Width	2.6 m (103 in.)
Height	43.3 cm (17 in.)
Weight	~17,500 kg (38,500 lb)
Material	Cast, high-tensile, semisteel
Tables	
Length	2.4 m (96 in.)
Width	3.7 m (144 in.)
Thickness	46 cm (18 in.)
Weight	~10,000 kg (22,000 lb)
Material	Cast, high-tensile, semisteel

Table IV-1 (Contd.)

Tested Loading Capacity	
Each Table	82,000 kg (180,000 lb)
Total	164,000 kg (360,000 lb)
Table Drive and Travel	
Separation	1.52 m (60 in.) (nominal)
Fast-closure (forward) Speed	25 cm/min (10 in./min) to 46 cm (18 in.) separation
Motor Rating	3/4 hp
Intermediate-closure (forward) Speed	5 cm/min (2 in./min for Pu operation and 15 cm/min (6 in./min) for U operation from 46 cm to 7.6 cm (3 in.) separation
Motor Rating	1/2 hp
Slow-closure (forward) Speed	0.42 cm/min (1/6 in./min) for Pu operation and 1.3 cm/min (1/2 in./min) for U operation from 7.6 cm to 0 cm separation
Motor Rating	1/4 hp
Fast-separation (reverse) Speed	76 cm/min (30 in./min)
Motor Rating	1 hp
Emergency Separation Speed	76 cm/min (30 in./min)
Type Motor	Air operated
Rating	0.9 hp
Matrix Assembly	
Overall Dimensions	
Tables Together	
Width	2.5 m (8 ft) (nominal--varies with load)
Length	2.5 m (96 in.)
Height	2.5 m (8 ft) (nominal--varies with load)

Table IV-1 (Contd.)

Matrix Assembly on One Table	
Width	2.5 m (8 ft) (nominal--varies with load)
Length	1.22 m (48 in.)
Height	2.5 m (8 ft) (nominal--varies with load)
Standard Tube Bundles	
25 Tubes/Bundle	5 rows and 5 columns
Size	1.2 m length (48 in.) 27 cm sq
Total Maximum Number Bundles	162
Maximum Number per Half	81
Maximum Number of Tubes	4050
Tube Dimensions	
Length	1.2 m (48 in.)
Width and Height	5.5 cm (2.17 in.)
Tube Material	Type 304 stainless steel/aluminum
Control and Safety Rods	
Dual Purpose Rods	
Number	5 per half
Drive Motor	27 V dc (fractional hp)
Maximum Reactivity Addition Speed	1 cm/sec
Scram Drive	Spring loaded
Length of Stroke	~61 cm (24 in.)
Scram Time	< 300 ms/$\frac{2}{3}$ of stroke
Approximate Maximum Reactivity Addition Rate	0.04 $/sec
Overall Dimension of Control Drawer	
Length	1 m (40 in.)
Width	5.2 cm (2.06 in.)
Height	5.2 cm (2.06 in.)
Wall Thickness	0.12 cm (0.048-0.051 in.)

Table IV-1 (Contd.)

Insertion Safety Rods	
Number	Up to 6 per half
Normal Drive	27 V dc electric motor (fractional hp)
Max Reactivity Addition Speed	1.5 cm/sec
Scram Drive	Loaded spring
Length of Stroke	76 cm (30 in.)
Scram Time	<300 ms/$\frac{2}{3}$ of stroke
Outside Dimensions of Standard ^{10}B Safety Rod	
Width	5 cm (2 in.)
Length	78 cm (31 in.)
Thickness	0.95 cm (3/8 in.)
Length of ^{10}B Area	71 cm (28 in.)
Neutron Source Assembly	
Number	1 per half
Neutron Source Strength	10^7 neutrons/sec (minimum)
Source	Am-Be
Source Drive Motor	Reversible, 115 V ac (fractional hp)
Time to Travel from Coffin to Full Insertion	1-2 min
Source Position	
Along Radial Direction	Core-blanket interface
Distance from Matrix Interface	
Location 1	Approximately 10 cm (4 in.)
Location 2	Approximately 20 cm (8 in.)
Source Coffin Size	61 dia (24 in. dia)

B. Reactor Instrumentation and Controls

1. Description of the Instruments

Since the details of the instruments, control circuitry and electrical interlocks have not been described in detail previously, these systems

are described below. The ZPR-6 and -9 critical facilities are equipped with the following instrument channels for startup and operation.

a. Channels 1 and 2 are BF_3 proportional counters with audible poppers and log count rate meters. These provide duplicate low-level trip circuits which remain in the trip condition until the count rates exceed a predetermined level. A schematic diagram is shown in Fig. IV-3.

b. Channel 3 shown in Fig. IV-4 consists of high sensitivity (3×10^{-13} amp/nv) uncompensated BF_3 ion chamber and a DC multirange linear electrometer with recorder and bucking current source. The bucking current can be used to cancel out part of the input current and provides a high level of accuracy when it is required to hold the reactor neutron level constant or return the reactor flux to a known flux for rod calibrations, etc.

c. Channel 4 consists of a BF_3 ion chamber and a DC multirange linear electrometer with a trip level which is adjustable from 5 to 105% of full scale on each range as shown in Fig. IV-5. Trip circuits are in the main power chain and in the dual-purpose rod and ^{10}B safety blade magnet chains. It is used as the main safety amplifier.

d. Channels 5 and 6 as shown in Fig. IV-6 and IV-7 respectively, are high-level safety trip-circuits, each with a BF_3 ion chamber and a DC multirange linear electrometer. The trip levels are normally adjusted to scram the reactor if the neutron level increases from 1/2 to 1 decade above the highest operating level established for the day.

e. Channels 7 and 8 as shown in Fig. IV-8 are duplicate log-n amplifiers and period meters with BF_3 ion chambers and three trip circuits in each channel. A single recorder simultaneously records the neutron level and the period for either channel. In both instruments a table trip will occur for periods of approximately 20 sec or less. This trip will stop the table's forward motion but does not interfere with table separation. An adjustable trip normally preset at 12 sec and a 5 sec backup trip are in the main control power chain, and in the dual-purpose rod and ^{10}B safety blade magnet chains.

f. Channel 9 shown in Fig. IV-9 consists of five ionization-type gamma monitors to indicate the levels of radiation at various locations for personnel protection. One monitor is located on top of the reactor, another is just inside the cell by the personnel door, a third in the control room on the south wall, and the other two are located in the normal and emergency exhaust, respectively.

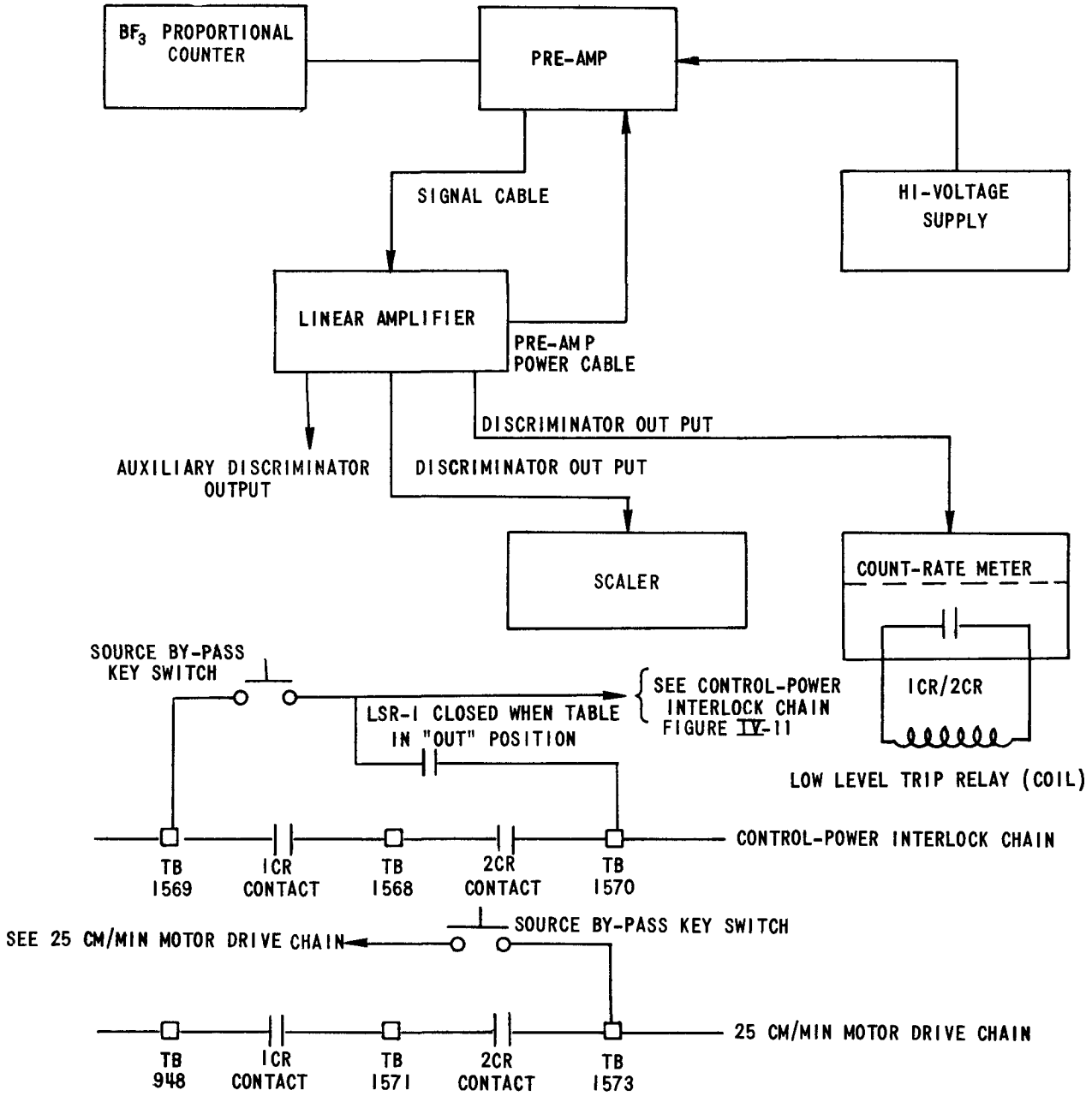


Fig. IV-3. Channels No. 1/No. 2. ANL Neg. No. 112-8115 Rev.1.

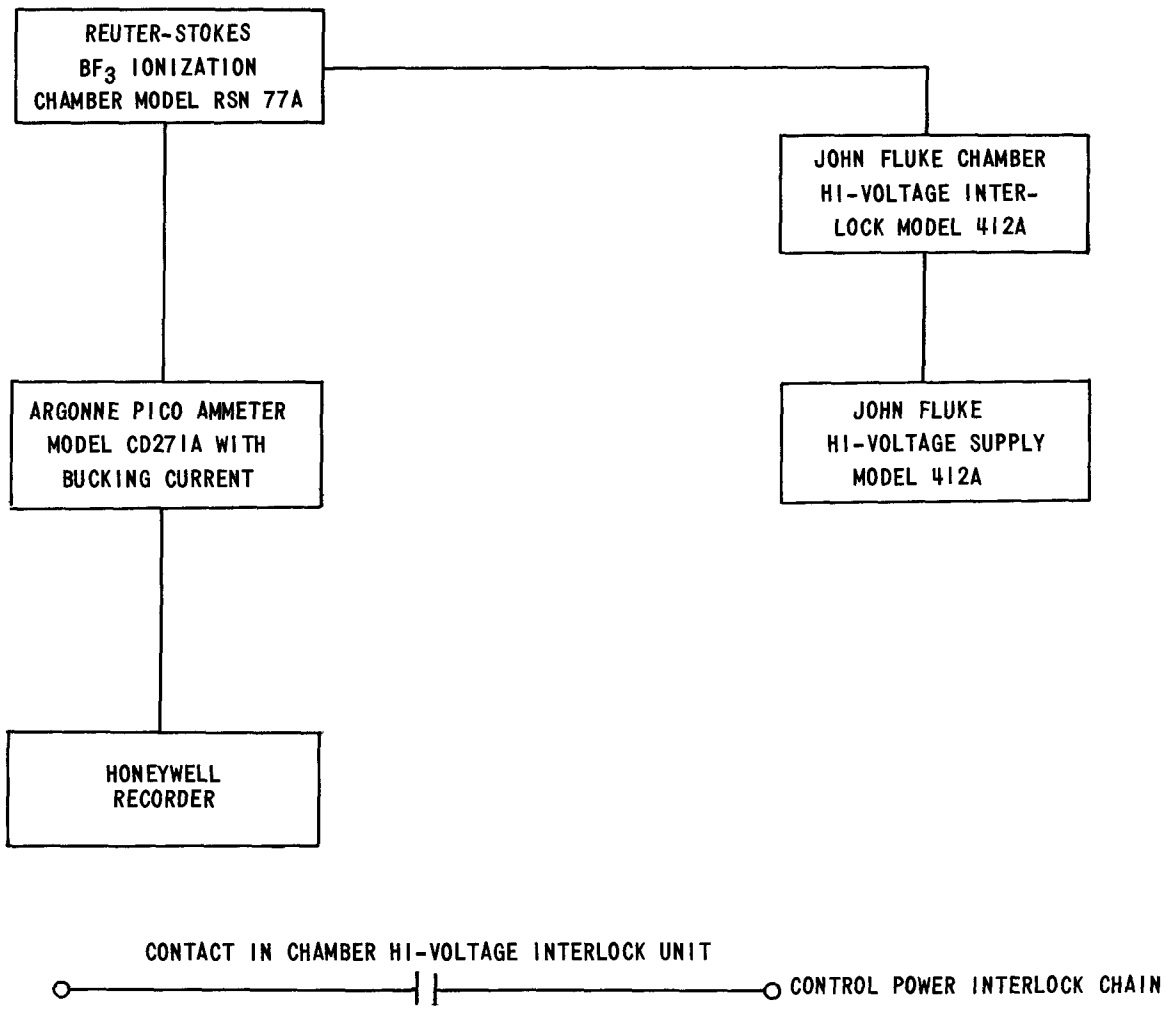


Fig. IV-4. Channel No. 3. ANL Neg. No. 112-8112.

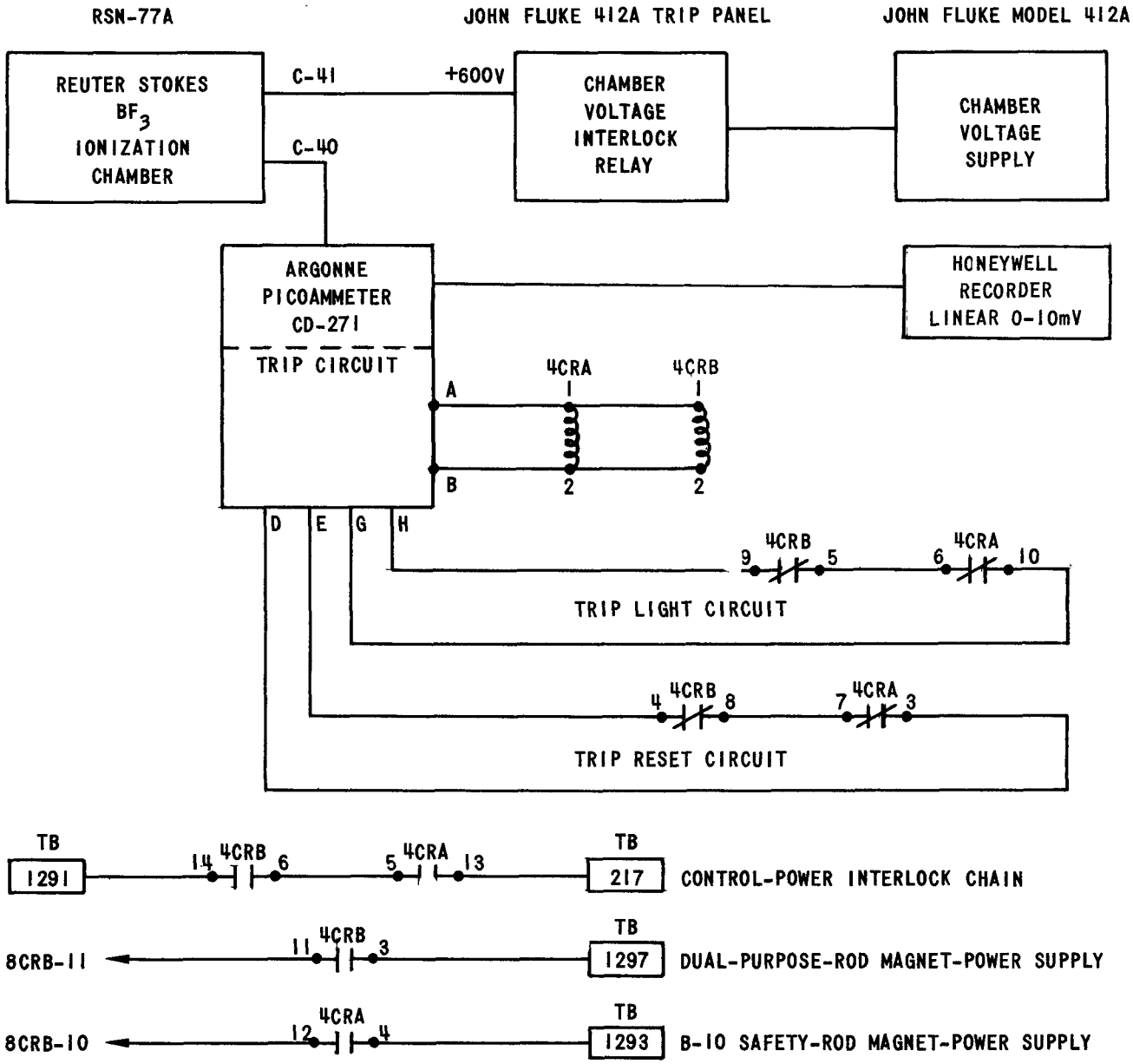


Fig. IV-5. Channel No. 4. ANL Neg. No. 112-8111 Rev.1.

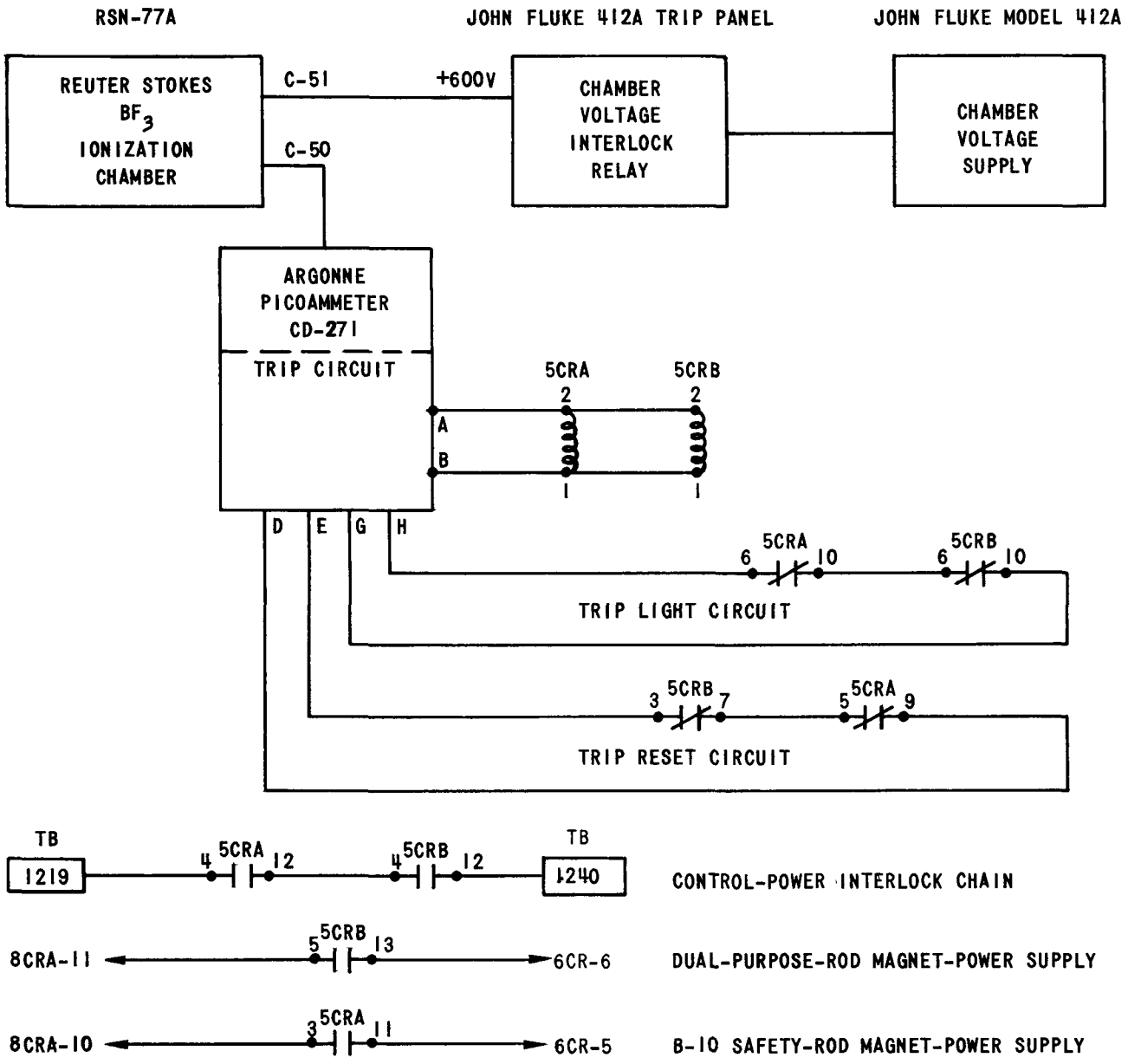


Fig. IV-6. Channel No. 5. ANL Neg. No. 112-8110 Rev.1.

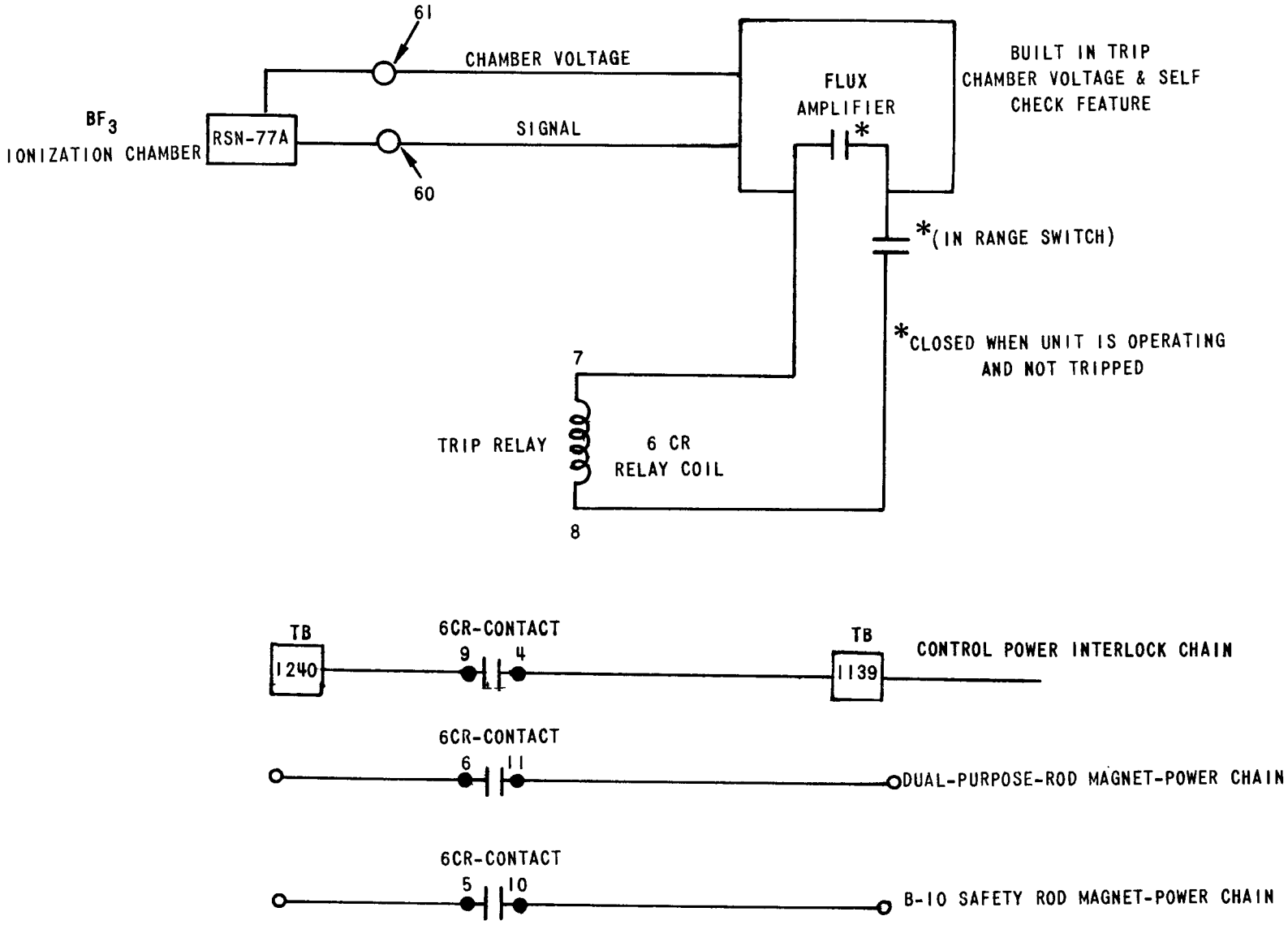


Fig. IV-7. Channel No. 6. ANL Neg. No. 112-8114 Rev.1.

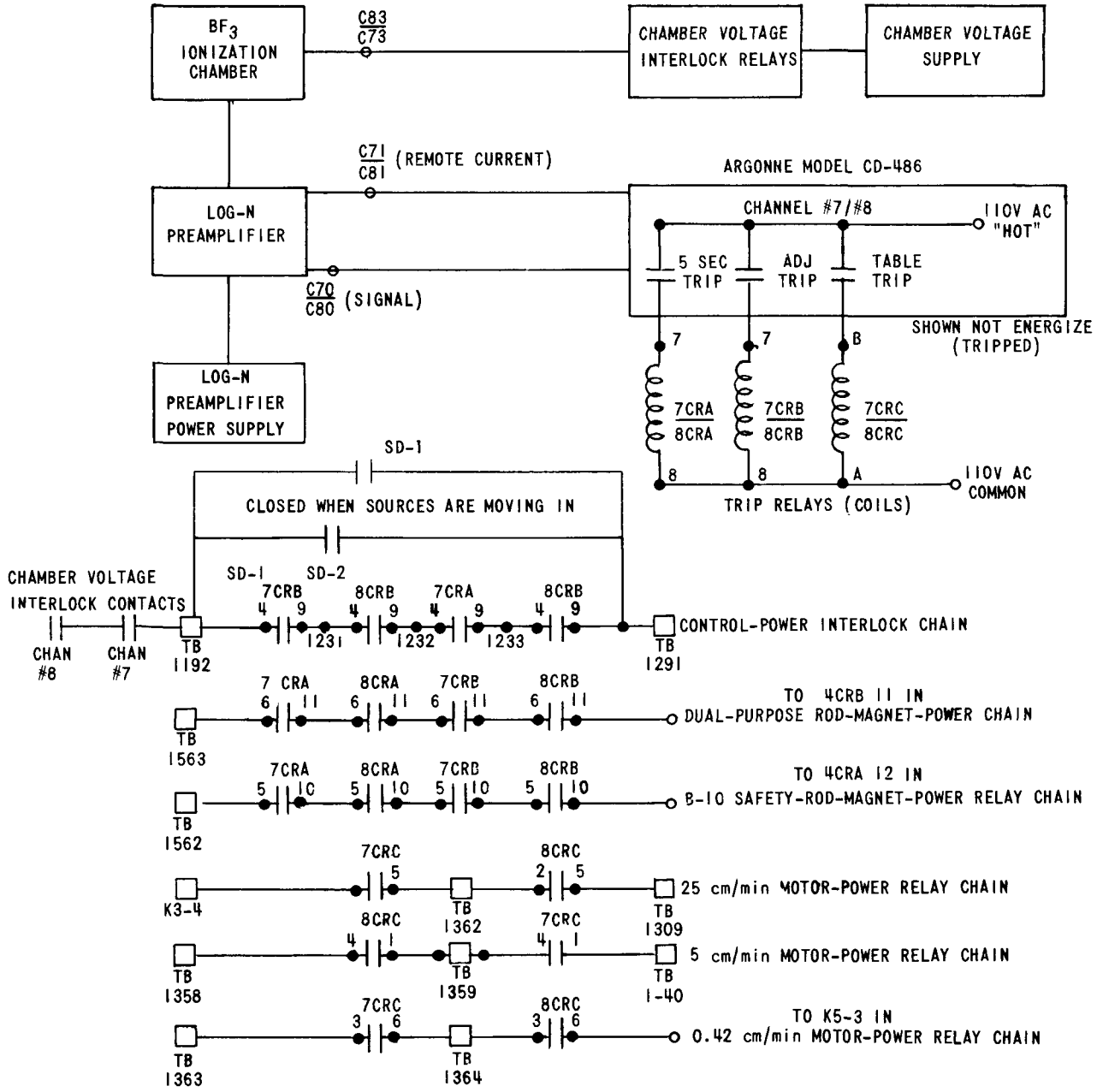


Fig. IV-8. Channels No. 7/No. 8. ANL Neg. No. 112-8108 Rev.1.

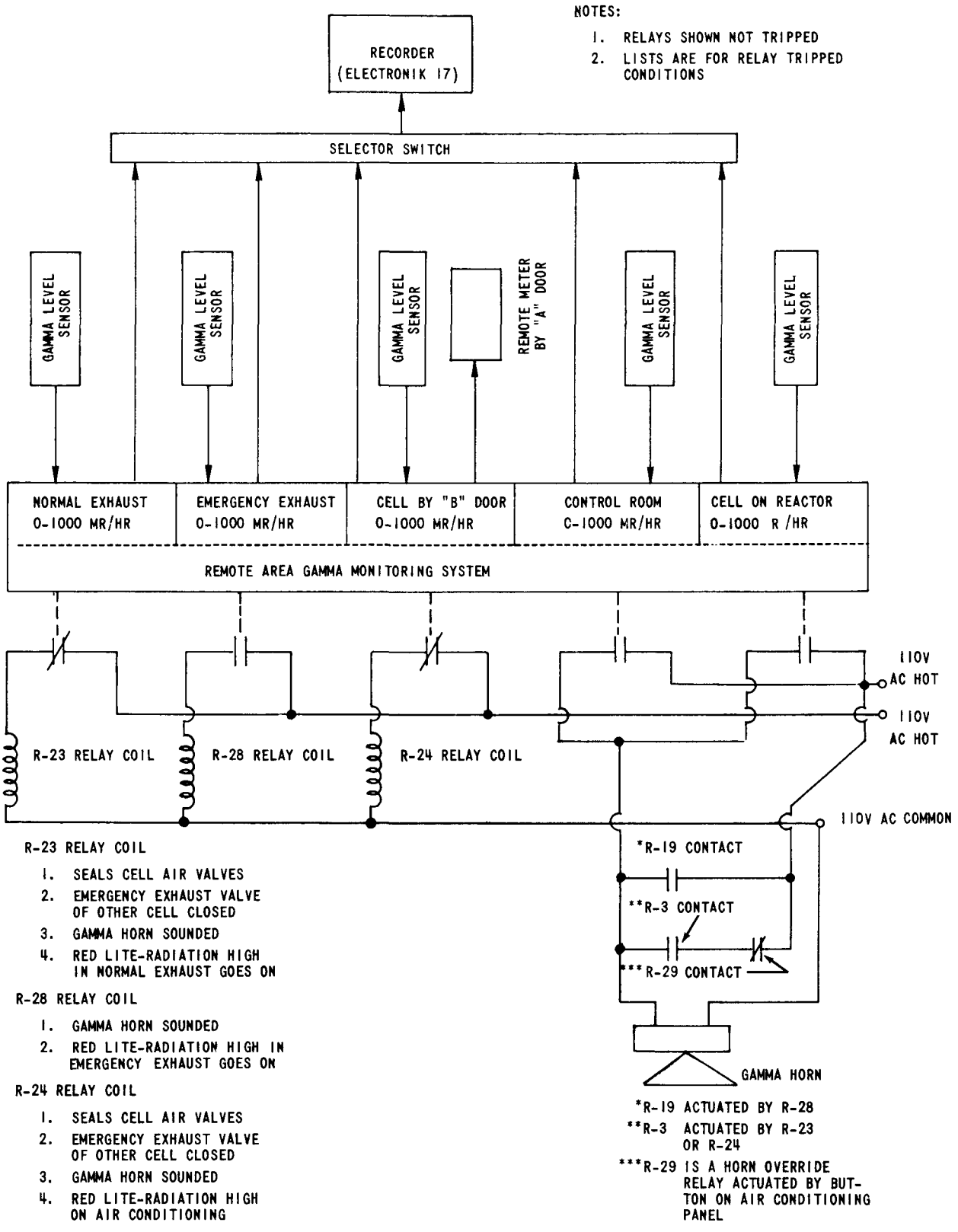


Fig. IV-9. Channel No. 9

In Figs. IV-3 through IV-9, the manufacturer and model numbers are given for the nuclear instruments for illustrative purposes only. These figures indicate instruments which currently are in use with the facilities. Periodically instruments will be modified or replaced by devices which have substantial improvements in their design and operating characteristics. This updating is carried out in order to maintain the highest reliability and performance characteristics attainable for the operating nuclear instrumentation.

Before any modifications are made to the nuclear instrumentation, the proposed changes will be reviewed by the Reactor Manager for safety considerations. The Reactor Manager may request further review and approval either at the Divisional or Laboratory management levels as discussed in Chapter VII if the change involves more than a routine maintenance or replacement and has reactor safety implications. If the modifications involve safety considerations outside of the scope of the approved SAR and addenda, they will be submitted to the Commission for further review and approval prior to the change.

2. Interlocks and Control System

The basic functions normally required for operation of the ZPR-6 and -9 reactors are as follows:

a. Moving the table halves "in" or "out."

b. Moving the dual-purpose rods "in" or "out."

c. Moving the ^{10}B safety blades "in" or "out."

In addition the following functions are included for safety:

d. Inserting the ^{10}B blades--scram.

e. Withdrawing the dual-purpose rods--scram.

f. Separating the table halves--scram.

Instruments and circuitry not directly used to accomplish these six functions are incorporated to provide system performance monitoring and/or police action to limit or initiate the above functions. The safety functions are achieved by:

a. Interrupting dc power to the ^{10}B safety blade magnets which in turn allows compressed springs to drive the ^{10}B blades into the core.

b. Interrupting dc power to the dual-purpose rod holding magnets which in turn allows compressed springs to eject the rods from the core, and by

c. Actuating a scram motor which drives the table halves apart.

Thus there are three separate and independent scram chains--one for the dual-purpose rods, one for the ^{10}B blades, and one for the scram motor drive.

It is to be noted that in the case of the ^{10}B safety blades and the dual-purpose rods, the release of compressed springs eject the dual-purpose rod assemblies and inserts the ^{10}B rods. The scram action (release of compressed springs) inherently overrides other rod drive modes. Subsequently the rods themselves cannot be driven without first duplicating in detail the entire startup procedure. The reason for this is in part, that the magnet assemblies and not the rods are driven by motors. Only after the springs are completely recompressed (which occurs when the rods and blades are in positions of minimum reactivity) and the rod/blade assemblies are magnetically latched to the magnet assemblies can the rods be moved. Only startup procedural interlocks are incorporated in the rod drive chains. It should also be noted that the rod/blade magnet chains actually carry magnet currents, i.e., no relay or other intermediate device is actuated. Thus breaking the rod/blade magnet chains directly interrupts the magnet currents.

Figure IV-10 depicts the requirements to sustain magnet current in the ^{10}B blade and dual-purpose rod magnets. The diagram is self-explanatory except for the "control power interlock chain" relay requirements which are explained below.

It should be noted that the schematic drawings of the control and interlock circuitry given by the figures in this chapter are descriptive of the present situation. They are to show the basic functions of the interlocks. Although the principles of operation of the control circuitry will be maintained, it may become necessary to change control circuitry from time to time in order to improve operating procedures or to include other required protective functions. Such changes as necessary may be introduced into the system following review by appropriate levels of supervision.

Each change before its introduction will be reviewed and approved by the Reactor Manager. The Reactor Manager may request additional Divisional or Laboratory level review as discussed in Chapter VII depending on the magnitude of the change. If the modifications involve safety considerations outside of the scope of the currently approved SAR and addenda, the modifications will be submitted to the Commission for further review and approval prior to the change.

Figure IV-11 is the "control power interlock chain" which insures that certain conditions have been established prior to reactor operation. If these procedures are violated control power cannot be obtained or is lost. This is reflected as a scram condition in the ^{10}B safety blade chain, the dual-purpose rod chain, and the scram motor chain; it also enforces procedural requirements associated with the table drive motors and the rod/blade system.

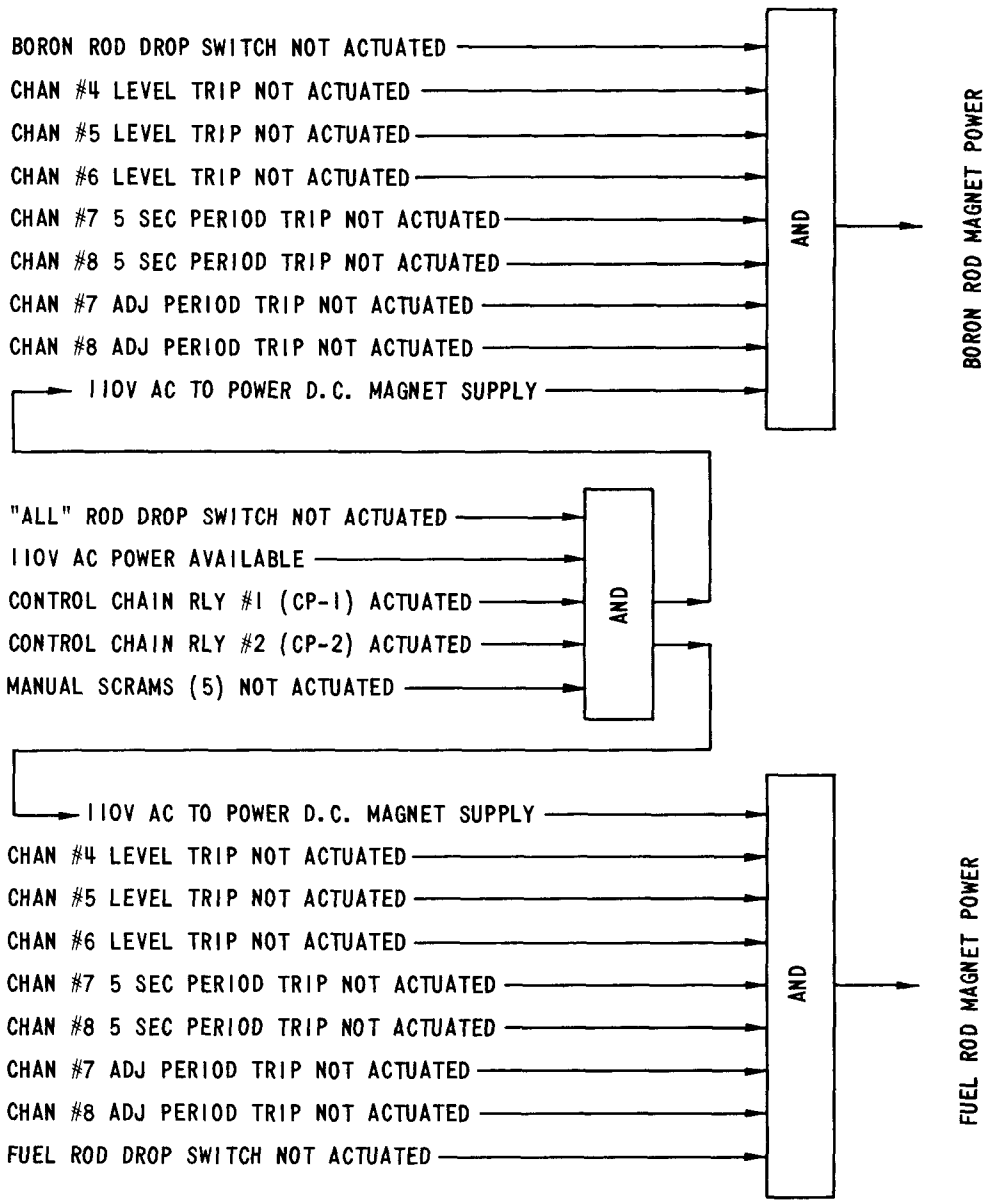


Fig. IV-10. ^{10}B Safety Rod and Dual Purpose Rod Magnet-power Interlock Logic. ANL Neg. No. 112-8122 Rev.1.

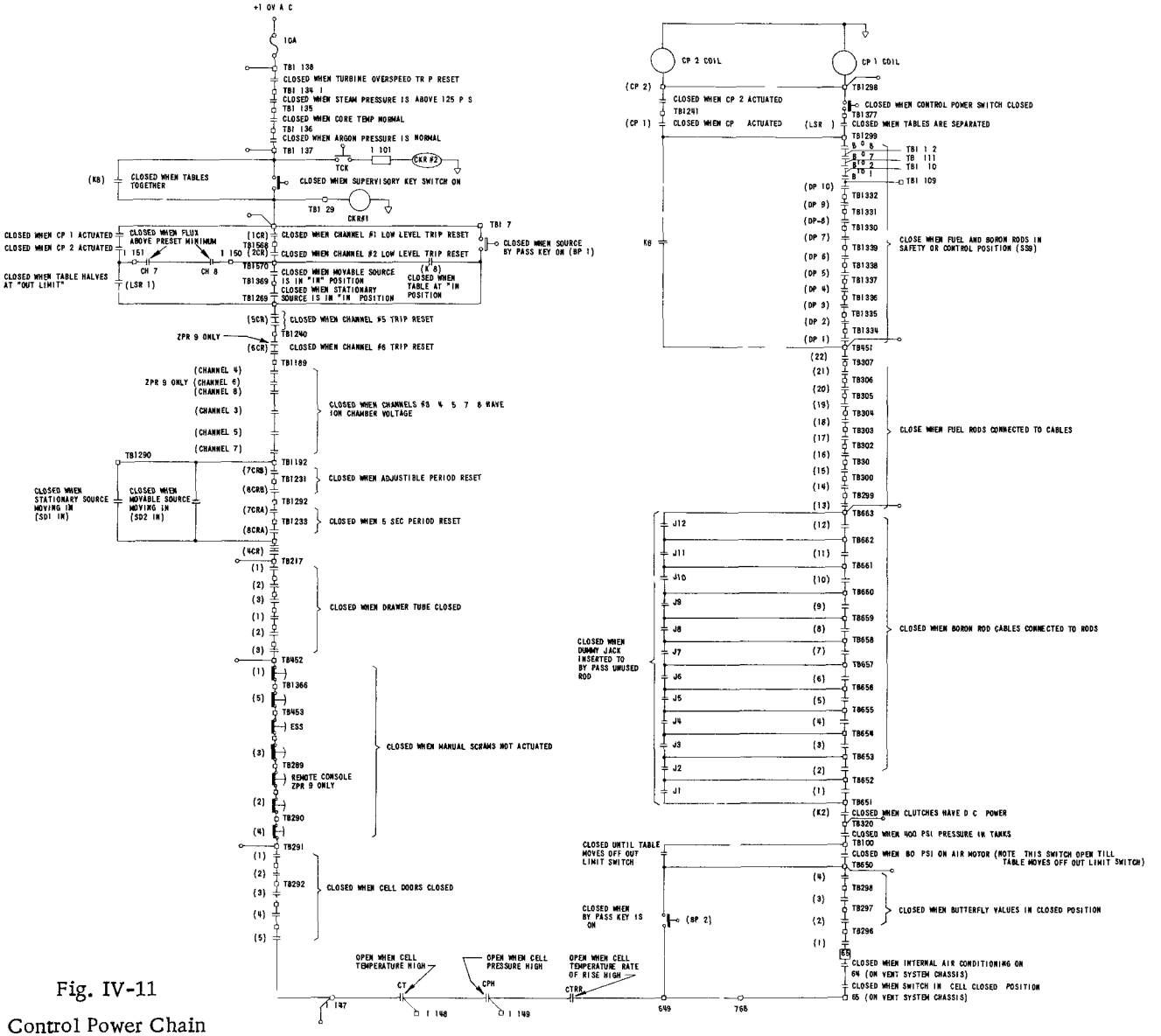


Fig. IV-11

Control Power Chain

The 0.42 cm/min motor drive interlock chain, the 5 cm/min motor drive interlock chain, the 25 cm/min motor drive interlock chain, are shown on Figs. IV-12 through IV-14, respectively.

The electrical interlocks in the table drive interlock chain are to fulfill the following conditions:

1. Table must move with a given speed only in a defined region of table movement.
2. Only one motor at a time may have power.
3. Table motion may occur only when the table trips are not actuated. (Scram motor will however operate upon a scram condition.)

Figure IV-15 shows the scram motor interlock chain, and Fig. IV-16 shows the air motor chain which backs up the electric motor to perform the table separation function after a scram signal occurs. It can be seen from the scram motor chain, that power is removed from the control power (CP-1, CP-2) relay which in turn actuates the scram motor and removes power from other drive motor relays. The electric scram motor is backed up by an air motor (powered from cylinders of compressed nitrogen) which will separate the halves if the electric scram motor is not turning within 0.5 sec after receipt of a scram signal.

The dual-purpose rod motor drive chain, the ^{10}B rod drive interlock chain, the ^{10}B rod drive circuit are shown in Figs. IV-17 through IV-19.

The rod drive and interlock chains have been designed to fulfill the following conditions.

1. All rods move to their positions of least reactivity worth upon a scram signal by spring operation.

2. All rods designated as safety rods (i.e. dual-purpose rods designated as safeties and ^{10}B designated as safeties must be in their positions of maximum reactivity worth before the table may be moved. The only differentiation between a "control" and a "safety" rod is that a rod designated as "safety" is moved to its position of maximum reactivity worth before the tables have been brought together. This is to insure that there is some shutdown available if the reactor had been accidently overloaded and the reactor started to go critical while the moveable table was moving toward the stationary one. The rods provide the fast shutdown on this facility. Once the tables are together there is no difference in principle between a rod designated as a safety or control rod since all rods scram on a scram signal.

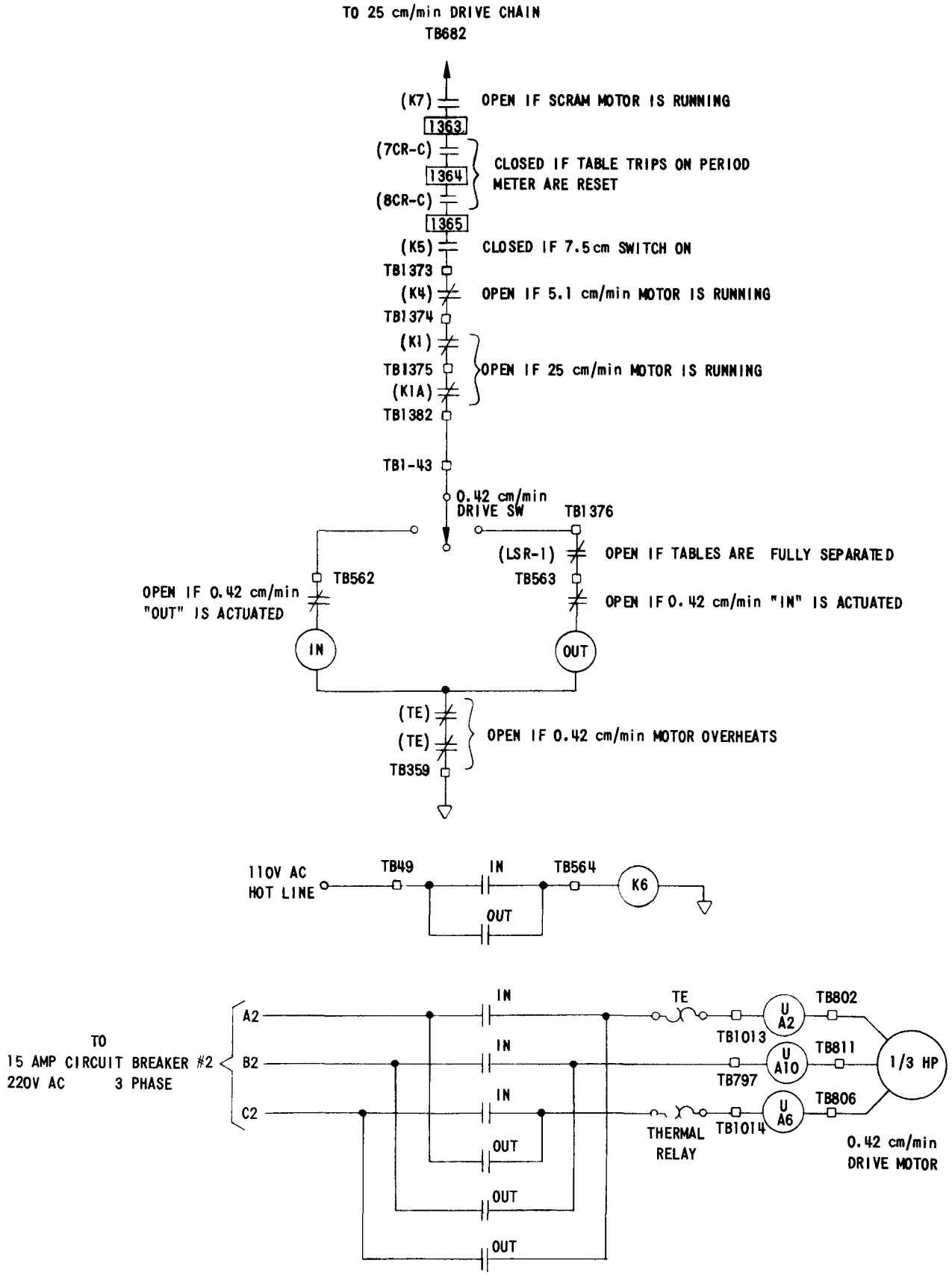


Fig. IV-12. 0.42 cm/min Table-drive Interlock Chain

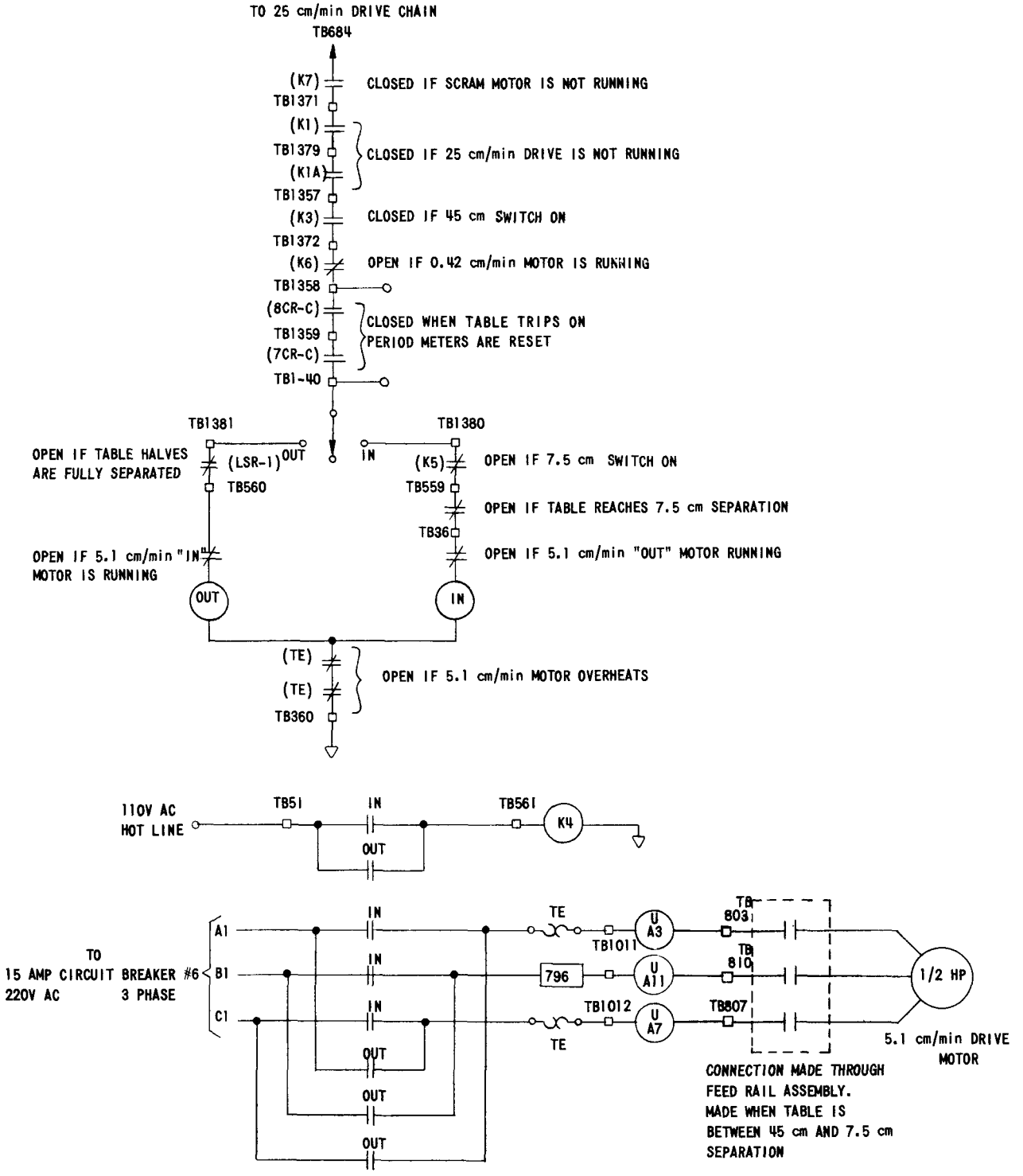


Fig. IV-13. 5.1 cm/min Table-drive Interlock Chain

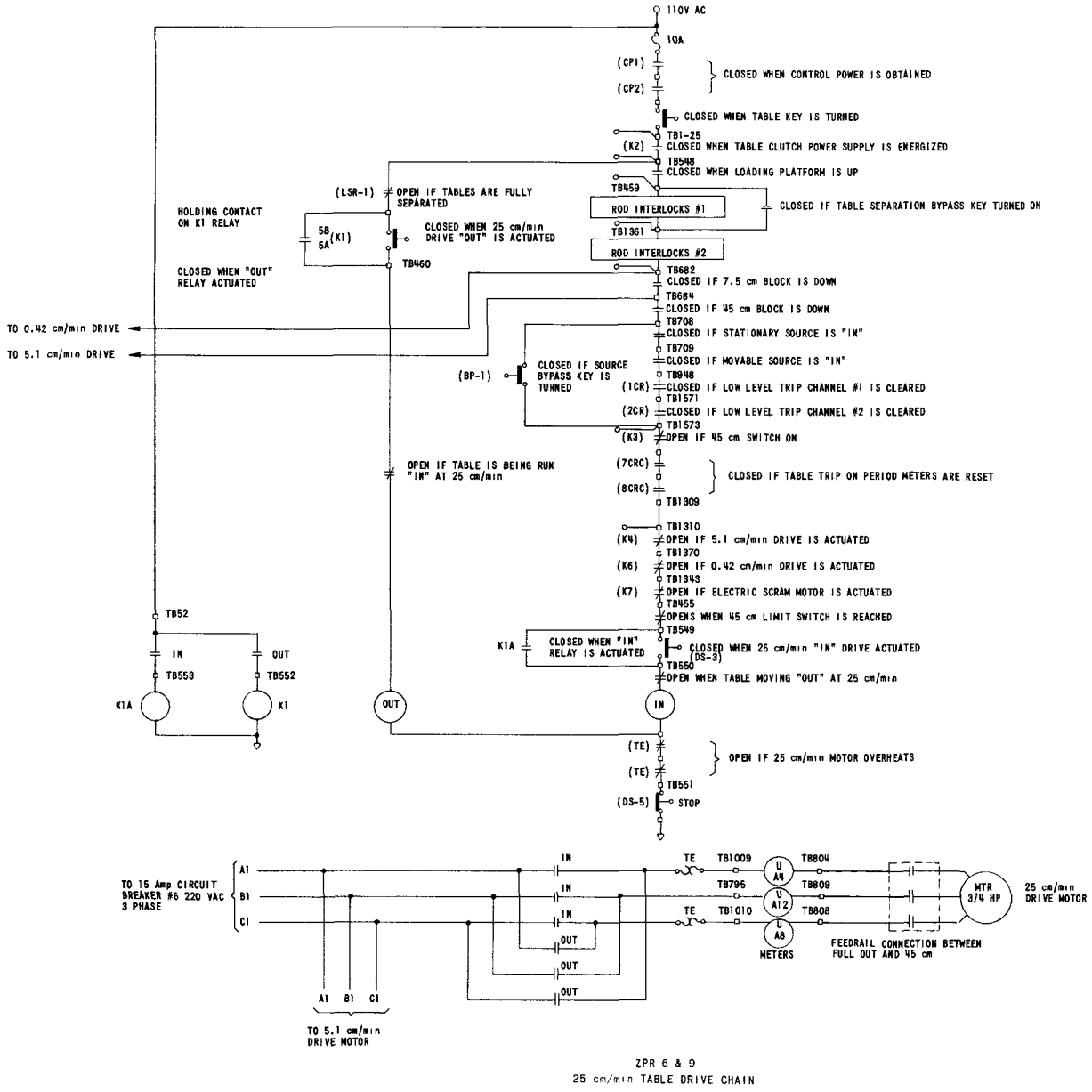


Fig. IV-14. 25 cm/min Table-drive Interlock Chain

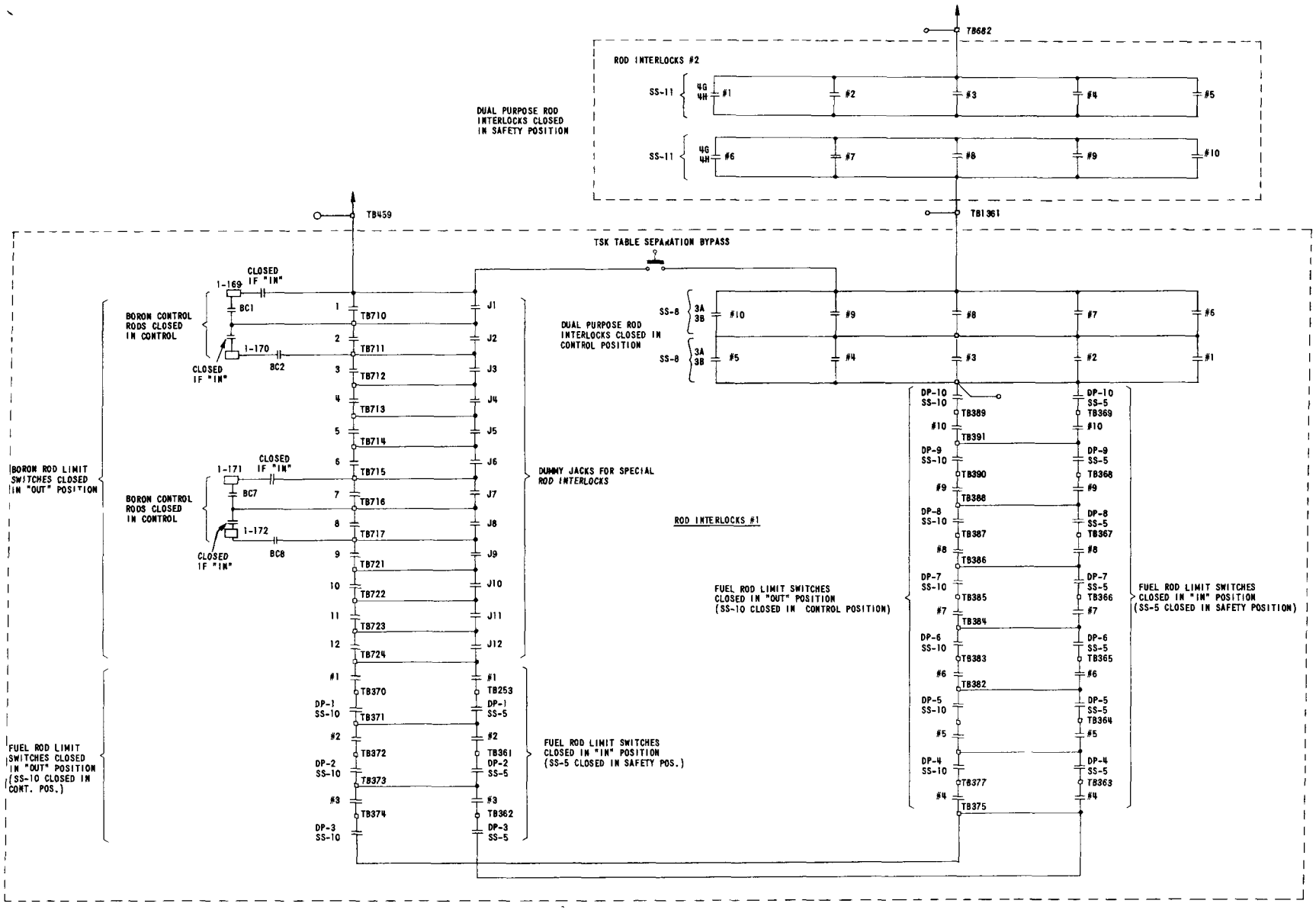


Fig. IV-14 (Contd.)

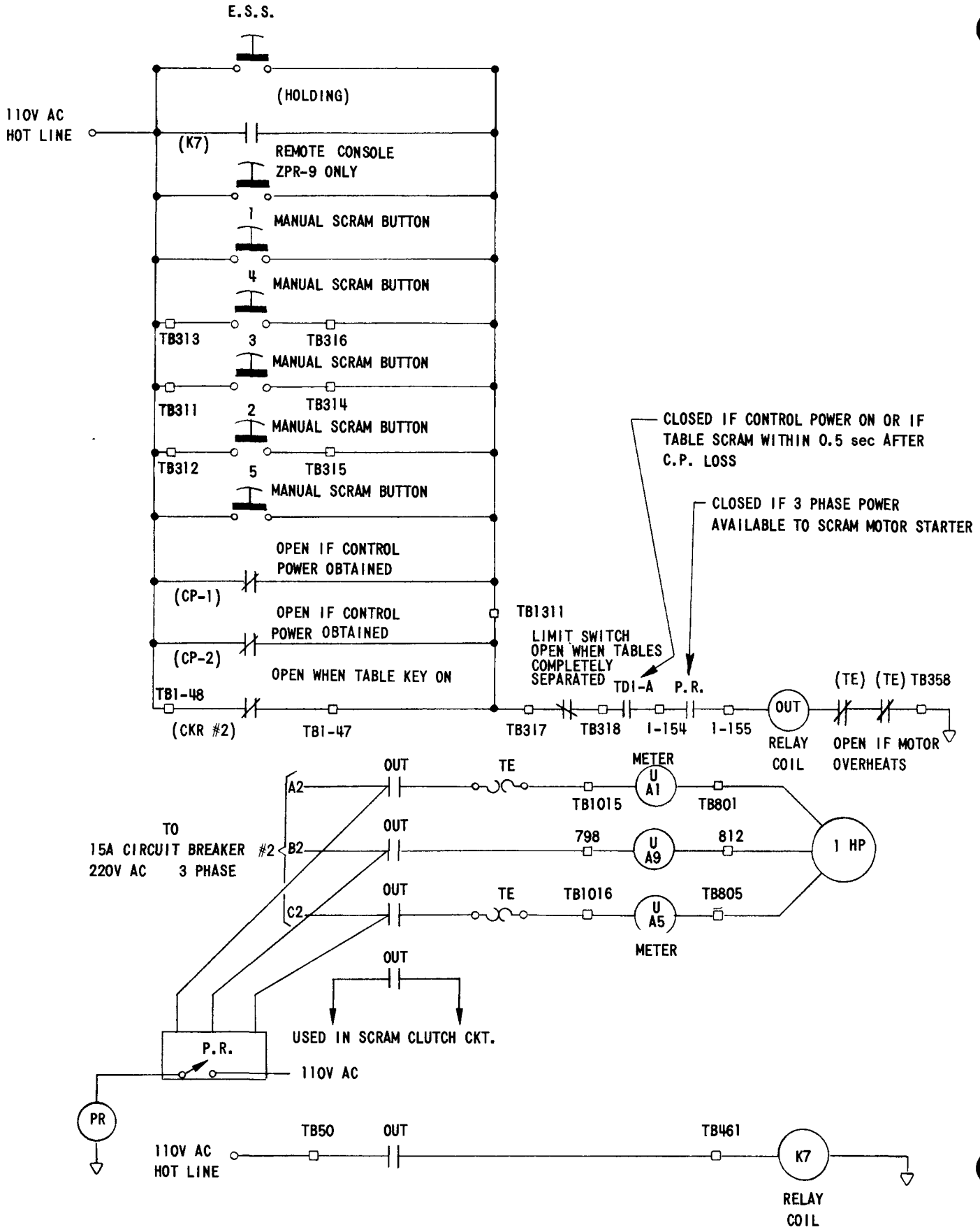


Fig. IV-15. Electric Scram-motor Interlock Chain

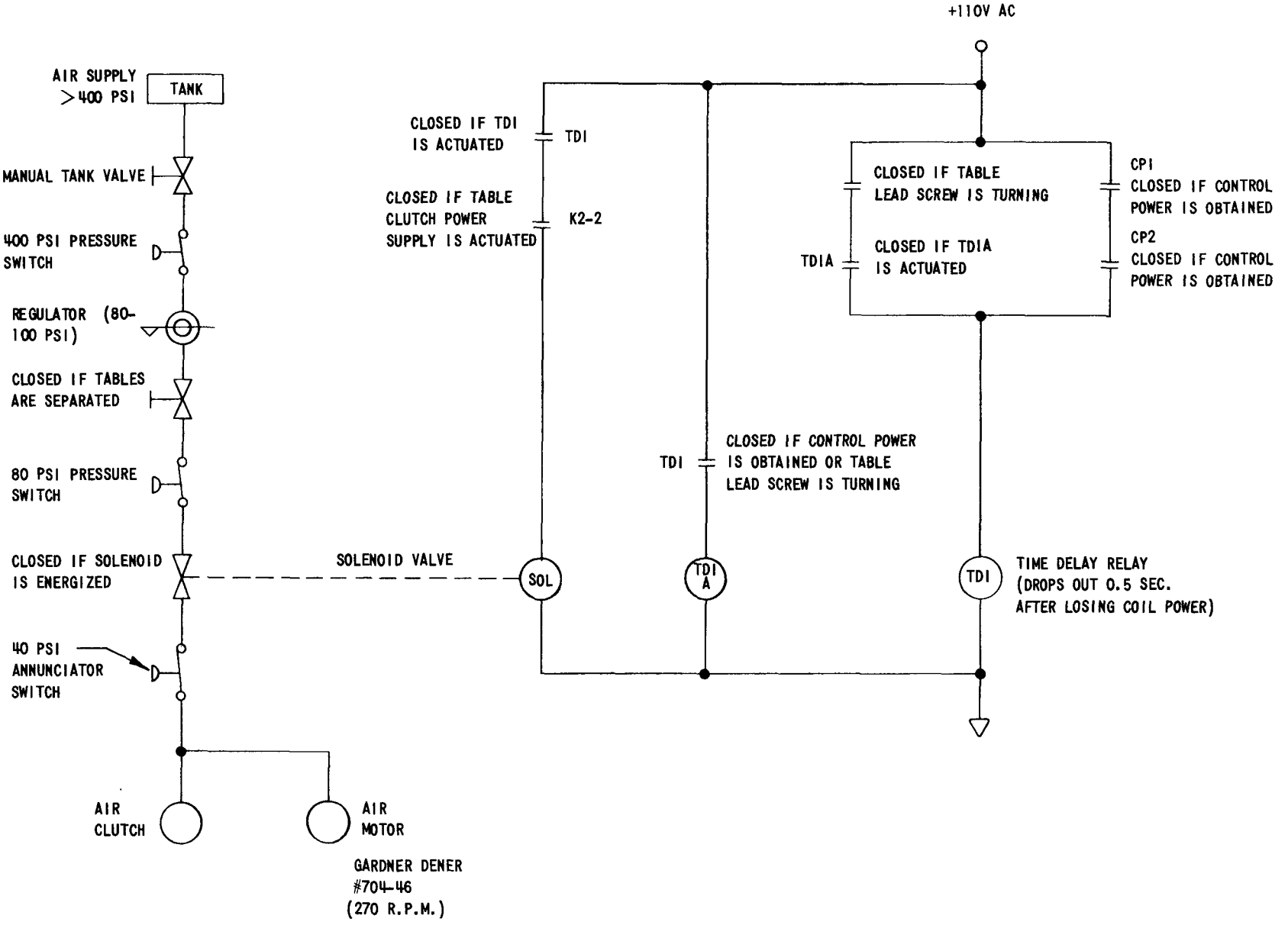
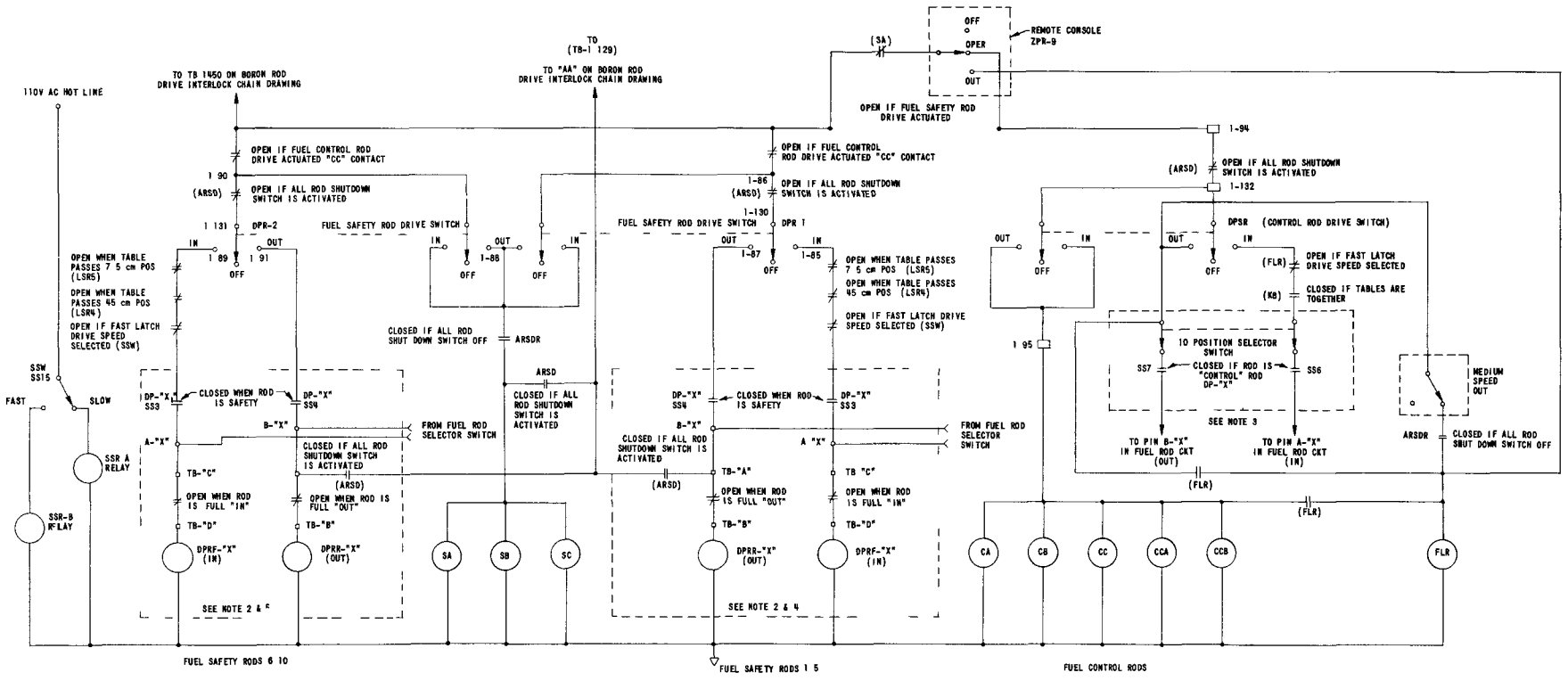


Fig. IV-16. Emergency Air-motor Drive Chain

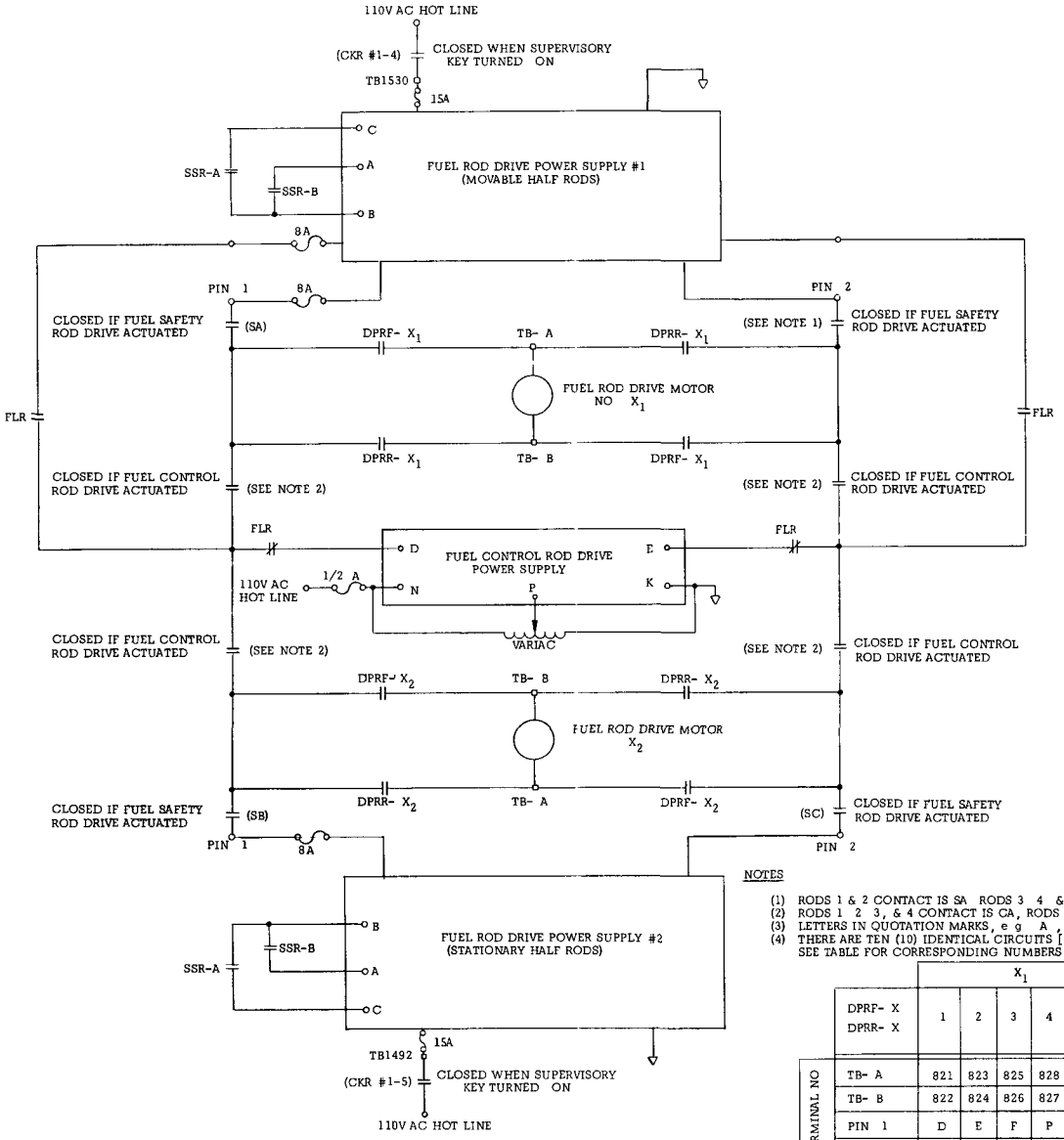


NOTES

(1) LETTERS IN QUOTATION MARKS e g "A" REPRESENT ROD NUMBERS OR TERMINAL BOARD NUMBERS
(2) THERE ARE TEN (10) IDENTICAL CIRCUITS (FIVE (5) FROM SWITCH DPR-1 AND FIVE (5) FROM SWITCH DPR 2) SEE TABLE FOR CORRESPONDING NUMBERS
(3) THERE ARE TEN (10) IDENTICAL CIRCUITS FROM SWITCH DPSR SEE TABLE FOR CORRESPONDING NUMBERS
(4) THESE FIVE (5) CIRCUITS CONSIST OF RODS 1 THROUGH 5
(5) THESE FIVE (5) CIRCUITS CONSIST OF RODS 6 THROUGH 10

	DPRR-"X" DRRF "X"	1	2	3	4	5	6	7	8	9	10
TERMINAL NO	TB "A"	486	487	488	489	490	396	397	398	399	400
	TB "B"	945	946	949	950	901	902	903	904	905	906
	TB "C"	481	482	483	484	485	491	492	493	494	495
	TB "D"	877	878	942	943	944	951	952	953	954	955

Fig. IV-17. Fuel Rod Drive Chain



NOTES

(1) RODS 1 & 2 CONTACT IS SA RODS 3 4 & 5 CONTACT IS SB
(2) RODS 1 2 3, & 4 CONTACT IS CA, RODS 5 6 7 & 8 CONTACT IS CB, RODS 9 & 10 CONTACT IS CC
(3) LETTERS IN QUOTATION MARKS, e g A , REPRESENT ROD NUMBERS OR TERMINAL BOARD NUMBERS
(4) THERE ARE TEN (10) IDENTICAL CIRCUITS [FIVE (5) FROM EACH FUEL ROD DRIVE POWER SUPPLY] SEE TABLE FOR CORRESPONDING NUMBERS

		X_1					X_2				
	DPRF- X DPRR- X	1	2	3	4	5	6	7	8	9	10
TERMINAL NO	TB- A	821	823	825	828	841	843	845	847	849	851
	TB- B	822	824	826	827	842	844	846	848	850	852
	PIN 1	D	E	F	P	R	D	E	F	P	R
	PIN 2	H	J	K	L	M	H	J	K	L	M

Fig. IV-17 (Contd.)

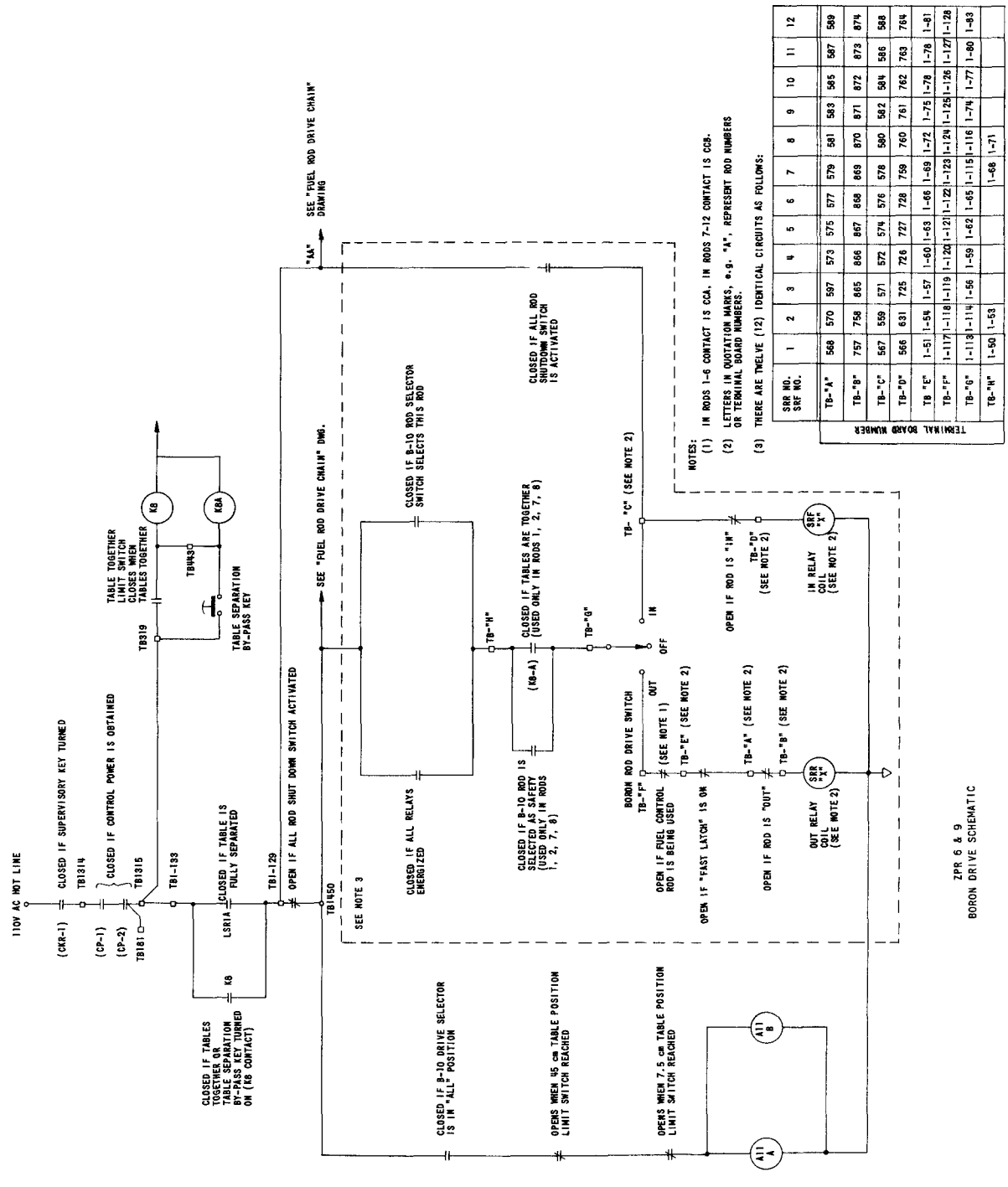
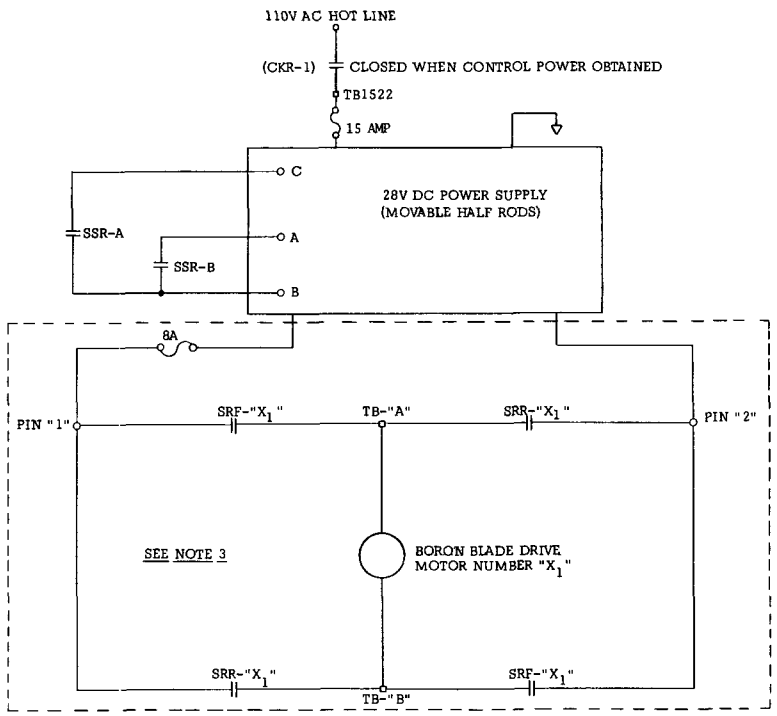
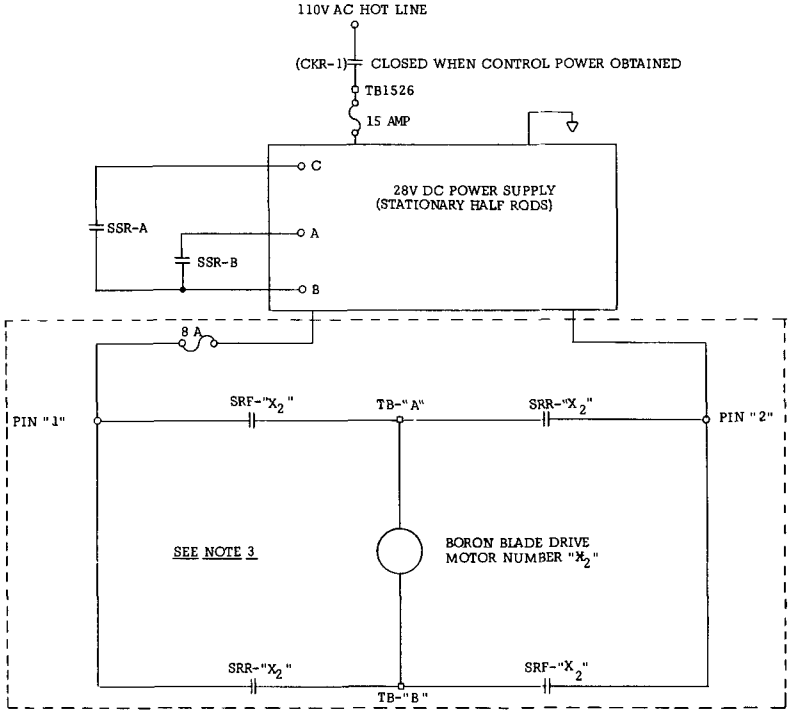


Fig. IV-18. ^{10}B Rod Drive Interlock Chain





NOTES:

(1) LETTERS IN QUOTATION MARKS e.g., "A" REPRESENT ROD NUMBERS OR TERMINAL BOARD NUMBERS.

(2) "X_1" REFERS TO RODS 1-6, "X_2" REFERS TO RODS 7-12.

(3) THERE ARE TWELVE (12) IDENTICAL CIRCUITS [SIX (6) FROM EACH POWER SUPPLY]
SEE TABLE FOR CORRESPONDING NUMBERS:

		"X_1"						"X_2"					
	SRR-"X" SRF-"X"	1	2	3	4	5	6	7	8	9	10	11	12
TERMINAL NO.	TB-"A"	879	882	884	886	888	890	773	775	777	779	817	819
	TB-"B"	880	881	883	885	887	889	774	776	778	816	818	820
	PIN "1"	D	E	F	P	R	S	D	E	F	P	R	S
	PIN "2"	H	J	K	L	M	N	H	J	K	L	M	N

Fig. IV-19. ^{10}B Rod Drive Circuit

3. A minimum of one dual-purpose rod per half must be in the core (position of maximum reactivity worth) before the moveable table may be moved.

4. A minimum of one dual purpose rod per half must be designated a "control" rod and must be in its position of least worth before the table is moved.

5. No rod motion may take place while the table is moving or between its completely together or separated positions.

6. Rods designated as "safety" may be gang driven to their positions of maximum worth when the tables are fully separated.

7. When tables are together only one rod at a time may be moved to increase reactivity.

8. Four of the ^{10}B rods may be designated as control rods.

In Fig. IV-20 is shown the schematic diagram for the source drive which is used to introduce the Am-Be startup source into the reactor before table motion or rod drive motion is initiated.

C. Power Feedrail System

The ZPR-6 and -9 utilizes limit switches and a two solenoid operated moveable block system as described in Ref. 1 to insure that the correct table speed motion is maintained in the three forward speed zones. During the five years of operation of these facilities where there have been over 1000 startups of each facility, there has been no instance when a table has moved at an incorrect speed. The limit switches and block system have operated as designed. Although the reliability of the system has been proven during the five years of operation in order to further increase the reliability of the table speed control, a power feedrail system has been installed on both facilities.

The power feedrail system is a device which allows power to be applied to the table drive motors only when the table is in a certain position. The forward table movement toward closure is accomplished with the use of three separate 220 volt 3 phase induction motors for the three different speed ranges. The fast speed (25 cm/min) drive motor and the intermediate speed (5.1 cm/min) drive motor will have power fed to them through a separate conducting rail system. Each conducting rail system provides power to its associated motor within a designated drive range; outside of this range no power is available because the conducting rails are terminated.

The power feedrail system consists of two 3-conductor rail assemblies (Feedrail Corp., New York, New York) and two trolleys with power take-off brushes as shown in Figs. IV-21 and IV-22. An actual photo of an

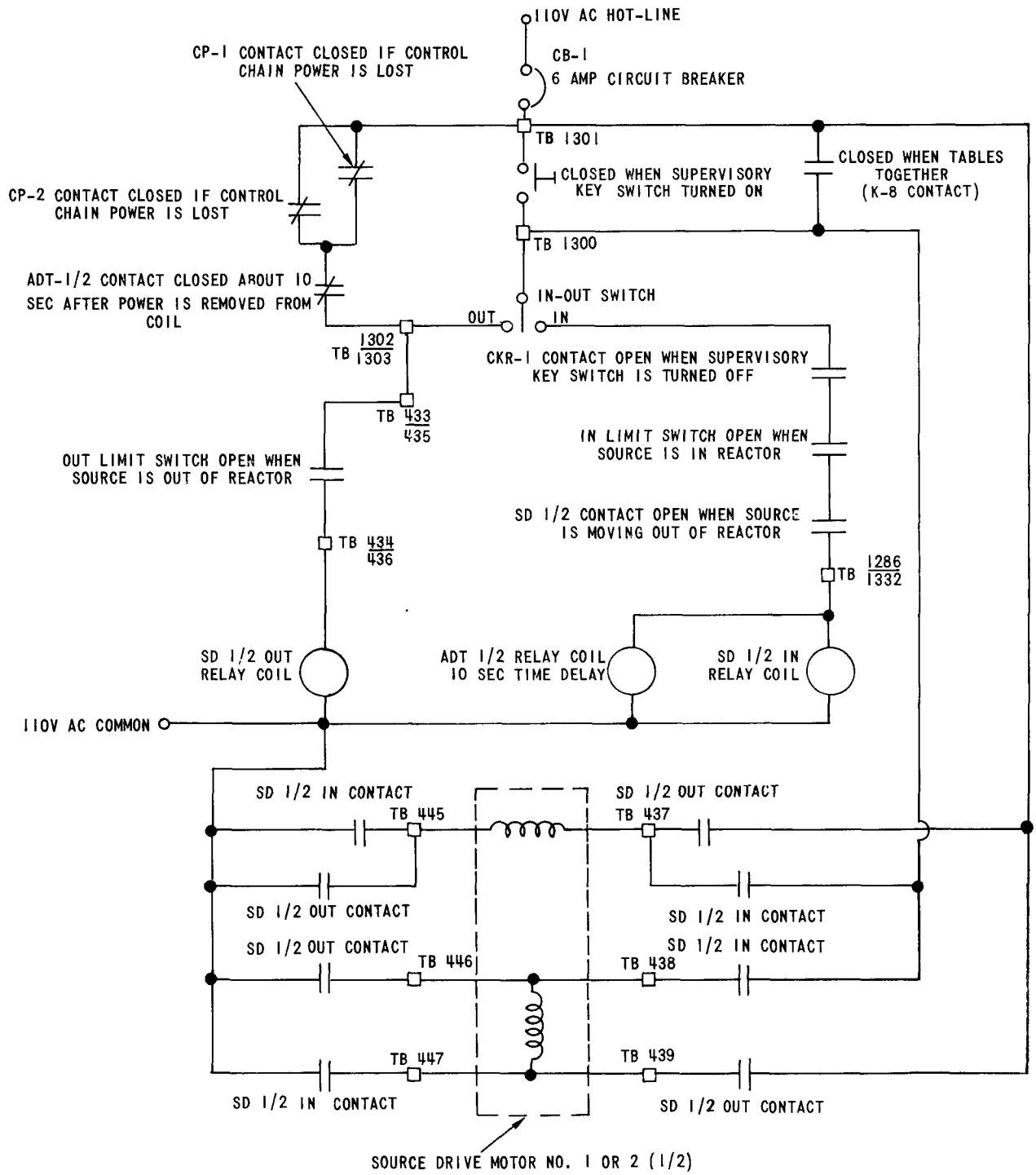


Fig. IV-20. Moveable/Stationary Source Drive. ANL Neg. No. 112-8120 Rev.1.

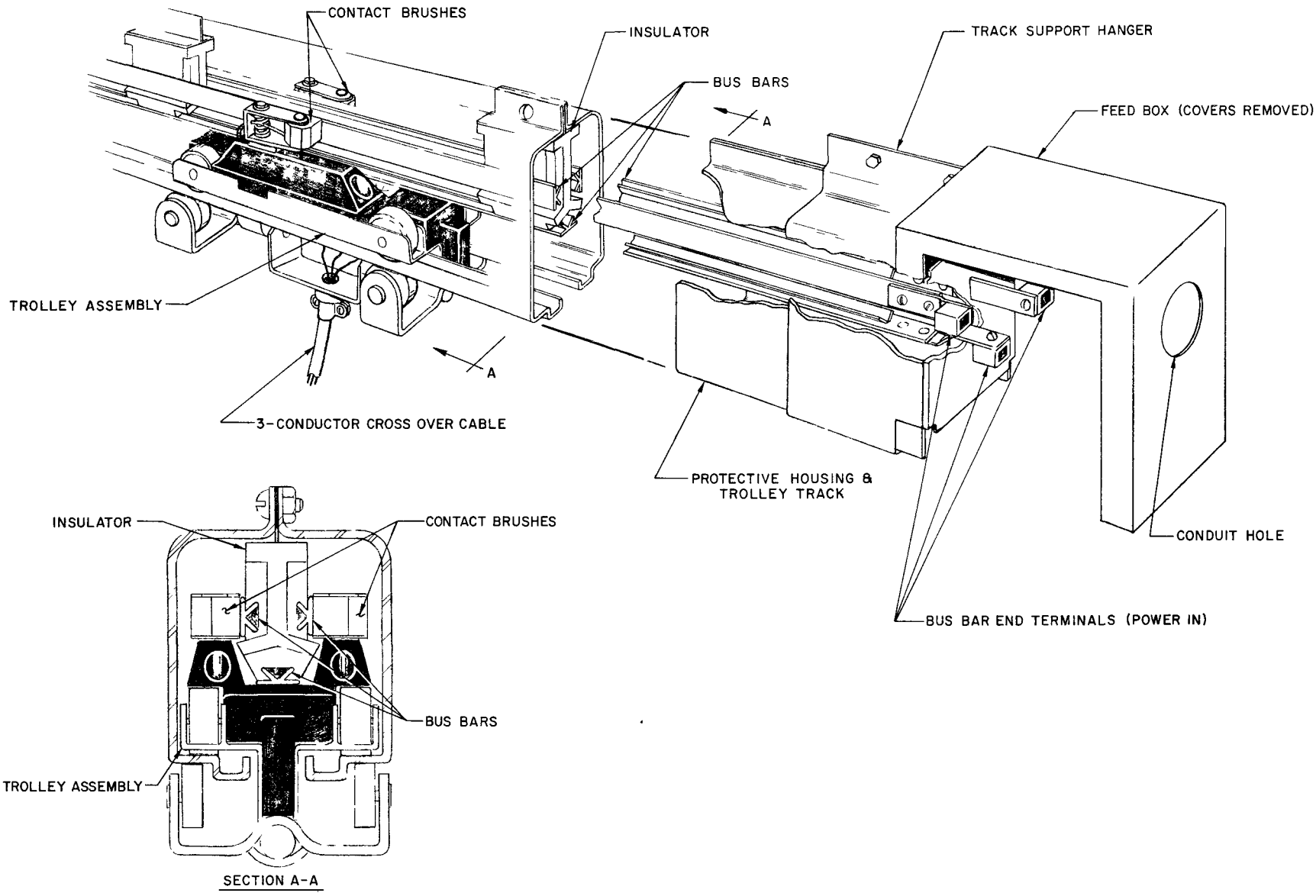


Fig. IV-21. Typical Feedrail Track and Trolley Assembly

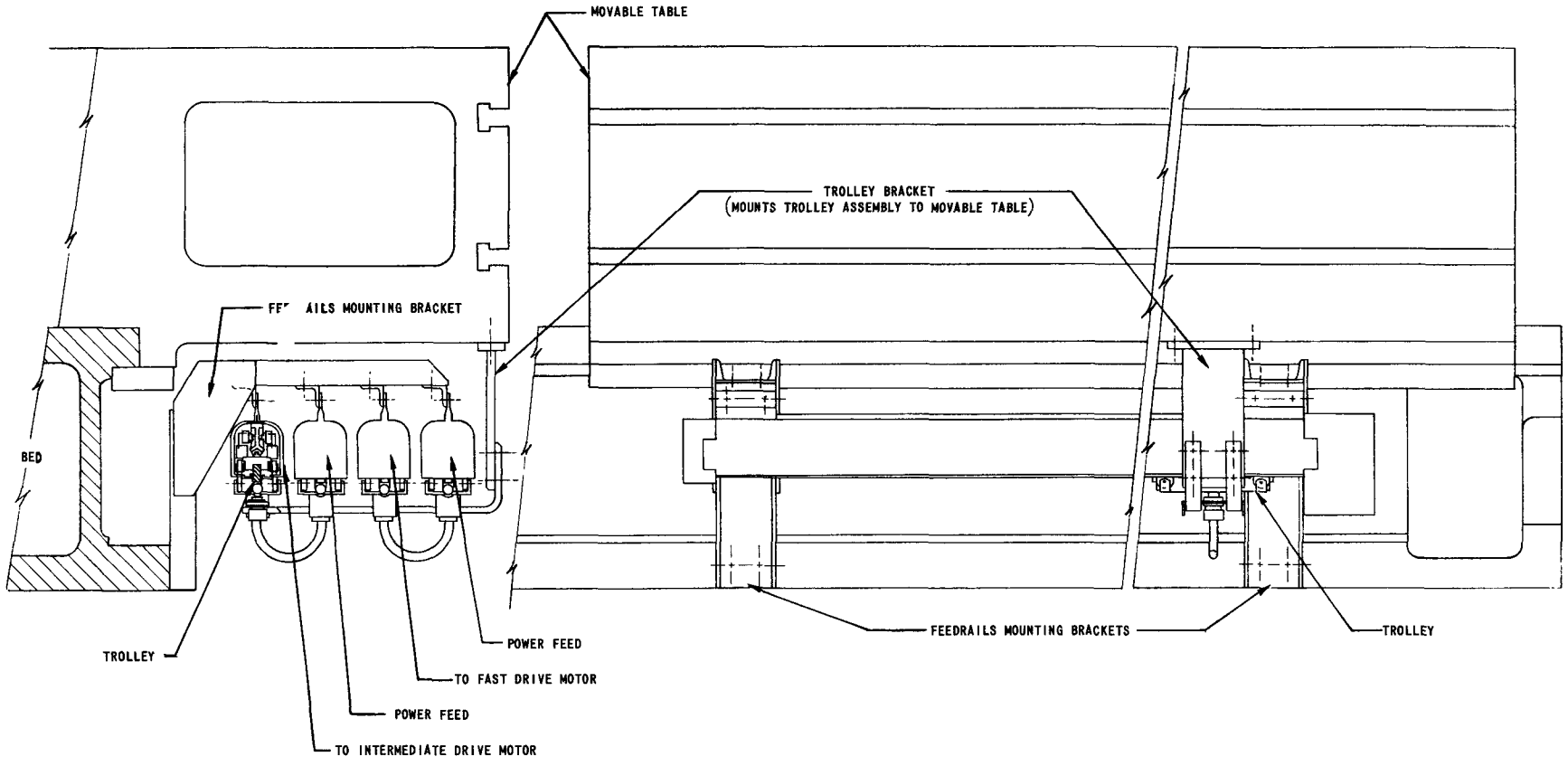


Fig. IV-22. Feedrail Assembly

installed system on ZPPR is shown in Fig. IV-23. Power is fed to a motor by routing it down the three rails (one rail for each phase) of a rail assembly, picking the power off with brushes contained in the trolley, transferring the power to the second trolley which is riding on a second set of three rails, and subsequently transferring power to this second set of rails through the brushes in the second trolley. Since the motor is electrically connected to the second set of rails, the physical extent of the second set of conducting rails limits the range over which motor power is available. No power can be transferred to the motor over that portion of travel where no power rails exist.

As can be seen the rails are located in such a manner that accidental shorting of the circuits is made difficult. A schematic diagram is given in Fig. IV-24 which shows how the motors are connected electrically to the feedrail system.

Only the fast and intermediate speed drive motors utilize the power feedrail system. The slow forward speed drive motor and the separation or scram motors are connected directly to the 220 volt 3 phase line.

D. Reactor Cooling System

The greater specific alpha activity of plutonium compared to ^{235}U results in the production of appreciably more heat energy. Plutonium containing 12% ^{240}Pu produces about 3.1 W/kg of plutonium and the 27% ^{240}Pu about 10 W/kg of plutonium. In terms of the plutonium-uranium, fuel alloy this is 0.9 and 3 W/kg of fuel, respectively.

For accurate reactivity measurements it is important to minimize the rate of temperature change in the core and stabilize the temperature distribution profile. A reactor cooling system has been designed to remove heat generated in the reactor core by the natural alpha activity of the plutonium. To cool the reactor core, cell air will be drawn across the core by blowers. Figure IV-25 shows the shrouding around the rear of the matrix and rod drives, and the flow patterns when the tables are separated and together. When the halves are separated, each half is cooled by air which is drawn from the center across the matrix tubes. When the halves are together the air flow is in one direction, being drawn across both halves.

A vertical view of the cooling system showing the location of the HEPA filters and blowers is shown in Fig. IV-26. All the air circulated through the core is passed through HEPA filters. The blower is interlocked so that it is turned off and air circulation stopped if the matrix temperature rises above a given set level.

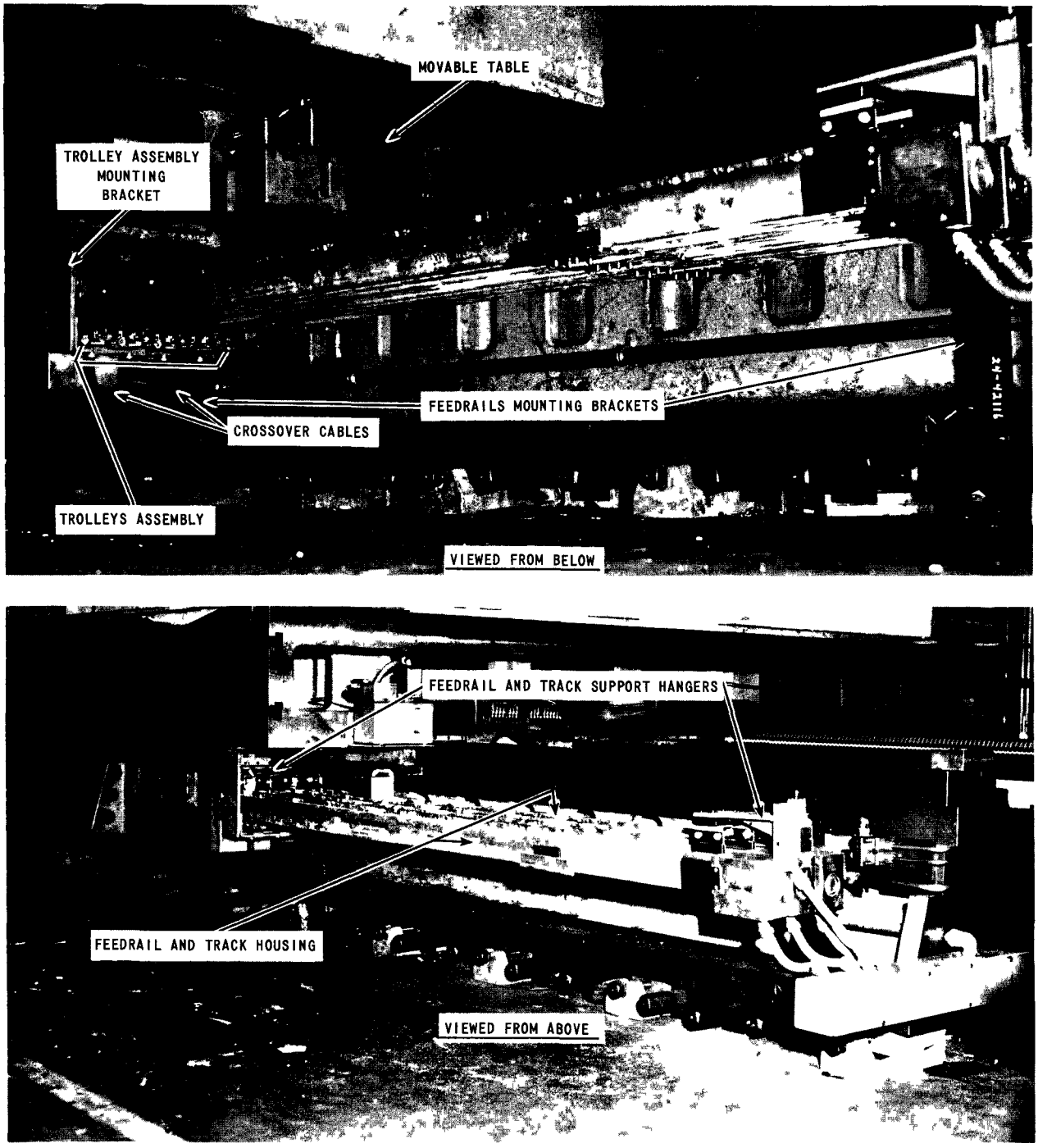


Fig. IV-23. Feedrail Installation

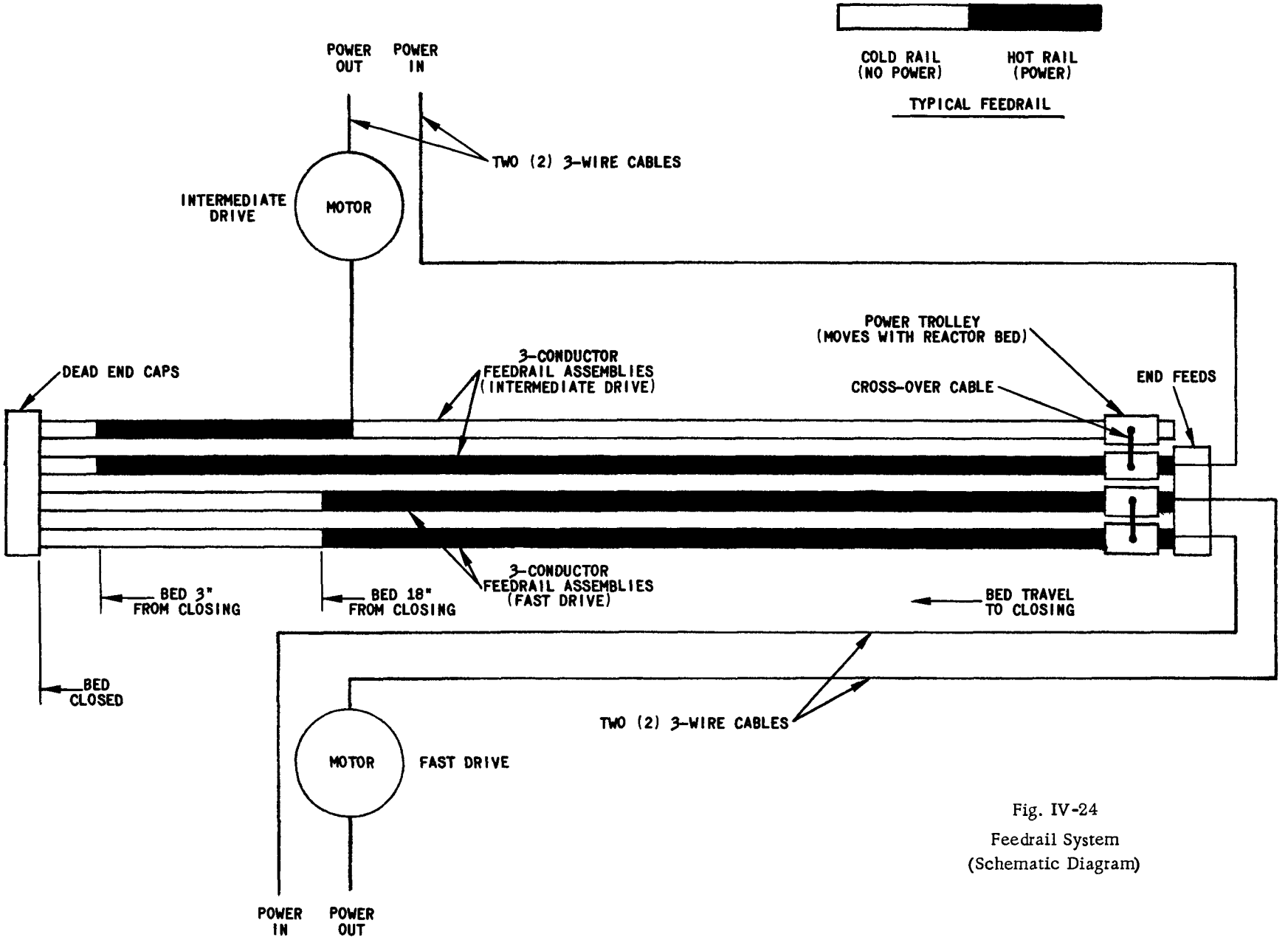


Fig. IV-24
Feedrail System
(Schematic Diagram)

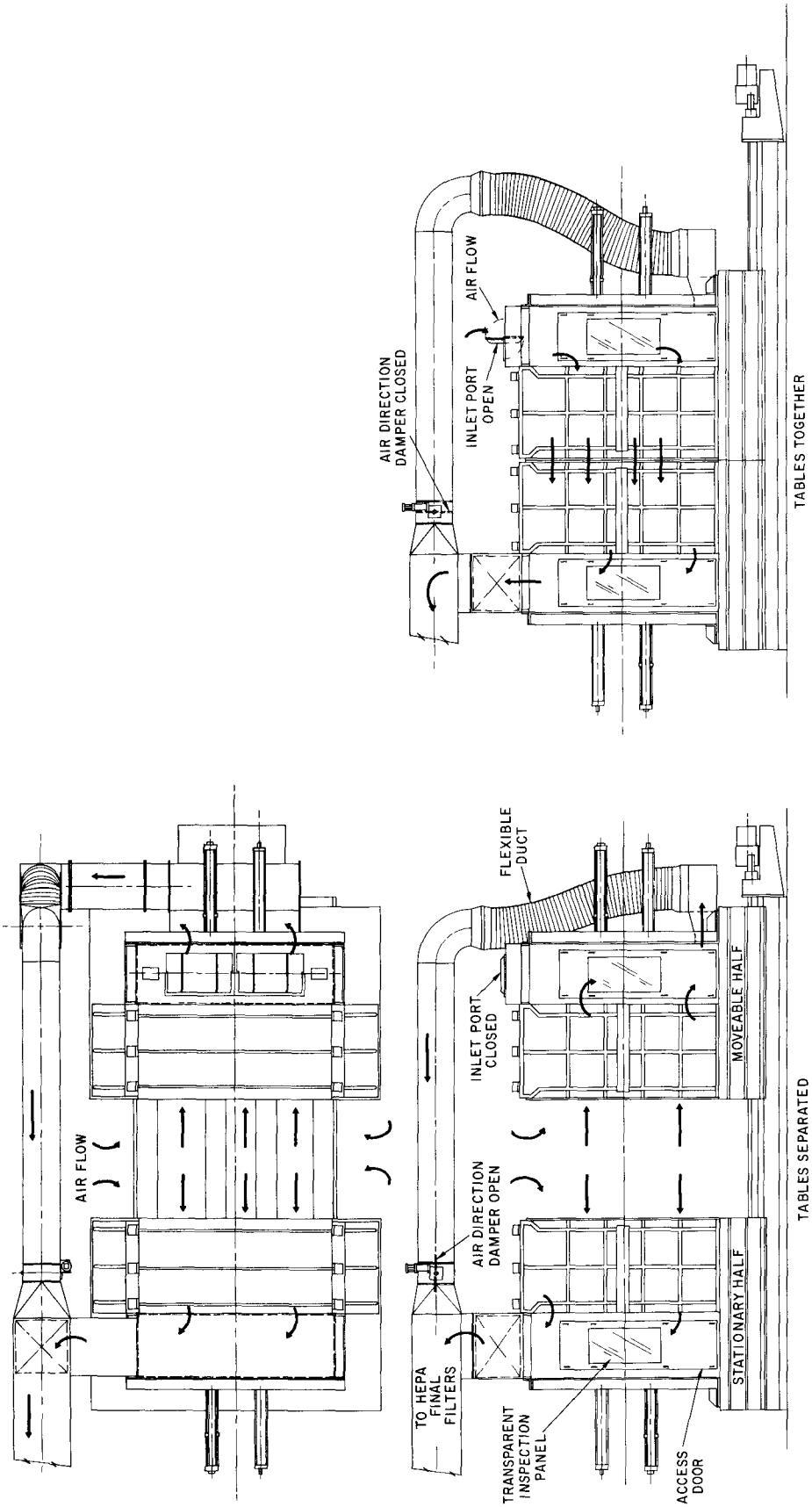


Fig. IV-25. Reactor Cooling System

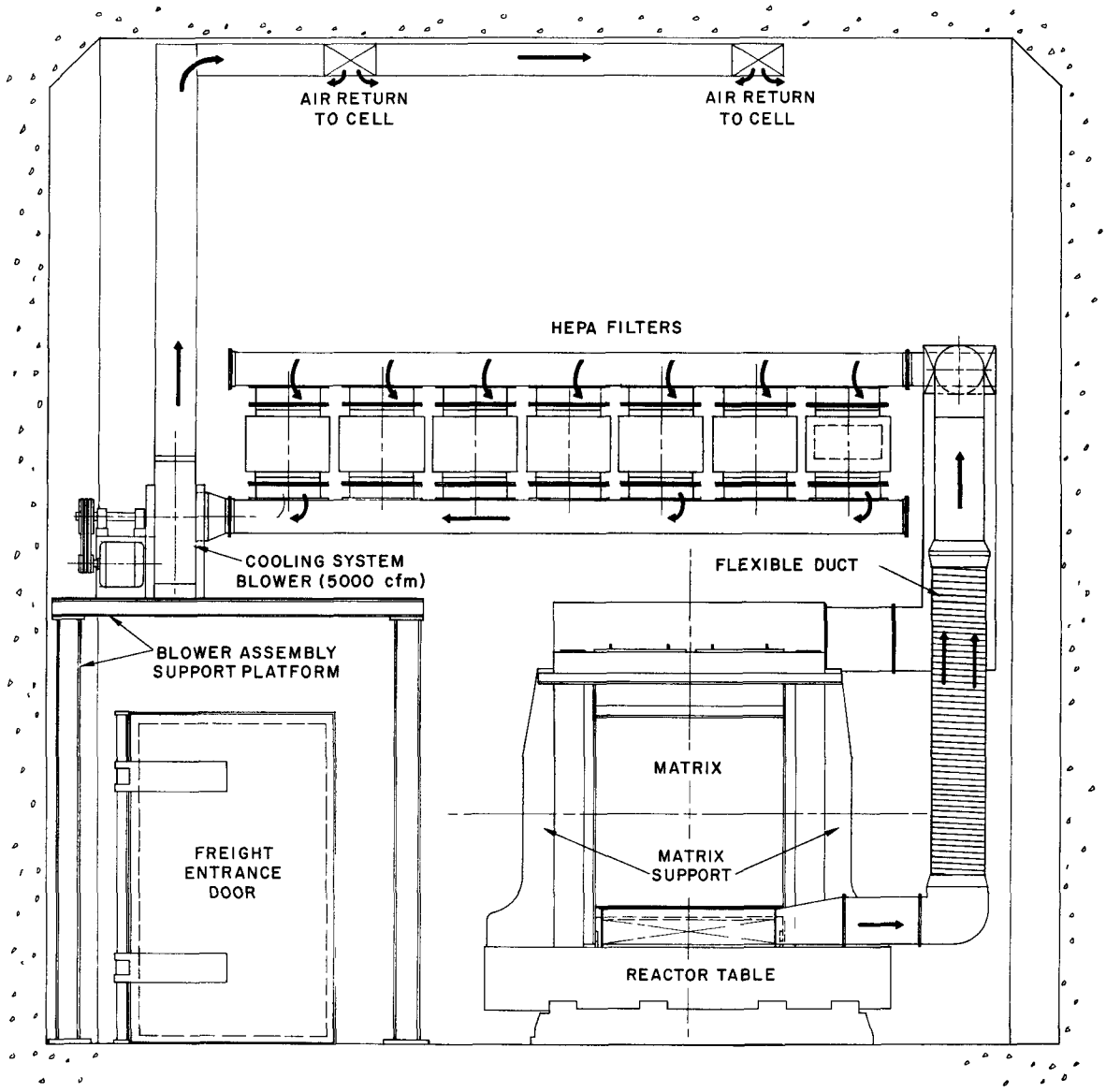


Fig. IV-26. Vertical View Reactor Cooling System

Flow tests have been made on a 5 x 5 tube (760 cm^2) cross section 2.4 m (8 ft) in length mockup using standard matrix tubes and drawers. The flow rate per matrix tube as a function of pressure drop across the 2.4 m length is given in Fig. IV-27. It is estimated that with a flow rate of about 5 cfm/matrix tube there is sufficient air cooling to maintain the core temperature to within about ±3°C. It is estimated that a large core might have a cross sectional area of about 1000 tubes hence a blower has been selected which would operate at 2400 liters/sec (5000 cfm) with a pressure drop of 0.06 bars (25 in. of H_2O).

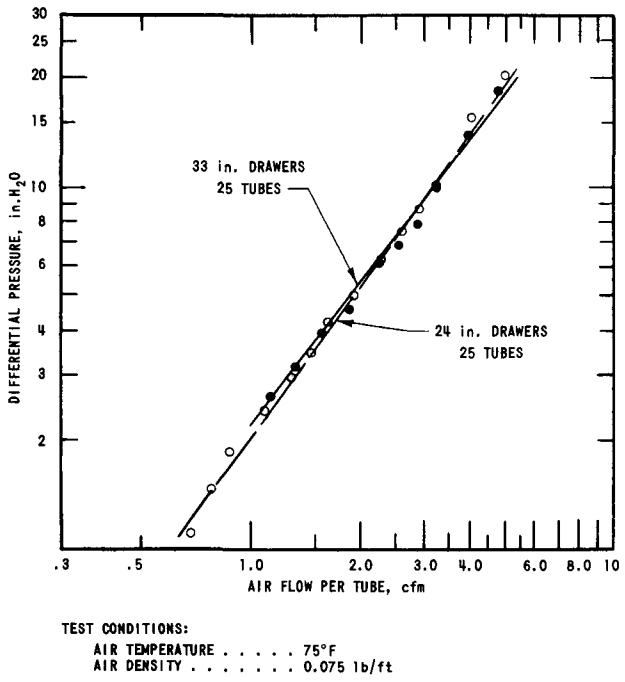


Fig. IV-27. Flow vs Pressure Drop across Loaded Matrix Tube

E. Radiation Monitoring System

Additional alpha monitoring devices are being placed in the cells, control rooms, vault, workroom, and in the exhaust systems for the detection of accidental plutonium releases. Typical examples of radiation monitors and survey instruments which will be used and their methods of application are outlined in the paragraphs below.

1. Continuous Alpha Monitoring System

A continuous alpha monitoring system consisting of ANL designed impactor type air monitors using passivated silicon detectors and a two channel pulse height analyzer has been developed and installed. This type of system will separate the ^{239}Pu alpha peak from the radon daughter peaks. Because of the pulse height distribution of the alpha activity from the radon daughters some of the counts will appear in the ^{239}Pu channel. In order to determine the real ^{239}Pu alpha content in the air it is necessary to correct the counting rate in the ^{239}Pu alpha channels using the value of the counting rate in the radon daughter channel. The alpha impactor and detectors will initially be located in the following areas:

(a) one on each reactor half in the plenum chamber of the cooling system,

(b) one on the reactor cell wall near the loading port from the workroom,

(c) one above the processing hood in the workroom,

(d) one above the loading hood in the workroom,

(e) one in the exhaust duct of the workroom hoods; and

(f) one in the exhaust duct of the plutonium vault.

The ^{239}Pu counting rate channel has a trip point so that when the counting rate exceeds a preset value there will be an alarm indication on the annunciator panel.

2. Low Level Alpha Activity Air Samples

Low air flow sampling points using a filter head will be located at the following places:

(a) above processing hood,

(b) above loading hood,

(c) above preloading table,

(d) fuel coating room,

(e) reactor cell near drawer ports,

(f) control rooms,

(g) plutonium vault,

(h) uranium vault,

(i) corridor between control rooms,

(j) corridor outside of air lock,

(k) stack; and

(l) isolation rooms (2).

These filter heads will allow air to be pulled through millipore filters for an extended period of time at a known flow rate. The filter paper will be removed from the filter head and the radon and thoron allowed to decay before the activity is counted for airborne alpha emitters. This will provide high sensitivity airborne plutonium monitoring capability.

3. Surface Monitors

Surface monitoring equipment will be used to survey plutonium fuel cans for gross alpha activity. Ten wire gas proportional alpha probe counters designed by the Laboratory will be used for monitoring the fuel plates and drawers in the different hoods in the wookroom as well as for personnel monitoring of hands and feet in the change rooms and in the reactor cells. They will be equipped with a count rate meter (visual output) and/or an earphone or speaker (audio output).

4. Miscellaneous Alpha Monitors

Alternating current operated alpha monitoring equipment will be available for checking air samples obtained from the confinement shell volume and from the stack. Air samples will be monitored from the confinement shell volume only in the event of an accidental discharge of plutonium in the cells. Air exhausting from the stack will be collected using a filter paper holder and monitored with ac operated equipment.

Portable equipment for floor monitoring will also be available.

5. Beta and Gamma Monitoring Equipment

a. Juno Survey Meters

Gamma radiation fields of ~5 mr/hr to 50,000 mr/hr can easily be measured with meters such as the Juno type survey meters manufactured by Technical Associates, Burbank, California. It is expected that massive quantities of plutonium could produce surface dose rates up to ~10 rads/hr for a dirty fuel, making it necessary to provide survey equipment to measure these radiation fields. The capability of measuring radiation fields up to 10^6 and 5×10^6 mr/hr using ionization type monitoring instruments with probe extensions of 4 and 12 ft is also available.

b. Low Level Beta-Gamma Survey Instruments

Survey instruments capable of measuring from background to 20 mr/hr levels will be used where levels are too low for the Juno meters to measure accurately. The Juno can reach no closer than 2 in. from the radiation source, while the low level probes can reach to within 1 in. from the source. Equipment such as model 107C survey instrument manufactured by Precision Radiation Instruments, Inc., Los Angeles, California, or equivalent, will be used for this purpose. The probe is sensitive to both beta and gamma radiation, but discrimination is easily accomplished by covering the probe with a beta shield.

F. Annunciator System

An annunciator system has been installed on the Control Consoles of each of the two facilities to provide the reactor operators with consolidated information regarding the status of the complete reactor system. Each annunciator panel consists of many annunciated points of which at least 53 points are common to annunciator panels on both the ZPR-6 and -9 consoles. These points present information on the status of the building pressure system, air locks and radiation monitors which could affect the operation of either reactor. The remaining points provide information only for the control room directly involved.

The annunciator information can be divided into the following categories:

(1) Radiation Monitor Status

(2) Nuclear Instrumentation Status

(3) Ventilation and Pressure System Status

(4) Door Status

(5) Miscellaneous Control Item Status

The layout of the complete annunciator is shown in Fig. IV-28. Separate sections are shown by use of a heavy black line.

The response of the annunciator for sections 1, 2, 4, 5, and 7 is as follows:

When an abnormal condition exists it will provide a flashing light and an audible alarm. The alarms will continue until acknowledged. If the field condition is still off normal, the acknowledgment will silence the audible alarm and cause the lamps to remain steady on. Reset to normal is automatic after field condition clears.

The sequence for sections 3 and 6 is as follows:

When an abnormal condition exists it will provide flashing lamps. Acknowledgment causes lamps to remain steady on. Reset to normal is automatic when condition clears regardless of operator action.

The annunciator design is as fail safe as possible. Where the read-out permits, at least two lamps are used so that burnout of one will not take a point out of operation. If one lamp fails the other comes on brighter, indicating the need for replacement. All lamps are overrated and are normally operated on dim to reduce thermal stress.

All field points are normally closed and are operated with a voltage applied. If an open occurs, or if a point is shorted to ground, an alarm condition will exist.

Reset and Test buttons for the annunciator panel are located on the reactor control console. All points on the annunciator can be checked through the use of the test button.

G. Confinement Shell Pressure Indication

As a part of the modifications to the ZPR-6 and -9 facilities a confinement shell as described in Chapter V has been constructed around the

SEC. 1	REACTOR GAMMA HIGH	CORE TEMPERATURE HIGH	CELL ALPHA HIGH	CELL GAMMA HIGH	CELL TEMPERATURE HIGH	CELL PRESSURE HIGH
	STATIONARY HALF ALPHA HIGH	MOVABLE HALF ALPHA HIGH	CONTROL ROOM GAMMA HIGH	NORMAL EXHAUST GAMMA HIGH	EMERGENCY EXHAUST GAMMA HIGH	
SEC. 2	CHANNEL 1 HIGH VOLTAGE LOW	CHANNEL 3 HIGH VOLTAGE LOW	CHANNEL 5 HIGH VOLTAGE LOW	CHANNEL 7 HIGH VOLTAGE LOW	ESS VENT CONTROL BYPASS	CELL AIR VALVE BYPASS
	CHANNEL 2 HIGH VOLTAGE LOW	CHANNEL 4 HIGH VOLTAGE LOW	CHANNEL 6 HIGH VOLTAGE LOW	CHANNEL 8 HIGH VOLTAGE LOW	SOURCE IN BYPASS	TABLE SEPARATION BYPASS
	CHANNEL 1 COUNT RATE LOW	CHANNEL 4 NEUTRON FLUX HIGH	CHANNEL 5 NEUTRON FLUX HIGH	CHANNEL 7 TABLE TRIP	CHANNEL 7 ADJUSTABLE TRIP	CHANNEL 7 BACKUP TRIP
	CHANNEL 2 COUNT RATE LOW		CHANNEL 6 NEUTRON FLUX HIGH	CHANNEL 8 TABLE TRIP	CHANNEL 8 ADJUSTABLE TRIP	CHANNEL 8 BACKUP TRIP
SEC. 3	INNER PERSONNEL DOOR OPEN	INNER FREIGHT DOOR OPEN	INNER DRAWER TUBE #1 OPEN	INNER DRAWER TUBE #2 OPEN	INNER DRAWER TUBE #3 OPEN	UPPER ESCAPE HATCH CLOSED
	OUTER PERSONNEL DOOR OPEN	OUTER FREIGHT DOOR OPEN	OUTER DRAWER TUBE #1 OPEN	OUTER DRAWER TUBE #2 OPEN	OUTER DRAWER TUBE #3 OPEN	LOWER EXCAPE HATCH OPEN
	RBV 51 OPEN	SBV 51 OPEN	RBV 52 OPEN	SBV 52 OPEN	SCRAM AIR PRESSURE LOW	TIMER OVERRIDE
						PLATFORM NOT UP
SEC. 4	ARGON PRESSURE LOW	STEAM PRESSURE LOW	-6" PRESSURE WARNING	-1" PRESSURE WARNING	MOVABLE HALF SHUTTER NOT OPEN	MOVABLE HALF DAMPER NOT CLOSED
		WORKROOM DUCT ALPHA HIGH	PLUTONIUM VAULT ALPHA HIGH	LOADING HOOD ALPHA HIGH	PROCESSING HOOD ALPHA HIGH	TIMER RELAY
SEC. 5	CONTROL ROOM PRESSURE HIGH	CENTRAL EXHAUST PRESSURE HIGH	WORKROOM PRESSURE HIGH	BASEMENT PRESSURE HIGH	PENTHOUSE PRESSURE HIGH	VAULT PRESSURE HIGH
			WORKROOM STANDBY FAN ON	BASEMENT STANDBY FAN ON		VAULT STANDBY FAN ON
	CENTRAL STANDBY FAN ON	TURBINE ON	EMERGENCY GENERATOR ON LINE	CENTRAL EXHAUST DAMPER #1 NOT OPEN		
	SHELL STANDBY FAN ON	TURBINE TEST		CENTRAL EXHAUST DAMPER #2 NOT CLOSED		SERVICE FLOOR DOOR #2 OPEN
SEC. 6	SERVICE FLOOR AIRLOCK 01 OPEN	SERVICE FLOOR AIRLOCK 02 OPEN	SERVICE FLOOR AIRLOCK 03 OPEN	MAIN ENTRANCE AIRLOCK OPEN	ZPR 9 CONTROL ROOM AIRLOCK OPEN	ZPR 6 TECH ROOM AIRLOCK OPEN
	A DOOR / B DOOR	A DOOR / B DOOR	A DOOR / B DOOR	A DOOR / B DOOR	A DOOR / B DOOR	A DOOR / B DOOR
	S.E. SHELL AIRLOCK OPEN	S.W. SHELL AIRLOCK OPEN	CARGO DOOR OPEN	WORKROOM-VAULT PASSAGE TO SHELL DOOR OPEN	SECOND FLOOR AIRLOCK OPEN	THIRD FLOOR AIRLOCK OPEN
	A DOOR / B DOOR	A DOOR / B DOOR	#1 / #2	#1 / #2	A DOOR / B DOOR	A DOOR / B DOOR
SEC. 7	WORKROOM TO VAULT DOOR OPEN	ESS DOOR OPEN	BYPASS PANEL OPEN	MAIN ARGON VALVE OPEN	ZPR-6 ARGON VALVE OPEN	ZPR-9 ARGON VALVE OPEN
	CENTRAL VACUUM SYSTEM OFF	AIR MOTOR SCRAM	CELL OXYGEN LOW	ARGON VALVE COVER REMOVED	ROD DROP	PHASE REVERSAL
	TURBINE OVER SPEED					

Fig. IV-28. Annunciator Panel

reactor cells. The confinement shell volume will normally be maintained during reactor operation at 0.994 bars (-2.5 in. of water with respect to atmosphere) pressure so that in the event of a severe nuclear accident particulate matter will not escape to the atmosphere. It has been estimated that a differential pressure of at least 0.998 bars (-1.0 in. of water) between the atmosphere and the confinement shell volume would be sufficient to prevent outleakage of particulate matter by diffusion through any small leaks or openings to the atmosphere. Although the exhaust fans of the confinement shell volume will maintain a pressure of 0.994 bars (-2.5 in. of water) within the volume, the actual differential pressure on a given wall of the confinement shell may vary because of wind currents. It has been estimated that a 0.998 bar (-1.0 in. of water) differential pressure may be maintained on the outside wall surface as long as the wind velocity does not exceed 62 mph. In addition the confinement shell has been designed to withstand differential pressures up to about 6 in. of water. These boundary conditions necessitate the installation of pressure measuring devices on the various surfaces of the confinement shell so that the reactor operators become aware of the condition when the negative 0.998 bars (-1.0 in. of water) is no longer maintained on a given surface or when the differential pressure exceeds 6 in. of water.

Because of variation in wind velocity and direction, the differential air pressure is measured at 49 selected points on the shell by means of differential pressure switches (F. W. Dwyer Manufacturing Co., Michigan City, Indiana); 17 of these switches are set to open when the differential pressure exceeds -6 in. of water (Model 1820-10-A-MIL); the remaining 32 switches are set to open when the differential pressure falls below -1.0 in. of water (Model 1820-2-A-MIL). The location of these switches on the confinement shell surfaces are shown in Fig. IV-29.

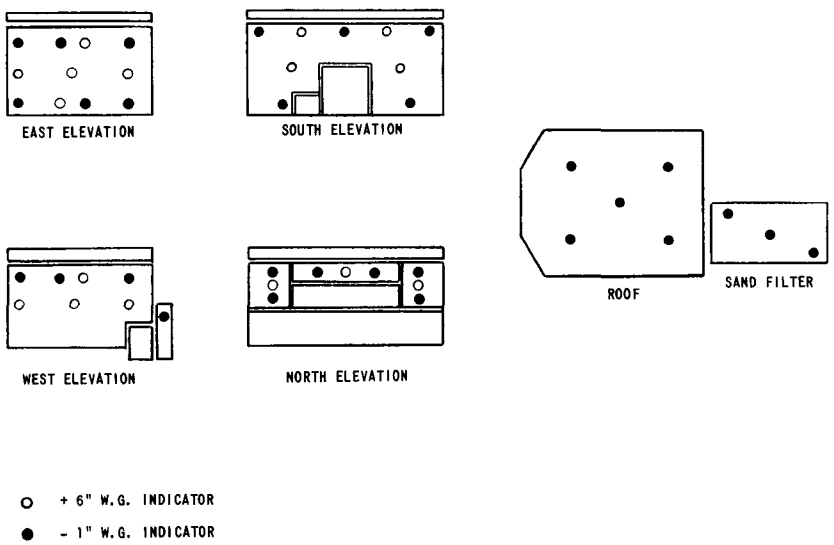


Fig. IV-29. Shell Pressure Switch Locations

If a differential pressure greater than -6 in. of water is detected by the pressure switches, an indication and alarm will appear on the control console. The shell fans will then be turned off to prevent possible damage to the shell. If the pressure drops below -1.0 in. of water an indication is displayed in the control room. The indications on the pressure switch panel on the control console and information from the wind velocity and direction detectors located on the roof and monitored at the control console will be used to decide if reactor operations should be terminated until such time as more favorable wind conditions exist.

The readout for the pressure switches will be on a panel with silhouette elevations of the building. Lights will be located on the silhouette at positions corresponding to the locations of the pressure switches on the building shell. If a pressure switch is activated a corresponding lamp will light on the Shell Pressure Readout and this fact will initiate an alarm signal on the main annunciator panel.

All circuitry is as fail safe as possible; consistent with good design practice. Two wires will be run to each switch and a potential applied so that any open or short to ground will cause an indication on the panel.

H. Temperature Monitoring System

Temperature monitoring of the ZPR-6 and -9 reactors will be with thermocouples and a multipoint recorder. Thermocouples will be placed in the matrix, in a few drawers adjacent to plutonium elements, and in the exhaust air duct for each half of the reactor. The reactor cooling system will be shutdown whenever the reactor core temperature goes significantly above normal operating temperatures indicating the possibility of a fire in the reactor.

The temperature recorder will be supplemented by a digital voltmeter which will be connected to measure the output of a number of thermocouples, including those monitored by the multipoint recorder. The digital voltmeter will serve to measure temperatures more accurately than will the multipoint recorder, whenever such a need exists.

REFERENCES

1. W. Y. Kato, G. J. Fischer, and L. R. Dates, "Safety Analysis Report, Argonne Fast Critical Facility (ZPR-VI)," ANL-6271 (December 1963).

2. W. Y. Kato and L. R. Dates, "Design and Construction of the Argonne Fast Critical Facility (ZPR-VI)," Proc. Physics of Fast and Intermediate Reactors, IAEA, Vienna, 1962.

3. L. R. Dates, "Design, Construction Details, and Preoperational Testing of an Argonne Fast Critical Facility," ANL-7195 (April 1966).

4. C. E. Cohn et al., "Safety Analysis for the ZPR-9 Facility [An Addendum to ANL-6271, Safety Analysis Report, Argonne Fast Critical Facility (ZPR-6)]," ANL-7166 (February 1966).

5. W. Y. Kato et al., "Preliminary Safety Analysis Report on the Use of Plutonium and ^{233}U in ZPR-6 and -9 [Addendum No. 3 to Safety Analysis Report, Argonne Fast Critical Facility (ZPR-6), ANL-6271]," ANL-7211 (to be published).

Chapter V

MATERIALS

A. General

In order to have a high degree of flexibility in the compositions and geometries which are studied on the Argonne fast critical facilities ZPR-3, -6, -9 and ZPPR, the Laboratory's inventory of fissile, fertile, structural and and coolant materials are in plate form. The plates which are used in the core have nominal thickness of 1/32, 1/16, 1/8, 1/4 and 1/2 in. with a height of 2 in. and of varying length in inch increments. Examples of such plates may be seen in Fig. V-1. Blocks having thicknesses of one and two inches are also included in the inventory for use in the blanket or reflector region of the reactors. The materials used in the Argonne fast critical facilities are of standard dimensions so that the materials may be used interchangeably in all four facilities. All of the materials are color coded or stamped with identifying marks for ease of identification.

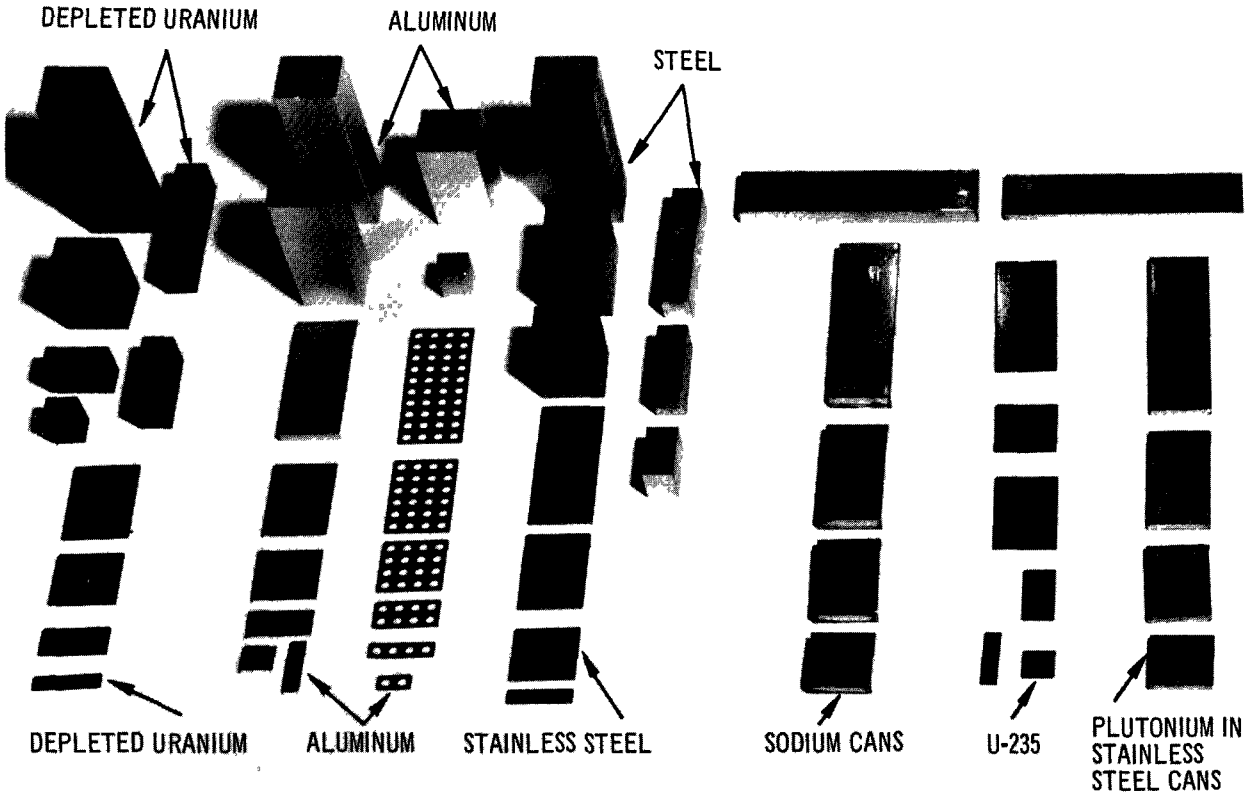


Fig. V-1. Plates and Blocks for ZPR-6 and -9

A typical drawer loading is shown in Fig. V-2 where three drawers are shown loaded with fuel, fertile material and sodium cans.

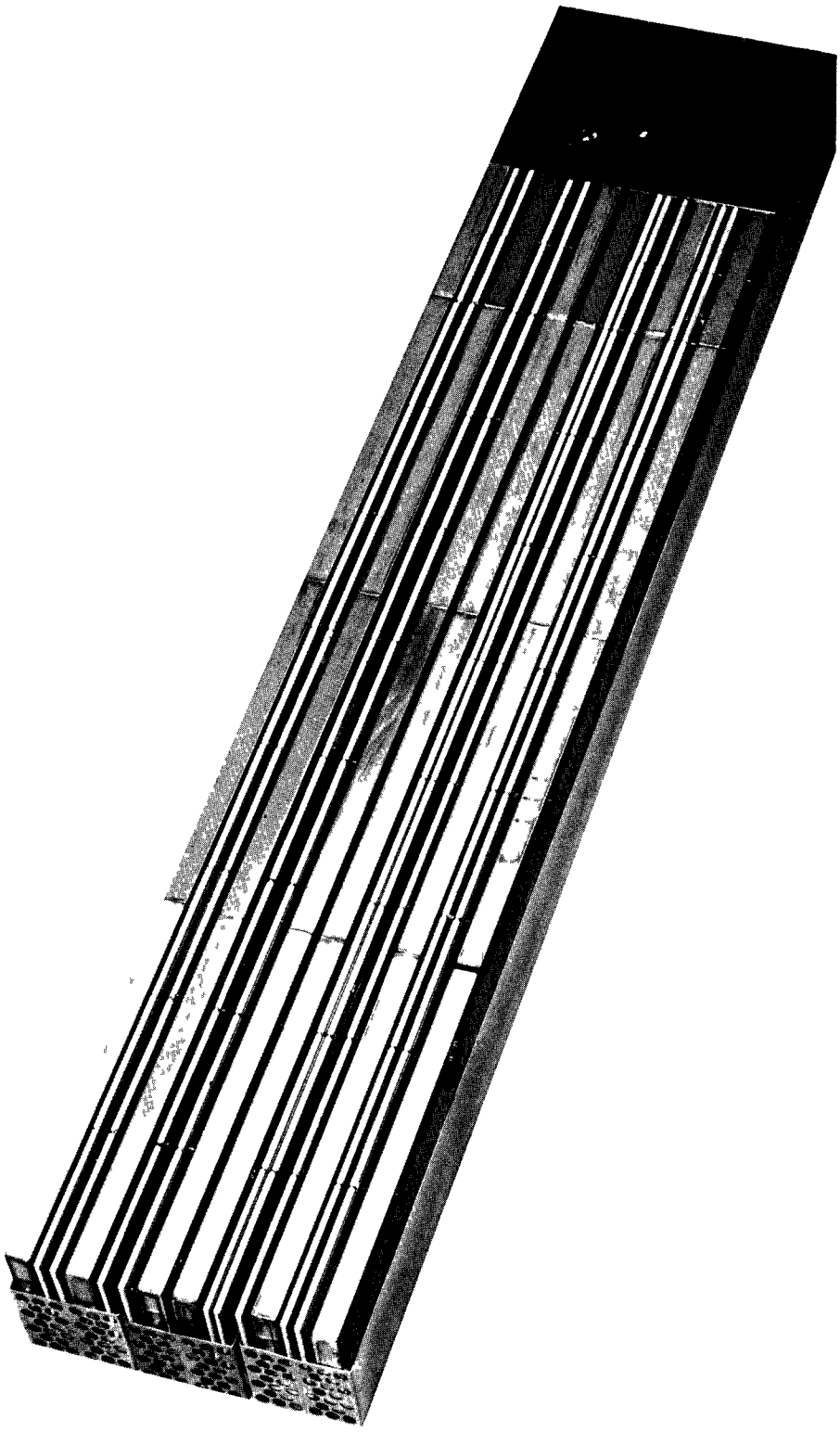


Fig. V-2. Three Loaded Drawers

B. Fuel and Other Materials

The Laboratory maintains a large inventory of fissile materials for use in ZPR-3, -6, -9 and ZPPR. The uranium fuel inventory is about 2300 kg of ^{235}U while the total plutonium inventory will be in excess of 3000 kg. The general dimensional and isotopic characteristics of the fuels available to ZPR-6 and -9 are listed in Table V-I.

Table V-1

FISSILE AND FERTILE MATERIALS

Material	Nominal Dimension, in.	Total Quantity, kg	Clad	Comments
		Heavy Metal		
1. 93% ^{235}U	(1/32, 1/16, 1/8) x 2 x (1, 2, 3)	2,560[a]	Unclad	
2. 20% ^{235}U	1/16 x 2 x (1, 2, 3)	180[a]	Unclad	
3. 31% ^{235}U	1/8	83[a]	Unclad	
4. 98.9 w/o Pu, 1.1 w/o Al	1/8	210[b]	SS	95% ^{239}Pu, 4.5% ^{240}Pu (Ref. 1)
5. 98.9 w/o Pu, 1.1 w/o Al		17.4[b]	SS	72.8 ^{239}Pu, 21.9 ^{240}Pu (Ref. 2)
6. 20 w/o Pu, 77.5 w/o U, 2.5 w/o Mo	1/4	175[b]	SS	91% ^{239}Pu, 8.5% ^{240}Pu (Ref. 2)
7. 28 w/o Pu, 69.5 w/o U, 2.5 w/o Mo	1/4	2,900[b]	SS	88% $^{239+241}Pu$, 12% ^{240}Pu
8. 36 w/o Pu, 61.5 w/o U, 2.5 w/o Mo	1/4	190[b]	SS	73% $^{239+241}Pu$, 27% ^{240}Pu
Depleted Uranium	1/16, 1/8	20,000	Unclad	
Depleted Uranium Blocks	2, 5	154,000	Unclad	
		Oxide		
5.7/1 U/Pu Rod Elements	0.375 diam	150[b]	SS	
2.4/1 U/Pu Rod Elements	0.375 diam	135[b]	SS	
5.1/1 $^{238}U/^{235}U$ Rod Elements	0.375 diam	150[c]	SS	
1.2/1 $^{238}U/^{235}U$ Rod Elements	0.375 diam	150[c]	SS	
5.7/1 U/Pu Plates	1/4	40[c]	SS	
Depleted Uranium Oxide Rod Element	0.375 diam	900[a]	SS	

[a] kg of total uranium.
[b] kg of total plutonium.
[c] kg of ^{235}U.

The uranium fuels have been used in ZPR-3, -6 and -9 for many years. They are basically rolled metal plates which are coated with about a 0.001 cm thick nonhydrogeneous KEL-F (3M Co.) trifluorochloroethylene polymer with red pigment (iron oxide). The coating is to reduce oxidation, to minimize alpha contamination, and to provide an easy means of identification. The 93% enriched uranium plates are colored red, while the 20% plates are red with white strips and the 31% enriched plates have a yellow pigment.

The plutonium undiluted by ^{238}U has been described in ANL-6504 (Ref. 1); thus only the new 28 w/o plutonium alloys will be described here. The largest fraction of the plutonium fuel for use on ZPR-6 or -9 will be

a 28 w/o Pu, 69.5 w/o U, 2.5 w/o Mo alloy in plate form which has been developed by the ANL Metallurgy Division. The plutonium alloy has the nominal overall dimensions of 0.64 x 5 x (10, 12.5, 15, 17.5, and 20 cm) (1/4 x 2 x (4, 5, 6, 7, 8) in.) and will be tightly canned in 0.015-in. thick stainless steel (Type 304) cans as shown in Fig. V-3. This particular alloy has been found by Kelman[3] to be highly resistant to air corrosion. Some of its other properties are listed in Table V-2. The nuclear properties of the fuel isotopes are given in Table V-3.

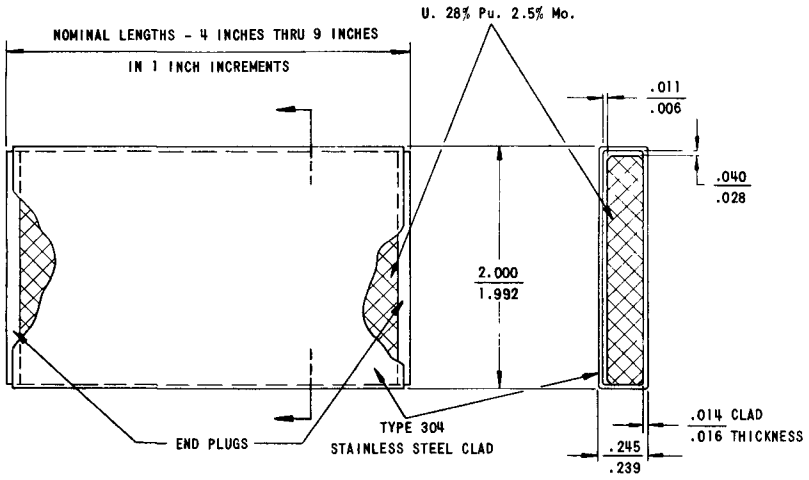


Fig. V-3. Plutonium Fuel Element

Table V-2

PROPERTIES OF URANIUM PLUTONIUM ALLOY [3,4]

Linear Expansion Coefficient

25 - 220°C		15.4×10^{-6} /°C
220 - 380°C		28.4×10^{-6} /°C
380 - 545°C		28.7×10^{-6} /°C
25 - 545°C	Average	23.6×10^{-6} /°C
545 - 620°C		0.39% Change in length
620 - 700°C		24.7×10^{-6} /°C

Density

18.3 g/cm^3

Mean Specific Heat

0.037 cal/g°C

Melting Point

880 - 980°C

The fuel elements are currently being fabricated using precision casting techniques. The plutonium alloy core plates are loaded into stainless steel cans and end plugs are welded in place. There is no more than 0.08 mm (0.003 in.) spacing between the uranium-plutonium alloy and clad in the axial direction. X-rays of randomly sampled fuel elements are made to insure that the fuel element manufacturer has met the aforementioned specifications. For a 10 cm (4 in.) long fuel element this results in a maximum loss of 48°C in the fuel expansion coefficient. Since there is contact between the core and end plugs, there is prompt transmission of the thermal expansion to the jacket ends and motion to the rest of the column of fuel.

Table V-3

NUCLEAR PROPERTIES OF FUEL ISOTOPES

Isotope	Half Life (yrs)	Spec. Activity (dis/sec g)	Watts/kg	Spontaneous Fission Half Life (yrs)	Neut/sec kg of Isotope	Delayed Neutron Fraction, β Fast Fission	Isotopic Comp. (%)	Watts/kg	Neut/sec kg of Pu	Isotopic Comp. (%)	Watts/kg	Neut/sec kg of Pu
U^{233}	1.62×10^{5}	3.51×10^{8}	0.27	--		0.0027						
U^{235}	7.1×10^{8}	7.93×10^{5}	5.5×10^{-4}	1.8×10^{17}	6.1×10^{-1}	0.0065						
U^{238}	4.51×10^{9}	1.23×10^{5}	8.3×10^{-5}	8.04×10^{15}	1.5×10^{2}	0.0157						
Pu^{238}	86.4	6.44×10^{11}	567	4.9×10^{10}	2.3×10^{6}		0.05	0.28	1.2×10^{3}	1	5.7	2.3×10^{4}
Pu^{239}	24,360	2.27×10^{9}	1.85	5.5×10^{15}	2.0×10^{1}	0.0021	86	1.58	1.7×10^{1}	54	1.0	1×10^{1}
Pu^{240}	6,580	8.39×10^{12}	6.85	1.2×10^{11}	1.0×10^{6}	0.0026	12	0.82	1.2×10^{5}	30	2.1	3×10^{5}
Pu^{241}	13.0	4.23×10^{12}	14	--			1.8	0.25	--	10.8	1.5	--
Pu^{242}	3.79×10^{5}	1.44×10^{8}	0.112	7.1×10^{10}	1.8×10^{6}		0.3	--	5.4×10^{3}	3		5.4×10^{4}
							After 2 yrs			After 2 yrs		
Am^{241}	458	1.20×10^{11}	105	--			0.2	0.21	--	1.2	1.1	--
							Total	3.1	1.3×10^{5}		11.4	3.7×10^{5}

Since one of the mechanisms for the shutdown of the reactor in the event of a nuclear accident is that due to fuel expansion and the propagation of the expansion along a fuel column, transient tests[4] were conducted on TREAT to determine the expansion characteristics of a fuel column. A fuel element containing depleted uranium but similar in design to the developed plutonium-uranium alloy fuel element was placed in a ZPR-3 drawer with a typical core loading. Results of the tests indicated that the fuel column expanded as expected and that fuel column expansion can be relied upon as a shutdown mechanism.

In addition to the 28 w/o Pu alloy fuel, the Laboratory will have about 500 kg of fuel which contains 36 w/o Pu having an isotopic content of 27% ^{240}Pu and 73% $^{239}Pu + ^{241}Pu$. This alloy has physical characteristics similar to the 28 w/o Pu alloy. Its air corrosion characteristics appear to be also similar to that of the 28 w/o Pu material.

In order to mockup carbide and oxide fuels and various structural and coolant materials the Laboratory maintains a large inventory of structural and coolant materials. The major materials which are available in the Laboratory's inventory are listed in Table V-4. Those materials such as sodium and Na_2CO_3 which react quickly with oxygen or water are canned in thin (0.015 in.) wall stainless steel cans. These materials are color coded to distinguish one from another easily.

Table V-4

MATERIALS INVENTORY

Material	Nominal Thickness, in.	Total Quantity, kg	Comments
	Structural and Coolant		
Na	1/4, 1/2	4,600	SS Clad
Na_2CO_3	1/4	3,000	SS Clad
U_3O_8	1/8, 1/4	40,000	
Fe_2O_3	1/8	3,263	
C	1/8	1,441	
Al_2O_3	1/4, 1/8	579	
BeO	1/8, 1/4, 1	2,028	
Mo	1/8	1,713	
Ni	1/4	5,034	
Nb	1/8	57	
SS	1/16, 1/8, 1/4	11,000	
V	1/16	13	
W	1/16, 1/8	1,915	
Zr	1/8	482	
Al			
(45% density)	1/8, 1/4	1,335	
(63% density)	1/8, 1/4	2,195	
(56% density)	1/8, 1/4	825	
(100% density)	1/16, 1/8, 1/4, 1, 2	7,785	
SS Void Cans	1/4, 1/8	70	
SS Void Simulation Frames	1/4	548	

REFERENCES

1. J. K. Long, "Safety Analysis of Plutonium Loadings in ZPR-III [Addendum to ANL-6408, Hazard Evaluation Report on the Fast Reactor Zero Power Experiment (ZPR-III)]," ANL-6504 (September 1963).

2. J. K. Long et al., "Safety Analysis of the Operation of ZPR-3 with Fuel Loadings Up to 430 kg of Plutonium (Addendum to ANL-6504, Safety Analysis of Plutonium Loadings in ZPR-3)," ANL-7049 (December 1965).

3. L. R. Kelman et al., "Metallic Plutonium Alloys for Fast Critical Experiments," Plutonium 1965, Proc. 3rd Intern. Conf. on Plutonium, London, Nov. 22-26, 1965, ed. A. E. Kay and M. B. Waldron, Chapman and Hall, London, 1967, pp. 510-524.

4. "Reactor Development Program Progress Report, February 1965," ANL-7017, pp. 25-27 (March 18, 1965).

Chapter VI

REACTOR CONTAINMENT

A. ZPR Building

The ZPR-6 and -9 are located in the west wing of the Reactor Physics Laboratory, Building 316-W, as shown in Fig. VI-1. The 316-W building consists of one service floor below ground level and 3 levels above. The service or basement floor, whose plan is shown in Fig. VI-2, contains 2 counting rooms, an experimental laboratory and a mechanical-equipment room. The first floor has 2 reactor cells, one other experimental cell, vault, vault workroom, reactor control rooms, offices, laboratories, and washrooms (see Fig. VI-3). The second floor, whose plan is shown in Fig. VI-4, has offices, laboratories, and a hot laboratory where radioactive materials may be handled. The third level houses air-exhaust fans and air filters as shown in Fig. VI-5.

B. Reactor Cells

The ZPR-6 and -9 facilities are currently located in blast resistant, highly reinforced concrete cells[1,2] as shown in the floor plan of Fig. VI-3. The interior dimensions of each cell are 12 m (40 ft) by 9 m (30 ft) and a height of 9 m (30 ft). There is 1.5 m (5 ft) of reinforced concrete between each cell and its control room. The other 3 sides of the cell, floor, and ceiling have 1.2 m (4 ft) of reinforced concrete as shielding. The center of the ZPR-6 facility is located approximately 5.5 m from the south wall, 3.4 m from the west wall and 2.1 m above the floor. The ZPR-9 facility is located the same distance from the south wall and above the floor but 3.4 m from the east wall. A summary of the physical characteristics of the cell is given in Table VI-1.

Both reactor cells were designed to withstand the blast effects of a bare charge of 45 kg (100 lb) of TNT exploded at the center of the reactor cell, as well as a long-time duration pressure of about 3.7 bars (47 psig). Strain gauge tests by S. H. Fistedis *et al.*[3] have been made on the reinforced concrete walls.

Using the measured strain gauge values obtained from the as constructed reinforced concrete walls, it is now estimated that continuous internal cell pressures should not exceed 3.4 bars (35 psig) which is slightly less than the design pressure of 3.7 bars (40 psig).

Analysis by Newmark and Hall[4] indicates that the cell as constructed will contain the blast effects equivalent to at least 86 kg of TNT at the center of the 2.5 m cube stainless steel matrix. Accepting minor cracking, it is estimated that the blast effects of at least 136 kg of TNT (1.5×10^8 cal energy release) could be contained.

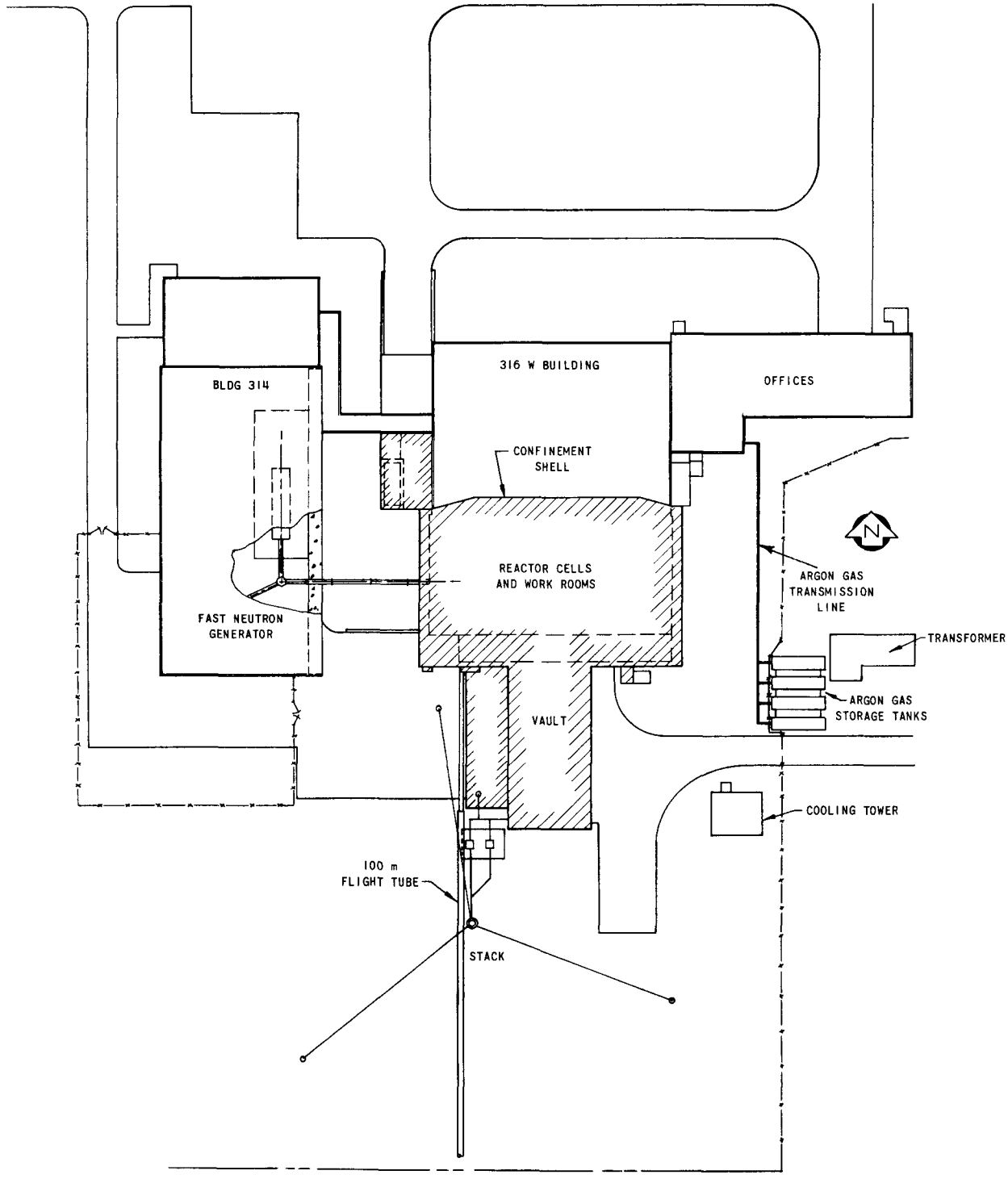


Fig. VI-1. Reactor Physics Laboratory, Bldg. 316-W

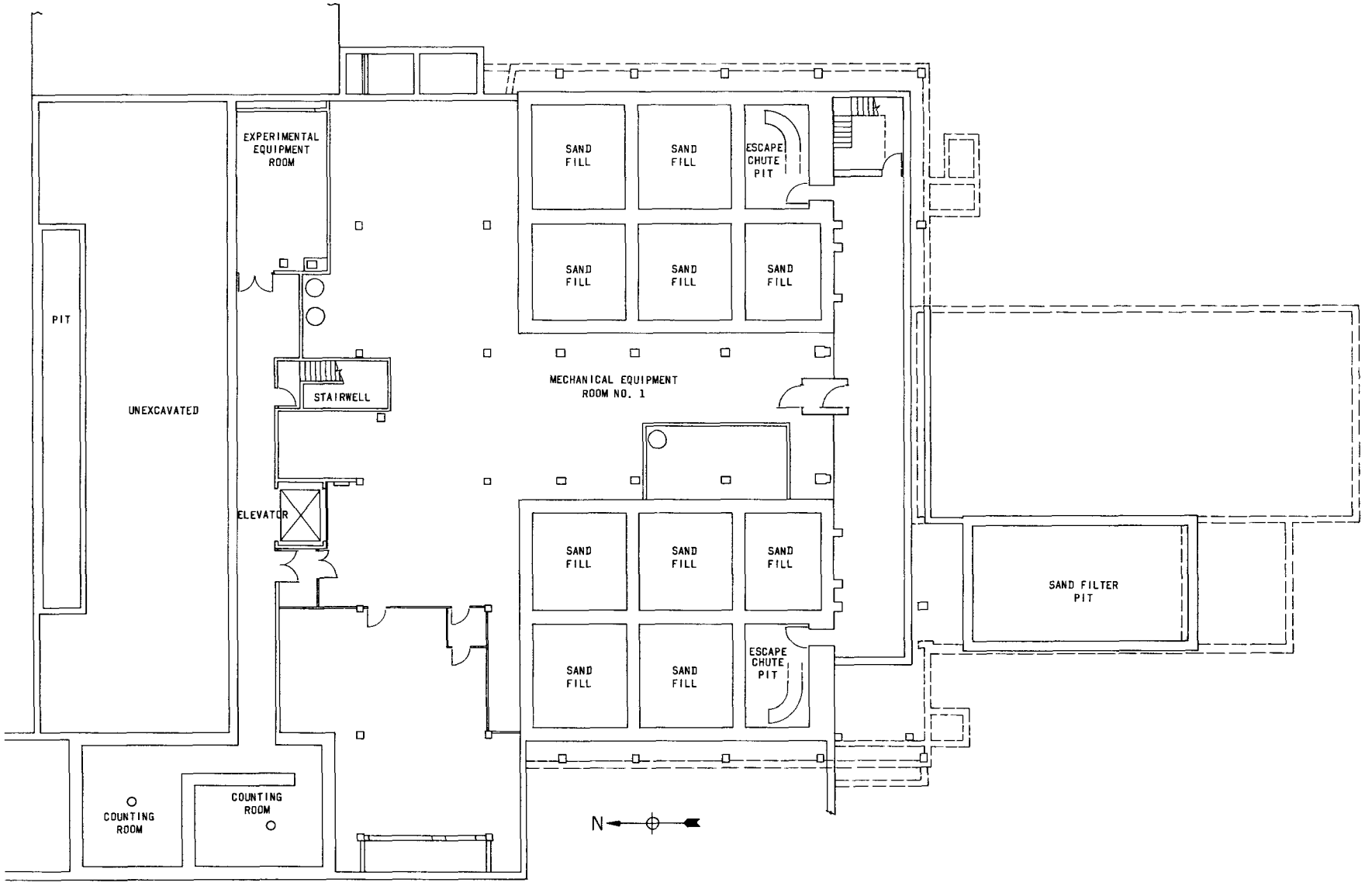


Fig. VI-2. Basement Floor Plan, Bldg. 316-W

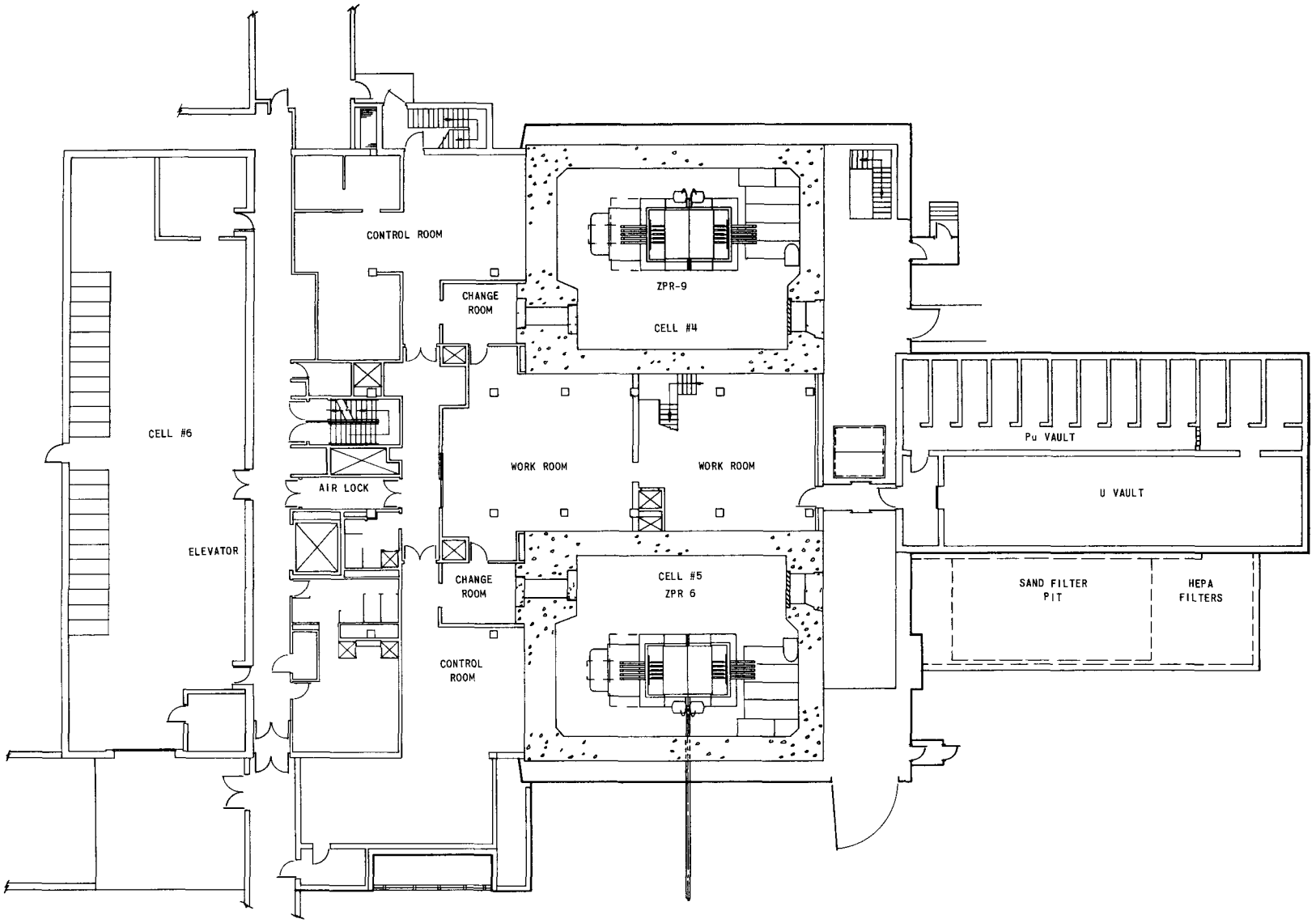


Fig. VI-3. First Floor Plan, Bldg. 316-W

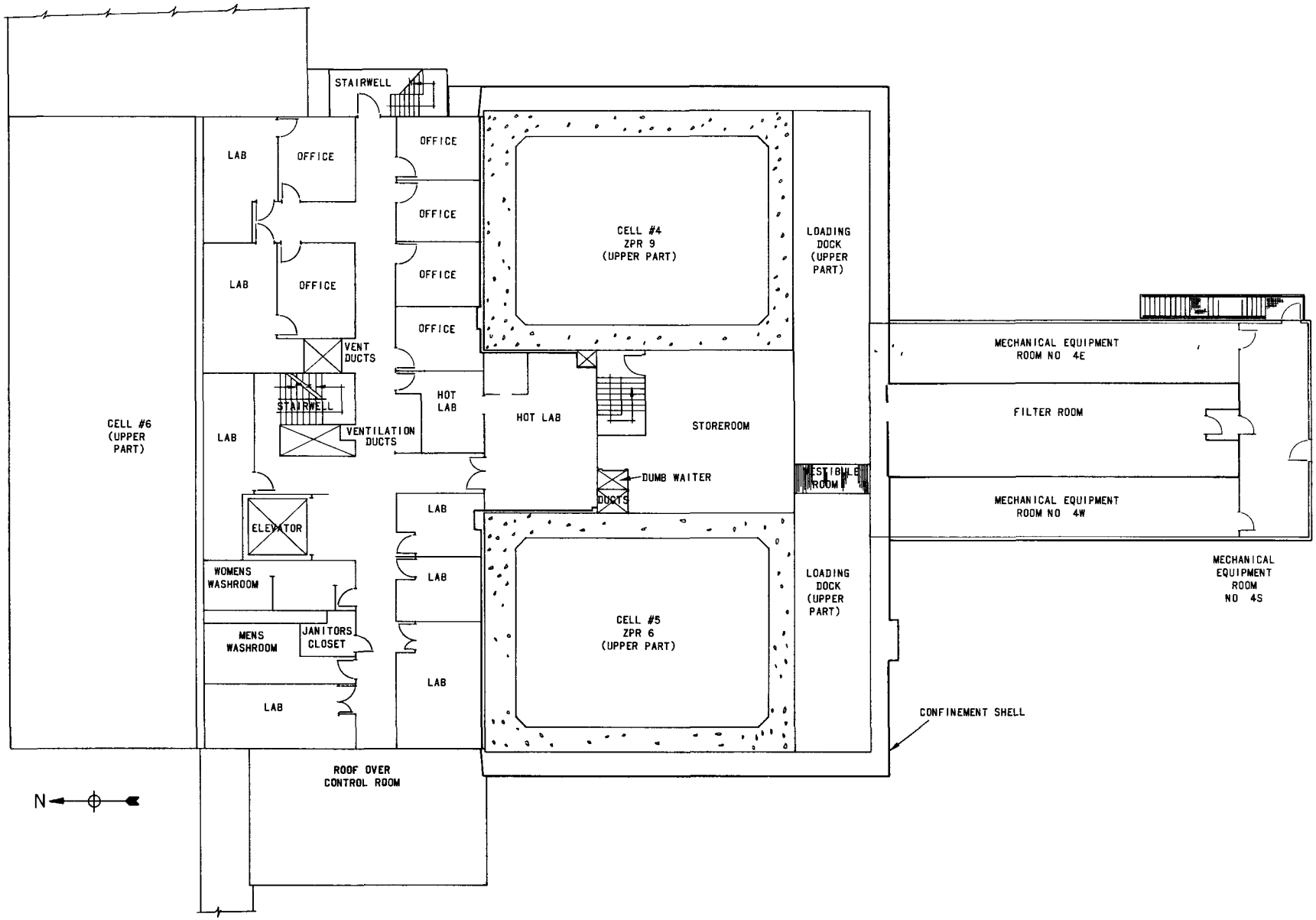


Fig. VI-4. Second Floor Plan, Bldg. 316-W

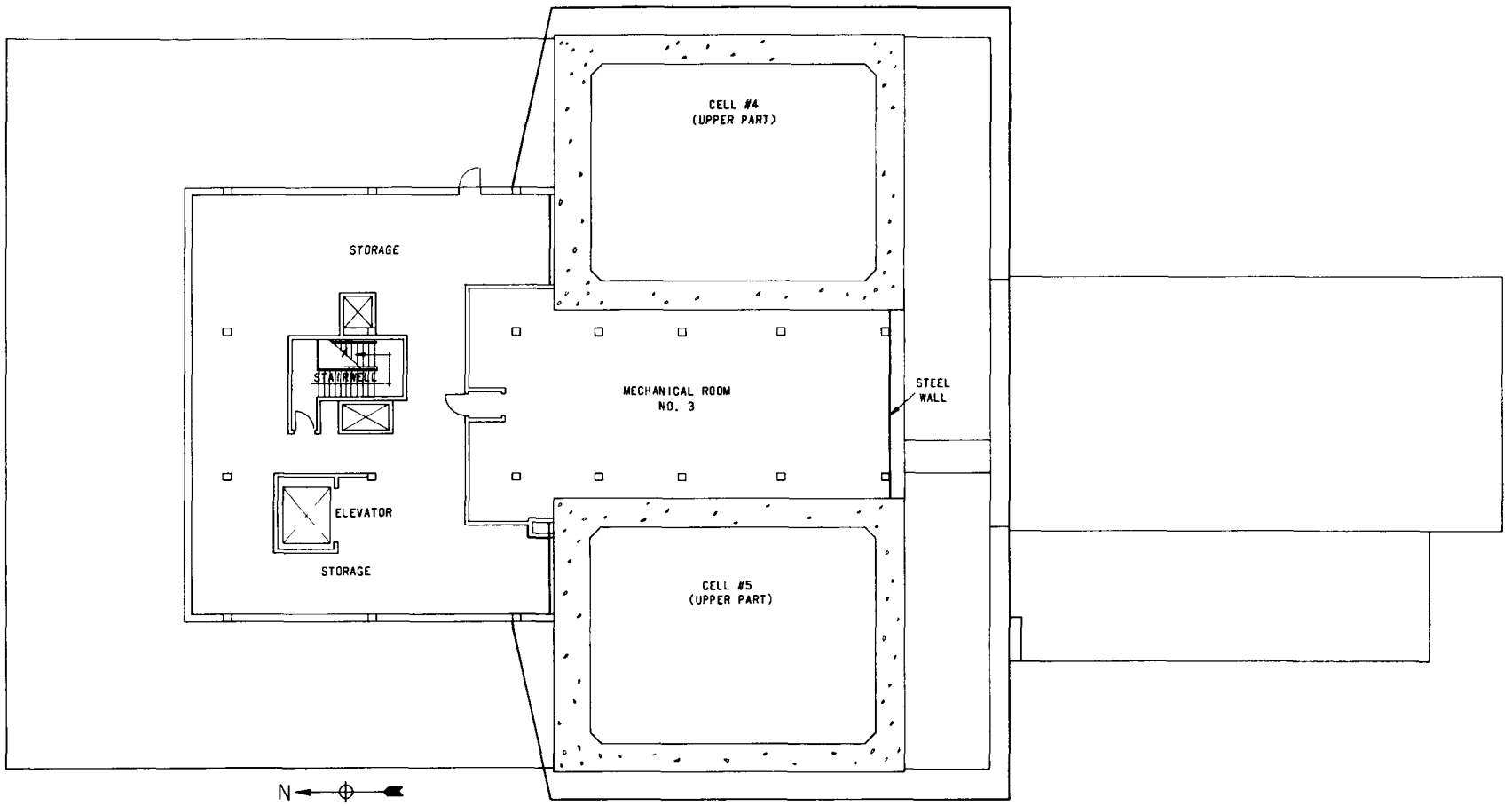


Fig. VI-5. Penthouse Floor Plan, Bldg. 316-W

Table VI-1

SUMMARY OF PHYSICAL CHARACTERISTICS OF CELLS

Interior Cell Size (Cells 4 and 5)	
Width	9 m (30 ft)
Length	12 m (40 ft)
Height	9 m (30 ft)
East, West, and South Wall Thickness	1.2 m (4 ft)
Ceiling Thickness	1.2 m (4 ft)
Floor Thickness	1.2 m (4 ft)
Wall Thickness between Control Rooms and Cells	1.5 m (5 ft)
Ventilating System (normal)	
Intake Pipe Size	61-cm diam (24-in. diam)
Intake Capacity	3500 cfm
Exhaust Pipe Size	61-cm diam (24-in. diam)
Exhaust Capacity	3500 cfm
Emergency Venting	
Exhaust Pipe Size	61 cm diam (24-in. diam)
Exhaust Pipe Size at Stack	137 cm diam (54-in. diam)
Exhaust Capacity at Stack	0-47,600 cfm
Floor-loading Limits	
Cells	3000 lb/sq ft
Vaults	500 lb/sq ft
Vault Workroom	500 lb/sq ft
Crane Capacities	
Cells	5 tons + 1 ton auxiliary
Vault	1 ton
Doorways	
Personnel Entrance Way Opening	2.1 m (7 ft) high x 0.91 m (3 ft) wide

In order to minimize the leakage of air from the reactor cell, the penetrations to the cell have been limited. In addition to those required by the service air lines, telephone, and electrical power conduits, the following penetrations of the cell exist:

1. 1 m x 2 m normal personnel entrance and a 0.8 m x 1.0 m personnel escape hatch;
2. 1.5 m x 3 m freight access door;
3. eighteen 3 in. pipe conduits for control wiring;
4. two 24 in. pipe penetrations for inlet and outlet of the air-conditioning system;
5. two 3 in. pipe for future use; two 2 in. pipe for future use; two 1 in. pipe for future use; three 6 in. pipe for future use;

6. three 3 in. pipe penetrations to be used as drawer ports for passing reactor drawers from vault workroom to the cell;

7. two 5 in. diameter penetrations for the Time-of-Flight tube and Proton Beam Transport Tube. (Only for ZPR-6 cell.)

All of the conduits containing wiring have an epoxy resin seal in a small section of the conduit to minimize leakage of air around the wires.

During the initial acceptance testing of the reactor cells it was found that the reinforced concrete construction of the cells was relatively porous. When the cells were pressurized to 10 psig, it was found that air was leaking out of a large area of the cell walls due to the porous nature of concrete. The leakage rate was found to be about 2.8 liters/sec (6 cfm). In order to decrease the leakage rate the interior walls of the cell were coated with a phenolic resin base paint (Carboline 305--manufactured by the Carboline Company, Saint Louis, Missouri). Following the painting the initial leakage rate from each of the cells was found to be about 1.5 cfm or 6% of the volume per 24 hr at 10 psig. Currently the leakage rate from the cells has been measured to be between 8 to 12% of the cell volume per 24 hr or 0.95--1.4 liters/sec (2-3 cfm) at 1.7 bars (10 psig) internal pressure. It is possible that in time the leakage rate may reach a figure of 2.8 liters/sec (6 cfm) with 1.7 bars (10 psig) internal pressure due to the deterioration of the paint. Leak checking will be done on an annual basis. If the leakage rate exceeds 6 cfm at 1.7 bars (10 psig) reactor operation will be terminated and action will be taken to reduce the leakage below the 6 cfm.

C. Normal Cell Air Supply and Exhaust System

The reactor cell has 2 separate systems for conditioning the air in the cell. When the reactor is not in operation and personnel have access to the cell, external air, after passing through filters and a conditioning system, is brought into the cell through a 60 cm (24 in.) diameter pipe located in the floor along the east wall of the cell as shown in the flow diagram of Fig. VI-6. The air is distributed in the cell by sheetmetal ductwork. The air flow through the pipe is controlled by two 24 in. pneumatically operated butterfly valves. The valve nearest the cell is a steel-seated butterfly valve constructed to withstand elevated temperatures and pressures. A rubber-seated butterfly valve follows the steel-seated one and acts as a positive seal for the system. Both inlet valves are located external to the cell in the maintenance room located in the basement of the building.

The steel-seated inlet butterfly valve has been designed to withstand momentary pressure impulses of 17.4 bars (255 psig) for a 1.7 ms duration and sustained pressures of 2.7 bars (40 psig). The steel-seated valve is also capable of withstanding a continuous elevated temperature of about 200°C (385°F) without deterioration of the valve seat or body.

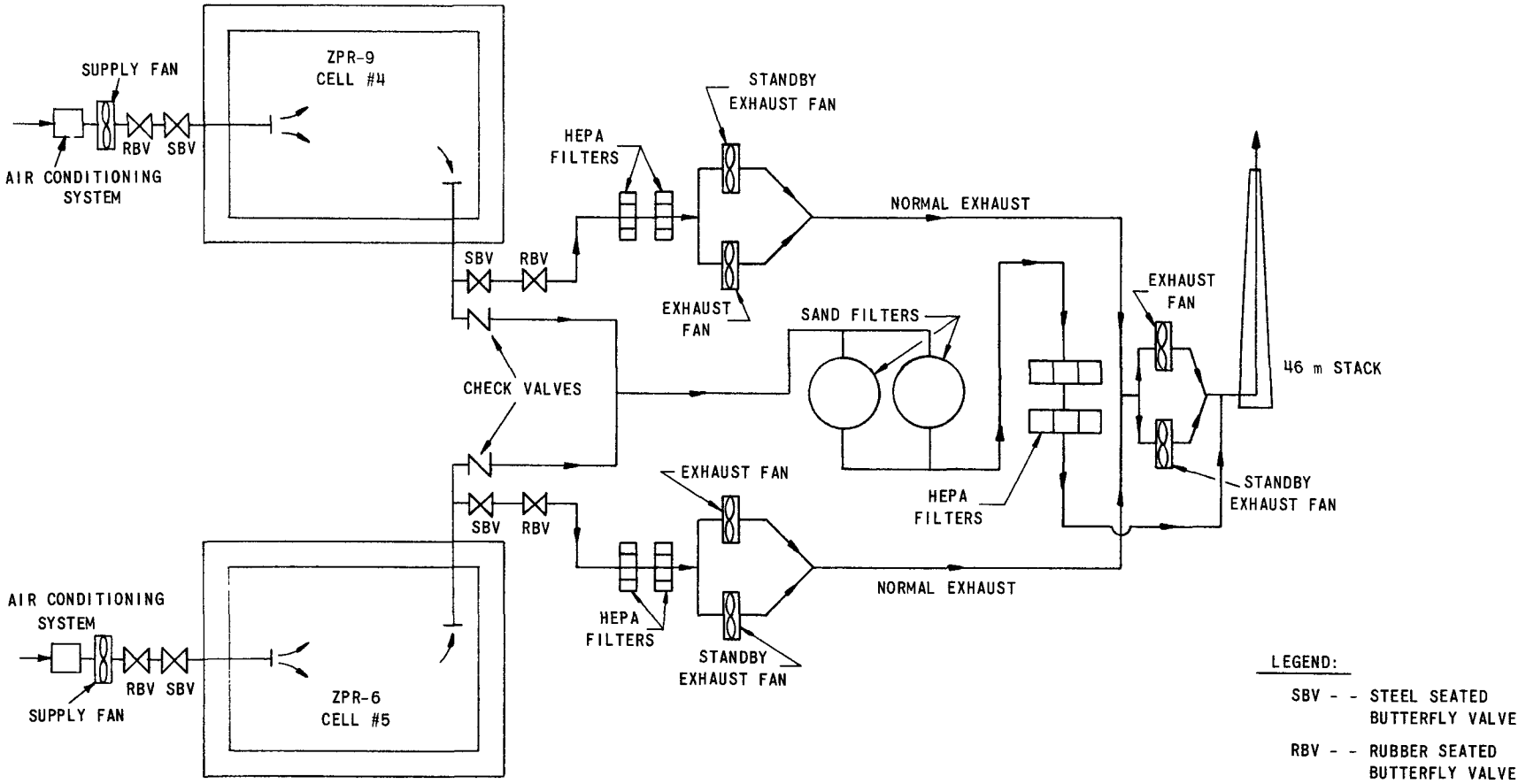


Fig. VI-6. Cell Air Supply and Exhaust Flow Diagram

The rubber-seated valves have been designed to withstand continuous temperatures of 79°C (175°F) without deterioration and long-time duration pressures of 2.7 bars (40 psig). The designed leak rate for this valve is less than about 140 liters (5 cu ft) per 24 hr when there is a differential pressure of 2.6 bars (28 psig) across the valve disk.

The cell air is exhausted through a 60-cm (24 in.) diameter pipe located near the ceiling in one corner of the cell. The air passes through 2 butterfly valves, one steel-seated and the second rubber-seated, and then through two banks of HEPA filters, fan, and up a 46 m stack. The butterfly valves function in a similar manner to those in the inlet duct. The exhaust butterfly valves are located in the fan loft in the third story of the building. The blast-resistance characteristic of the steel-seated exhaust butterfly valve is limited to momentary pressure impulses of 17 bars (255 psi for a 1.7 ms duration). This limitation is due primarily to the valve disk. The valve is capable of withstanding temperatures of 200°C (385°F) without deterioration of the valve seat or body. The rubber-seated exhaust butterfly valve has the same temperature and pressure characteristics as the inlet rubber-seated valve.

The air from the cell after flowing past the 2 butterfly valves enters a bank of high-efficiency AEC filters which will operate up to a temperature of about 120°C (250°F). These filters will pass approximately 1700 liters/sec (3500 cfm) with maximum gas temperatures of about 120°C (250°F).

The 24 in. butterfly valves in the inlet and exhaust air-supply system are controlled from the control console. Electrically energized pneumatic valves control the position of the butterfuly valves by allowing air pressure to activate the piston driving the valve. An air reservoir is provided near the piston to move it in such a direction as to close the valve in the event of failure of the air supply. These valves require external air pressure to open the valve. Failure of the air supply with the valve closed means that the valve would remain in the closed position unless manually opened.

The inlet and outlet butterfuly valves must be closed in order to satisfy one of the reactor-startup interlock conditions. This is accomplished by actuation of valve-control switches at the console. With the external air-supply and exhaust system shutdown, an internal air-conditioning system consisting of heaters and cooling coils maintains the preset temperature. Cooling is accomplished by circulating a refrigerant (mono-chlorodifluoromethane, [Freon]) through coils located in the cell. The refrigerant is piped into the cell from compressors located in the basement. The refrigeration capacity of the cooling system is 15 tons (3000 Btu/min--7.5 x 10^5 cal/min heat-removal capacity). Heating is accomplished with the use of electric heaters.

D. Modifications

In order to improve the containment characteristics of the reactor cells the following modifications to the reactor cells have been made:

1. construction of a steel confinement shell system around the reactor cells;
2. installation of a sand filter emergency venting system;
3. increased storage capacity of argon gas to 3.4×10^6 liters (120,000 ft^3);
4. installation of a steam driven turbine generator as a standby emergency power source; and
5. construction of a new plutonium and uranium storage vault.

Although a nuclear accident which would release substantial amounts of plutonium or fission products within a cell is considered incredible, a backup confinement system has been constructed as insurance against the release of radioactive particulate leaking to the atmosphere from the cell. The confinement system consists of a steel shell around the exterior of the southern portion of Building 316-W, and the creation of three different static pressure zones inside the building in areas adjacent to the reactor cells. The areas adjacent to the reactor cells are maintained at a negative pressure by special fans as shown in Fig. VI-7 the flow diagram for building supply and exhaust systems in order to prevent leakage of any particulate matter to the atmosphere. The air from these negative pressure areas is exhausted through double banks of high efficiency absolute (HEPA) filters and up a 46 m (150 ft) stack.

An emergency venting system through a set of sand filters and HEPA filters is provided as an insurance against any long term buildup of pressure in the cells. The amount of argon gas stored outside of the cells has been doubled to 3.4×10^6 liters (120,000 ft^3) in order to be able to bring the oxygen content in the cell down to about 1% in the event there is a metal fire in a cell.

A steam driven turbine generator of 250 kW capacity has been installed as a source of emergency power in the event that the normal power source is lost. The emergency generator has been installed in order to insure that adequate power is available for reactor instrumentation and controls and the operation of the exhaust fans necessary to maintain the required negative pressures in the various parts of the confinement system.

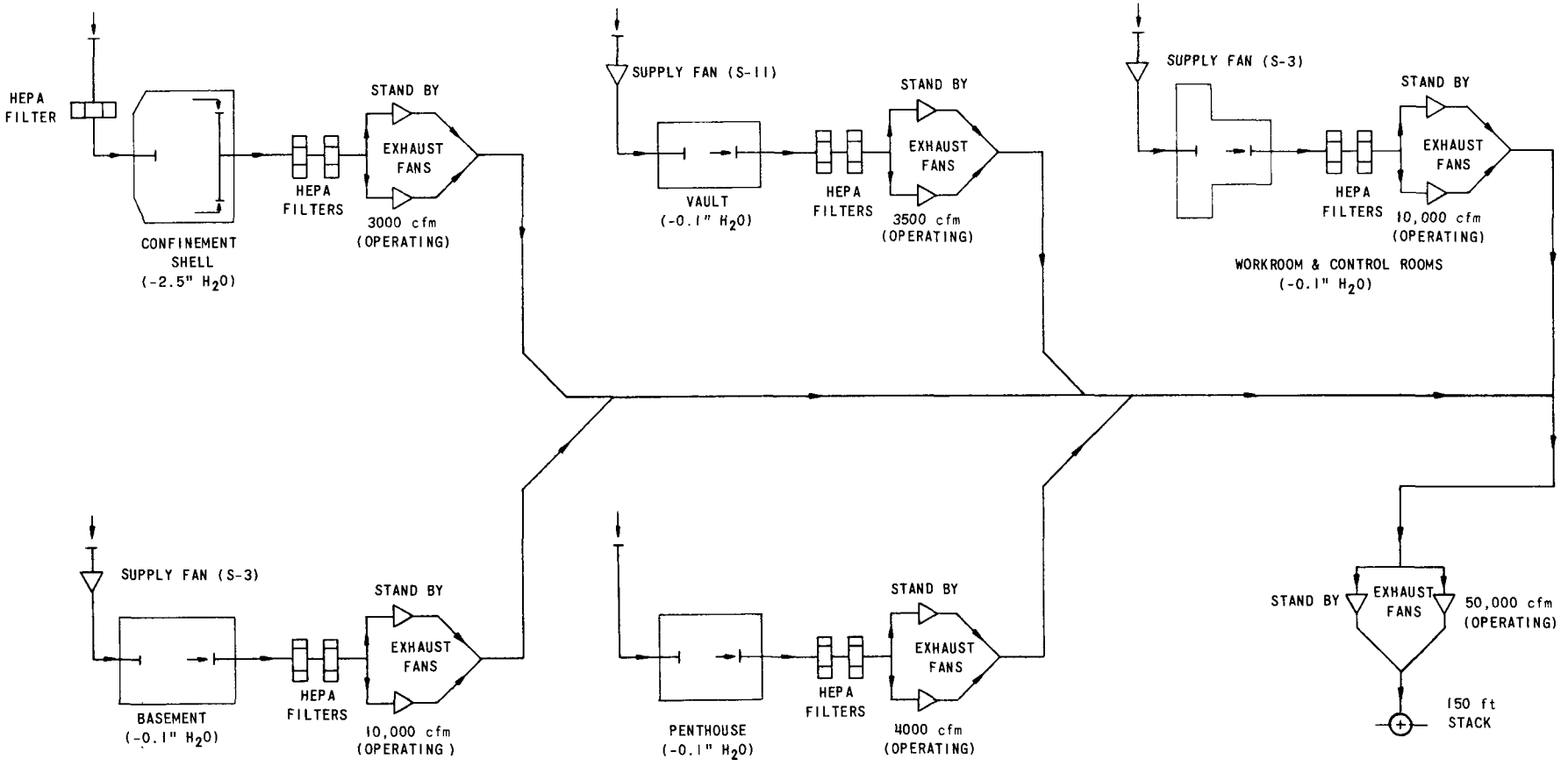


Fig. VI-7. Flow Diagram of Supply and Exhaust Systems for Bldg. 316-W

1. Confinement System

A confinement shell as shown in Fig. VI-8 which encloses the south portion of Building 316-W, both reactor cells, loading dock, existing vault, new vault and the sand filters and HEPA filters for the emergency venting system has been constructed.

The steel shell is constructed of plate steel 0.47 cm (3/16 in.) thick welded on a structural steel frame which is supported on existing building columns and on new footings. The shell is entirely free of the cells and the steel plate extends below grade.

The steel shell which encloses the reactor cells has been designed to withstand the following natural phenomena:

Snow Load	30 lb/ft^2 on roof
Wind Load	30 lb/ft^2 on vertical surfaces

(Corresponding to effective wind velocities of about 177 km/hr [110 mph])

Exhaust Differential Load	5 lb/ft^2 on all surfaces
Total Load	35 lb/ft^2 on all surfaces

The confinement shell is also designed to maintain structural integrity under normal rapidly varying atmospheric pressures. The shell is sealed to the existing building surfaces except the portions below grade. All of the personnel doors which provide access to the shell volume are air lock type doors so that personnel may enter the confinement shell without loss of air pressure. There will be a freight entrance on the west side of the building which will be normally closed and alarmed.

The steel shell provides a ventilated air space of about 1.2 meters (40 in.) between the shell and the existing cells and vault as shown in the vertical plan views of Figs. VI-9 and VI-10. This space is maintained at a pressure of about 0.994 bars (-2.5 in. of water) with respect to the atmosphere during reactor operation by exhausting the air inside the shell through two sets of HEPA filters and up a 46 m (150 ft) stack. By maintaining a 0.994 bars (-2.5 of water) pressure inside the shell it will be possible to have at least 0.998 bars (-1.0 in. of water) between the atmosphere and the shell volume as long as the wind velocity on the outside surface does not exceed 62 mph. An air space has been enclosed between the top surface of the shell and a new roof. Louvers are installed in all four walls of this air space in order to reduce the effect of any negative pressures due to wind velocity on the shell roof.

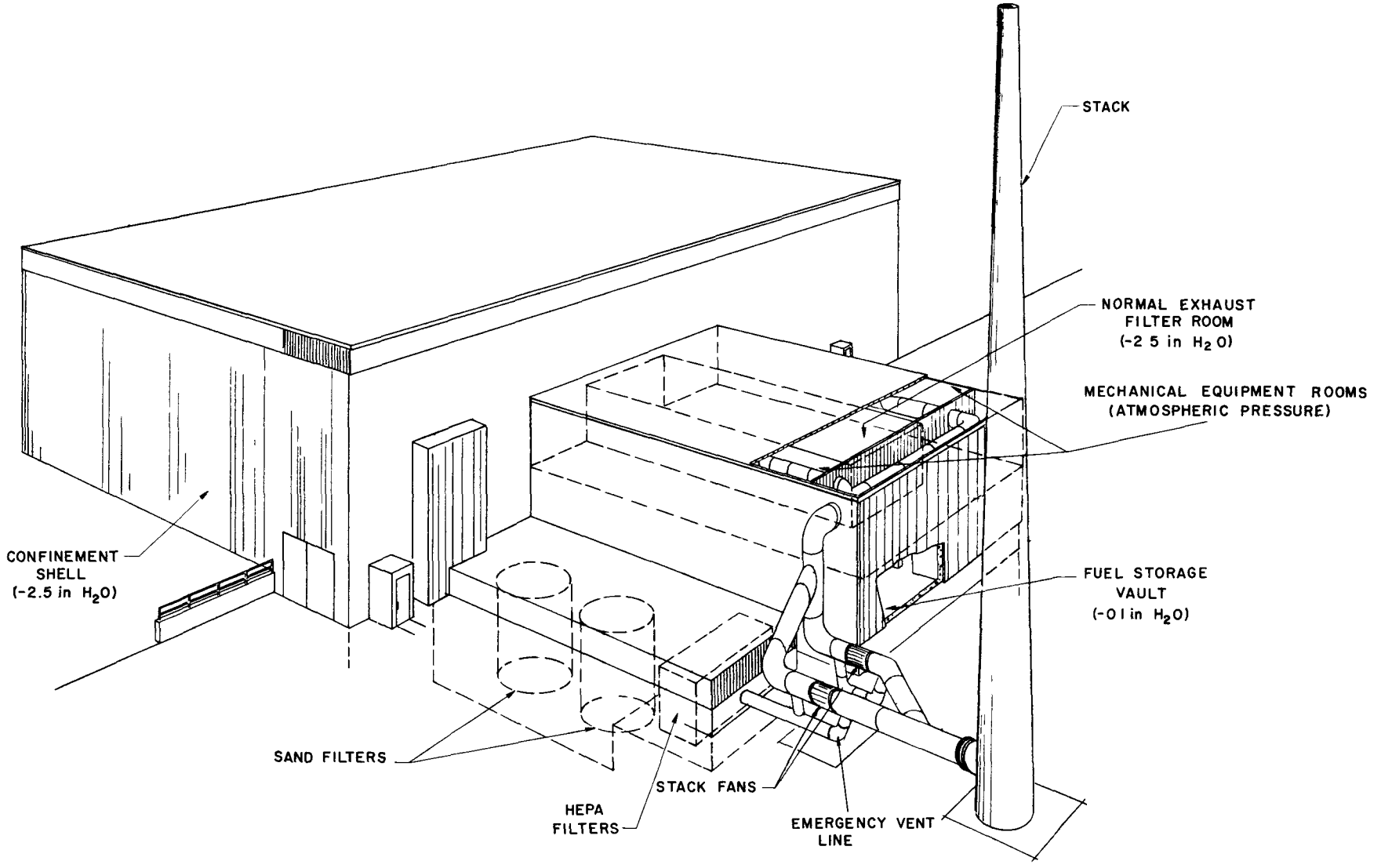


Fig. VI-8. Confinement Shell

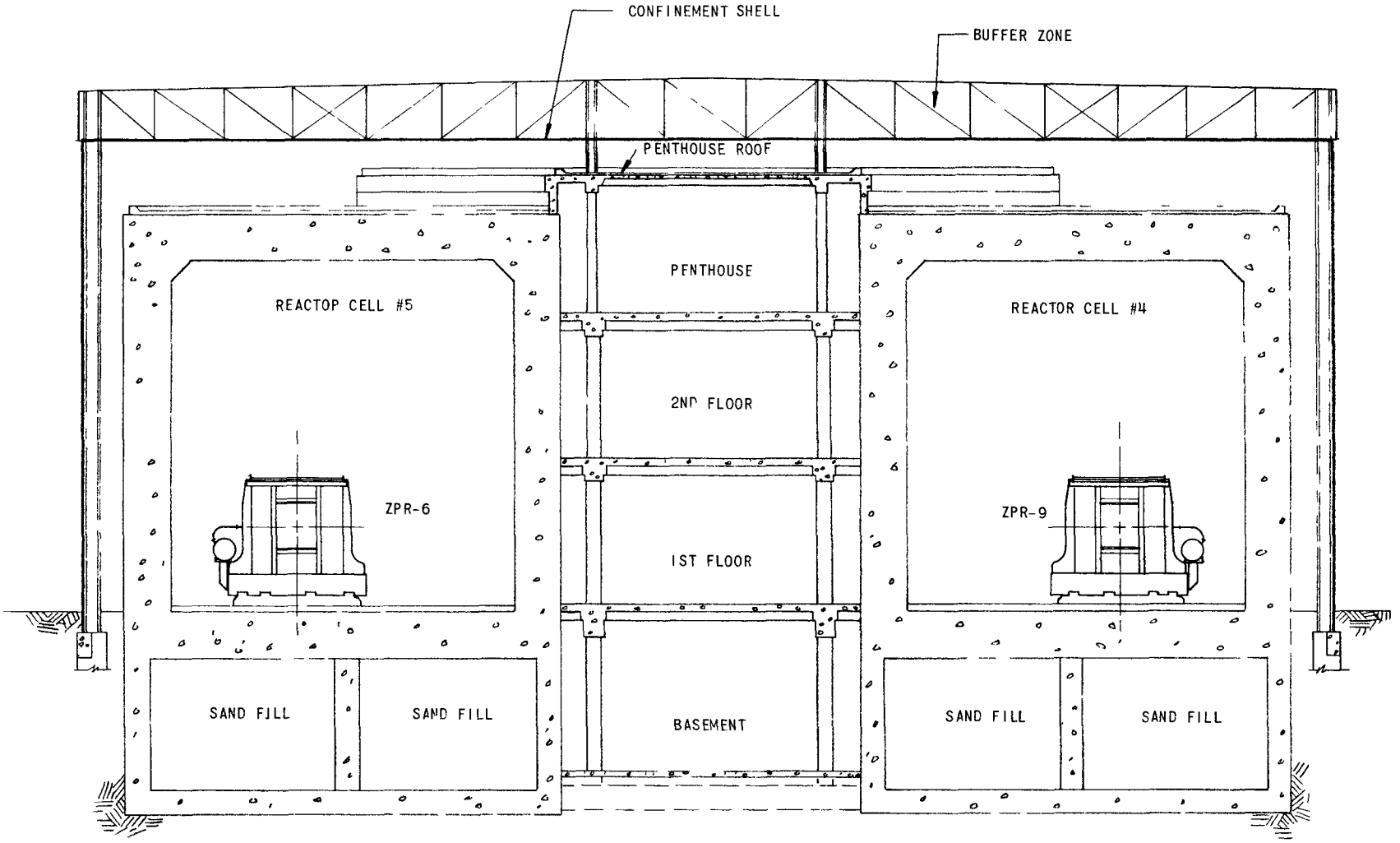


Fig. VI-9. Confinement Shell Section, East-West Section

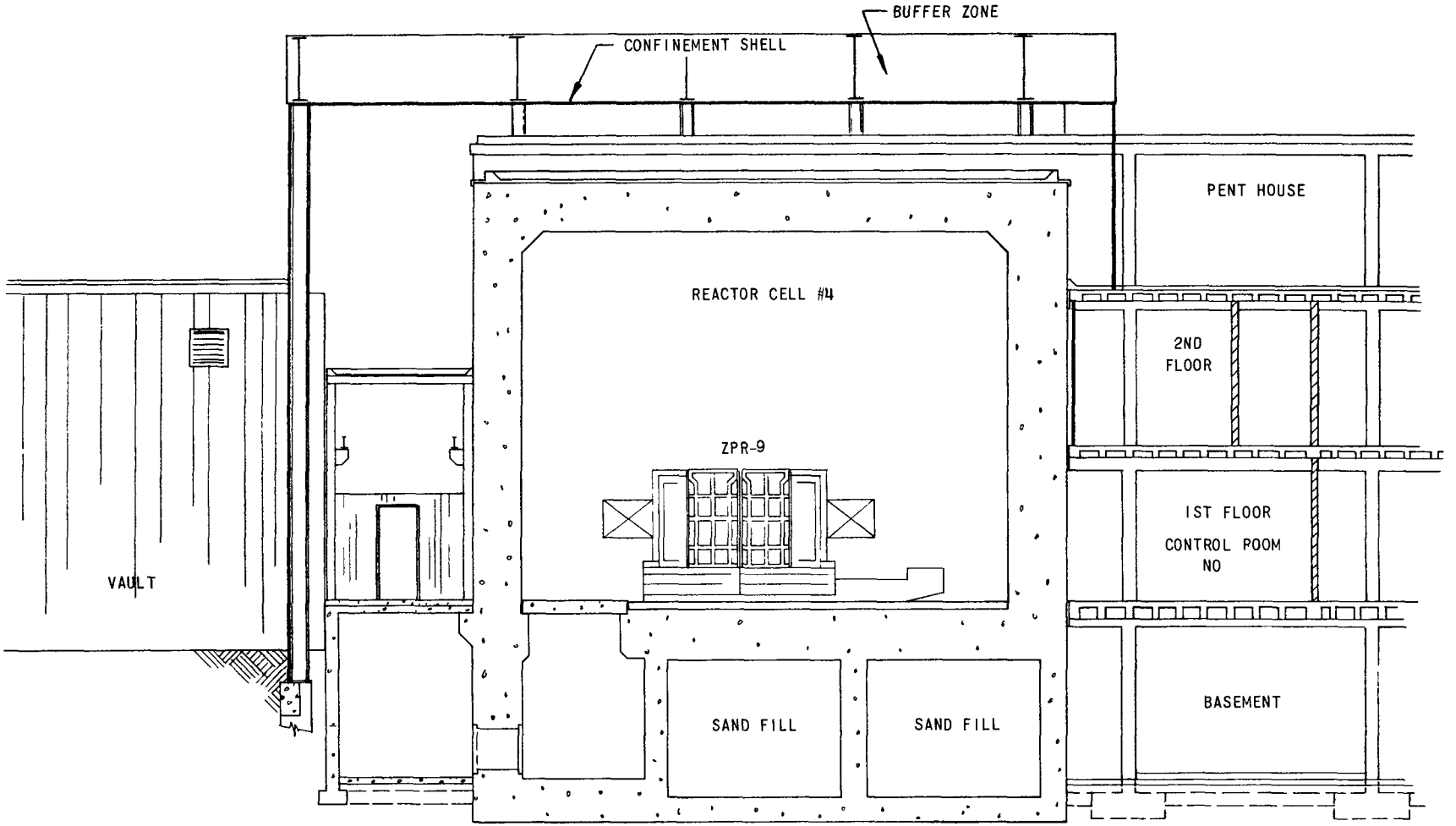


Fig. VI-10. Confinement Shell Section, North-South Section

A modulating control damper located in the shell inlet system is used to vary the exhaust flow in order to maintain the 0.994 bars (-2.5 in. of water) pressure in the shell volume due to atmospheric pressure changes or changes in the leakage rate into the shell volume. The confinement shell exhaust system utilizes a fan having a 1400 liters/sec (3000 cfm) capacity to maintain the negative pressure as shown in the flow diagram Fig. VI-7. In addition to the normal duty fan, a standby fan having the same capacity has been installed. The standby fan comes into operation when the pressure within the shell starts to increase above a preset value due to either abnormal conditions of the normal fan, sudden changes in the leakage rate into the shell volume, or rapid changes in the atmospheric pressure. With both fans in operation there is an exhaust capability of 2800 liters/sec (6000 cfm). It is estimated that the leakage into the shell volume from leaks in the sealant between the shell and masonry construction will not exceed about 240 liters/sec (500 cfm). With only one fan in operation there is thus about a factor of 6 additional fan capacity to maintain the correct negative pressure in the shell volume.

Since the shell does not fully enclose the reactor cell, a steel wall is provided in the penthouse, second floor office and laboratory area which isolates the reactor cells from the rest of Building 316-W area. The volume enclosed by the steel walls and reactor walls is directly connected to the volume of the confinement shell and kept at a 0.994 bars (-2.5 in. of water) pressure. The zones adjacent to the reactor cells on the first floor which contain the control room, workroom, vault, fan room in the penthouse, and the service floor will be kept at 0.9997 bars (-0.1 in. of water) negative pressure with respect to the atmosphere. These areas are considered suspect areas and the air from these zones will be exhausted through two banks of HEPA filters and will be discharged up the 46 m stack. The different static pressure zones in Building 316-W are shown in Figs. VI-11-14. These areas are protected in all cases from exposure to outside walls subject to negative wind pressure by an outer zone or buffer zone. For the most part this is formed by the remaining building areas, but in a few cases a new buffer zone has been constructed with sufficient openings from it to the existing outer zone to assure the same pressure. The pressures in the suspect areas adjacent to the cell is maintained at a static pressure of at least 0.9997 bars (-0.1 in. water) lower than the pressure in the buffer zone. The fluctation in outer zone pressures relative to atmospheric will be reduced by adding a static pressure controlled damper in the major exhaust system serving this zone. This will facilitate the pressure control of the suspect areas.

For convenience of construction, the area south of the cells at the basement level as shown in Fig. VI-11 is maintained at the same negative pressure as the exterior shell zone. A steel partition between the cells isolates this area from the rest of the basement. The south vault walls

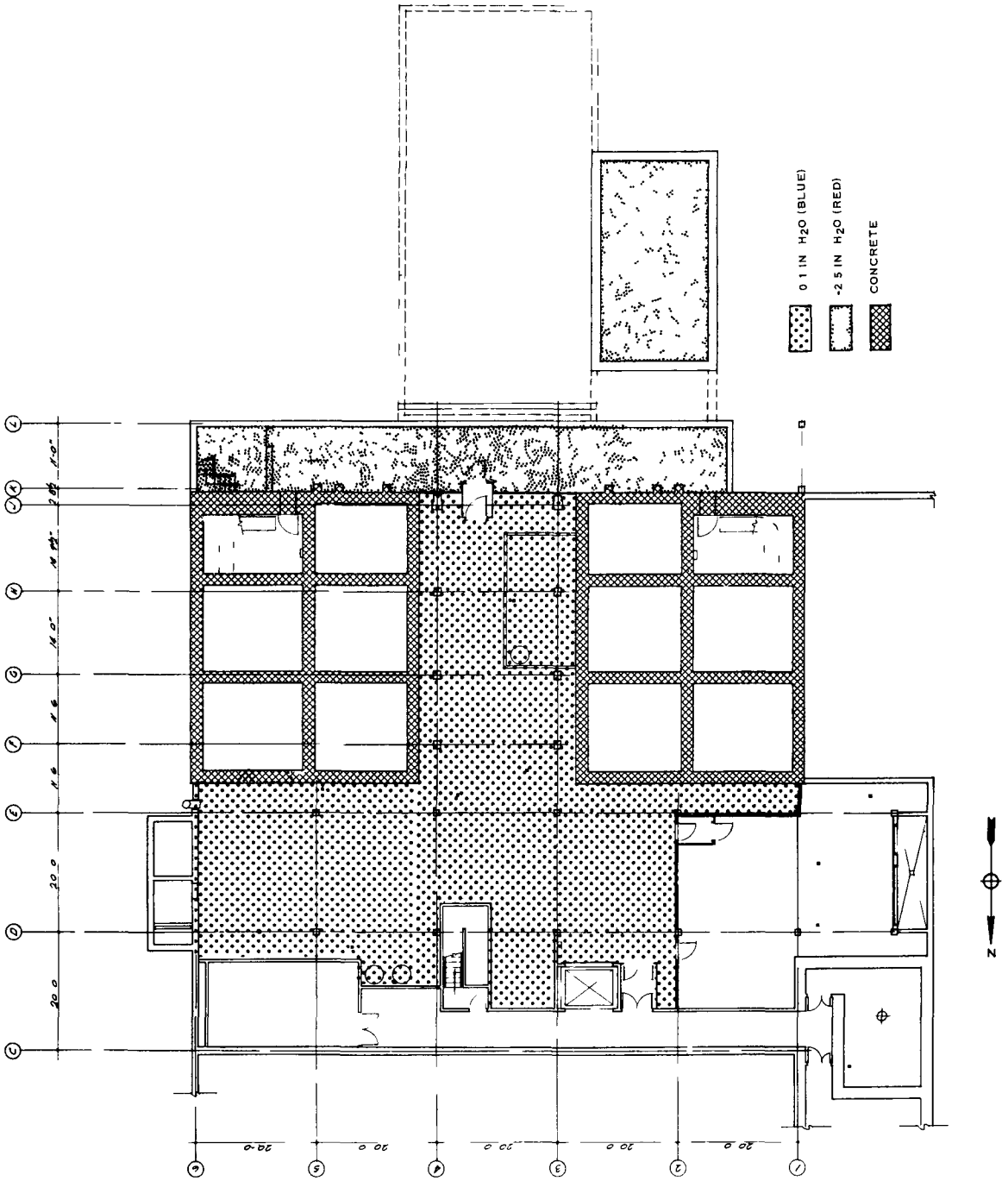


Fig. VI-11. Static Pressure Zones (Basement floor plan)

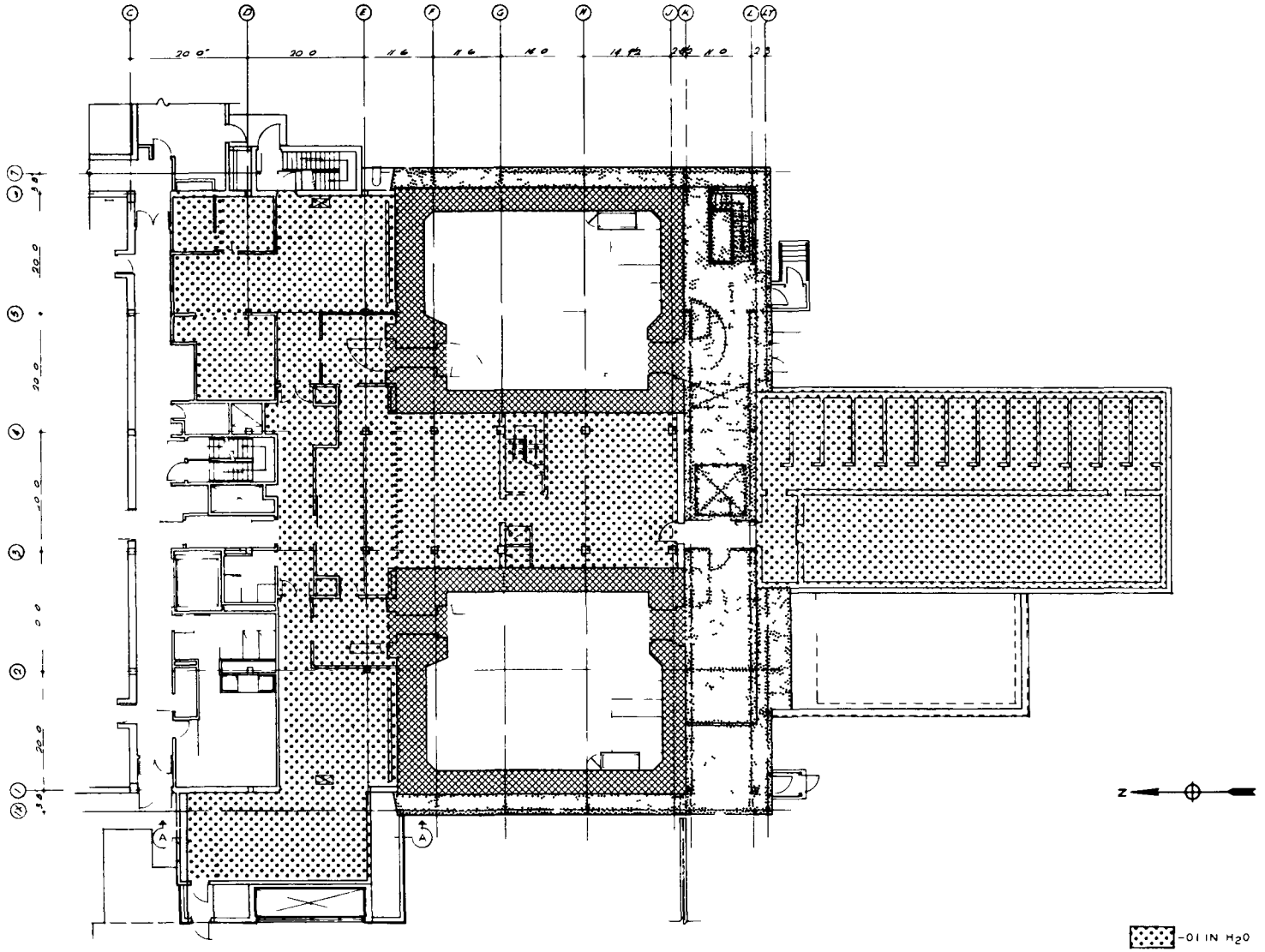


Fig. VI-12

Static Pressure Zones (First floor plan)

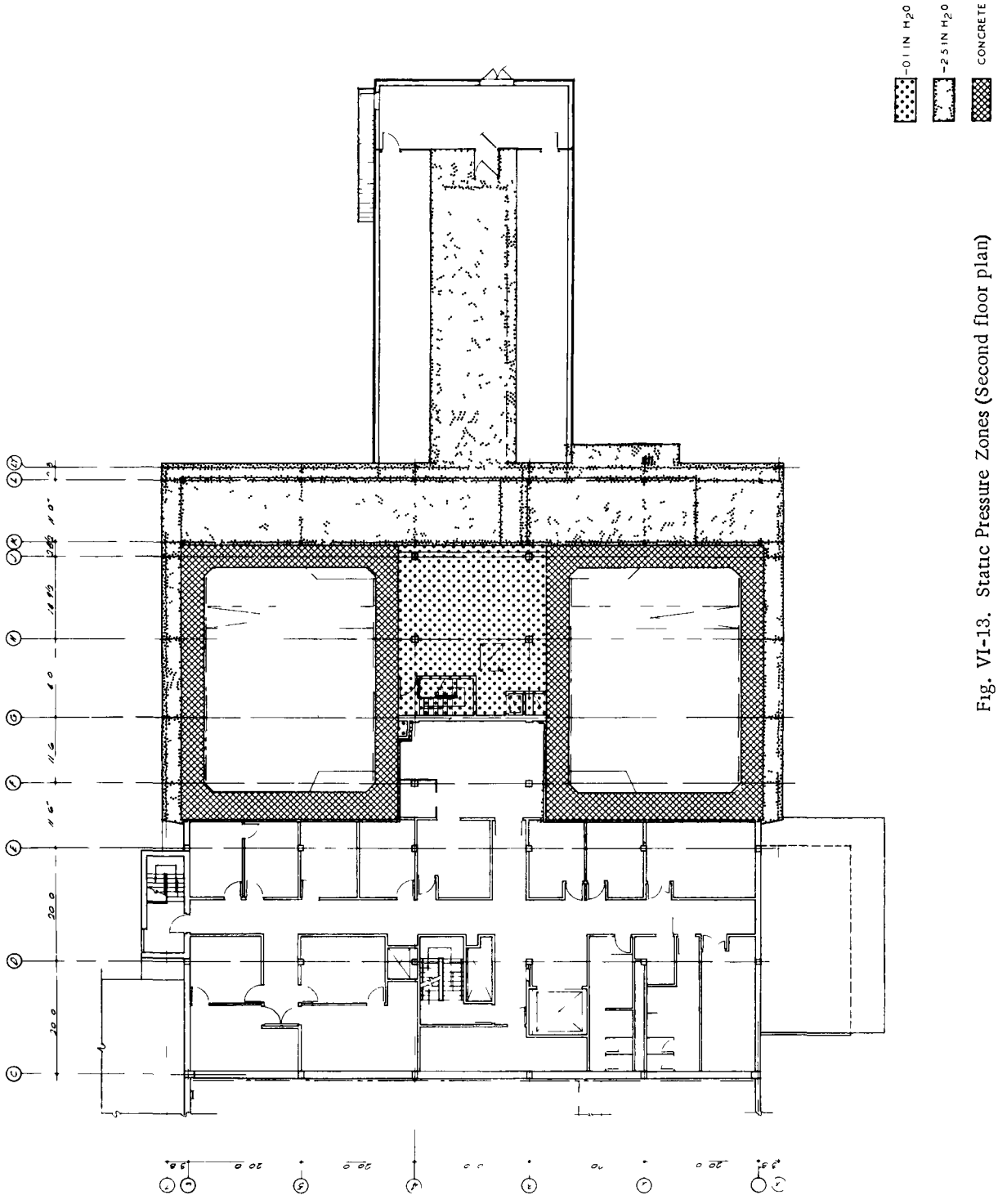


Fig. VI-13. Static Pressure Zones (Second floor plan)

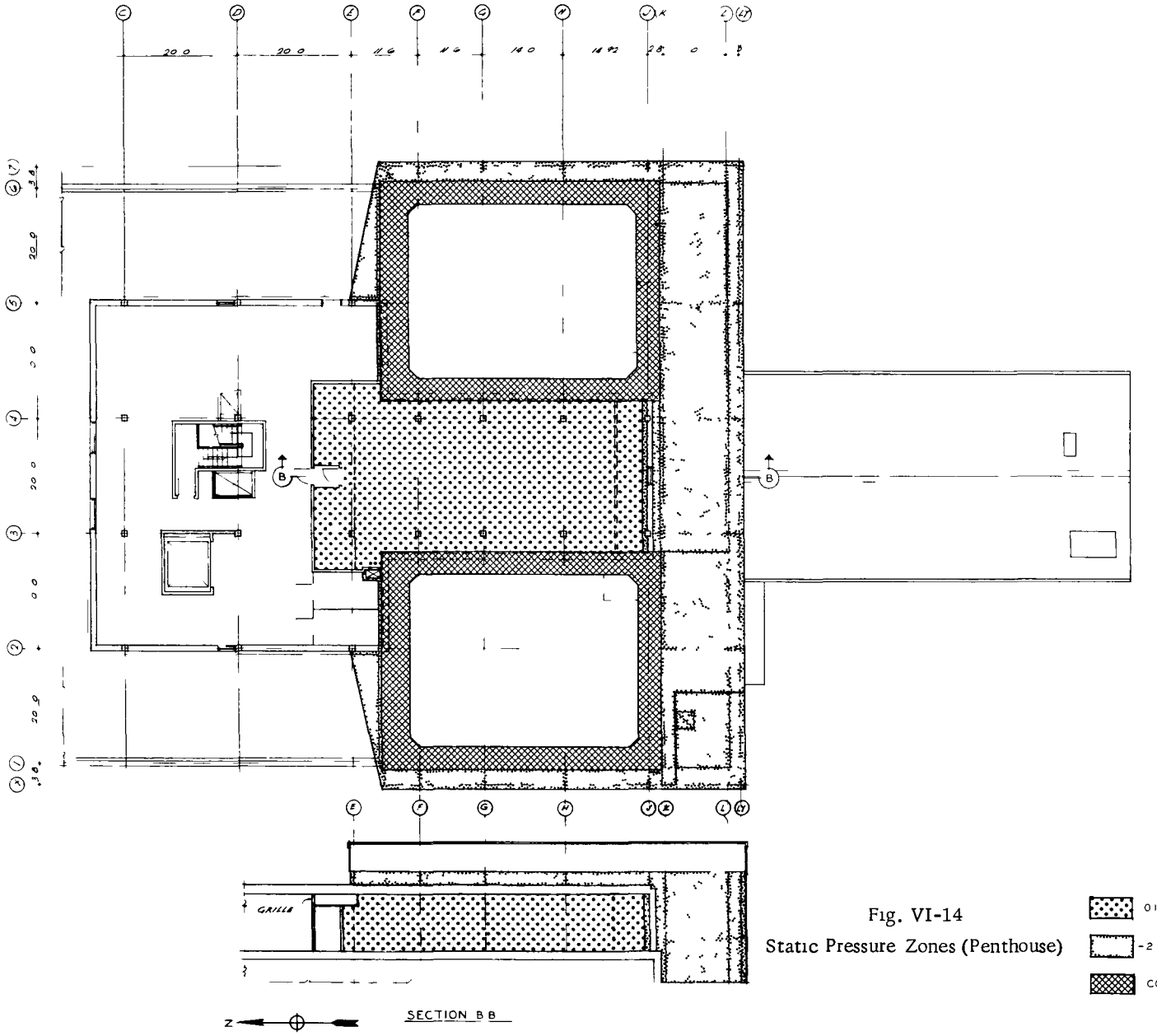


Fig. VI-14
Static Pressure Zones (Penthouse)

forms part of the interior wall of the low pressure shell zone on the first and second floors. A new steel partition between the cells at the south end of the penthouse as shown in Fig. VI-14 forms part of the interior wall of the low pressure shell zone at the penthouse level.

In the basement, Mechanical Room No. 1 is designated as a suspect area since it is adjacent to the cells. A part of this room surrounding the building air conditioning unit is partitioned off and included in the buffer zone. This is necessary to prevent in-leakage of suspect air into the unit which supplies air to both suspect zone areas and buffer zone areas. The unit which ventilates the basement is also included within this area which is partitioned off. Sufficient air from this unit is supplied within the area to maintain its integrity as part of the buffer zone.

All of the HEPA filters and sand filters are located in either a suspect area or within the extension of the confinement shell over the new vault. The location within the shell of the HEPA filters for the cell emergency venting and purging system, provides additional protection in case of the extremely remote possibility of overpressure within this system causing an out-leakage of unfiltered and contaminated air. Double banks of HEPA filters (two in series) are used in all suspect exhaust systems in order to increase the attenuation factor for removal of particulate from the exhaust. The second filter also serves as a backup filter in the event that the primary filter is damaged in the event of some unforeseen event.

In order to insure that the proper negative pressures are maintained during operation of the facility, secondary standby fans are incorporated in the exhaust systems of the reactor cells, confinement shell volume, work room, vault, basement mechanical room (Mech. Rm. No. 1), Penthouse mechanical room (Mech. Rm. No. 3), vault mechanical room (Mech. Rm. No. 4), and the central suspect exhaust line to the stack as shown in Fig. VI-7. In addition to the redundancy of fans, direct drive fans are used in order to increase reliability of the exhaust systems. As a further precaution, motor control centers for the normal fans are separated from the motor control center for the standby fans. Emergency power from the emergency steam driven turbine generator is available to the fans in these exhaust systems in the event of a power failure.

2. Emergency Exhaust System

An emergency high temperature exhaust system which is a special part of the ventilation system has been constructed and installed for use by both reactor cells. The emergency exhaust system has been designed to rapidly relieve pressures built up in the cells due to a rapid metal fire. A schematic diagram of the system is shown in Fig. VI-6. Figures VI-15 and VI-16 show the plan and elevation view of the system. As the diagram indicates, it consists of a 24 in. diameter exhaust pipe which bypasses the

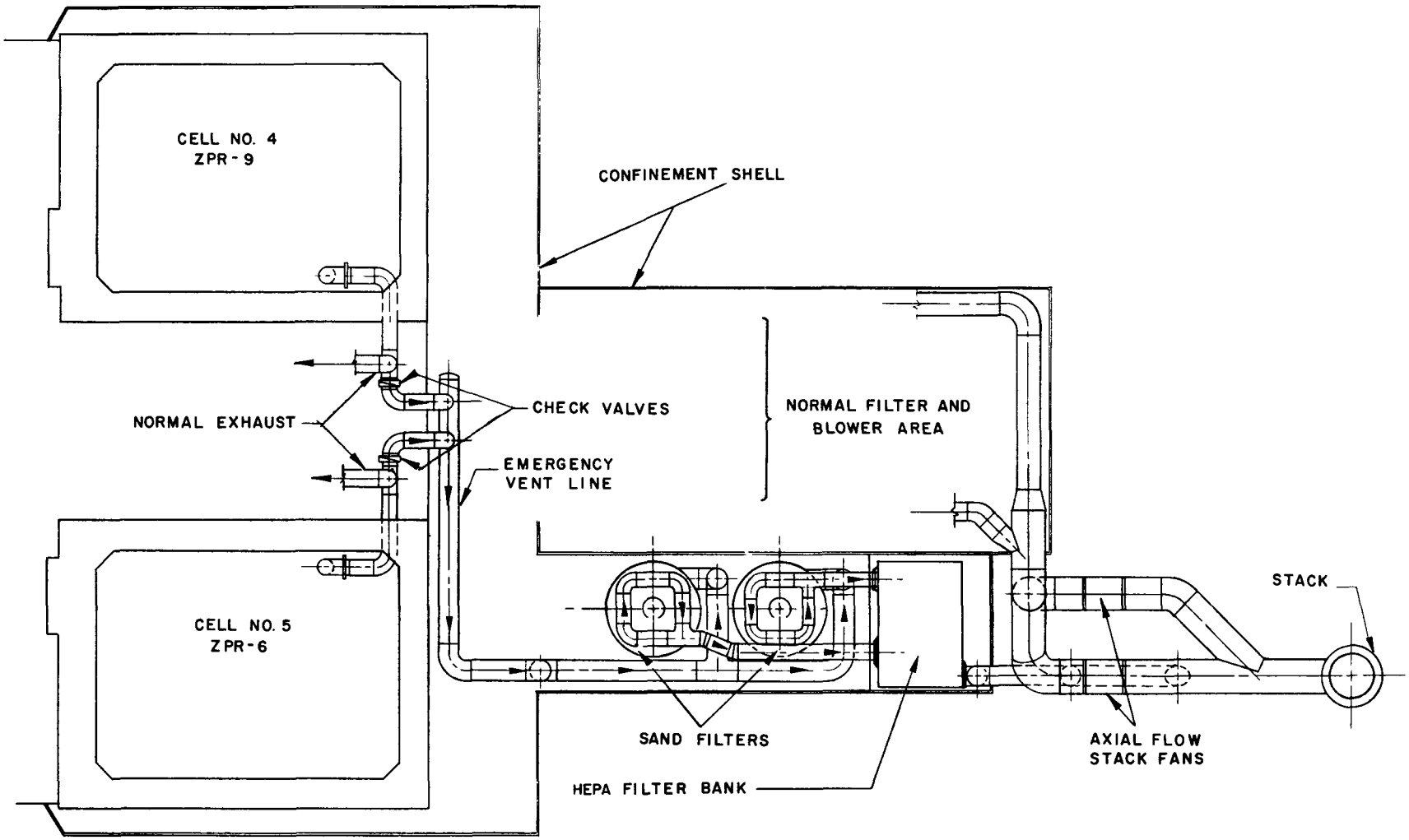


Fig. VI-15. Venting System (Plan view)

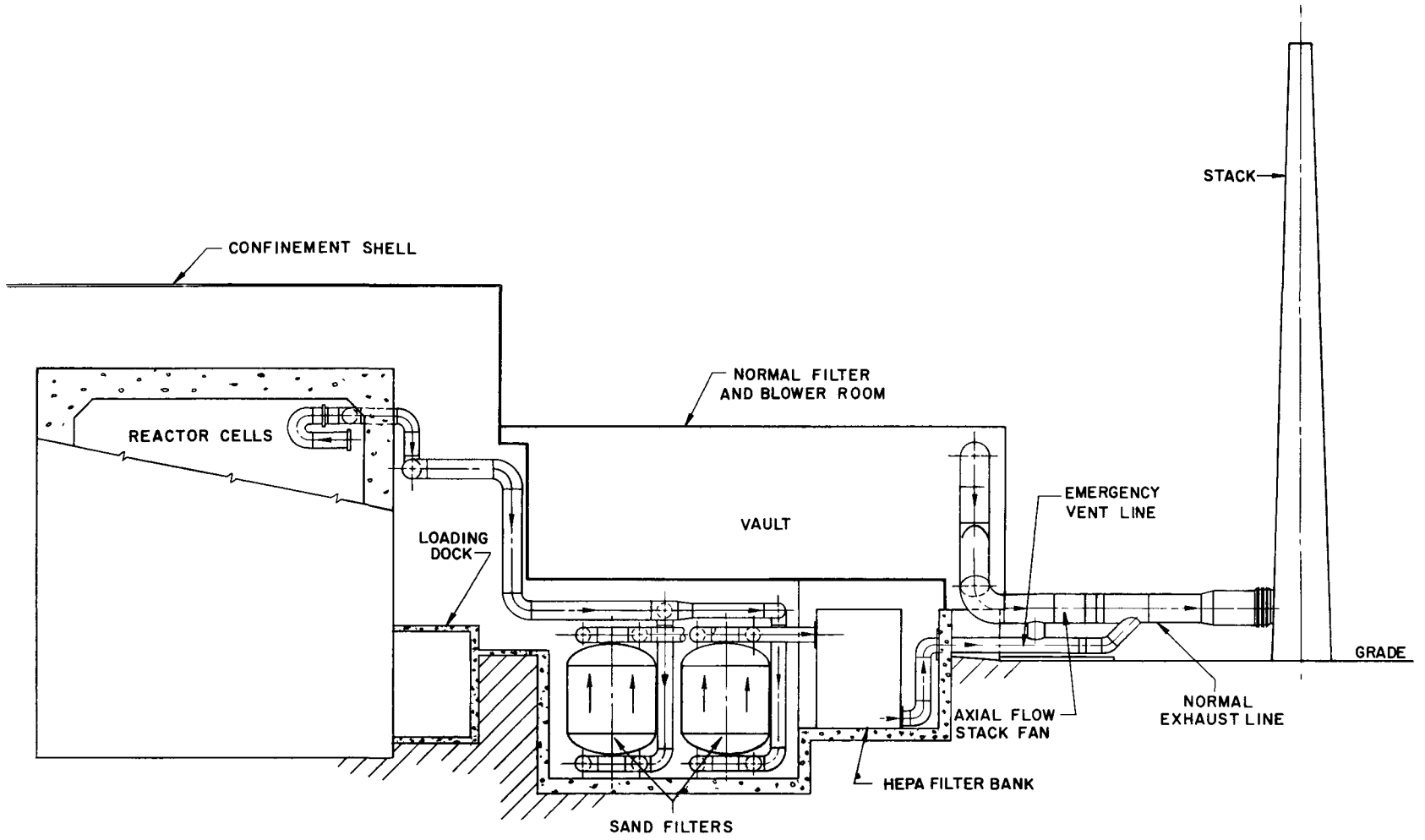


Fig. VI-16. Venting System (Elevation)

normal exhaust system and is connected to sand filters and a double bank of HEPA filters. The check valves, sand filters, two banks of HEPA filters and the 46 m (150 ft) stack are also a part of the emergency exhaust system.

The check valve is a one-way spring loaded, silicon rubber-seated valve as shown in Fig. VI-17 which allows gases to flow only in one direction and are sufficiently leak-tight so there is little backflow. The purpose of the check valve is to allow rapid venting to the sand filter but also to prevent any contaminated gas flow into the other cell. It is assumed that the need for the use of the emergency venting system would not occur simultaneously from both reactor cells. The check valves are (Mission Valve Company, Model No. 15SSSCIC-246) rated at 50 psig service at 260°C (500°F) maximum. They have a 150 lb USASI raised face flanges with ASTM-A216 WCB steel body, plates and stops. They utilize 316 stainless steel pins; Inconel x-springs and silicon rubber seats. They have minimum torque springs which require 1.3 in. of water differential pressure across the valve to open fully. Tests have indicated that the valves will begin to open at a pressure differential in the normal direction with as little as about 0.2 in. of water. The valves were specified to seal bubble tight against reverse flow with differential pressures across the valve from 2 psi to 27 psi. In tests the valves would close bubble tight against back flow with as little as about 6 in. of water differential pressure.

Two large sand filters as shown in Fig. VI-18 have been constructed and installed as a part of the emergency venting system. The sand filters have a 3.6 m (12 ft) diameter and are composed of succeeding finer mesh aggregate ~1.8 m in depth followed by a 75 cm (30 in.) thick layer of 20 to 50 mesh sand. The purpose of the sand filters is to act as a heat sink for hot gases as well as to remove particulate matter. The sand filters have a very large heat capacity in order to handle large volumes of high temperature gases. There is approximately 148 tons of sand and gravel in the two filters. Assuming a specific heat of 0.2 cals/gm/°C there is a heat capacity of about 2.6×10^7 cals/°C in the sand. As shown in Fig. VI-15, there are two 12 ft diameter sand filters in parallel. Each sand filter has been designed to handle about 7100 liters per second (15,000 cfm) of air with a pressure drop of 0.7 bars (10 psi). The maximum flow capacity for the two sand filters is thus about 14,000 liters per second or 30,000 cfm with a 0.7 bars (10 psi) pressure drop. The design of the sand filter was based on flow tests of a 75 cm (29 in.) thick sand filter whose results are shown in Fig. VI-19.

Attenuation tests as described in Appendix B using samples of sand used in the filter have shown that attenuation of between 10^3 and 10^4 for particles into 0.02-0.06 micron range and higher for larger sizes exists at flow velocities up to 10 meters per second (200 ft per minute). Maximum penetrations appear to occur at the higher flow velocities. In addition to the

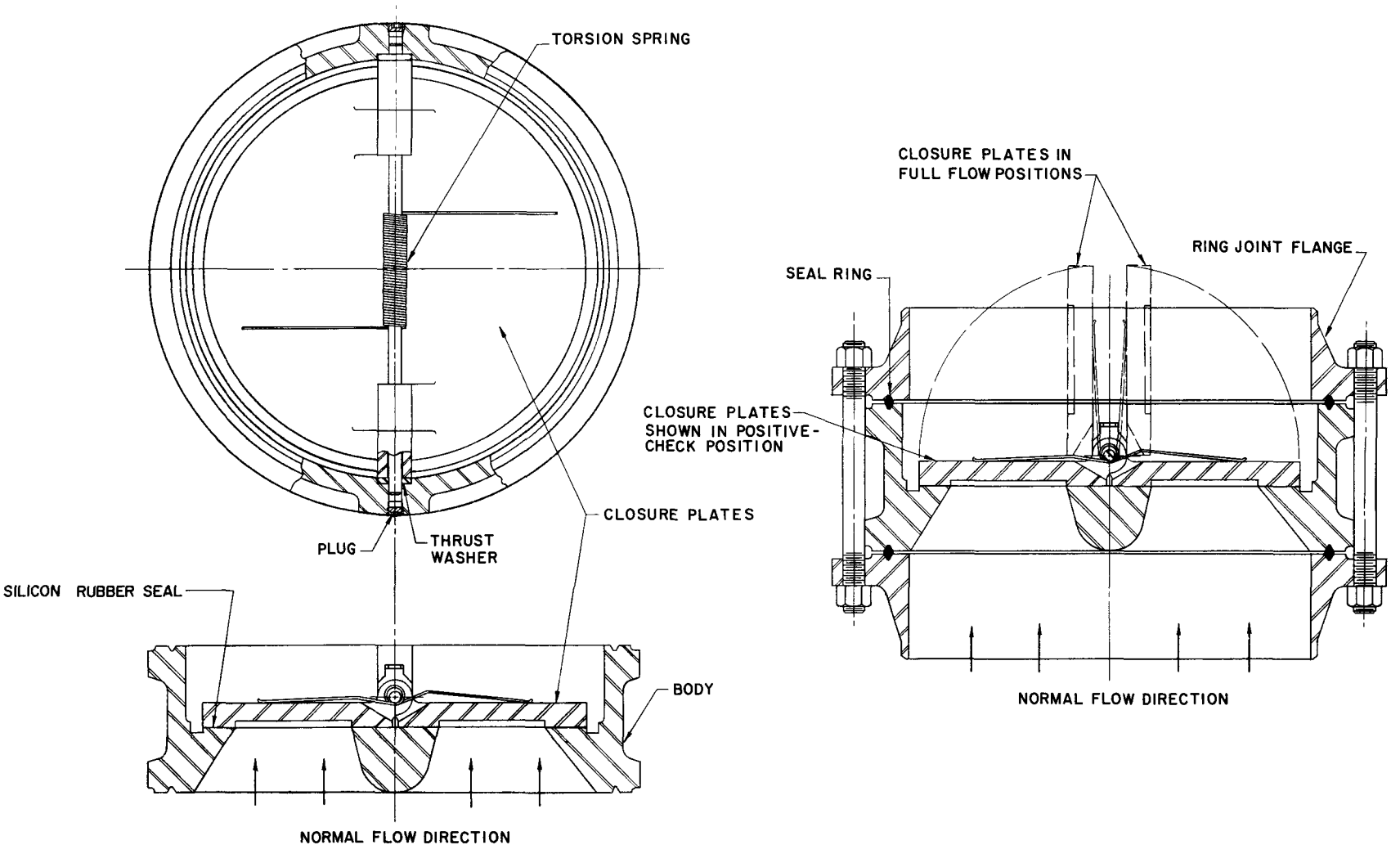


Fig. VI-17. Duo-check Check Value

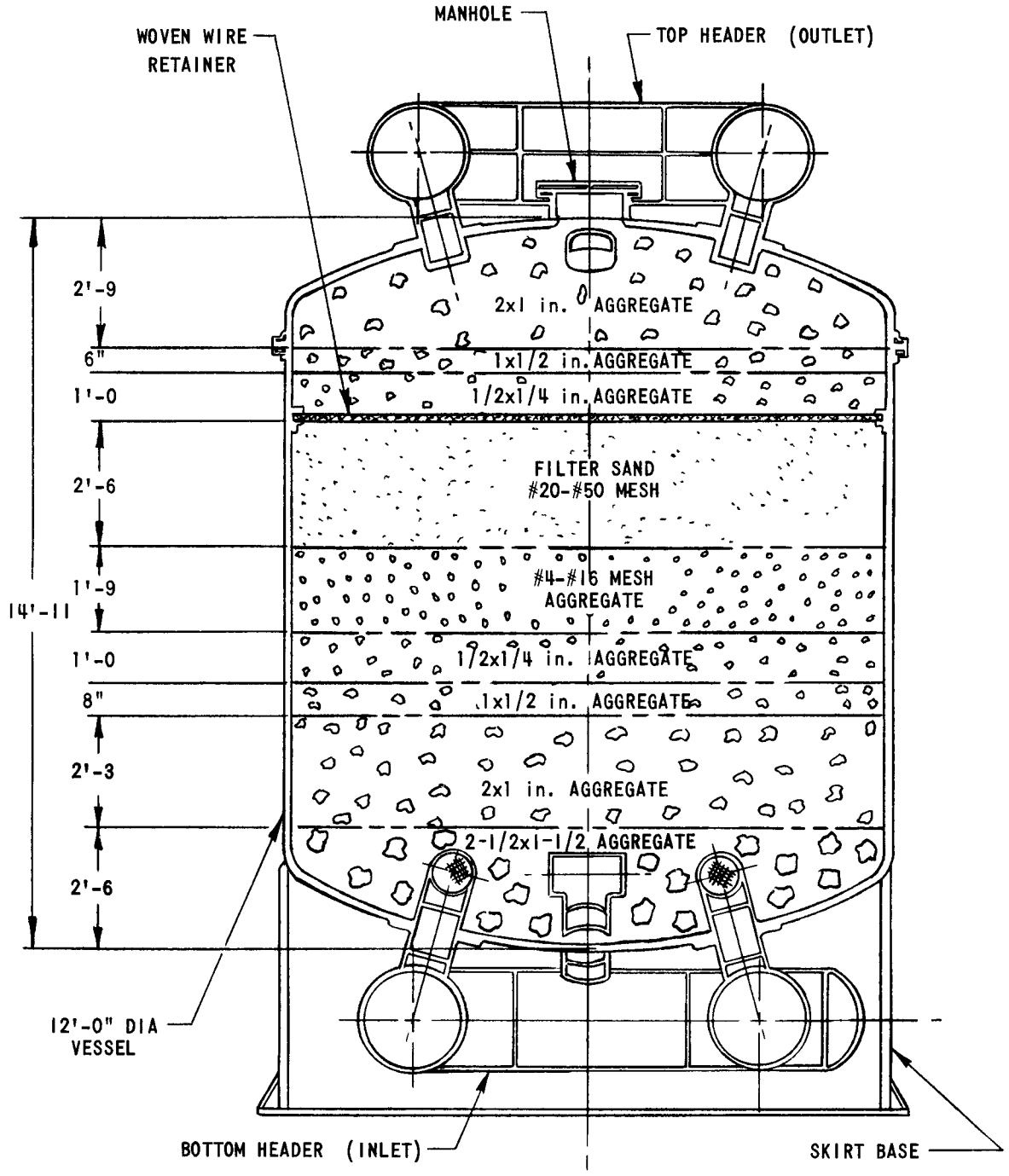


Fig. VI-18. Sand Filter

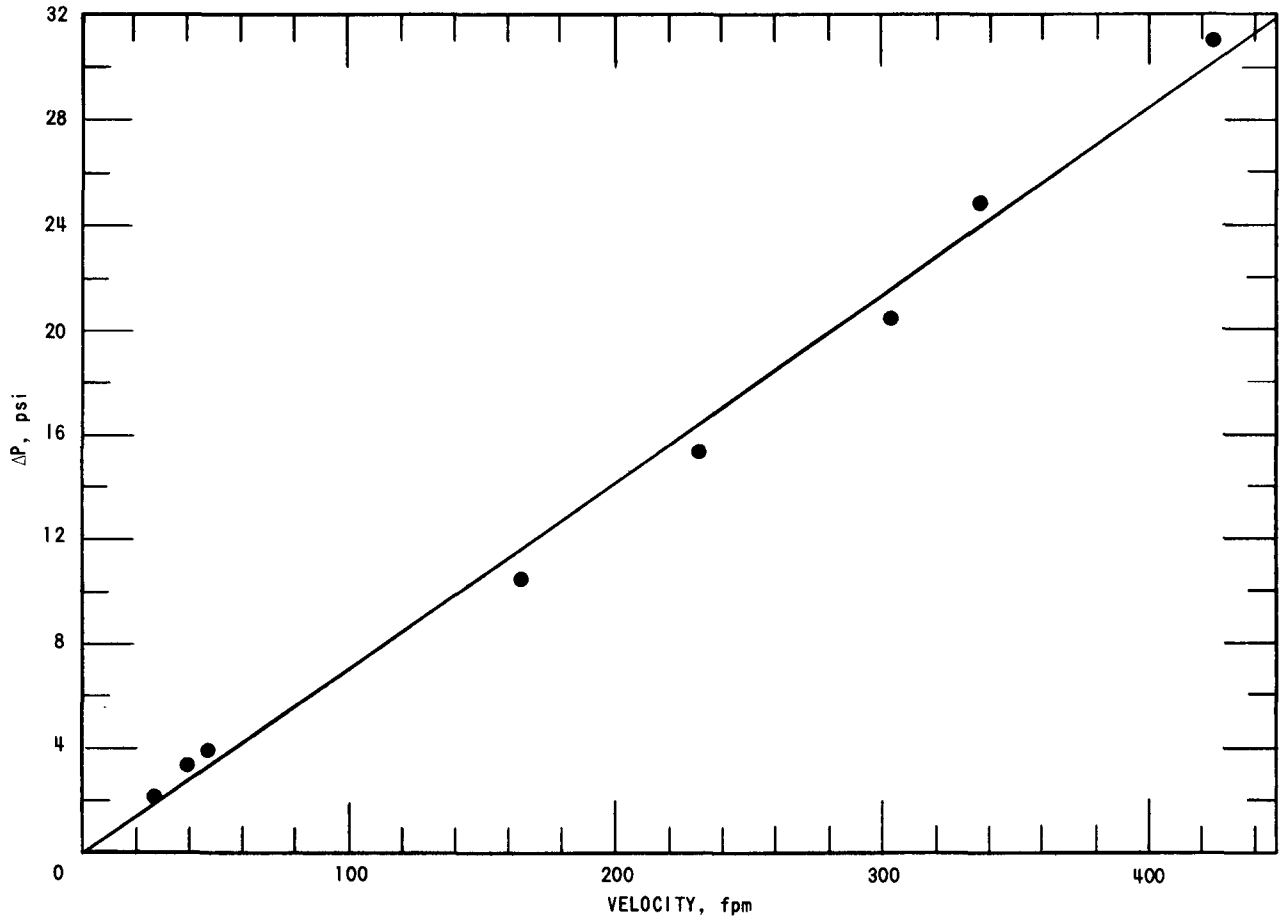


Fig. VI-19. Sand Filter Characteristics

sand filters, there are two banks of high efficiency HEPA filters in the emergency exhaust system to act as a backup for removing fine particulate matter which might not have been removed by the sand filter. Measurements of attenuation of typical filters and one set of AEC filters in series using dioctylphthalate (DOP) as aerosol as well as fumes of uranium and plutonium have shown that attenuation factors about 10^6 and higher are achieved for 30 in. of sand followed by AEC type filters. With an additional HEPA filter bank, it is estimated that the filter system would have decontamination factors of between 10^7 and 10^8. In the emergency exhaust system, there is however no attempt to remove noble gases or volatile radioiodine.

Calculations have indicated that initial heat releases as high as about 1.3×10^7 calories per second with maximum pressure buildup of about 20 psig can be handled by the cell with the use of an emergency exhaust system.

3. Argon Purge System

In order to have a method for extinguishing metal fires, a 3.4×10^6 liter supply of argon gas in high pressured tanks is stored outside of the

reactor facility for use in displacing the oxygen in the cell. High pressure tanks which store the argon gas are connected in parallel to a manifold. The gas is piped in the manifold to the reactor cell walls in a single 4 in. pipe. The schematic diagram of the argon purge system is shown in Fig. VI-20. Table VI-2 lists the function and location of the valves and pressure switches shown in Fig. VI-20. The argon gas is piped through the reactor cell walls in two 2 in. diameter pipes. Feeder lines from the 2 in. pipes in the cell wall distributed the gas around the reactor. The argon gas may be introduced into the reactor cell by an operator at the control console. This is accomplished by opening the main valve which is an electrically controlled pneumatically operated valve with a switch on the control console. A second switch must be opened to insure that the gas goes into the proper reactor cell. All valves in the argon purge system are operable manually as well as electrically.

The purpose of the argon system is to decrease the available oxygen content of the cell rapidly in order to minimize the combustion of such metals as uranium, plutonium, or sodium. The introduction of 3.4×10^6 liters (120,000 cu ft) of argon gas into a cell would decrease the oxygen level from about 20% to 1% which is below the levels for which a self-sustaining metal fire can exist. The argon gas can be introduced into this cell at an initial flowrate of about 36,000 cfm and an average rate of about 6000 cfm so that it would be possible to decrease the oxygen content to about 1% in about 20 minutes. The argon gas would be introduced in the cell if a metal fire occurs which is sufficiently large so that it cannot be put out by conventional MET-L-X type fire extinguishers.

4. Emergency Power Source

An auxiliary steam driven turbine generator which will provide 250 kW electrical power for reactor control instrumentation, exhaust fans, and air compressor in the event of a power failure has been installed in Building 316-W. The replacement of the existing diesel generator by a steam driven turbine generator increases the reliability of the emergency power source. The steam for the steam turbine is provided by the Laboratory's steam plant lines. Interlocks are provided so that the reactors are operable only when the available steam pressure for the turbine is above preset values. The turbine generator will provide power within about 20 seconds of a power failure. The Laboratory steam plant also has auxiliary electrical power generators which will maintain steam pressure even if there is a lengthy power failure.

The purpose of the emergency power source is to have available an alternate source of power for the exhaust fans in order to maintain a negative pressure in the confinement shell volume as well as to provide air pressure to operate the inlet and exhaust valves for the cells. A list of all components which are connected to the emergency power source is given in Table VI-3.

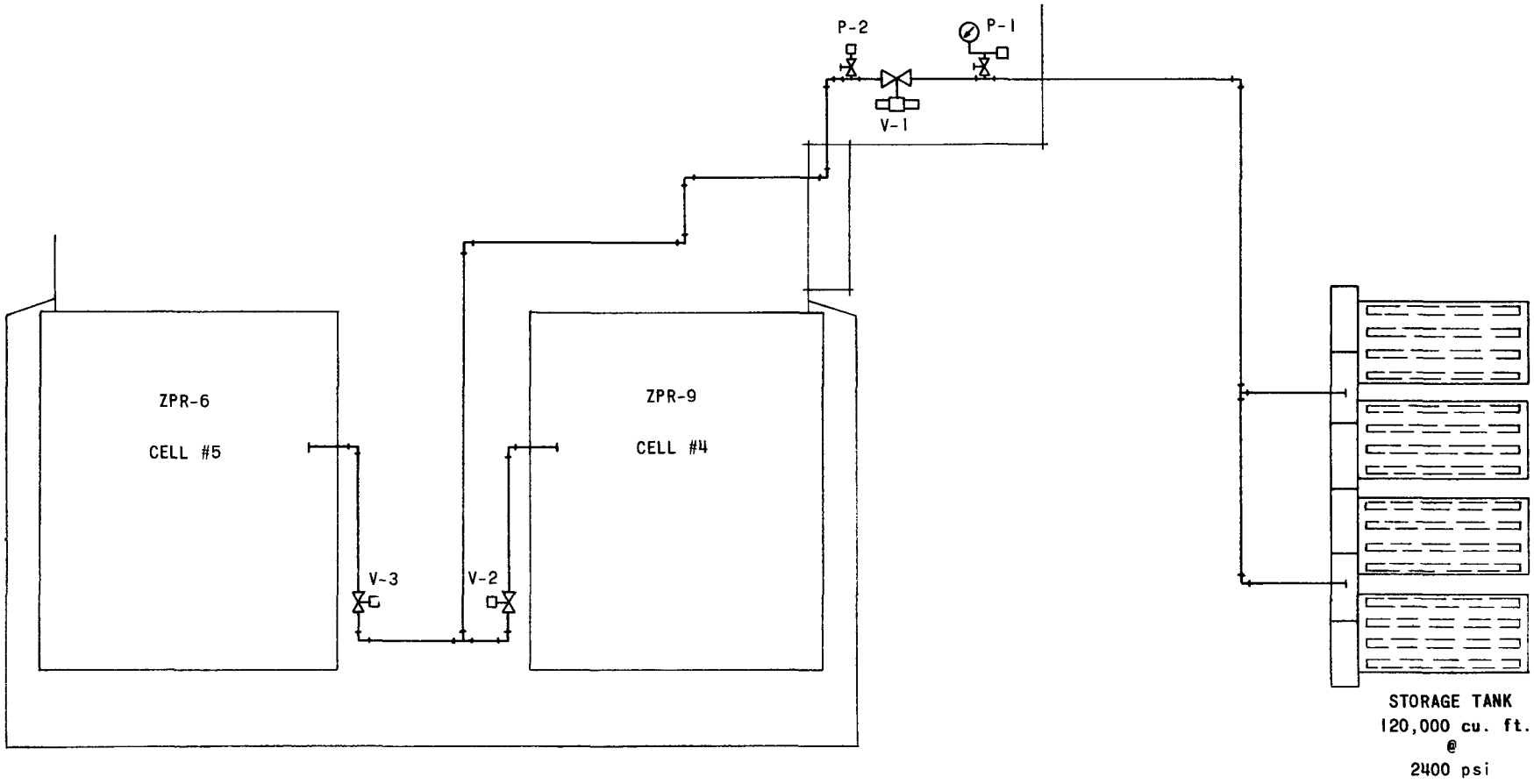


Fig. VI-20. Argon Purge System

Table VI-2

ARGON PURGE SYSTEM COMPONENTS

Item	Type	Function	Monitor	Control
P-1	Pressure Switch	Monitors tank pressure	Equipment room (gauge) ZPR-6 and -9 control rooms (indicator light)	Equipment room
P-2	Pressure Switch	Monitors pressures in lines to reactor cells	ZPR-6 and -9 control rooms (indicator light)	Equipment room
V-1	2 in Ball Shut-off Valve and Operator	Normally closed--releases argon gas into reactor cell on demand	ZPR-6 and -9 control rooms, emergency surveillance room, equipment room	ZPR-6 and -9 control rooms, emergency surveillance room, equipment room
V-2	2 in Ball Shut-off Valve and Operator	Normally closed--releases argon gas to ZPR-9 cell when opened on demand	ZPR-9 control room, emergency surveillance room, basement	ZPR-9 control room, emergency surveillance room, basement
V-3	2 in. Ball Shut-off Valve and Operator	Normally closed--releases argon gas to ZPR-6 cell when opened on demand	ZPR-6 control room, emergency surveillance room, basement	ZPR-6 control room, emergency surveillance room, basement

Table VI-3

COMPONENTS CONNECTED TO EMERGENCY POWER SOURCE

1. Operating Nuclear Instrumentation for ZPR-6 and -9
2. Cell TV Monitoring Systems
3. Cell Air Conditioning Control System
4. Cell Air Supply Valves
5. Cell Air Exhaust Valves
6. Argon Gas Purge System Valves
7. Exhaust Fans for
   - a Confinement Shell
   - b Vault
   - c Workroom
   - d Service Floor
   - e Penthouse
   - f Control Rooms
   - g Central Suspect Air System
8. Building Air Compressors
9. Two Light Outlets in Each Reactor Cell and Each Control Room
10. Cell Vacuum Pumps for Alpha Monitors
11. Central Vacuum System Pumps
12. Emergency Surveillance Station Equipment

E. Emergency Surveillance Station

In order to fulfill the criterion 11 of the AEC General Design Criteria for Nuclear Power Plant Construction Permits that controls for the engineered safeguards must be accessible even in the event of a severe nuclear accident an Emergency Surveillance (ES) station has been established. This station is located outside of the negative pressure areas described in Chapter VI in the southwest corner of Cell 6 as shown in Fig. VI-3. The purpose of this station is to provide a location for the reactor operators to observe the control console instrumentation and to operate the cell ventilation controls or the argon purge system in the event it is necessary to evacuate the control rooms.

In the ES Station are located TV monitors so that it is possible to observe both ZPR-6 and -9 reactor cells and their respective control rooms. The TV cameras located in the control rooms are positioned so that all of instrumentation including the dials and annunciator panels are observable and readable. Scram buttons which actuate scram operation for each of the reactors, remote controls for each of the cell ventilation system (fans, valves, etc.), and controls for the argon gas purge system are located in the ES Station. One neutron detecting channel from each of the reactor cells is also located in the ES Station.

F. Time-of-Flight Experiment

The spectrum measurements by time-of-flight techniques in ZPR-6 will require two 5 in. diameter beam holes penetrating the cell concrete containment structure and the confinement shell. Pulses of protons or deuterons will be ducted through the hole in the west wall to a target in the center of the assembly and neutrons whose flight time over a fixed path length is to be measured will be ducted out of the cell through the hole in the south wall. Both penetrations are to be made by core boring through the existing concrete walls. The penetrations and associated equipment are designed to withstand momentary overpressures of 250 psig for a period of about 3 msec and static overpressures of 50 psig.

1. Time-of-Flight Tube

As far as possible and practical, the bulk of the neutron flight path will be evacuated to approximately 1 mm of mercury or better. That part of the path which is inside of the reactor cell will consist of short sections of evacuated tubes having end windows of 0.0075 in. Mylar. Their length has been chosen so that the sections can fit between the matrix and the mounting plate and between the mounting plate and the wall with a minimum amount of effort and maximum flexibility. A 4 to 6 in. air gap is provided near the wall for insertion of neutron beam filters. The flight tube from the outside of the cell wall to its termination at 100 m as shown

in Fig. VI-21 will be welded steel pipe which will maintain its integrity under both vacuum conditions and 50 psig overpressure at 800°F. The first 67 ft of the flight tube will be 12 in. diameter pipe while the remaining 261 ft will consist of 24 in. pipe. Collimators which define the geometry of the neutron beam, will be located at $3\frac{1}{2}$, $62\frac{1}{2}$ and $262\frac{1}{2}$ ft from the cell wall. Each of the collimators will be 4 ft long and fit inside the pipe. Sections of the pipe near the collimators will be removable in order to facilitate small adjustments in collimator locations.

Fifteen ft long by 10 ft wide detector stations will be located at 50 and 100 m from the exterior cell wall. A short section of the flight tube through these stations will be removable. For the initial measurements an 18 in. square by 5 in. thick plastic scintillator will be used as a neutron detector and will be totally contained within a steel box which terminates the flight tube at either the 50 or 100 m position. One of the two vacuum pumps for evacuating the flight tube will be located in the 50 m station while the second will be inside the cell to evacuate the portion of the flight tube within the cell.

2. Valves

Shock arresting, normally closed valves will be provided for each of the two cell penetrations. One will be located immediately adjacent to the inside surface of the cell wall and the other between the steel containment shell and the concrete cell wall for the time-of-flight tube. Both valves will be of the normally closed type which can be operated remotely. In the closed position the valves will maintain integrity against a 250 psi blast loading of 3 msec duration. The valve does not have to operate under these conditions and may be damaged but will maintain its integrity. Under normal conditions the valve will close in approximately 1/2 to 1 second either under manual control or when the pressure in the flight tube exceeds 50 μ. All valves will revert to their closed position upon detection of the temperature above 120°F or a pressure above 0.5 psig in the cell. The valve will also maintain its integrity against either a positive static pressure of 50 psi at 800°F or a negative pressure (vacuum) of 15 psi in the tube. Damage to the seal under such conditions can be tolerated provided leak tightness is maintained.

The 5 in. penetrations of the cell wall will be as shown in Fig. VI-22. This design insures against any change in the original structural and strength designs of the cell. It also provides an air tight penetration.

3. Proton Transport Tube

Short bursts of protons or deuterons generated by the positive ion accelerator will be transported through a 4 in. stainless steel pipe extending from the accelerator analyzing magnet to the center of the reactor.

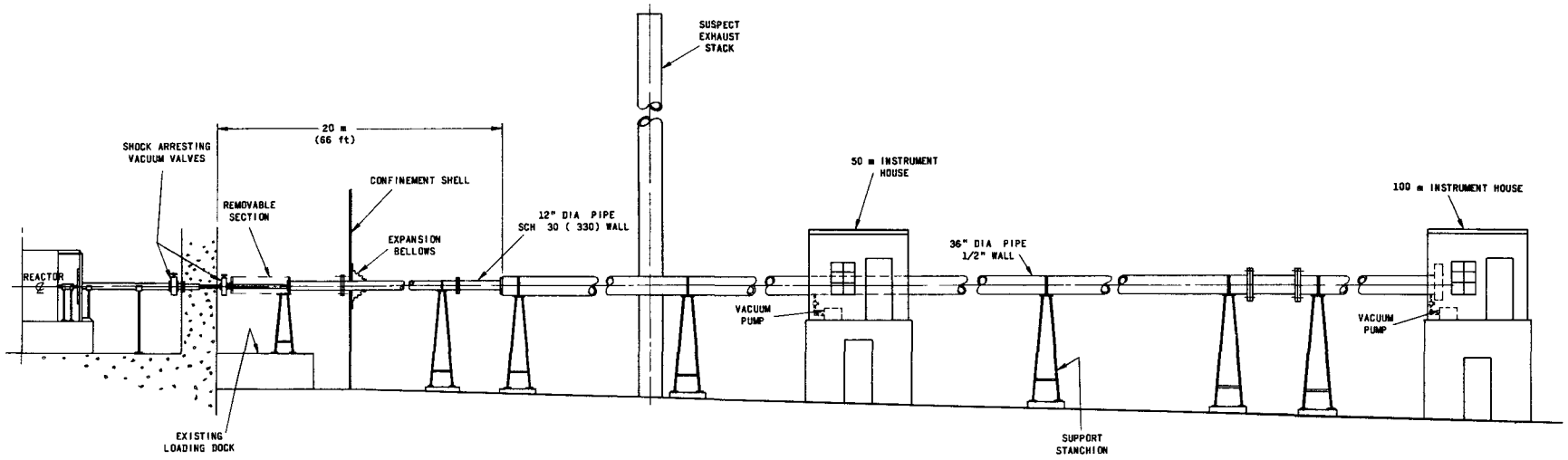


Fig. VI-21. Time-of-Flight Tube

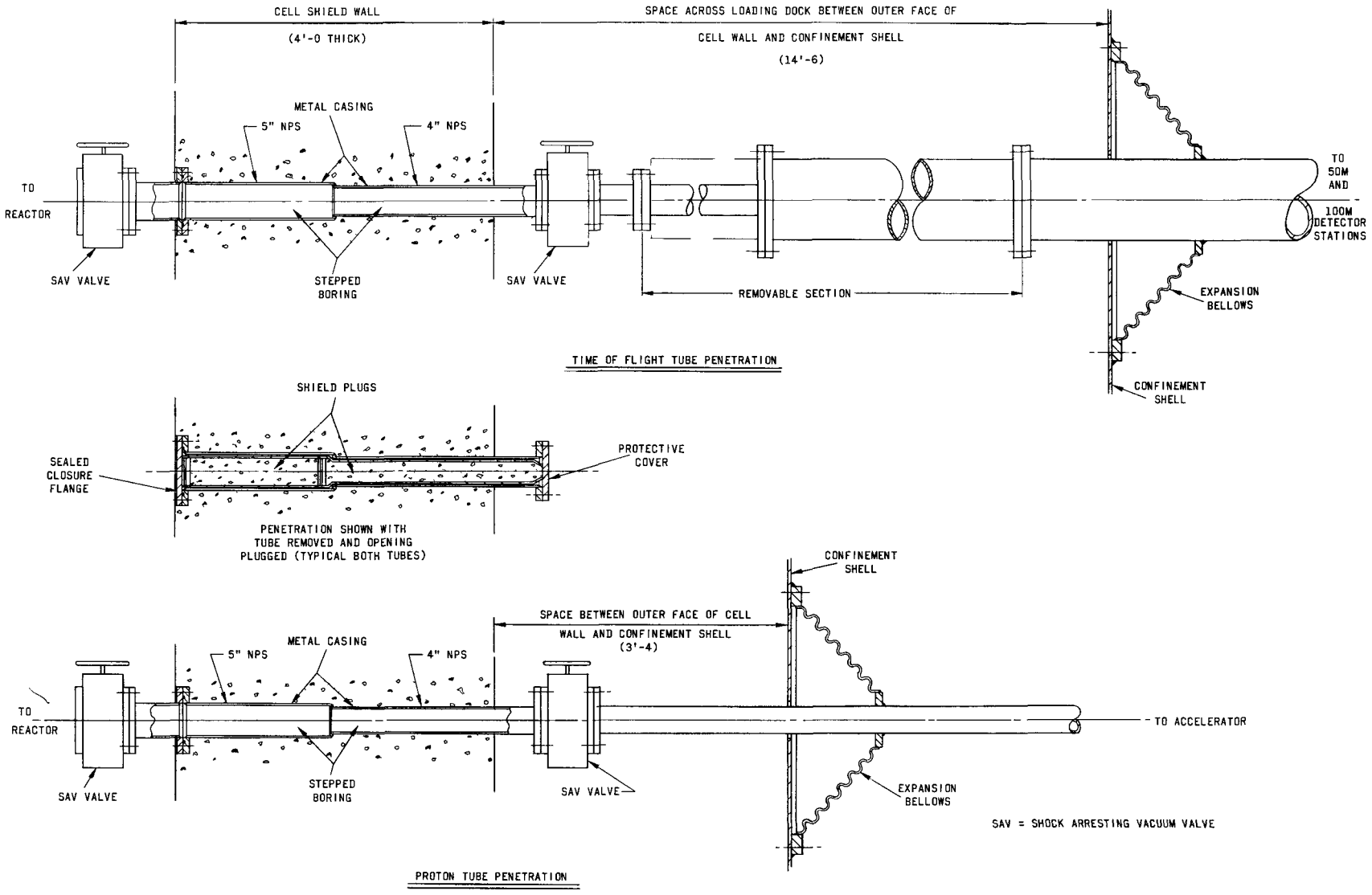


Fig. VI-22. Cell Penetration for TOF Tube and Proton Tube

Steering and focusing of the beam will be accomplished by bending magnets and quadrupole lenses located in the space between the outside shell of the reactor cell and the fast neutron generator building. Cell penetration for the source beam will be identical to that for the neutron time-of-flight tube penetrations.

REFERENCES

1. W. Y. Kato et al., "Safety Analysis Report, Argonne Fast Critical Facility (ZPR-VI)," ANL-6271 (December 1963).

2. W. Y. Kato et al., "Preliminary Safety Analysis Report on the Use of Plutonium and ^{233}U in ZPR-6 and -9 [Addendum No. 3 to Safety Analysis Report, Argonne Fast Critical Facility (ZPR-6), ANL-6271]," ANL-7211 (to be published).

3. S. H. Fistedis, A. H. Heineman, and M. J. Janicke, "Testing of Containment Capabilities of Reinforced Concrete-shielded Enclosures," ANL-6664 (March 1963).

4. W. Y. Kato, op. cit., p. 185.

Chapter VII

REACTOR MANAGEMENT

A. Philosophy

Operation of the facility in a safe manner is the primary concern of all personnel associated with the facility. The first line of defense against accidents is based on the selection of personnel qualified by education and training for the management and operation of the facility. The exercise of careful and cautious judgment on the part of these individuals is expected at all times. The second line of defense consists of the fail-safe design of the various components of the facility and the safety circuits associated with the nuclear instrumentation. The routine maintenance of the equipment and preoperational check procedures serve to minimize the possibility of mechanical or electrical failure of the components of the system. Care in preplanning of the experiments and agreement by at least two qualified persons in regard to loading changes serves to minimize the introduction of unsafe experiments and conditions or faulty operating procedures.

B. Management

Administratively, the ultimate responsibility for the safe, competent, and effective utilization of the Laboratory's facilities rests with the Laboratory Director. Major programmatic and safety matters are channeled to him from the Reactor Manager through the Head of the Fast Reactor Experiment Section, the Associate Division Director and the Director of the Applied Physics Division.

The following categories of personnel are directly associated with the facility: Reactor Manager, Assembly Coordinator, Reactor Supervisor, Operator, and Trainee.

The title Reactor Manager is vested in one person appointed by the Division Director, and in the absence of this person from the Reactor Physics Laboratory building, in an Alternate. He is charged with the overall responsibility for the safe management of the facility. His qualifications and responsibilities are outlined below:

1. Qualifications of Reactor Manager

a. Knowledge and understanding of fast reactor physics

b. Reactor Supervisor status

2. Responsibilities

a. Safe management of facility

(1) enforces observance of all operating procedures and safety rules

(2) insures that all experiments meet the requirements outlined in SAR and other authorizing documents

(3) reviews all experiments from safety point of view

(4) cognizant of measurements being conducted on day to day basis

(5) insures the availability of trained operating personnel when required

b. Reviews and approves the following operations

(1) all complete loading changes

(2) loading changes which can result in reactivity additions greater than $0.80

(3) daily, weekly, monthly, semiannual and annual checklist

(4) use of interlock bypass keys and operating keys

(5) daily operation

c. Training and supervision of operating personnel

(1) recommends the classification of personnel as Reactor Supervisor, Operators and Trainees

(2) supervises personnel in the operation and maintenance of the facility

(3) supervises the training of new personnel in the operation of the facility

d. Maintenance of facility

e. Recommends desirable changes to operating procedures and rules based on operating experience.

In the absence of the Reactor Manager, a person designated Alternate Reactor Manager takes over these responsibilities. The Alternate Reactor Manager must meet the same qualifications as the Reactor Manager. In the absence of both the Reactor Manager and the Alternate Reactor Manager, higher levels of supervision (e.g., Head of FRE Section or Associate Division Director) may assume the responsibilities of the Reactor Manager provided they meet the qualifications given above. ZPR-6 and -9 have separate Reactor Managers.

The Reactor Manager of each of the ZPR-6 and -9 facilities reports to the Head of the Fast Reactor Experiments Section. The Head of the Fast Reactor Experiments Section has the responsibility for the experimental program on ZPR-6 and -9. His basic responsibilities are outlined below:

1. Responsibilities of the FRE Section Head

   a. Experimental program on ZPR-6 and -9

      (1) sees that the experiments carried out on ZPR-6 and -9 are consistent with and meet the requirements of the Laboratory's experimental fast reactor physics program

   b. Sees that the Reactor Managers carry out their assigned responsibilities

   c. Reviews and approves experimental proposals by the Assembly Coordinator to insure that they meet programmatic needs and that they may be carried out in a safe manner

   d. Reviews recommendations of the Reactor Manager.

The Head of the Fast Reactor Experiments Section in turn reports in matters of safety to the Associate Director of the Applied Physics Division. The Associate Division Director has the responsibility for supervision of all of the critical facilities on the Illinois Site under the jurisdiction of the Applied Physics Division from the safety point of view. The Associate Division Director in turn reports to the Applied Physics Division Director who in turn is responsible to the Laboratory Director. The responsibilities of the Associate Division Director in matters of reactor safety are outlined below.

1. Responsibilities of Associate Division Director

   a. Safe operation of all critical facilities under the jurisdiction of the Applied Physics Division on the Illinois Site

      (1) insures the observance of all operating procedures and safety rules

      (2) reviews all unusual experiments brought to his attention by the heads of the experimental sections or Reactor Manager

      (3) with the assistance of a safety inspection group conducts monthly inspection of each of the critical facilities with special emphasis on the mechanical and electrical components

   b. Reviews operating personnel recommendations of the Reactor Manager.

The Laboratory Director, with the advice of the Laboratory Reactor Safety Review Committee, reviews and authorizes procedures and practices for the safe operation of the critical facilities within the Laboratory. A chart of the Laboratory organization showing the line of responsibility for the safety of the reactors is shown in Fig. VII-1.

The Reactor Safety Review Committee is composed of staff members from various divisions of the Laboratory and from varied professional disciplines. The purpose of this committee is to advise and to make recommendations to the Laboratory Director at his request on the safety aspects of the operation of various nuclear facilities at Argonne. It maintains a continuing review of the current operations of the Argonne reactors by giving consideration to changes made in reactor equipment and operating procedures, by considering special nuclear hazards involved in proposed reactors or nuclear multiplication experiments, by periodic inspections, and by devoting special attention to reactor hazards.

The term Assembly Coordinator is used to designate the person responsible for the planning and conduct of experiments on a particular assembly. His qualifications and responsibilities are outlined below:

1. Qualifications of Assembly Coordinator

   a. Knowledge and understanding of fast reactor physics

2. Responsibilities

   a. Plans and conducts experiments on ZPR-6 or -9

      (1) designs core loading (enrichment, plate arrangement, volume percentages of fissile, fertile, structural and coolant materials)

      (2) prepares assembly and detailed drawer loading charts

      (3) he or his designated representative checks drawing and reactor loadings to see that the facility is loaded in accordance with plan

   b. Insures that proper analysis and reporting is made on the measurements conducted on this assembly.

The Assembly Coordinator is designated for a particular assembly by the head of the appropriate section of the Applied Physics Division. The Assembly Coordinator may be a member of such sections as: Fast Reactor Experiments Section, Experimental Reactor Physics Section, or Critical Experiment Analysis Section.

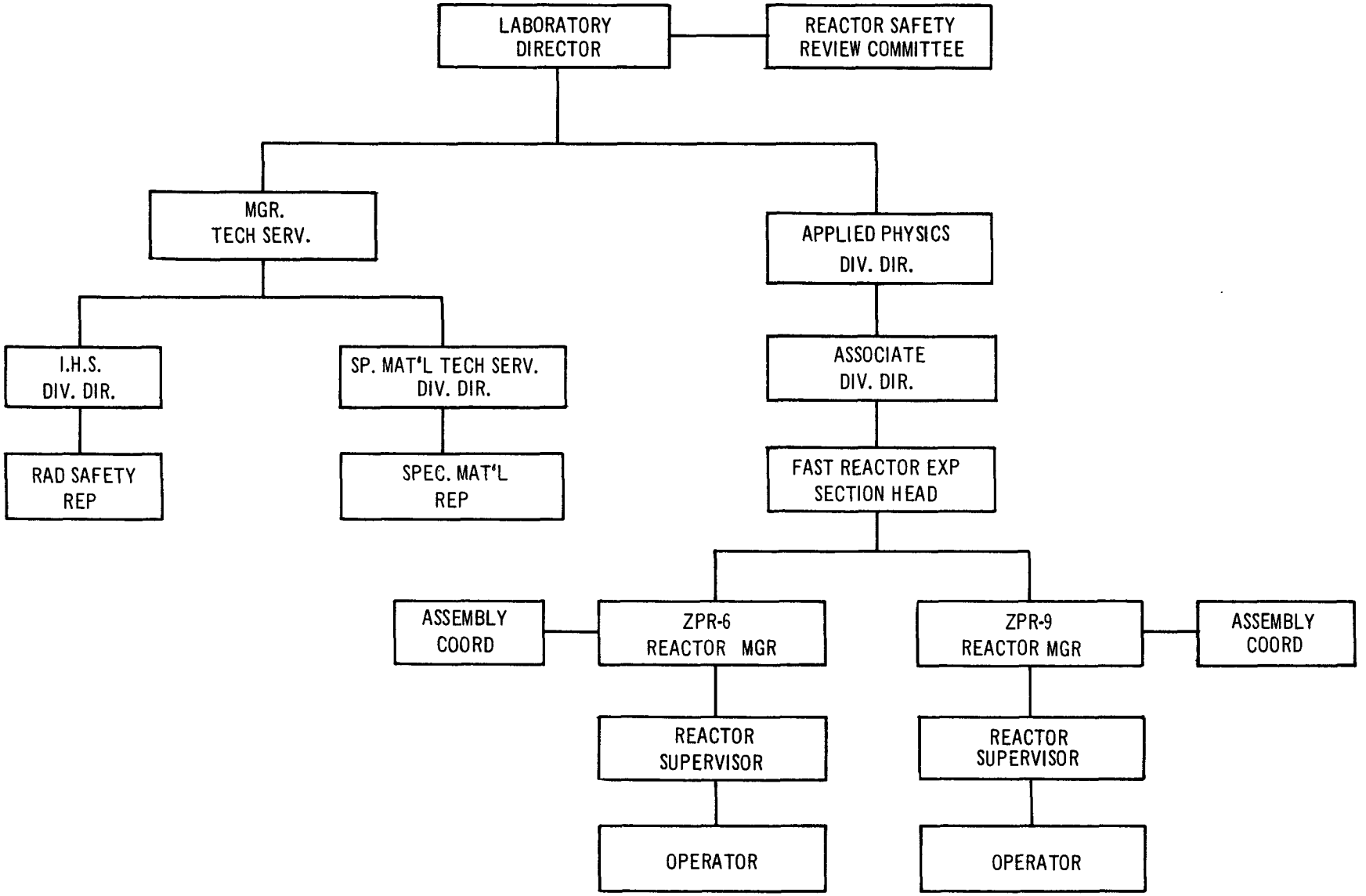


Fig. VII-1. Laboratory Organization for Reactor Safety

A Reactor Supervisor is a person who has sufficient ability and experience in the operation of a particular facility that he has been given the authority to supervise the actual operation of that facility. His qualifications and responsibilities are outlined below.

1. Qualifications of Reactor Supervisor

   a. Knowledge of reactivity effects as applicable to this facility

   b. Familiar with mechanical, electrical and electronic components of facility

   c. Knows how to operate facility and recognize malfunctions

   d. Understand operating and emergency procedures

   e. Experience in reactor operation

2. Responsibility

   a. Assures safe operation of the facility at all times whenever a member of the operating group

   b. Ascertains that all reactor control mechanisms and safety devices are functioning properly

   c. Sees that all operating rules and procedures are followed

   d. Informs operating personnel on decisions on conduct of experiments

   e. Makes judgments on the safety of a particular loading change

   f. Reports daily all changes in core loading less than $0.80 to Reactor Manager.

An Operator is an individual who has had some experience in the routine operation of zero-power reactor facilities and has a clear understanding of the necessary procedures to be undertaken upon the occurrence of abnormal situations on the system. His qualifications and responsibilities are outlined below.

1. Qualifications of Operator

   a. Knowledge of--

      (1) operating procedure and rules

      (2) reactor operation

      (3) maintenance of facility

(4) techniques used in usual reactivity measurements

(5) emergency procedures

b. Experience in actual operation of facility

c. Able to recognize malfunction of equipment or unusual readings of instruments

2. Responsibilities

a. Operation and maintenance of facility

b. Observe all operating procedures and rules

c. To take prompt required action in emergencies.

The Reactor Supervisors and Operators are appointed by the Applied Physics Division Director upon the recommendation of the Reactor Manager, and approval by the FRE Section Head, and the Associate Division Director.

In addition to those directly associated with the facilities the Special Materials Representative, and the Radiation Safety Representative play a key role in the operation of the facility.

The Special Materials (SM) Representative is responsible for storing, issuing and accounting for all fissile and fertile materials in Building 316 not in the reactor. He has responsibility for the fissile material while the fuel is in the vault. He is also responsible for enforcing all rules regarding prevention of accidental criticality for the vault.

The Radiation Safety Representative is responsible for making periodic surveys for alpha and beta contamination and neutron and gamma ray hazards in Building 316. He is responsible for issuing film badges and dosimeters and maintaining accumulated radiation doses records for personnel in the Applied Physics Laboratory.

C. Administrative Procedures

As a consequence of the extreme flexibility available in critical assemblies, all of the operating procedures cannot be precisely specified in advance for every possible type of experiment. For this reason, a number of administrative controls of a very general nature have been established for ZPR-6 and -9.

1. Unusual Experiments

The general procedure followed for any given program of measurements is to submit an experimental proposal to the Reactor Manager for

review and approval from the point of view of reactor safety. Following the Reactor Manager's approval, the routine experiments are then carried out by the Assembly Coordinator and staff. If the experiment is deemed "unusual," in the judgment of the Reactor Manager the proposal and pre-experiment ancillary analysis relative to the pertinent hazards will be submitted to the Associate Division Director for review. He in turn may submit it for information or further review to the Division Director, who in turn may submit it to the Laboratory Director. If it is considered to fall outside the scope of authorized operation the Laboratory Director would submit it to the AEC for approval.

2. Checkout and Preventative Maintenance

Daily, weekly, monthly, semiannual, and annual checkout and preventative maintenance procedures have been established as a means of insuring proper operation of all the mechanical interlocks and instruments and preventing costly and extensive repair or replacement of worn or damaged components. It is by this means that gradual changes in operating conditions are detected and the safeguards instituted to control the hazards of critical experiments are maintained.

3. Keys

There are fourteen keys associated directly or indirectly with reactor operations. Thirteen of them, plus some duplicates, are kept in a key cabinet in the control room, which is always locked whenever there is sufficient fuel in the reactor to produce criticality under any condition. The Reactor Manager, his alternate, Head of the FRE Section and the Associate Division Director, and the Laboratory locksmith are the only persons holding the key-cabinet key and, except for the locksmith, have complete control over all locks and lock switches. A Table Key is issued on a temporary basis to the Assembly Coordinator for the duration of his responsibilities. All other keys are normally under the immediate control of the Reactor Manager or his alternate.

4. Personnel Access

No personnel are allowed in the cell when the reactor is critical. Exceptions must be reviewed and approved by the Laboratory Director. At all times when fuel is present in the reactor or in the storage rack in the cell, no individual may be in the cell by himself; a minimum of two persons is required. It is normally required that the emergency personnel ("E") door and one of each of the personnel and freight doors be closed. There are four situations when both personnel doors or both freight doors may be opened simultaneously when the reactor is not in operation. These situations are:

a. When large components which do not fit in the airlock door must be taken into or out of the reactor cell from the control room or building exterior.

b. In order to accommodate large groups of visitors both doors may be left open for rapid transit. Visitors are always escorted by personnel associated with the facilities.

c. Maintenance or repair of doors.

d. When all the fuel has been removed from the reactor and cell.

The Reactor Manager must approve the simultaneous opening of the personnel doors or the freight entrance. Only operation directly related to the requirement of simultaneous opening of either the freight or personnel doors will be permitted when these doors are open. The simultaneous opening of these doors will be kept to a minimum consistent with operational requirements. During operation all doors as well as the six ports to the vault workroom must be closed. The cell, control room, vault, and vault workroom are within the 316-West FH area and access to these areas is restricted to authorized personnel by a Security Division guard at the FH area entrance. The number of people in the reactor control room is to be kept at a minimum during reactor operations since uninformed visitors or others not immediately involved in the operation may affect the hazards of an experiment if by no other means than to provide a distraction to the operators.

D. Operating Rules

The current general set of rules for the operation of the ZPR-6 and -9 are summarized in this section for convenience.

These rules conform to procedures established and in current use at the critical experiment facilities on the Illinois Site. It must be recognized that such procedures may change with time and are always subject to improvement. When changes are proposed in such general procedures, they will be carefully reviewed and will be approved by appropriate Laboratory management only if a clear advantage exists in the changed procedures. Thus, these rules, although current and of considerable duration, must be considered to some extent as illustrative of attitude, rather than as firm commitments.

For the operation of the facility, two separate and different keys are required to open two locks. One key, called the Supervisory Key, allows the main power to be turned on at the control console. The second key, called the Table Key, controls the power to drive the movable table. The Supervisory Key will be controlled by the Reactor Manager or a delegated Reactor Supervisor when not inserted in the control console.

The Table Key will be controlled by the Assembly Coordinator or his designate. The two-key system will serve as a check that a minimum of two responsible persons are cognizant of the operation of the facility. The back of the control console is locked except when access is required for maintenance.

1. General Rules

(a) There must be two independent mechanical means for achieving shutdown. For ZPR-6 and -9 these are (1) the dual-purpose rods and the ^{10}B safety blades, and (2) the movable table.

(b) The presence of a minimum of two persons is required during any change in reactor loading.

(c) The initial criticality for new cores must always be approached by a series of discrete steps such that the critical conditions can be predicted by extrapolation.

(d) Prior to the first operation of the day except during periods of extended continuous operation, the reactor must be checked out and inspected by a member of the operating crew. The Reactor Manager or his alternate must be informed of the general program of measurements to be performed and must approve operation on a daily basis

(e) Before each run, personnel must be evacuated from the cell and all doors and ports to the cell must be closed.

(f) Any member of the operating crew is urged to stay away from the reactor controls if he does not feel qualified to assume the responsibilities which go with reactor operation. Preoccupation due to personal circumstances or physical disposition will be honored as good reasons to refrain from participating in the running of the reactor.

(g) Interlock bypasses can only be used under the immediate supervision of the Reactor Manager or his alternate and must be recorded in the log book for each instance of use.

(h) Final reactivity adjustment as criticality is approached is normally through control rod motion. If criticality is approached during safety rod or table motion because of a loading error, that approach to critical will be terminated and excess reactivity removed.

(i) The control console must be attended as long as the Supervisory and Table Drive Keys remain inserted in the console key switches. At no time during the operations is the man operating the controls to leave them unattended, nor is he to partake in other activities beyond the direct

manipulations of the controls until he is relieved. Nor is any unauthorized person (any person not specifically named on the current crew member list) to be allowed to operate the reactor controls.

(j) Occasionally it may be necessary to bring the reactor tables partially or completely together with fuel in the reactor matrix (for maintenance) while personnel are in the reactor cell. (All work of this nature must be done under the direct supervision of the Reactor Manager.) Before this is done, sufficient fuel must be removed from the reactor and measurements made to assure that the maximum k_{eff} is no greater than 0.9; confirming measurements must be made with the halves together and all rods in their most reactive positions.

(k) A predetermined minimum neutron induced counting rate must exist on channels 1 and 2, before moving control rods or safety rods/blades.

(l) All required facility instruments and interlocks must be operating properly during reactor operations.

(m) All ^{10}B safety blades and dual-purpose rods designated as safety rods must be available for the introduction of negative reactivity before the movable half is driven towards the stationary half.

(n) Any sudden changes in apparent multiplication or reactivity must be diagnosed immediately or the reactor shut down.

(o) The movable table must be driven to its maximum separation before the reactor cell is entered.

(p) An up-to-date set of records covering the operation of each system must be available at all times.

(q) An assembly area containing fissile material may not be entered for fire-fighting purposes unless the individuals are accompanied by a Reactor Supervisor or Area Emergency Supervisor.

(r) Hydrogenous materials are not to be stored in the reactor cell.

(s) The cell containment must not be breached (e.g., opening of both personnel doors) unless prior approval is obtained from the Reactor Manager. Any containment breach is to be of minimum duration. The Reactor Manager will not authorize a containment breach during reactor operation or whenever such a breach can significantly compromise the safety of the facility.

(t) The Reactor Manager will review the operations log book weekly during periods of operation and note compliance by initialing and dating in the log book.

(u) All modifications of the mechanical and electrical components of the facility will be approved in advance by the Reactor Manager, and this approval indicated by initialing and dating of the descriptive entry made in the maintenance log book or an equivalent form.

(v) The Reactor Manager is responsible for the safe operation of his facility and as a part of this responsibility will insure that those involved in the actual operation are informed of all changes affecting the safety of the operation.

(w) The unalloyed Pu fuel will be limited to zones for which the reactivity of the zone is no more than 25% of the total reactivity as defined on p. 21.

(x) The ports for the neutron time-of-flight tubes and charged particle tube will not be opened when the reactor has a $k_{eff} > 0.98$.

(y) The cell must not be entered until it has been ascertained that the airborne Pu concentration is below prescribed levels.

(z) The ZPR-6 and -9 facilities will only be operated when the pressure within the confinement shell is at least -1.0 inch of water and the other suspect areas are less than -0.1 inch of water with respect to the atmosphere.

2. Reactivity Requirements

(a) Shutdown Reactivity: A minimum of 2% $\Delta k/k$ must be available due to the action of all dual-purpose and ^{10}B rods.

(b) Dual-Purpose Control/Safety Rod Selection: The reactor must be subcritical by at least 0.5% $\Delta k/k$ when the reactor halves are together and the control rods are in positions of minimum reactivity.

(c) Control Rod Worth: The maximum worth of any one control rod must not exceed $3.

(d) Maximum Excess Reactivity: The maximum excess reactivity, with all rods in their most reactive positions, must not exceed $0.80. The reactivity restrictions for ^{235}U and Pu cores are summarized in Table VII-1.

(e) The reactivity addition rate, whether by control rods, or table motion, etc., may not exceed $0.04/sec when within 2% Δk of critical.

(f) Reactivity may be added in only one manner at a time.

TABLE VII-1

REACTIVITY LIMITATIONS

		U^{235} Cores	Pu Cores
1.	Degree Subcritical (Subcritical Reactivity) Due to All Rods in Their Least Reactive Positions	2% Δk ($3 for β_{eff} = 0.0063)	2% Δk ($6.00 for β_{eff} = 0.0033)
2.	Degree Subcritical With Table Halves Together and All Control Rods Out	0.5% Δk (80¢ for β_{eff} = 0.0063)	0.5% Δk (1.50 for β_{eff} = 0.0033)
3.	Maximum Worth of One Control Rod	$3.00 (1.9% Δk for β_{eff} = 0.0063)	$3.00 1% Δk for β_{eff} = 0.0033)
4.	Maximum Excess Reactivity with All Rods at Their Most Reactive Positions	80¢ (0.5% Δk for β_{eff} = 0.0063)	80¢ (0.27% Δk for β_{eff} = 0.0033)

3. Loading Changes

There are three types of operation of the fast critical facilities in which the type and number of operating personnel required at the console differ. The three types are when the reactor is in a confirmed state, unconfirmed state, or two operator operation state.

(a) Confirmed State

The reactor is defined to be in a "confirmed" state when the following requirements are satisfied:

1. The shutdown margin has been experimentally established as 2% Δk/k or greater.

2. The maximum excess reactivity is known to be less than $0.80.

3. The increase in reactivity over that of a previously operated confirmed reactor is not greater than $0.30.

4. No changes in control/safety rod or instrument performance/location, or in interlock/control circuitry have been made since last having a confirmed core.

Examples of confirmed operation of ZPR-6 and -9 are given below:

1. foil activation or nuclear emulsion irradiation;

2. flux traverses or fission ratio measurements by means of foils or small counters;

3. danger coefficient measurements when small reactivity changes are involved;

4. pile-oscillator-type measurements after the reactor has been made critical with the oscillator sample in the reactor;

5. sodium-void coefficient measurements when small reactivity changes are involved.

For a confirmed operation a minimum of one Reactor Supervisor and one Operator is required at or near the control console observing the operation of the facility.

(b) Two Operator Operation

Under very restricted conditions the ZPR-6 and -9 facilities may be operated with only two Operators at the control console. The physical presence of a Reactor Supervisor would be required only during startup and the initial phases of confirmed operation. The following conditions must be met before two operator operation may be initiated.

1. Reactor must be in a confirmed state

2. The experiment to be conducted on the reactor must be one of the following four:

   a. Foil irradiations at a constant power level for periods of time in excess of 30 minutes.

   b. Rossi-alpha and spectrum measurements usually performed in subcritical state.

   c. Material replacement measurements made with the sample changer or with the single drawer Doppler apparatus and in which a prescribed sequence of experimental operations are repetitively performed. These measurements do not necessarily have to be central measurements.

   d. Fission ratio and fission distribution measurements using small fission detectors with some sort of a traverse mechanism.

3. The reactivity addition associated with these experiments cannot exceed \$0.30 relative to the unperturbed state. This will require at least one measurement of the most reactive material worth or Doppler sample.

4. The Reactor Supervisor must confirm that the maximum available excess reactivity is not greater than \$0.80 and that a 2% $\Delta k/k$ shutdown margin is available at all times.

5. The Reactor Supervisor must make sure that the experiment and its execution is understood by the operating crew and that a routine procedure has been established for the experiment.

6. When in the judgment of the Reactor Supervisor, his physical presence is no longer required for the safe conduct of the experiment, he may leave the control room but not the Reactor Physics Laboratory building taking the Supervisory key with him. The Supervisory key is interlocked in such a way that once the reactor is in an operating state, its removal will not cause a scram.

7. The fact that all of the above conditions have been satisfied and the telephone number in the Reactor Physics Laboratory at which the Reactor Supervisor can be reached will be recorded in the log book by the Reactor Supervisor's signature. It will be his responsibility to have a minimum of two Operators at the console at all times during operation.

(c) Unconfirmed State

For all other circumstances the reactor is in an "unconfirmed" state.

Examples of changes which cause the reactor to be in an unconfirmed state are:

1. Addition of ^{233}U, ^{235}U, ^{239}Pu, etc., which can result in a positive reactivity change in excess of $0.30.

2. Rearrangement of core or blanket materials which can result in a positive reactivity change in excess of $0.30.

3. Changes in location of nuclear sensors used in facility operation.

4. Changes in location of control and/or safety rods.

Loading changes which can result in reactivity additions greater than $0.80 must have the approval of the Reactor Manager or his alternate.

Two Reactor Supervisors must sign all Reactor Loading Records before the intended loading changes are made. Use of a pink Reactor Loading Record implies an unconfirmed core or experiment, whereas use of a white chart implies a confirmed core or experiment. The signatures of the two Reactor Supervisors implies that they agree the core is confirmed or unconfirmed according to the color of the chart used.

A minimum of two Reactor Supervisors is required at or near the control console observing the operation of the reactor when it is in an unconfirmed state.

4. Rules for Loading and Storing the ZPR-6 and -9 Fuel Drawers

(a) The Reactor Loading Records must be approved by two Reactor Supervisors before the intended loadings are made.

(b) Loading of the fuel into the drawers will normally be done only in the vault workroom and under the supervision of the Assembly Coordinator or his designated representative.

(c) Normally only numbered drawers are to be used to hold core materials; unnumbered drawers may be used for counters.

(d) No more than three fuel drawers may be in the process of being loaded or unloaded at any one time in the workroom.

(e) No more than 6 kg of enriched uranium or 4.6 kg of contained Pu will be allowed on the loading table or processing hoods at any one time, and all drawers containing fuel must be removed from the work area as soon as they are loaded.

(f) During all loading and unloading operations, the Assembly Coordinator will be responsible for the safe and proper handling of the fuel for which he is charged. All materials and drawers must be kept in their designated storage locations when not being loaded or in the reactor.

(g) In all core changes involving the addition of fuel, it is the responsibility of the Assembly Coordinator to insure that the change in multiplication is observed. It is required that either channel 1 or 2 and its associated popper be operating during all fuel loading changes.

(h) The Assembly Coordinator or his delegate must check each drawer as it is loaded against the proper Drawer Master, and then sign the Reactor Loading Record before the drawer leaves the workroom.

(i) After the loaded drawer is inserted into the reactor matrix the name of the individual who checked the matrix loading is also recorded on the assembly loading chart.

(j) No fuel bearing drawers may be stored in the reactor cell without first informing the Reactor Manager or his alternate and then only on the criticality-safe storage racks provided. Storage other than in the reactor matrix is always to be of a temporary nature, i.e., not for more than a few days.

(k) Loading changes for only one facility may be made in the workroom at a given time.

E. Security

The west wing of the Reactor Physics Laboratory is divided into three areas from the standpoint of security. The offices, laboratories, and counting rooms in the basement level, first floor, and second floor are considered open areas where personnel assigned to the building and visitors, after identification by the receptionist, may move freely. The southern portion of the first floor, which includes two control rooms, reactor cells, vault, and vault workroom, is designated the FH area, and access is strictly controlled. All doors leading out of the FH area are alarmed. In order to gain access to the FH area, personnel must be on an access list and be identified by a security guard at the entrance to the FH area. Access to the FH area will be limited to those personnel having direct relationship with the Zero Power Reactor facilities. Visitors may have access to the facilities only under escort The third area which consists of the basement service area (Mechanical Equipment Room No. 1) and the penthouse service area (Mechanical Equipment Room No. 3) are also limited access areas. The entrance doors to these areas are normally locked during reactor operation. Access to these areas by authorized personnel may be accomplished by obtaining a key for these areas from the Security Guard.

The interior entrance door to the FH area, vault workroom doors, vault and the exterior doors to the reactor cells and control rooms are protected by an alarm system. The opening of any of these doors in an unauthorized manner activates the Laboratory security alarm system. Steel bars installed in the ventilating duct systems going into the FH area will prevent unauthorized entrance to the area through these openings

All exterior doors to the Reactor Physics Laboratory except personnel entrance doors outside of the FH area, are normally locked. The key for these doors is under the supervision of the Building Superintendent and may be used when special access is desired. During nonworking hours, access to the Reactor Physics Laboratory building outside of the FH area is gained by obtaining a key for the building from a guard post.

F. Fuel Storage Management

The fissionable material used as fuel for ZPR-6 and -9 will be stored either in the vault for the reactor cell or in the facility.

The SM (Special Materials) representative issues fuel, in units not to exceed 6 kg of ^{235}U/or 4.6 kg of plutonium from the vault to the workroom when drawer-loading operations are in process. Fuel will, in general, be brought out of the vault in the birdcages. No more than three drawers may be in process of either loading or unloading at any one time. During the unloading operation, no more than three or the number of drawers corresponding to 6 kg of ^{235}U content or 4.6 kg of plutonium whichever is less, will be unloaded at a given time. The ^{235}U or plutonium will be restored to the vault in birdcages immediately after removal from the drawers.

Chapter VIII

OPERATION

A. Loading

Prior to commencement of loading, the Assembly Coordinator will plan carefully the type of assembly to be constructed in the ZPR-6 or -9 facilities. He will have available preliminary theoretical calculations of such parameters as critical mass and worth of fuel drawer to guide him in the design of the loading. In large cores in which the fuel drawer worth is small, he will add the proper number of insertion-type safety rods to each half of the assembly so that there will be a minimum of 2% $\Delta k/k$ shutdown available due to the action of the dual-purpose rods and insertion safety rods. The basis for the selection of the number and location of these safety rods will be provided by calculations for that assembly prior to loading or from previous experiments on a similar assembly. The worth of the safety rods to meet above requirements will be verified after criticality is attained.

The Assembly Coordinator will see that first a Reactor Loading Record (Fig. VIII-1) for each half of the facility is completed. This will designate the type of loading to go into each matrix tube. The Drawer Master (Fig. VIII-2) will then be made out for each type of drawer; these charts indicate the composition in terms of fuel, blanket, and structural material in the drawer. These charts are used to load and check the drawers which are numbered to correspond to their location in the reactor.

After obtaining the approval of the required number of Reactor Supervisors for the Reactor Loading Record, the Assembly Coordinator or his designated representative will supervise the loading of the drawers based on the Drawer Master in the vault workroom.

The Special Materials (SM) Representative will release fuel from the vault upon presentation of a properly authorized Drawer Master. The following steps are illustrative of the procedures taken by the SM representative and loading crew during loading operation.

1. Storage containers (birdcages) for the fuel will be removed from the vault and brought to the workroom as shown in Fig. VIII-3. Up to 30 unopened birdcages may be temporarily stacked against the south wall. They will be stored there until the particular fuel plate size they contain is needed in the drawer loadings. The actual number of birdcages stacked in readiness will be determined by the size of the loading or loading change and the different types of fuel needed in a loading. The level of neutron and gamma radiation existing in the room may also be a limiting factor. A Special Materials Representative will be responsible for these activities.

ZPR-6

UNCONFIRMED REACTOR LOADING RECORD

(Two Reactor Supervisors Required for Reactor Startup)

Approved By _____ Date _____ Assembly No. _____

_____ _____ Loading No. _____

	DRAWER NO.			STATIONARY HALF				MOVEABLE HALF				ASSEMBLY
				MASTER	KILOGRAMS		CH'K'D	MASTER	KILOGRAMS		CH'K'D	LOADING
	S M	ROW	COL.	NO.	U^{235}	$Pu^{239}+Pu^{241}$	BY	NO.	U^{235}	$Pu^{239}+Pu^{241}$	BY	CH'K'D BY
1												
2												
3												
4												
5												
6												
7												
8												
9												
10												
11												
12												
13												
14												
15												
16												
17												
18												
19												
20												
21												
22												
23												
24												
25												

TOTAL DRAWERS LOADED WITH FUEL _____ ESTIMATED REACTIVITY CHANGE _____

TOTAL Kg $Pu^{239}+Pu^{241}$ IN CORE _____ NET CHANGE IN Kg $Pu^{239}+Pu^{241}$ _____

TOTAL Kg U^{235} IN CORE _____ NET CHANGE IN Kg U^{235} _____

RP-71 (4-7-69)

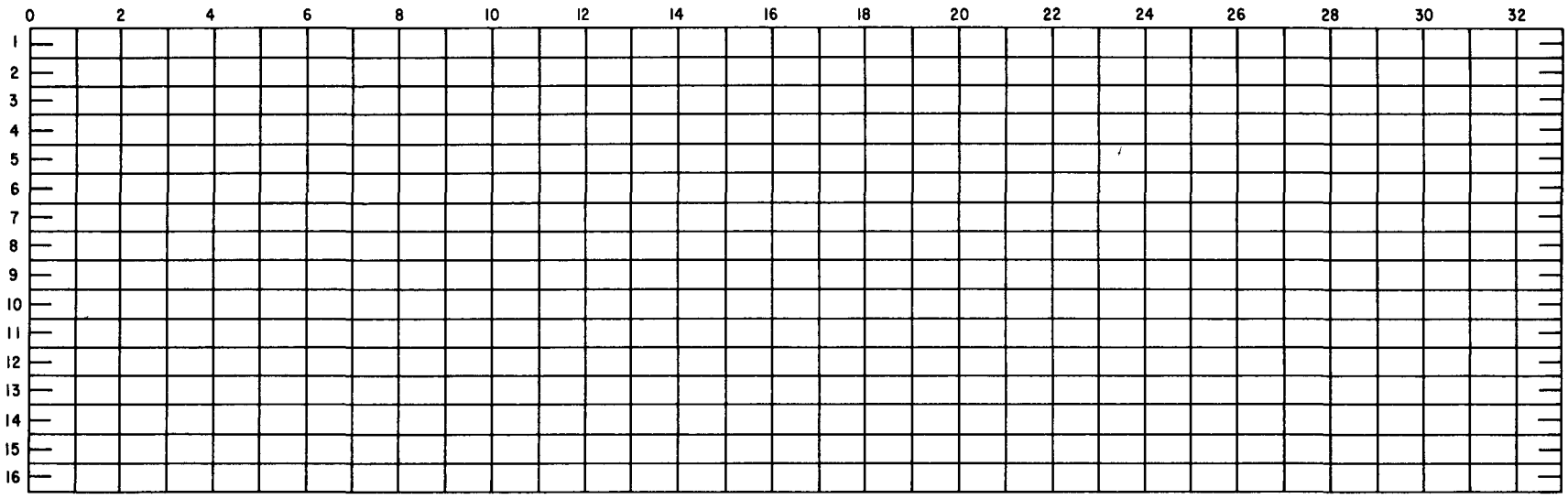
Fig. VIII-1

ZPR-6

DRAWER MASTER

Master No. _____ Drawer Type _____ Date _____

Assembly No. _____ Loading No. _____ Reactor Supervisor _____



INVENTORY

MT'L THICKNESS / COLOR	U^{235}			U^{238}																		Pu-Al ALLOY			Pu-Mo-U ALLOY		
	NO. PCS.	GROSS WT.	25 WT.	NO. PCS.	CORE WT.	25 WT.	BLNKT. WT.	NO. PCS.	CORE WT.	BLNKT. WT.	NO. PCS.	CORE WT.	BLNKT. WT.	NO. PCS.	CORE WT.	BLNKT. WT.	NO. PCS.	CORE WT.	BLNKT. WT.	NO. PCS.	GROSS WT.	239+241 WT.	NO. PCS.	GROSS WT.	239+241 WT.
1 x 1/2 x T																									
2 x 1/2 x T																									
1 x 1 x T																									
2 x 1 x T																									
2 x 2 x T																									
2 x 3 x T																									
TOTAL WTS.																									

TOTAL U^{235} IN CORE _____ kg

TOTAL $Pu^{239} + Pu^{241}$ IN CORE _____ kg

Fig. VIII-2

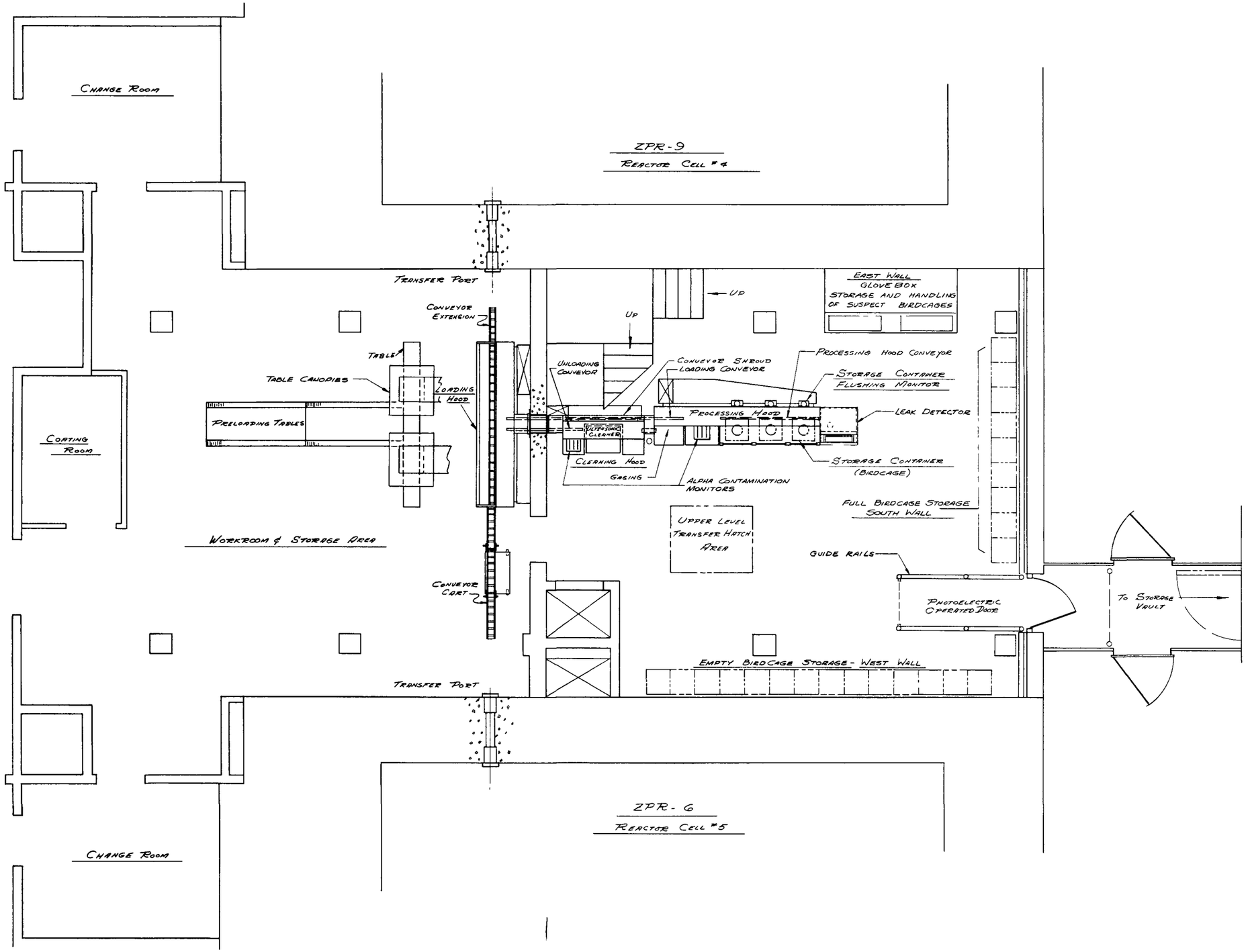


Fig. VIII-3
Vault Workroom Layout

2. As certain fuel plates are needed in the drawer loadings, the birdcage containing that size will be drawn from the storage stack and moved into the processing hood.

3. Each birdcage entering the processing hood will be first connected to the flushing and monitoring system before being opened to the atmosphere. This is accomplished by making connections to special fittings. The air within the birdcage will be purged through a monitoring station where alpha activity will be monitored. If the birdcage shows no contamination it will be opened and its contents will be placed on the process hood conveyor as needed in the loading operations. Empty birdcages will be identified as such and promptly moved to the storage area along the west wall. Birdcages indicating a suspect condition, either visually or through the flushing monitoring system, will be resealed and moved into a glovebox which is located against the east wall. There it will be opened for a complete examination and further monitoring of individual fuel plates. Ruptured or suspect fuel plates will be bagged out of the glovebox in sealed polyethylene bags and forwarded to Reclamation or the Materials Science Division for disposition.

4. Fuel plates will be transported to the end of the processing hood conveyor where they are removed for dimensional checking using go-no-go gauges. If a plate is found to be oversize or otherwise deformed, or if it appears to be damaged by mishandling, it will be surveyed and if clean placed in the helium leak detector to determine if the integrity of the jacket has been violated. (The final weld which closes the stainless steel jacket is done in a chamber filled with helium-argon gas under about 1/2 atmosphere of negative pressure therefore, helium can be detected if a plate jacket is cracked, punctured or otherwise damaged to the point of leaking.) All acceptable plates will be placed on the loading conveyor for transport to the loading hood. Contaminated or leaking plates will be sealed in polyethylene bags. Damaged plates and plates which fail the go-no-go gauge test will be held separately from the balance of the plates and repaired if possible. The accountability and handling of all plates not accepted for drawer loadings will be controlled by the Special Materials Representative.

5. Acceptable fuel plates will be removed from the loading conveyor at the delivery point in the loading hood. In this hood, the fuel plates are loaded into drawers which have been preloaded with nonfissionable materials on the preloading tables adjacent to the loading hood.

6. The loaded drawers are pushed along a roller fitted conveyor path to the transfer ports through which they are delivered into the reactor cell. Final inspection of the loaded drawer will be made while the drawer is on the conveyor extension for ZPR-9 and on the conveyor cart for ZPR-6.

Each drawer is numbered to correspond with the number of a matrix tube to aid in the correct loading of the assembly. The Reactor Loading Record (Fig. VIII-1), indicates the proper loading for each tube.

During and after loading, a current Master Loading Chart, showing a cross section of each half of the assembly, is kept in the control room. It shows the complete loading of the assembly at all times. The Assembly Coordinator is responsible for seeing that all changes in assembly loading are shown on these charts immediately after such changes are made.

The blanket may be loaded directly into the reactor in the reactor cell since this material is stored there. The large blanket pieces can be loaded directly into the matrix tubes, the drawers being omitted in this section, if infrequent changes are anticipated.

Unloading operations will be, in general, the reverse of loading operations, the SM representative checking all fuel plates back into the vault as the drawers are unloaded. Each drawer, however, need not be checked against its Drawer Master. No more than 6 kg of ^{235}U or 4.6 kg of contained plutonium will be unloaded from the drawers at any given time. The following steps are illustrations of the procedures used for the unloading of fuel.

1. Matrix drawers withdrawn from the reactor will be delivered through the transfer ports to the loading hood via the roller conveyor.

2. Fuel plates will be removed from each matrix drawer, checked for contamination, and placed on the unloading conveyor for transport to the cleaning hood.

3. Contaminated and/or dirty plates will be loaded into a basket, which is designed to hold a limited number in a fixed configuration, then placed in an ultrasonic cleaner.

4. Cleaned plates will then be transferred into the processing hood where they will be gauged again for size conformance. Leak testing will be repeated on any plate which has changed in size during its time in the reactor. All clean, leak tight and properly sized fuel plates will then be placed on the processing hood conveyor and returned to a position where they may be reloaded into birdcages.

5. The empty birdcages will be brought to the processing hood as needed and reloaded with the fuel plates. Each birdcage will be marked in such a way so that only plates of a particular size and of a particular alloy may be loaded into it. Such loadings will be supervised by the Special Materials Representative.

6. Loaded birdcages will be returned to the vault for storage.

B. Reactor Startup

A number of interlock requirements must be met before the reactor can be operated. Certain requirements should be satisfied before

others. The following is an acceptable ordering of administrative and electromechanical-interlock requirement satisfaction, and control manipulation to operate the reactor.

Administrative Requirements

1. The Operators will perform the daily reactor-maintenance check and submit the checkout form to the Reactor Manager or his alternate to obtain approval for the day's operation.

2. The Reactor Supervisor will inspect the log book for any unusual conditions prevailing from previous runs and begin entries in the log book.

3. Visually check the reactor cell to make sure that no personnel are in the cell and for obvious hazards.

4. The Reactor Supervisor will select from one to four fueled rods and up to two ^{10}B rods in each half as control rods. (See Chapter VII for rod worth requirements.)

Interlock Requirements

1. Reset instrument level and period trips, and if necessary, the high voltage interlocks on the chamber power supplies.

2. Drawer transfer ports in both the cell and the vault workroom must be closed.

3. All personnel and freight doors into the cell must be closed.

4. Air conditioning must be in "closed" position, in order to isolate the cell air from the building air in case of an accident, and turned on.

5. Emergency table drive air pressure must exceed 400 psig and the movable table must be in its farthest out position.

6. Boron safety rods must either be installed or bypassed by special dummy jacks.

7. All of the dual-purpose rods must be connected and either in the "control" or "safety" position (not in the "off" position).

8. Argon gas pressure must be greater than a preset value.

Startup Procedure

1. Turn on the Main Power ("Supervisory") Key to provide the electric power for the control chain.

2. Insert the startup sources into each half of the reactor and reset the low-level trip interlocks on channels 1 and 2.

3. Depressing the control power "on" switch will acquire control power.

4. If poison-type safety blades are used, they are latched and driven to their full "out" position. Dual-purpose safety rods are latched and driven to their full "in" position. (See Chapter VII--for rod worth requirements.)

5. Turning the Table Key on, and depressing the 25 cm/min "in" button will provide power to drive the table from its full "out" position (152 cm) to the first stop at 45 cm at a rate of 25 cm/min.

6. When the table stops at the 45 cm position, the block can be raised and the table driven to the second stop at 7.5 cm at a rate of 5.1 cm/min.

7. At the 7.5 cm position, the second block can be raised and the table driven to its full closure position at a rate of 0.42 cm/min.

8. With the tables together, the control rods can be driven in, one at a time, to achieve criticality.

9. Either shortly before, during, or after criticality is achieved, the startup sources are removed. The exact time to do this is largely a matter of experience.

10. Any of the following occurrences will cause the reactor to scram:

    a. removal of the Table Key;
    b. loss of console electrical power;
    c. initiation of instrument trip circuits caused by:
        (1) too high a flux level (channels 4, 5, and 6);
        (2) a period faster than a preset value of about 10 sec (channels 7 and 8); on an adjustable period trip circuit.
        (3) a period shorter than a fixed 5 sec (channels 7 and 8);
        (4) loss of voltage on any ionization chamber;
    d. loss of voltage for carriage drive clutches;
    e. opening of any of the five cell doors;
    f. opening of any of the six drawer ports;
    g. opening of any of the four butterfly valves associated with the air handling system;
    h. safety rods not connected and/or suitably bypassed if they are not being used;
    i. operation of one of the manual scram buttons;
    j. low steam pressure;
    k. low argon gas pressure.

C. First Approach to Critical

During an approach to criticality with a new composition or geometry, the first loading--for ^{235}U fueled cores--has one-third to one-half of the calculated critical mass. The lower value is used for those cores which

are substantially different from cores previously studied. The loading of the facility is always started by filling the dual-purpose control/safety rod drawers with the proper amount of fuel and diluent material first.

After the assembly has its proper initial number of drawers filled with fuel, blanket, or structural material, multiplication measurements are started. Multiplication measurements are made after each step in loading, with the control rods both in their in and out positions.

After each multiplication measurement, a new estimate of the critical mass will be made graphically. A reasonable amount of fuel, taking into consideration the estimated worth of the control and safety rods, is next inserted into the reactor. This amount is approximately one-third to one-half of the difference between the new estimate of the critical mass and the present fuel loading for ^{235}U fueled cores during the early stages of the multiplication. When the reactor is close to critical an amount of fuel is introduced into the core such that the total fuel in the reactor is a little greater than the estimated critical mass with the control rods in their most reactive position so that the reactor will go critical using the control rods.

The increment of fuel added to the reactor during all stages of the approach to critical will be such that the total amount of fuel in the reactor will be less than the estimated critical mass with the control rods in their least reactive position.

For Pu fueled systems the initial loading will not exceed one-third of the calculated critical mass. Subsequent loadings prior to achieving an indicated multiplication of ten will not exceed one-half of the difference between the estimated critical mass and the mass of the existing fuel loading. Loadings subsequent to achieving a multiplication of ten will be no greater than one-half the difference between the present fuel loading and the estimated critical mass with all control rods in their least reactive position.

D. Reactor Shutdown

The reactor will be shut down by pushing the manual scram button, or as a check of the interlock circuits, by reproducing any of the occurrences in step 10 under Reactor Startup. The dual-purpose rods are usually driven out of the core and the boron rods into the core before the scram.

The instrumentation will be observed so that all of the dual-purpose rods and the blade insertion rods are known to be in their least reactive positions following scram operation. Also, the separation of the two halves will be followed with the position indicators and/or on the closed circuit TV. (Both facilities have a red monitoring light to indicate everything is in its least reactive state.)

The recorders will be turned off, the keys removed from the console, and returned to the Reactor Manager and the Assembly Coordinator or put into the key cabinet through the slot provided.

The necessary entries will be checked and completed in the log book.

E. Limitations on Power Level

As with the ^{235}U cores, the average power level and duration of a run will be determined by the experiment in progress. The integrated power is normally limited to a few watt hours per day in ^{235}U systems for practical reasons. The SAR (Ref. 1) specified a maximum average power level of 500 W and estimated that the total integrated power would not exceed 2 kW hr per year. The resulting Pu buildup under these conditions is negligible.

The integrated power for any given run will be kept as low as is consistent with data acquisition and fuel handling requirements, and accessibility to the reactor core following the run. The average fission fuel power density shall not exceed 50 W/kg of fissile fuel for cores containing up to 200 kg of fissile fuel and 10 W/kg of fissile fuel for cores containing more than 200 kg of fissile fuel. The total integrated power shall not exceed 500 kW hr during any single extended run. In addition the maximum operating matrix temperature will be limited to 95°C.

F. Time-of-Flight Spectra Measurements

Neutron spectrum measurements using the 100 m time-of-flight tube will be made only when the reactor is known to be subcritical with a k_{eff} no greater than 0.98 with all rods in their most reactive position. The following procedure will be followed for time-of-flight measurements.

(1) When time-of-flight spectra measurements are not being made the flight tube and charged particle beam tube penetrations through the cell walls will be filled with shielding plugs and covered by flange plates. The cell containment will be maintained by these flange plates backed up by valves located on the outside of the cell.

(2) Before the shield plugs and flange plates are removed, the reactor will be loaded to critical and the core "confirmed."

(3) Using the measured rod calibration and edge worth of fuel and core materials as a basis for an estimate, sufficient fuel and core materials will be removed from the core so that the reactor's k_{eff} is no greater than 0.98 with the control rods in their most reactive position.

(4) Following the removal of fuel and/or core materials, the reactor will be assembled in the usual manner and confirmed that it is subcritical with all of the rods in their most reactive position.

(5) The inner steel valves for the two penetrations will be put in place.

(6) The flight tube and charged particle beam tube will then be evacuated and tested for leaks.

(7) If these systems are leak tight then the system is considered ready for time-of-flight spectra measurements.

(8) The reactor will then be reassembled and pulsed using an auxiliary neutron source.

(9) Following the completion of the time-of-flight measurements, the inner valves will be removed, the concrete shield plugs will be inserted into the hole in the concrete wall, and flange plates put back in place.

G. Procedures

In Chapters VII, VIII and IX, a proposed set of procedures and rules for startup, operation, shutdown, and maintenance of the ZPR-6 and -9 facilities have been outlined. These rules and procedures conform to those established and in current use at the critical facilities on the Illinois Site. They have been based on past experience with fast critical experiments and are anticipated to fulfill administrative requirements for the operation of these facilities. They should, however, be considered illustrative of the principles used for the safe operation of these facilities rather than firm commitments. It should be recognized that these procedures and rules may change with time in order to facilitate the operation of these facilities for reactor physics experiments. When modifications or additions to these rules or procedures are proposed, they will be incorporated into these procedures only after careful review and approval by appropriate Laboratory management.

Chapter IX

TEST AND MAINTENANCE PROCEDURES

A varied testing program has been developed to insure that all of the components of the ZPR-6 and -9 facility are in good working order. Depending on the particular component, it may be tested or checked on a daily, weekly, monthly, semiannually or annual basis. The areas which are checked may be divided into three major areas: Reactor, Cell, and Confinement Shell. It is proposed in the following sections to describe the tests and frequency of tests of the various components of the entire facility.

A. Reactor Facility

1. Nuclear Instrumentation

As a part of the daily checkout of the reactor all of the nine channels of operating instruments are checked. This is accomplished by placing a neutron source near the detector and observing the response of the amplifier. The operation of the trip circuits associated with these channels is also observed. Instruments and/or trip circuits which have been repaired are checked out in detail with a current source, period generator or appropriate test equipment.

The nuclear instruments calibration is checked in a detailed manner on a semiannual basis.

2. Electrical Circuits

Once each year the entire control circuitry is inspected in an attempt to locate broken or burned wires, faulty or deteriorating relays or switches. All of the contacts of relays are inspected during this annual inspection to insure that there are no welded, pitted or burned contacts.

The operation of the emergency power source (steam turbine generator) will be checked quarterly under full load conditions to insure that the power source comes on to line within the specified 20 sec interval.

Scram operation (loss of control power) by the operation of scram buttons, opening of doors, drawer ports is checked on a monthly basis.

The fire alarm system associated with the reactor cells and control rooms is checked on a monthly basis.

3. Mechanical Components

The dual purpose rod drives and safety blade drives are inspected once a month. The rod drive speeds and the scram times are

also checked on a monthly basis. The drive speeds and scram times are compared with average values obtained over a period of time.

The movable table drive speeds, both closure and scram operation are checked on a monthly basis. The table speed must conform to established values in the proper range. The motor current is also checked to see that there is no overload on the system.

The argon purge system pressure is visually inspected on a daily basis to insure that an adequate capacity is available.

The argon purge system is checked semiannually by opening the valves for a short period of time and allowing some of the stored gas into the cells.

The bottled nitrogen supply pressure for the scram air motor drive is checked daily to insure that an adequate supply of air is available for scram operation without power.

The loading platform drive mechanism and neutron source drive mechanisms are inspected on a monthly basis and serviced as needed.

The alignment of the rod drives, matrix assembly are checked on a semiannual basis.

B. Reactor Cell

1. Cell Leakage Test Procedure

a. Acceptable Leakage Criterion

The leakage of air from the reactor cells is considered acceptable if the leakage rate is less than the equivalent of 1% per hour of the contained gases as measured over a nominal 4-hr period at 10 psig.

b. Test Procedure

(1) Install Test Equipment

A pressure detector and 24-hr chart recorder are connected to the cell for continuous pressure monitoring.

(2) Establish Cell Integrity

(a) Secure flange covering normal cell exhaust system.

(b) Close normal personnel entrance, personnel escape hatch, freight access door, and drawer ports.

(3) Pressurize Cell to 10 ± 0.5 psig

The No. 1 and 2 compressed air supply system is used since it provides oil-free room temperature air and an easily regulated flow rate.

(4) Monitor Test Equipment

The pressure chart recorder and cell temperature are monitored to insure proper operation and to determine the variation of the leakage rate with time.

c. Analysis

The acceptance criterion is applied to the data in the following manner:

Ideal Gas Law: $P_1V_1/n_1T_1 = P_2V_2/n_2T_2$

where

P_1 = Initial Cell Pressure (Absolute)

T_1 = Initial Cell Temperature

n_1 = Initial Gas Mass in Cell

P_2 = Final Cell Pressure (Absolute)

T_1 = Final Cell Temperature

n_2 = Final Gas Mass in Cell

$V_1 = V_2$ = Cell Volume.

Therefore:

$$P_{Final} \geq [1.000 - 0.010\,x\,(hr)]\left(T_{Final}/T_{Initial}\right) P_{Initial}$$

must be satisfied so that $n_2/n_1 \geq [1.000 - 0.010\,x\,(hr)]$.

d. Testing Frequency

The reactor cells will be tested in accordance with this test procedure on an annual basis.

2. Normal Exhaust System

The valves and fans are usually operated on a daily basis but a detailed inspection of these components will be conducted on an annual basis.

The pressure drop across the HEPA filters will be checked on a monthly basis. Filter replacement will occur if the pressure drop across the filter exceeds normal operating values.

The efficiency of the HEPA filters will be checked on an annual basis using standard dioctylphthalate (DOP) tests.

3. Emergency Exhaust System

Leakage Tests

The initial testing of the leak tightness of the system will be conducted as follows:

a. Sand Filter Section

The sand filter section is defined as that part of the exhaust system extending from Cells 4 and 5 up to, but not including, the HEPA filters (see Fig. IX-1). The sand filter tanks have been tested under a pressure of 50 psig, and no pressure loss was detected for a holding time of 30 min. The entire sand filter sections will be leak tested according to the following procedure:

1. Blank-off flanges will be installed at points BP No. 1, 2, 3, 4, 5, and 6 as shown in Fig. IX-1. The check valves are propped open.

2. A pressure gauge will be installed at SP No. 1, or any other convenient SP location.

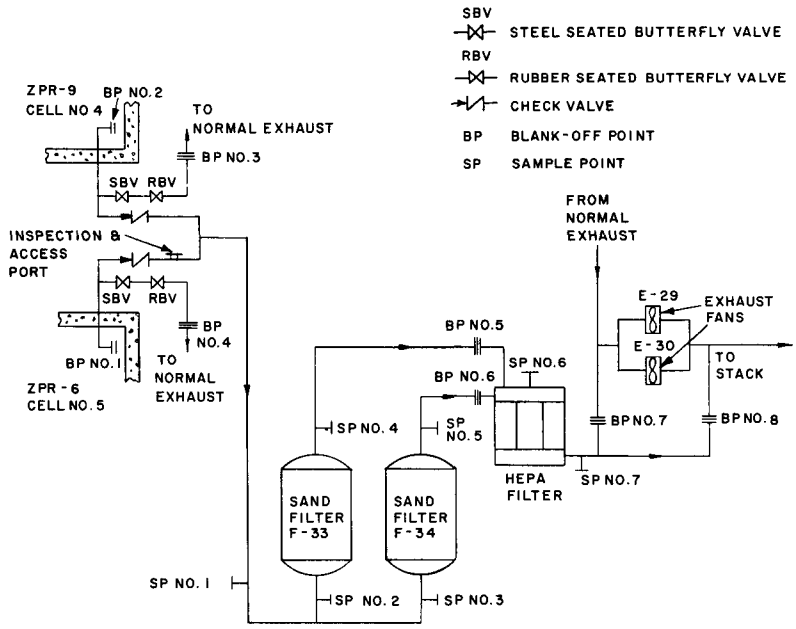


Fig. IX-1. Flow Diagram of Emergency Exhaust System

3. Air will be injected into the closed system through a convenient SP point by compressor.

4. The system will be pressurized to the desired test pressure.

5. Upon reaching the test pressure, the compressors will be shut off. Pressure decay will then be observed for a reasonable period of time.

The leakage rate can be obtained from the expression:

$$L = \Delta V/[\bar{P}_{(t)} \times \Delta t], \qquad (1)$$

where

$L \equiv$ leakage rate (ft^3/psig-min)

$\Delta t \equiv$ time interval (min)

$\Delta V \equiv$ change in volume of air (STP) in sand filter section during Δt

$\bar{P}_{(t)} \equiv$ average pressure (psig) in sand filter system during the the interval Δt.

The differential volume of air is computed from

$$\Delta V = V_i - V_f,$$

where subscript "i" refers to quantities at the beginning of the time interval Δt, and subscript "f" refers to quantities at the end of the time interval. V_i and V_f are given by:

$$V_i = V_0[(P_0 + P_i)/P_0]$$

$$V_f = V_0[(P_0 + P_f)/P_0]$$

where P_0 is atmospheric pressure, P_i and P_f are the measured gauge pressures at the beginning and end of the time interval, and V_0 is the volume of air in the sand filter section under no pressure (estimated to be 2000 $\pm$ 400 ft^3). The average pressure during Δt is given by $(P_i + P_f)/2$. Thus, the leakage rate is given by:

$$L(ft^3/\text{psig-min}) = 2V_i[1 - (P_0 + P_f)/(P_0 + P_i)]/(P_i + P_f)\,\Delta t. \qquad (2)$$

The sand filter section will be pressurized to initial pressure of 10, 20, and 30 psig. The pressure decay will be observed during a period of at least one hour for each of the three initial pressures.

The volume, V_i, will be checked by observing the pumping rate of the compressor during the pressurization of the sand filter section and the time required to fill the section.

A leakage rate less than 1.2 (ft^3/psig-min) is considered acceptable.

b. HEPA Filter Section

The HEPA filter section is defined as that part of the exhaust system included between the pairs of blank-off flanges BP No. 5, 6, 7, and 8 (see Fig. IX-1). This section will be leak tested according to the following procedure:

1. Blank-off flanges will be installed at points BP No. 5, 6, 7, and 8.

2. A water manometer will be installed at SP No. 6.

3. Gas will be injected into the closed system through SP No. 7. The gas will be leaked from gas tanks at a rate required to maintain a constant pressure (approximately 2 in. of water) in the system.

4. The time required to empty the gas tanks will be observed.

The leakage rate is given by the volume of gas leaked into the system divided by the time to empty the tanks and the pressure maintained in the system. A leakage rate of about 100 ft^3/min-in. of water is considered acceptable.

c. Filter Efficiency Tests

(1) Sand Filters

The efficiency of the sand filters will be tested with the following procedure:

a. Install blank-off plates BP No. 2, 3, 4, 5, and 7 (see Fig. IX-1). The air flow path is now restricted to sand filter F-34.

b. A flow rate of 10-15 fpm through the sand filter shall be established with the reactor cooling fan in Cell No. 5. The cooling fan will be connected to the emergency exhaust system at BP No. 1.

c. DOP compressed air generated aerosol will be introduced upstream of the sand filter (in Cell No. 5 near BP No. 1 location).

d. Upstream air samples will be taken at location SP No. 1, 2, or 3 and downstream samples will be taken at location SP No. 5.

A forward light scattering aerosol photometer will be used to obtain an on-line analysis of these samples. [Minimum sensitivity is 0.002% penetration (5×10^4 attenuation).]

e. The penetration will be calculated by:

$$\frac{\text{Downstream \% aerosol reading} \times 100}{\text{Upstream \% aerosol reading}} = \text{\% penetration.}$$

A minimum attenuation of 1×10^3 or 0.10% penetration is considered acceptable. The above procedure is repeated for sand filter F-33 with blank-off plate BP No. 6 in place of BP No. 5 in Step a and SP No. 2 and 4 replacing SP No. 3 and 5 in Step d.

(2) HEPA Filters

The efficiency of the HEPA filters in the Emergency Exhaust System, Confinement Shell Exhaust, Cell, Vault, Workroom, Control Room, and Basement Exhausts will be tested on an annual basis. The procedure to be used, shown below, for the Emergency Exhaust HEPA filters is illustrative of the procedure to be followed for the efficiency test of the filters.

a. Install blank-off plates BP No. 5, 6, and 8.

b. The confinement shell exhaust system will be shut off and the air flow into the system will be produced by Fans E-29 and E-30.

c. A flow rate of about 20% of the rated filter flow, or about 2400 cfm, will be established. A full flow orifice pressure drop measurement and velometer traverse will be used for this measurement.

d. DOP compressed air generated aerosol will be introduced into the system at location SP No. 6.

e. Upstream air samples will be taken at location SP No. 6 and downstream samples will be taken at location SP No. 7. Samples will be analyzed as described in Section 1.

A minimum attenuation of 1×10^4 or 0.01% penetration is considered acceptable.

C. Confinement Shell Leakage Rate Test Procedure

The leakage rate of air into the confinement shell will be tested on an annual basis as follows:

1. The inlet makeup air to the confinement shell will be temporarily closed off. This may be accomplished with a sheet metal plate and sealing

tape, locating the plate at the duct inlet to the HEPA filter for the confinement shell inlet. These filters are located in the southeast corner of the Mechanical Equipment Room No. 2 above the vault.

2. The following measurement of leakage flow will be made. A calibrated intake orifice assembly with a 4-fit duct extension will be attached to the inlet side of the HEPA filter. (A flange is provided for such alterations.) Air flow through the calibrated orifice will be measured at -2.5 in. w.g. An additional flow measurement, using a calibrated rotating vane-type anemometer, will be made to check the results. Once experience is gained in making these flow measurements, only the more satisfactory of these two measurements will be used.

3. All doors leading to the confinement shell will be locked.

4. Fans E-21 and/or E-22 (confinement shell exhaust fans) will be operated to lower the pressure within the shell to -2.5 in. w.g. The sensing connections for the static pressure regulator for Fans E-21 and E-22 must be connected in their normal positions and the set point adjusted to -2.5 in. w.g. The time it takes to achieve this pressure will be noted.

5. The air flow will be measured and if less than 1000 cfm at -2.5 in. w.g., the leak tightness of the shell will be considered adequate. If this test fails, a checkout of the confinement system leakage will be conducted as described on the following page.

Checkout of Confinement System Leakage

In the event that the flow tests show a flow rate greater than desired, a systematic search for leaks will be conducted by the negative test method described as follows:

1. As for the leakage rate test, the inlet makeup air to the confinement shell will be closed off.

2. Exhaust Fans E-21 and E-22 will be operated to produce a negative pressure of -4.0 ± 0.25 in. w.g. To increase the capacity of the above fans during test conditions, the filter elements for the inlet HEPA filter may be removed. Also if more fan capacity is desired, Exhaust Fans E-21 and/or E-32 (vault exhaust fans) can be used. This requires the removal of filter elements from filter F-31 and closing off the exhaust duct coming from the new vault area. This duct closing may be done with a sheet metal plate and sealing tape on the inlet side of the fire damper. Access to the fire damper may be obtained through the access panel opening below the damper.

3. The negative test pressure in the confinement shell must be under control at all times. The inlet dampers for Fans E-21 and E-22 and the static pressure regulator connected to them may be used for this purpose. The pressure regulator must be reset to the desired test pressure (here -4.0 in. w.g.). If Fans E-31 and/or E-32 are used, their inlet dampers must be temporarily connected to the static pressure controller which normally serves only Fans E-21 and E-22.

4. All doors leading to the confinement shell will be locked except for the double doors leading from the filter room to the Mechanical Room 4-S.

5. The rushing of air in large leaks can be easily heard. Initial inspection of the containment shell will be for these large leaks. Smaller leaks can be found by soap bubble testing.

6. After the leaks are repaired, another leakage flow rate test will be made.

7. During a -4.0 in. w.g. test, a periodic inspection of filters in the Filter Room should be made, checking for failures in the tape membrane between filters and ducts.

Chapter X

EMERGENCY PROCEDURES

A. Alarms

The basic audible alarms in the Reactor Physics Laboratory are essentially the types listed in the Laboratory's Policy and Practice Guide, Health and Safety Chapter, Part II Safety Engineering, Section C, Appendix, May 1965 and are summarized in Table X-1.

In addition, there is a Klaxon type alarm (which sounds in the reactor cell and the control room) connected to the local radiation area monitors, a local high temperature detector which sounds a buzzer when sections of the annunciator panel are activated in the control room. The buzzer alarm is also actuated if the cell air pressure becomes excessive.

B. Responsibility

Any individual encountering an emergency situation shall initiate a "DIAL 13" on the telephone. The DIAL 13 automatically alerts all personnel normally involved in an emergency by a Group Alerting System. An emergency situation is one where there is a possibility of hazards to personnel or property such as a fire or radiation incident. He will provide to those personnel on the Group Alerting System all pertinent information as to the type of emergency and exact location. It will also be his responsibility to contact the Area Emergency Supervisor or one of his alternates and the building's IHS representative (Industrial Hygiene and Safety--Radiation Safety) if radiation is involved.

C. Objectives

The principal objectives of emergency action undertaken after an accident are to:

(1) Insure safety of all personnel, and if practical, the equipment and facilities.

(2) Prevent radioactive contamination of the local surrounding area.

The means of securing safety for personnel consists of accountability for and, if necessary, evacuation of all personnel.

D. Action

(1) Everyone hearing any alarm listed in Table X-1 shall take the action prescribed in the table.

Table X-1

AUDIBLE ALARMS

SIGNAL	CAUSE	ACTION	
		General Laboratory Personnel	Signal Response Personnel Who Have Duties and Instructions in Connection with Given Signals
Intermittent resonating horn (at least 108 db at a distance of 10 ft. on horn axis)	Accidental nuclear reaction.	Evacuate as rapidly as possible.	Clear area. Follow radiation safety instructions.
Steady horn or klaxon.	High radiation; toxic or explosive gas release.	Evacuate area immediately.	Investigate cause immediately; correct if hazard is tolerable.
Steady tone over PA system and outdoor siren.	Alert. Emergency situation requiring PA announcement.	Move close to PA speakers to receive instructions. Evacuation of area or site may be necessary.	Emergency personnel carry out previously assigned instructions.
Warbling tone over PA system and outdoor siren.	Tornado or enemy attack imminent.	Take cover immediately in nearest shelter.	Emergency personnel carry out previously assigned instructions.
Bell, buzzer, or electronic device steady or intermittent.	Malfunction, leak, high or low pressure fire, timing device, break period, etc.	Follow instructions of responsible authorities in area.	Investigate cause; take appropriate action.

Table X-1 (Contd.)

SIGNAL	CAUSE	ACTION	
		General Laboratory Personnel	Signal Response Personnel Who Have Duties and Instructions in Connection with Given Signals
Repetitive gong.	Moving objects such as crane, indoor vehicles, heavy door, etc.	Be alert.	None.
Siren - on vehicle.	Emergency vehicle.	Be alert, yield right-of-way by stopping on road shoulder.	Emergency personnel carry out previously assigned instructions.
Siren - inside building.	Emergency which requires evacuation of building or area.	Evacuate building or area, follow instructions of monitors.	Emergency personnel carry out previously assigned instructions.
Coded electronic sounds over PA system or radio (used only during other than regular day shift such as paging tone).	Need to contact individual paged.	Individual paged: call predetermined phone number.	None.
Signal distinctly different from those above.	To be determined by originator but not to include causes listed above.	None.	To be determined by originator.

(2) If a reactor is in operation at the time of an evacuation alarm, the reactor shall immediately be "scrammed" and the area evacuated of personnel.

(3) An accident may be indicated by the fact that the reactor scrams, and/or by an audible alarm in the control room which is initiated by an increase in cell air pressure, cell air temperature (to 40°C), or reactor radiation level (2 R/hr).

Upon detecting an increase in pressure or high rate of temperature rise (>45°F/min) or high abnormal radiation levels in the cell, the automatic controls on the air conditioning control system will close or maintain closure of the inlet and exhaust valves. If the cell pressure exceeds a few inches of water, air will flow through the emergency venting system tending to decrease cell pressure.

If only the area radiation monitor alarm or the cell air pressure or temperature alarm is activated in one of the control rooms, a member of the operating crew shall:

(a) Determine the origin and cause of the alarm.

(b) Determine the radiation level in the control room by noting the appropriate indicators.

(c) If the radiation level is greater than 100 mR/hr in the control room, he shall promptly:

1. Scram the reactor and assure cell closure.
2. Evacuate control room area and if possible, announce the accident over the building public address system.
3. Dial 13 from outside the control room.
4. Notify Area Emergency Supervisor.
5. Go to Emergency Surveillance Station in Room N-121 to observe radiation levels in cell and control room and status of reactors. Take appropriate action with the air conditioning system and argon purge system.

(4) If the radiation levels are less than 100 mR/hr, he may remain in the control room and follow the course of action prescribed under the following section.

E. Action in Case of a Fire

The following procedures will be followed in the event of a fire regardless of where it occurs:

(1) Immediately leave the area where a fire is suspected.

(2) Scram the reactor if the fire is in the building.

(3) Dial 13 on the telephone and notify the Group Alerting System personnel of what has happened.

(4) Notify the Area Emergency Supervisor.

(5) Shut down the air conditioning system for that part of the building where the fire occurs. In the case of the reactor cells, the inlet and exhaust valves should be closed if not closed automatically.

(6) Small fires in the confinement shell, vault, or reactor cells, will be extinguished by the fire department personnel using the proper type of fire extinguishers. Fire department personnel will enter the cells, or vault areas only with protective clothing and special breathing apparatus under the authority and instructions of the Area Emergency Supervisor or Reactor Supervisor.

(7) Large metal fires in the cells will be extinguished with the introduction of argon gas into the cell. The cell will be exhausted through the emergency venting system.

F. Action in a Nuclear Accident Condition (see also material under D)

Upon indication of an accident in the reactor cell, the following course of action is to be followed, but not necessarily in the sequence given:

(1) Scram the reactor if it is operating.

(2) Insure cell closure by turning Cell Open/Close switch to Cell Close (located on air conditioning control panel).

(3) Dial 13, and give name, extension, and building location and as much detail as may be available.

(4) If inside cell during a loading accident, do not attempt to rectify the cause of the accident, but leave the cell immediately by either the personnel chute or freight door, whichever is faster.

(5) Determine severity of accident from indicators on air conditioning control panel and Channel 9 area monitors.

(6) Announce accident and any request for assistance over building public address system (microphone on west wall of control room for ZPR-9 and on the north wall for ZPR-6 by the entrance doors).

(7) If cell entry is necessary to remove personnel:

(a) Enlist aid of emergency personnel who are equipped with self-contained breathing apparatus and protective clothing.

(b) Secure radiation survey instruments (located by cell personnel door).

(c) Observe radiation levels in cell, emergency exhaust lines and status of reactor with control instruments.

(d) If radiation levels are at an acceptable level, the reactor is shut down, and there is no buildup in cell pressure; personnel may enter the cell for removal of injured persons.

(8) If there is an increase in the cell temperature and pressure indicating a probable metal fire introduce argon gas into the cell as follows:

(a) Open argon-inlet valve to Cell 5 for ZPR-6 or Cell 4 for ZPR-9 (control button next to air conditioning control panel).

(b) Open main argon-supply valve by manually operating the main valve. (Make sure the argon goes into the proper cell.)

(c) With the normal supply and exhaust valves closed, the cell air will flow out through the emergency venting system.

IMPORTANT: All personnel must be out of the cell before argon is injected into the cell.

G. Action of Emergency Personnel

A DIAL 13 automatically alerts all people normally involved in an emergency. The notification of an emergency may originate from a telephone, radio, messenger, or alarm source. When a DIAL 13 call is received at the switchboard, the Emergency Operator will activate a Group Alerting System, which simultaneously connects the following:

(1) Emergency Coordinator
(2) Alternate Emergency Coordinator
(3) Fire Protection Department
(4) Security Division's Central Station
(5) Industrial Hygiene and Safety Division's Radiation Safety Office
(6) Industrial Hygiene and Safety Division's Safety Engineering Office
(7) Plant Services Division's Administrative Office
(8) Public Information Office
(9) Communications and Special Services Department
(10) Meteorology

The Health Division is alerted only when the emergency report indicates injuries, illness, or criticality alarm.

The DIAL 13 caller will be questioned by the Fire Alarm Operator, who will use a prepared list of questions to get information necessary for emergency response. After these questions are answered, the caller may give other details and anyone on the conference circuit, after identifying himself, may ask the caller for additional necessary information.

Each DIAL 13 which is received is monitored by Communications and Special Services for evaluation and action as required.

During off-shift hours the DIAL 13 caller will be connected (by means of the Group Alerting System) to the Fire Alarm Operator, Security's Central Station, the Health Division, and Meteorology only.

Each of the units follows established emergency procedures as given by ANL Emergency Handbook to protect personnel and facilities and to bring the emergency situation under control. It is the responsibility of the Area Emergency Supervisor to notify the Laboratory Emergency Coordinator in the event of a major emergency such as a nuclear accident. The Emergency Coordinator or his designated alternates acts in accordance with established procedures as outlined in the ANL Emergency Handbook and has the primary responsibility for coordinating the actions of emergency personnel for controlling the situation.

Personnel in the Reactor Physics Laboratory (Building 316) which contains ZPR-6 and -9 are required to evacuate the building if an accidental or uncontrolled nuclear reaction occurs which activates the radiation area monitor (RAM-C) (indicated by an intermittent resonating horn--108 db at 10 ft on the horn axis). In addition, personnel may be instructed by the Area Emergency Supervisor or other appropriate personnel to evacuate the building by means of the building public address system in the event of a fire, or accidental release of radioactivity, toxic gases or chemicals. Personnel have been instructed to evacuate to the parking lot north of the building or upwind of the affected area or may be instructed to drive upwind and away from Building 316. There are wind socks around the building to indicate wind direction.

In the event that evacuation of neighboring facilities is necessary, this would be ordered by the Laboratory Emergency Coordinator using the Laboratory wide public address system or with the assistance of the plant protection forces and Area Emergency Supervisors. The suspect areas would be cordoned off and roadblocks established by the plant protection personnel.

Table III-1 gives a list of on-site facilities within 1000 meters of the ZPR-6 and -9 facilities with an estimate of the number of employees in each building during a normal working day. Figure III-2 shows a map of the Laboratory Site and the facilities located close to the Reactor Physics Laboratory (Building 316). It is estimated that Building 316 (Reactor Physics Laboratory) can be evacuated within one to two minutes after a nuclear accident. The facilities within 200 to 600 meters of Building 316 can be evacuated within 15 minutes. It is estimated that the entire Laboratory could be evacuated within 30 minutes.

Evacuation of areas outside of the Laboratory boundaries would be carried out by action of the Chicago Operations Office--AEC Security Division Director following notification by the Laboratory Emergency Coordinator.

Chapter XI

MAXIMUM CREDIBLE ACCIDENT

It is worthwhile at this point to briefly review the arguments presented in the original Safety Analysis Report[1] in discussing the maximum credible accident with ^{235}U-fueled assemblies.

It was explained then that three general lines of defense against accidents are provided. These are:

(1) The use of caution in preparing an assembly for investigation. The loading operations are broken into steps which are separately checked, so that a major loading error would require a significant error in the initial calculations, blind acceptance of the preloading evaluation by the Reactor Manager, and the Assembly Coordinator and the absence of suitable pre-critical multiplication checks. It is not believed that such a complete breakdown of administrative procedures is credible. It is equally unreasonable to assume that lesser errors in loading will not occur. While such minor errors in loading could increase the probability of an accident, they could not themselves cause an accident.

(2) The observance of caution in approaching criticality and in operating the reactor for experimental runs. It is considered incredible that the reactor would be operated in such a way that all established procedures are completely disregarded. Therefore, as is the case for the loading procedures, it is necessary to postulate a string of minor operational blunders rather than single operational violation which could lead to trouble.

(3) The use of trip and interlock circuits. Both the steps in bringing the assembly to criticality and operating it in a critical condition are limited by a string of interlocks and neutron-monitoring instrumentation with trip circuits. Monitoring instrumentation and trip circuits are checked daily to minimize undetected malfunctions. It is not considered credible that all, or even a majority, of the trip and interlock circuits of the system will be inoperable at any time; however, accidents must be considered that may arise from a partial failure of the circuits.

Under normal circumstances, each of these areas of malfunction represents a separate and independent contingency. Common safety practice considers that an accident which has double contingency is properly secured, and on this basis it is believed that the critical assembly and its operating procedures together constitute a safe system. The important considerations that went into the selection of the maximum credible accident in ANL-6271 were the following:

(1) The accident must occur under plausible experimental conditions. Any highly unusual procedure that would require a considerable engineering redesign effort was eliminated from consideration, and it was argued that any modification that leads to an accident must be one that can be accomplished with relative ease.

(2) There must be a logical reason why period and neutron level trip circuits do not function at the proper time. It was considered incredible that reactor operation would proceed with no instrumentation or trip circuits in operation.

(3) Under normal experimental conditions, there are only three basic ways by which reactivity addition can be accomplished mechanically:

(a) movement of the tables,

(b) insertion of the dual-purpose rods, and

(c) removal of the boron-10 safety rods.

An accident involving table movement was selected because this movement gives the maximum possible reactivity insertion rates.

This line of reasoning brought about the selection of an illustrative mechanism for the maximum credible accident. That case, one of a variety of possibilities that were thought through for the appropriate combination of malfunctions, was typical of the worst case that was found. The postulated course of the accident was as follows: It was assumed that an experiment to measure the effect of the gap between the halves for a 50-liter core was under way. It was then postulated that the following unusual circumstances occurred. First, the two period amplifiers with their associated trip circuits were bypassed, and the linear flux-level trip as well as both high level safety trips were set approximately two decades above their normal trip level settings, thereby, preventing tripping until the neutron density reached a much higher level than would be usual. In combination with a malfunction of the 7.6 cm limit switch, that would allow the forward intermediate speed drive to be actuated, the operator mistakenly drove the halves toward each other. As a result of this compounding of unusual events the moveable table moved toward the stationary one at a rate of 0.25 cm/sec from the 1-cm position, so that reactivity was added at a high rate, and a nuclear excursion followed. As the period trip circuits were bypassed the excursion continued, and because the level and safety trips were set about two decades above their usual operating points the excursion was not sensed until the reactor had become prompt critical. The almost vertically rising power following prompt critical then actuated the level trip circuit or the safety trip circuits and scram action was initiated. (Scram operation starts the removal of the dual-purpose safety rods and, initially, stops forward motion of the table. The table then starts moving backward a short time after the scram operation has started.) The excursion was

turned by the expansion of the enriched uranium plates and the power dropped half a decade or so below its peak value, whereupon the removal of the dual-purpose rods by the scram operation stopped any further increase of the power level.

The neutronics parameters were then chosen to give the largest energy-density release, and the result was considered to be the maximum credible accident. All of the main elements of this discussion are unaltered by the proposed conversion. The arguments hold with equal force for plutonium-fueled assemblies. The results of a similar accident analysis will be somewhat different in detail, because of the differences in fuel characteristics, reactivity addition rates, delayed neutron fraction, and improved knowledge of some of the parameters. One additional assumption must be made: Because the new power feedrail system will not allow table movement at speeds other than prescribed for any given table position, the postulated gap-worth measurements must be for gaps greater than 7.6 cm. It could be argued that the very particular set of circumstances postulated for this accident in themselves strain the bounds of credibility. However, the particular sequence considered above was meant only to be illustrative of the combination of malfunctions that would be necessary to give an accident of this kind.

If one chooses to disregard this particularized sequence of events, and to postulate other mechanisms leading to accidents, the end point is the same: namely, that the accident analyzed amounts to an overestimate of the maximum accident that is even remotely credible. The same initiating mechanism would, for example, be present if one chooses to consider credible, against the evidence of all experience with the assemblies of this general type, an accidental overloading sufficient to bring the reactor critical in the intermediate speed range. To reach criticality with the halves separated by 7.6 cm, the minimum separation for which the intermediate speed drive can be operable, an overloading of approximately 80% (~60 kg) in critical mass would be necessary for the 50-liter uniformly loaded cores, and an overloading of approximately 30% (~400 kg) would be necessary for the 3500-liter uniformly loaded cores. Accidental overloads of this magnitude would grossly violate all relevant operating procedures. Even then the postulation of an accidental malfunction of each successive trip soon puts the accident in the incredible class. Proper operation of any one of the five trips would lead essentially to no excursion at all.

The above is therefore considered to be the maximum credible accident, maximum in the sense that the bounds of credibility of initiating mechanism are at their maximum and maximum also in the sense that the neutronics parameters have been chosen so that the excursion results in the maximum possible energy release.

Because a 50-liter ^{235}U-fueled core was analyzed for its maximum credible accident in the original Safety Analysis Report, a 50-liter plutonium

fueled core will also be analyzed in this report so that a direct comparison can be made for the two cases, with only the fuel changed. However, such a core has little practical interest. Much more typical of the cores of interest to the program are the larger more dilute assemblies, so a 3500-liter core was chosen as the reference case. In it, the maximum quantities of (about 1400 kg of $^{239}Pu + {}^{241}Pu$) of plutonium are present.

The excursion calculations used the R-101 kinetics code[2] which solves the usual one-energy group space-independent reactor kinetics equations:

$$\dot{n} = \frac{k_{ex} n}{\ell} - \sum_{i}^{N} \dot{c}_i + S$$

$$\dot{c} = -\lambda_i c_i + \frac{\beta_i}{\ell}[1 + k_{ex}]\, n$$

where

n = neutron density

c_i = precursor density

S = source

β_i = effective delayed neutron fraction of the delayed group i

λ_i = decay constant of delayed group i

k_{ex} = excess reactivity

ℓ = prompt neutron lifetime.

In the option used for these calculations the excess reactivity is given as:

$$k_{ex} = At + C \ln [1 + D \int (n - n_0)\, dt] + E \int n\, dt$$

where

A = table reactivity addition or subtraction rate

C = Doppler coefficient

D = temperature to $\int n\, dt$ conversion constant

E = expansion coefficient.

The term involving the coefficients C and D in the above expression allows the Doppler effect to be put in with a $1/T$ dependence, shown by experiment to represent the actual Doppler coefficient more closely than the assumption of a simple linear dependence with temperature as assumed in ANL-6271.

The core parameters, the gap worths, and the relevant feedback coefficients were based on the values listed in Chapter II. The actual values of the constants used in the calculations are listed in Table XI-1. The constant A includes the reactivity addition rate when the movable table is adding reactivity and the reactivity subtraction rate due to rod movement subsequent to the scram. The two components of A are included at appropriate times during the calculation.

Table XI-1

CONSTANTS FOR EXCURSION CALCULATIONS

	Fuel Plates Only Heated		All Materials Heated	
System	3500 ℓ	50 ℓ	3500 ℓ	50 ℓ
A_T	7.9×10^{-4}	2.2×10^{-3}	7.9×10^{-4}	2.2×10^{-3}
A_R	-0.218	-0.158	-0.218	-0.158
C	-2.5×10^{-3}	-1.4×10^{-4}	-8.6×10^{-3}	-2.1×10^{-4}
D	4.11×10^{-7}	4.53×10^{-7}	4.38×10^{-8}	1.74×10^{-7}
E	-4.80×10^{-10}	-1.01×10^{-9}	-5.12×10^{-11}	-3.86×10^{-10}

$A \equiv A_T + A_R$.
$A_R \equiv$ Reactivity Subtraction Rate due to Rod Motion in $\Delta k/k\ sec^{-1}$.
$A_T \equiv$ Reactivity Addition Rate due to Table Motion in $\Delta k/k\ sec^{-1}$.
$C \equiv$ Doppler Coefficient $= \frac{\Delta k/k}{\int dT/T}$.
$D \equiv$ Conversion Factor Is $cm^3(neutron\ sec)^{-1}$.
$E \equiv$ Expansion Coefficient Is $\Delta k/k\ cm^3(neutron\ sec)^{-1}$.

Calculations were done for two separate cases for each of the two core types considered: the first assumed that only the plutonium-uranium-molybdenum fuel plates were heated by the excursion, and the second assumed that all materials were heated uniformly. The actual case will lie somewhere between these two extremes.

As the initial neutron density in an excursion will depend on whether it is assumed the excursion starts with the reactor at operating powers or during a startup, the effect of this on the resulting excursion has been examined for initial neutron densities of 10^{-4} n/cm^3, 10^{-2} n/cm^3, and 1 n/cm^3, where a neutron density of 1 n/cm^3 corresponds to 100 W for the 3500-liter core and to 7 W for the 50-liter core. The resulting excursion proved to be largely insensitive to the value chosen for this parameter. However, the excursions from an initial neutron density of 10^{-2} n/cm^3 appeared to be slightly worse than the other two. The excursions shown in Figs. XI-1 to XI-4 are therefore shown starting from this neutron density. In these

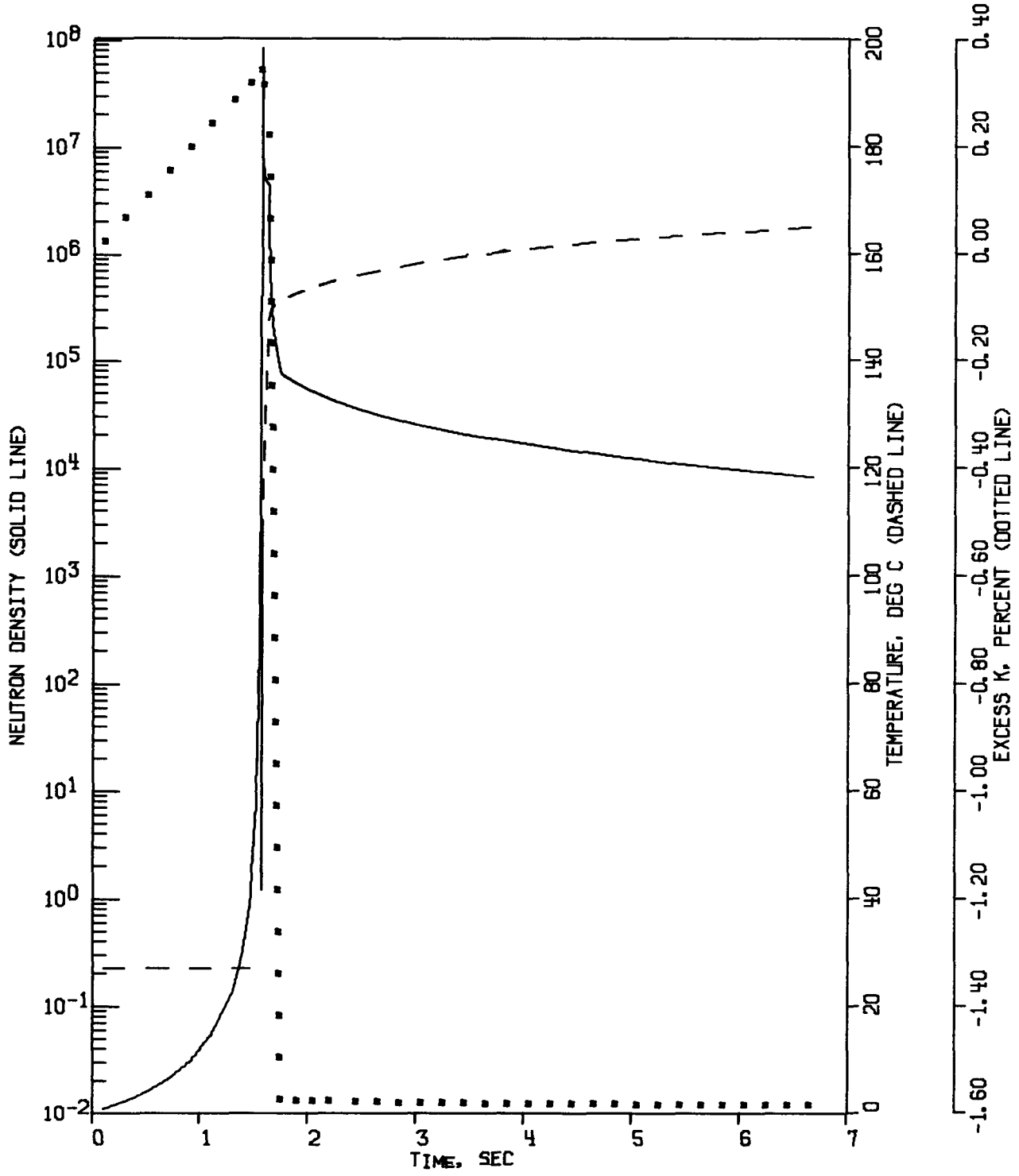


Fig. XI-1. Maximum Credible Accident Analysis, 50 liter Core, Fuel Plates Only Heated, Reactivity Insertion Rate 0.65 $/sec

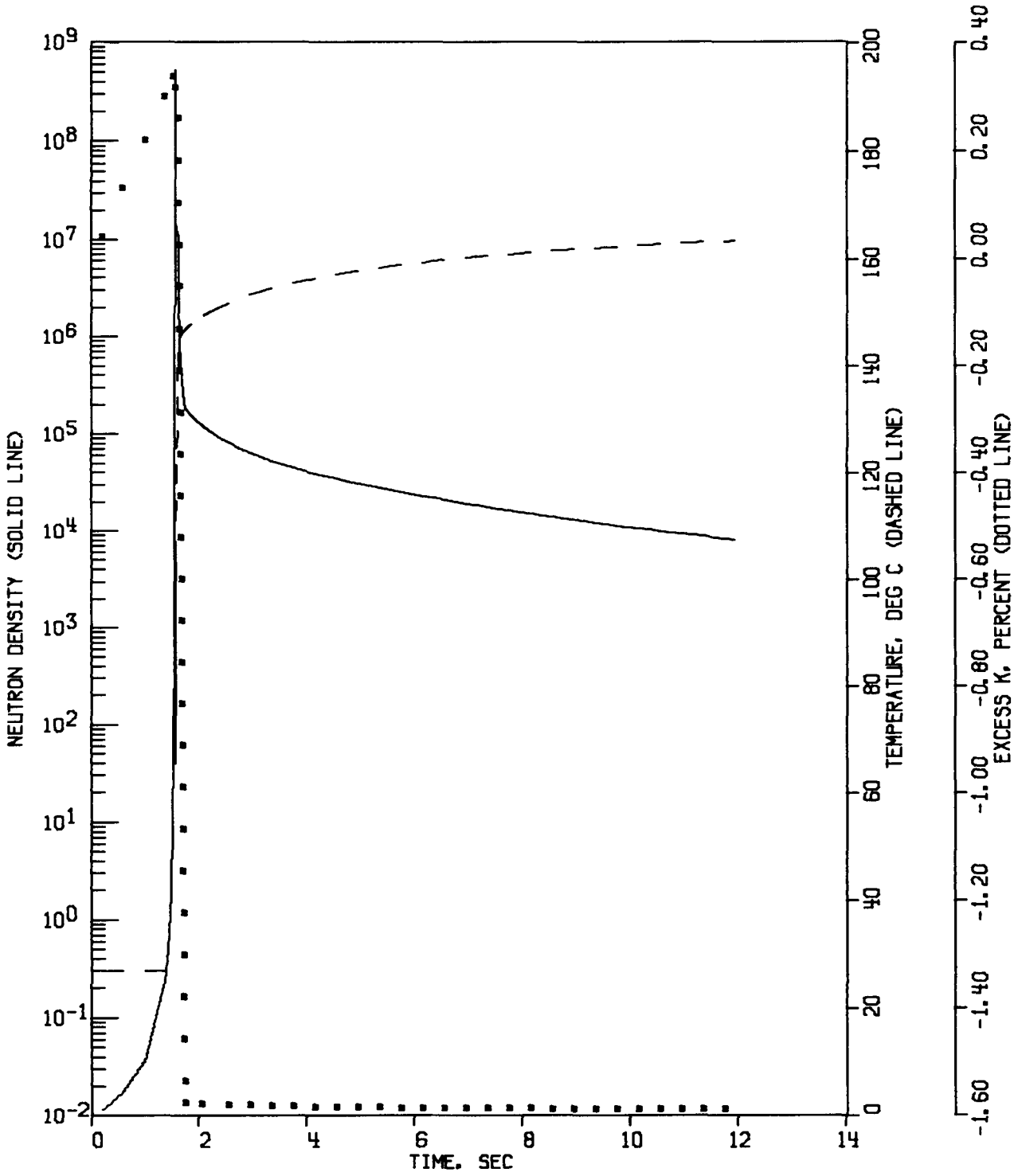


Fig. XI-2. Maximum Credible Accident Analysis, 50 liter Core, All Materials Heated, Reactivity Insertion Rate 0.65 $/sec

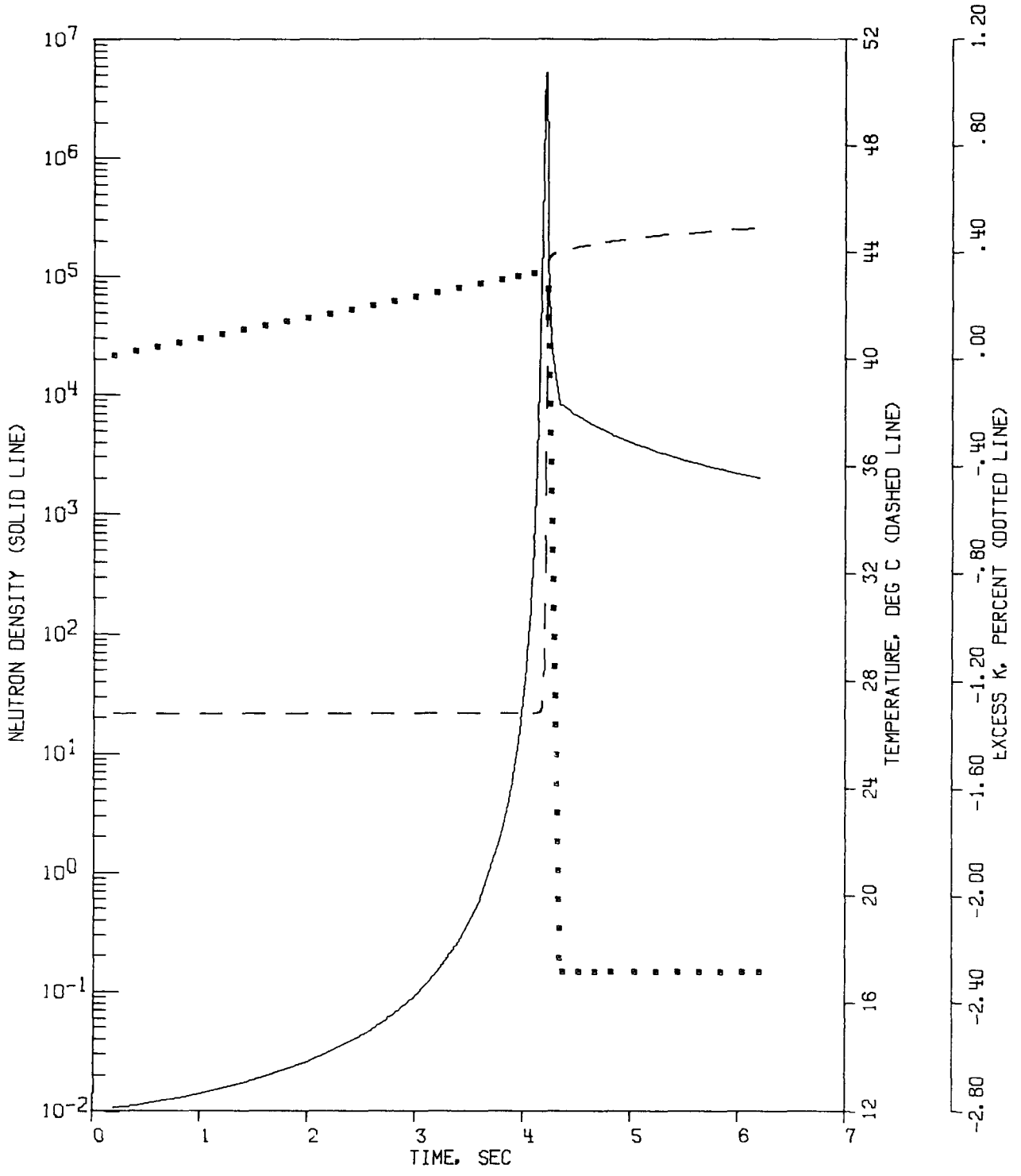


Fig. XI-3. Maximum Credible Accident Analysis, 3500 liter Core, Fuel Plates Only Heated, Reactivity Insertion Rate 0.24 $/sec

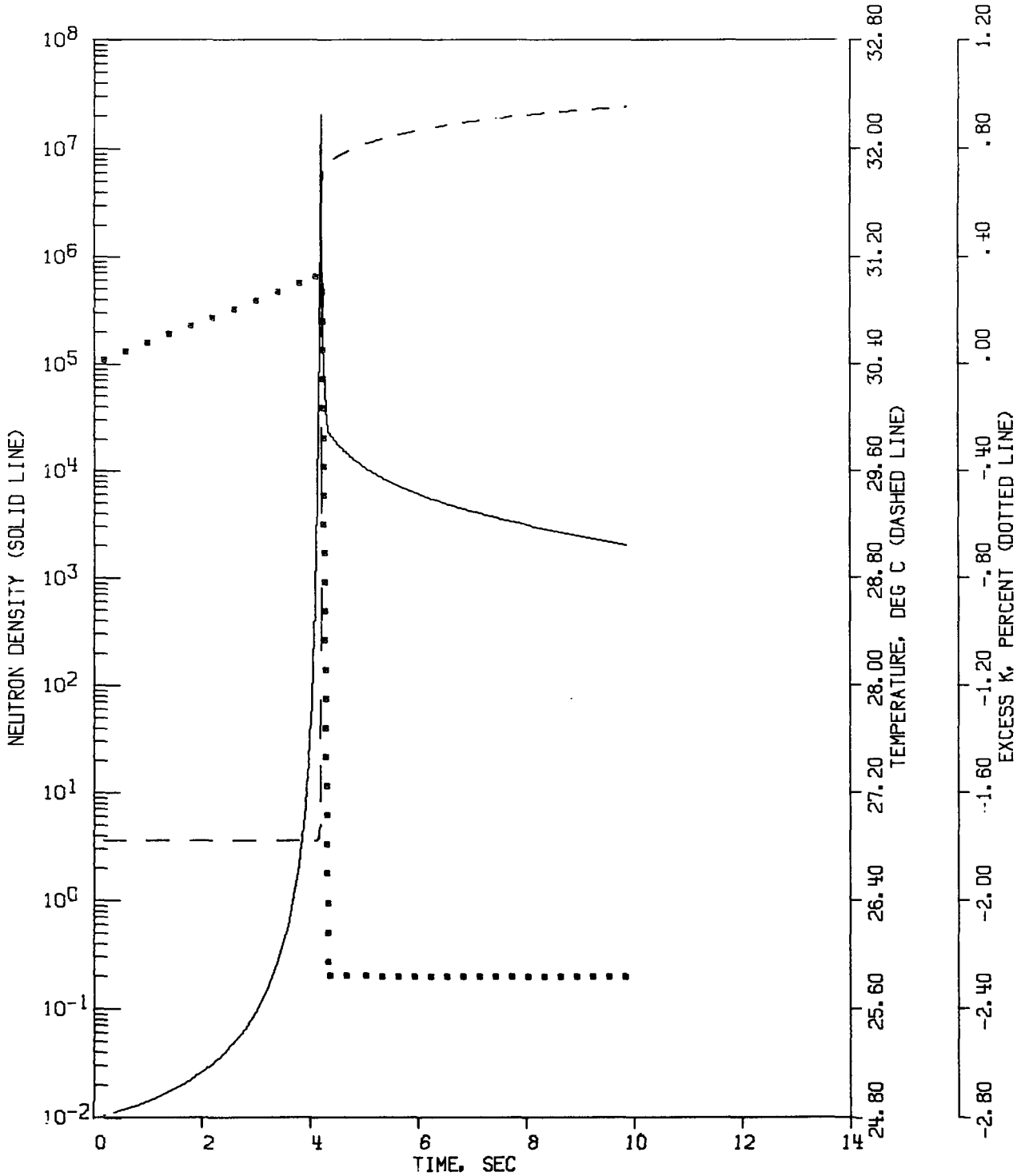


Fig. XI-4. Maximum Credible Accident Analysis, 3500 liter Core, All Materials Heated, Reactivity Insertion Rate 0.24 $/sec

figures the solid line represents the neutron density in n/cm^3, the dotted line represents k_{ex}, and the dashed line represents the average core temperature in °C.

Examination of Figs. XI-1 to XI-4 shows that in each case the neutron density rises sharply as prompt critical is approached and exceeded, the excursion is turned by the Doppler and/or expansion coefficients of the fuel, and the triggering of the safety rods then stops further reaction. In the cases where the reactor reached prompt critical the trip was assumed to take place at this point, since once the reactor reached prompt critical the power rises so rapidly that the delay times from the time that the trip occurs to the time that the safety rods start moving in (taken as 90 ms) makes this assumption slightly conservative.

The 50-liter core cases are shown in Figs. XI-1 and XI-2. For this reactor the expansion coefficient is the relevant feedback mechanism; the Doppler coefficient contributes very little. It can be seen that for the case of only the fuel plates heating, the average core temperature rises to approximately 165°C, and for the case of all materials heating the temperature increases to approximately 163°C. The peak-to-average power in this core is 1.2, so that the maximum temperatures at the center of the reactor are 194 and 192°C, respectively. These temperatures are far below either the ignition point (603°C) or the fuel-iron eutectic point (650°C). There is, therefore, no possibility of fuel ignition.

In these calculations and all subsequent calculations the reactivity feedback due to expansion is assumed to take place only after an initial temperature increase of 48°C. This corresponds to an assumption of a 3-mil gap between the fuel plate and its clad, on the conservative assumption that all fuel plates are the minimum length of 4 in.

Analogous results for the 3500-liter carbide core are shown in Figs. XI-3 and XI-4. In this case the excursion is turned by the Doppler effect, before the expansion feedback has a chance to enter. The temperature increases to about 45°C for the case of only fuel plates heating, and to about 32°C for the case where all materials are heated.

The results are summarized in Table XI-2. Comparison of these results with the values calculated for the ^{235}U-fueled systems in Ref. 1 illustrates the beneficial safety effects of the assuredly negative Doppler effect in these systems, and the reduction in table speed, resulting in lower reactivity addition rates.

The sensitivity of the observed results to the values calculated for the feedback parameters were tested by repeating the calculations assuming feedback coefficients one-half and twice the calculated values. Because the

important coefficients have been at least partially checked by experiment, this ±100% variation is considered to be far outside the range of probable error. The two worst cases, for feedback coefficients half their normal value, for the case of the fuel only heating, are shown in Figs. XI-5 and XI-6 for the 50-liter and 3500-liter cases, respectively. It is seen that this results in an increase in the average temperature of the fuel from approximately 165°C, for the normal coefficient, to approximately 215°C for the halved coefficient. For the 3500-liter case, the average fuel temperature increases from approximately 45°C to approximately 60°C. Thus, the results of the analysis using normal coefficients are not changed significantly by this very substantial reduction in the value assumed for the feedback coefficients. The fuel still remains very substantially below the ignition point.

The cores discussed above were uniformly fueled cores. For non-uniformly fueled (zoned) cores it is possible that the region which contains fuel undiluted by ^{238}U may have a small (or zero) expansion coefficient of reactivity. The restriction (see Chapter II) that the fuel undiluted in ^{238}U may only have a maximum of 25% of the total reactivity of the core, however,

Table XI-2

ANALYSIS OF MAXIMUM CREDIBLE ACCIDENT

System	Peak Power, MW	Total Energy Release, fissions	Average Fuel Temperature, °C	Maximum Fuel Temperature, °C
3500 liter Fuel Plates Only Heated	0.38	3.8×10^{17}	45	57
3500 liter All Materials Heated	1.76	1.1×10^{18}	32	36
50 liter Fuel Plates Only	1.85	2.2×10^{17}	167	230
50 liter All Materials Heated	4.50	5.2×10^{17}	155	213

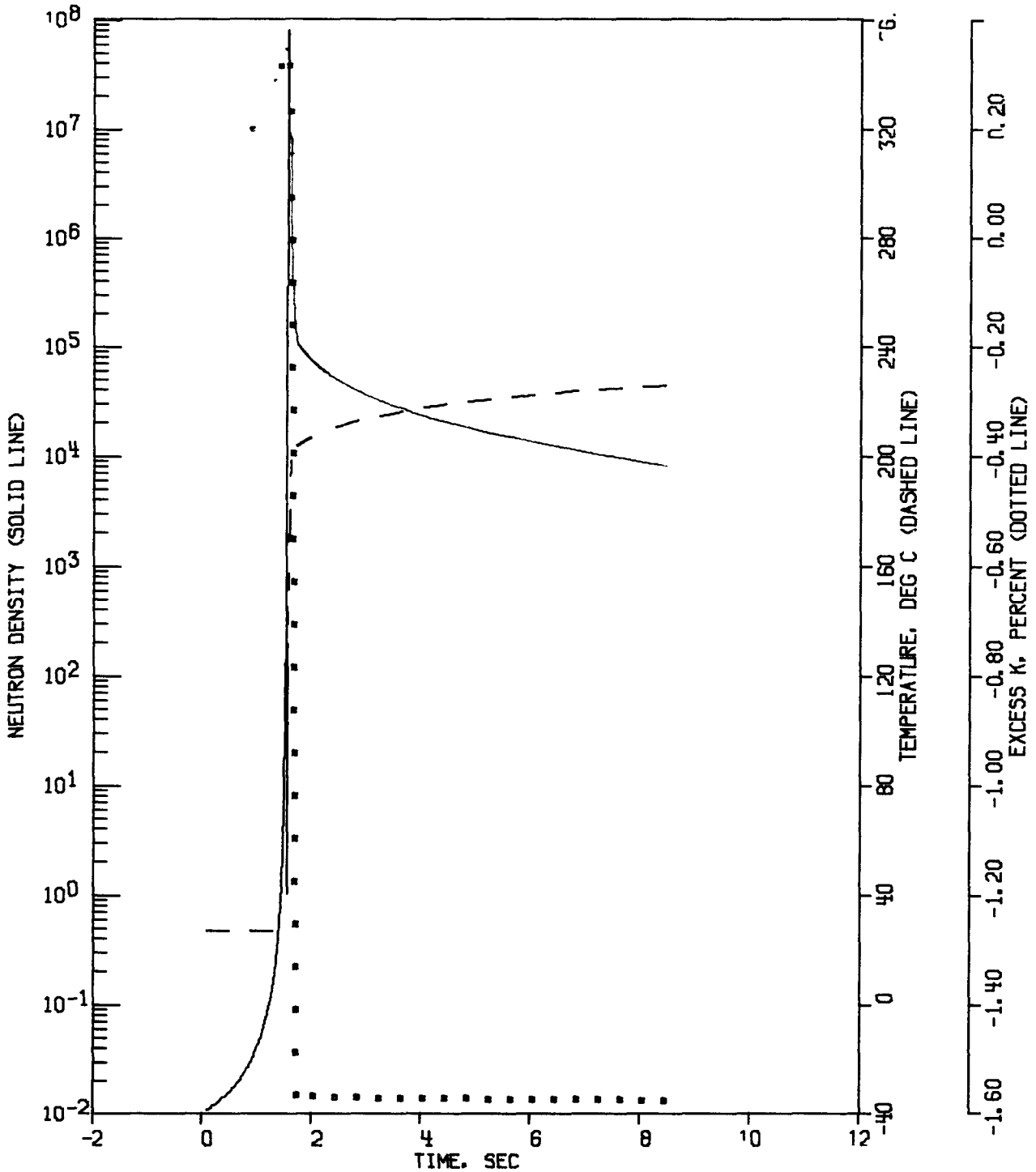


Fig. XI-5. Maximum Credible Accident Analysis, All Negative Feedback Coefficients Halved, 50 liter Core, Fuel Plates Only Heated, Reactivity Insertion Rate 0.65 $/sec

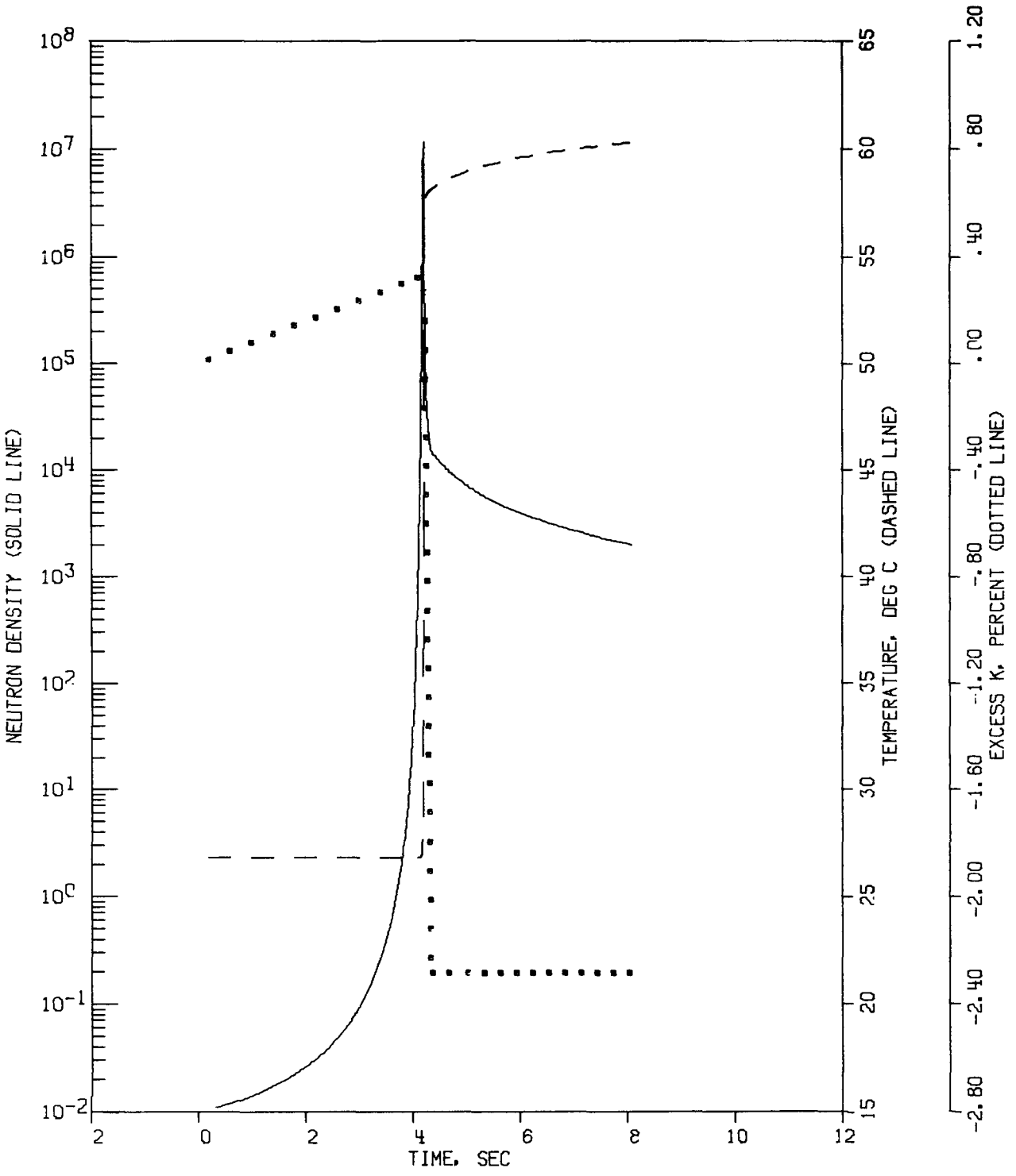


Fig. XI-6. Maximum Credible Accident Analysis, All Negative Feedback Coefficients Halved, 3500 liter Core, Fuel Plates Only Heated, Reactivity Insertion Rate 0.24 $/sec

means that the total feedback coefficient of the system may be reduced to about 75% of its usual value. This reduction of the feedback coefficient is only one-half of the arbitrary reduction of a factor of two which was shown above to have a minor effect on the results of the maximum credible accident analysis, so the above conclusions will also hold for the proposed nonuniform fueled (zoned) cores.

The conclusion therefore is that the Maximum Credible Accident can result in a moderate increase in the fuel temperature, but the maximum fuel temperature does not approach the ignition point and there is no likelihood of release of activity.

The foregoing accident analysis is considered the Maximum Credible Accident since the initiating mechanism represents the outer limits of credibility and because the neutronic parameters were chosen so that the excursion results in the maximum possible energy release or maximum possible temperature rise in the fuel. Although the MCA is in reality not a very credible situation it is proposed to use this method of analysis to determine whether a particular experiment is within the boundaries of this Safety Analysis Report. Experiments will be conducted on ZPR-6 and -9 using plutonium or uranium fuels whose MCA analysis results in maximum fuel temperatures no greater than about 200°C which is the maximum temperatures attained for the 50-liter MCA analysis. This is in keeping with the ground rules as outlined previously in ANL-6271 (Ref. 1) for uranium fueled systems.

REFERENCES

1. W. Y. Kato, G. J. Fischer, and L. R. Dates, "Safety Analysis Report, Argonne Fast Critical Facility (ZPR-VI)," ANL-6271 (December 1963).

2. J. J. Kaganove, "Reactor Kinetics Code," ANL Program Library R-101, (October 7, 1964).

Chapter XII

SAFETY ANALYSIS--DESIGN BASIS ACCIDENT

A. Introduction

In order to assess the type and magnitude of an accident which may be contained by the ZPR-6 and -9 reactor cells and confinement structures, a severe nuclear accident which is designated the Design Basis Accident (DBA) is hypothesized and analyzed. The DBA analyzed in this chapter is considered highly incredible and virtually impossible except for a deliberate sabotage attempt carried out by a minimum of two highly competent technically knowledgeable individuals who are familiar with the facilities. The DBA analysis is carried out to illustrate the type of severe nuclear accident which may be contained by the cells and associated structures. It does aid in establishing the adequacy of the reactor cells and associated confinement shells and facilities. The DBA considered below is representative of a class of very severe nuclear accidents; however, it is not the worst imaginable or conceivable excursion or accident. Conservative parameters were chosen in order to illustrate the ability of the cells and associated safeguards to contain an extremely severe hypothetical nuclear accident.

The previous chapter discussed an accident caused by criticality occurring while the halves were closing at the intermediate speed, with the results of the accident compounded by a postulation of severe coincidental failures of the appropriate trip circuits in addition to operator error and operator inattention. It was argued that the successive failure of the trip circuits necessary to give any significant excursion at all was not very credible. The subjective nature of credibility may allow some degree of credence to be attached to a situation where one trip fails, or two trips fail coincidentally. But the more coincident failures that one postulates, the less credible the situation becomes, and a failure of all controls is considered incredible. It is asserted that as one moves along the line of decreasing credibility one reaches a point well before the failure of all controls where the bounds of credibility end.

The compounding of:

(1) Large fuel overloading of the reactor caused by operator judgment error

(2) Inattention on the part of two reactor operators

(3) Failure or bypassing of period trip circuits

(4) Erroneous setting of three high level trip circuits is considered almost beyond the realm of credibility. However, in order to produce a severe nuclear accident which approaches the limits of ZPR-6 and -9 reactor cell containment capability, the failure of all controls is postulated.

The results of an intermediate closure accident discussed in Chapter XI as the MCA is considered with the additional hypotheses that no trip circuits are operable and that the operators continuously add reactivity (i.e., keep the moveable table moving toward closure) ignoring the several indicators of neutron flux level and its rate of change.

For purposes of detailed discussion, the 3500-liter carbide mockup is used as the reference case. Discussion of every possible core is not feasible, and in any case, the repetitive detail would tend to obscure rather than enlighten. The 3500-liter carbide core is considered to be typical of the cores that are of sufficient interest to the fast reactor program to represent likely mockups on these assemblies. The proportions of constituent materials for this core are reasonably similar to all but the smallest cores of this type. (See Table II-1.) The nominal 3500-liter core was selected as the reference for discussion of a design basis accident because its core composition is typical of those to be studied and it is the largest proposed core containing the greatest amount of plutonium.

B. Analytical Techniques

The DBA is analyzed using a modified form of the R-101 kinetics code coupled to several subroutines to calculate the neutronic excursion, the temperature distribution within the reactor, the metal burning rates, and the cell pressure, as functions of time. Expansion, Doppler and fuel flow reactivity feedback terms are calculated in the subroutines and appropriately introduced into R-101 reactivity equations.

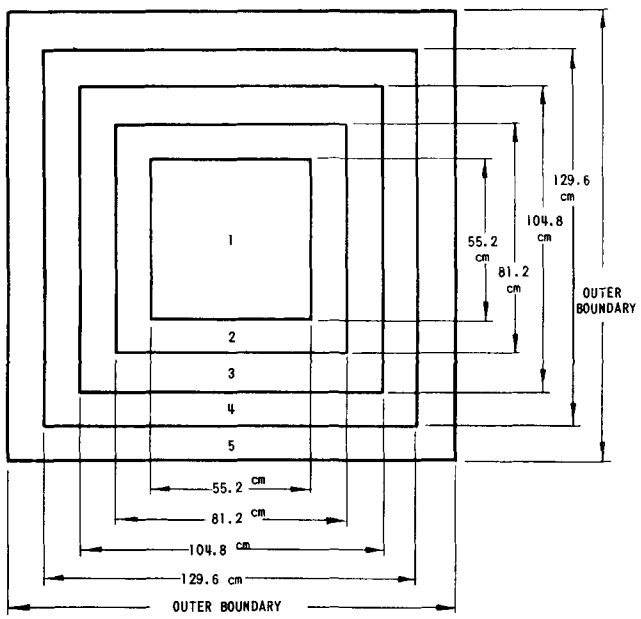


Fig. XII-1. Reactor Zone Used in Design Basis Accident (DBA) Calculation

As previously stated the R-101 code is basically a one energy group, point reactor, kinetics code. However it is possible to simulate spatial reactivity effects by dividing the reactor into zones and applying appropriate flux, importance, and statistical weights to each zone. The larger the number of zones, the more accurately the actual reactor is represented. To adequately represent the 3500-liter carbide core, five zones were selected. Radial and axial dimensions of the zones selected are shown in Fig. XII-1.

The neutron density and hence power density in zone "j"

is related to the average neutron density in a one energy group bare homogeneous core by the following equation.

$$\Phi_j = f_j\left(\frac{V_t}{V_j}\right)\bar{\Phi} = M_j\bar{\Phi}$$

where

$\Phi_j \equiv$ average neutron flux in zone "j"

$f_j \equiv$ fraction of total reactor power in zone "j"

$V_j \equiv$ volume of zone "j"

$V_t \equiv$ total reactor volume

$\bar{\Phi} \equiv$ average neutron flux in the reactor

$M_j \equiv$ a constant for zone "j."

The reactivity change due to temperature can be related to zonal temperature by noting that the perturbation expression given by Meneghetti can be written as follows.[1]

$$\frac{\Delta k}{k} = \frac{\sum_{\substack{\text{All Energy}\\ \text{Groups}}}^{i} \sum_{\text{All Zones}}^{j} \int_{\text{Zone Volume}} \Phi^* \left\{ \int_{\text{Temperature}} \left(\frac{\partial P}{\partial \theta} - \frac{\partial L}{\partial \theta} \right) d\theta \right\} \Phi \, dV}{\sum_{\substack{\text{All Energy}\\ \text{Groups}}}^{i} \int_{\substack{\text{Reactor}\\ \text{Volume}}} \Phi^* P \Phi \, dV}.$$

Here the denominator of the right-hand side is the importance weighted production integral and the other terms are as follows:

$\Delta k \equiv$ reactivity change due to temperature

$\Phi \equiv$ reactor flux (column vector)

$\Phi^* \equiv$ neutron importance (column vector)

$P \equiv$ neutron production matrix

$L \equiv$ neutron loss matrix

$\theta \equiv$ temperature

$dV \equiv$ incremental volume.

The indices and arguments have been omitted for simplified presentation.

When the above equation is contracted to one energy group to be compatible with the R-101 kinetics code, we have the following:

$$\frac{\Delta k}{k} = \sum_{\text{All Zones}} W_j \int_{\text{Temperature}} \left(\frac{C}{\theta_j} + E\right) d\theta_j$$

where

$$W_j \equiv \text{Statistical weight of zone "j"} = \frac{\int_{\text{Volume of j}} \Phi^* \Phi \, dV_j}{\int_{\substack{\text{Reactor} \\ \text{Volume}}} \Phi^* \Phi \, dV_r}$$

$$\frac{C}{\theta_j} \equiv \text{Doppler feedback term } \frac{1}{P}\left(\frac{\partial P}{\partial \theta} - \frac{\partial L}{\partial \theta}\right)_{\text{Doppler}}$$

$$E \equiv \text{Expansion feedback term } \frac{1}{P}\left(\frac{\partial P}{\partial \theta} - \frac{\partial L}{\partial \theta}\right)_{\text{Expansion}}.$$

Zonal fluxes, temperatures, and feedbacks are thus related to the average neutron flux as calculated in R-101. The expansion coefficients, the Doppler coefficients, the statistical weights, and the M_j's are calculated by use of the one dimensional, multienergy-group diffusion theory code MACH1.[2]

The Doppler coefficient was evaluated by calculating the change in k_{eff} using cross section sets with cross sections appropriate for 300 and 500°K, respectively. A θ^{-1} Doppler reactivity temperature dependence was assumed and the Doppler coefficient calculated as follows:

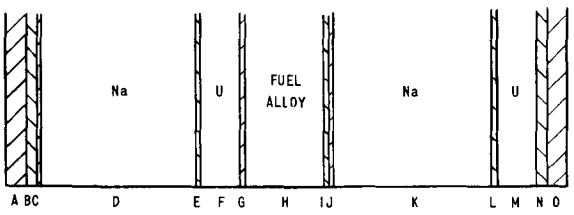
$$C = \frac{\Delta k_{eff}}{\ln \frac{500}{300}}.$$

Separate evaluations of the fuel and ^{238}U Doppler coefficients were made.

The expansion coefficient was evaluated by calculating the change in k_{eff} resulting from the redistribution of fuel in the axial direction. A linear relation between temperature and fuel expansion was assumed. Since the length of a column of fuel, L, is known, a linear expansion coefficient of $dL/d\theta$ of 1.54×10^{-5} is assumed, and $\Delta k/k$ for an expansion of δL cm (1 cm in this case) is calculated from MACH1, we can evaluate the expansion coefficient as follows:

$$E = \left(\frac{1}{L}\frac{dL}{d\theta}\right)\left(\frac{\delta k}{k} \Big/ \frac{\delta L}{L}\right).$$

To calculate the material temperatures within a given zone, the plates which comprise the loading are assumed to have temperature gradients only in the transverse direction. This is a reasonable approximation since relatively small temperature gradients exist in the axial and vertical directions, and in addition since the transverse plate dimensions are small compared to the axial or vertical dimensions, heat flow is primarily in the transverse direction. Figure XII-2 shows the region and regional materials comprising the cells and zone.* Boundary conductivities and gap conductances are initially assigned to the contact surfaces between regional materials. As expansion occurs, the contact conductances are recalculated and set equal to appropriate values. When melting occurs the contact conductivity is increased to 14 calories cm^{-2} sec^{-1} $(°C)^{-1}$ ($\sim 10^5$ Btu hr^{-1} ft^{-2} $(°F)^{-1}$). Material temperatures are calculated at a number of nodal points within the material and these are subsequently averaged to obtain an average material temperature. Material temperatures are maintained constant as latent heats are supplied for melting or boiling. The boundary conductance from the adjacent cell into the first and last region of the cell under consideration is set equal to zero and the heat capacity of a boundary region is reduced to half its normal value to account for heat flow into the region from adjacent cells.



REGIONS	cm	INCHES
A = MATRIX TUBE	0.1524	0.060
B = DRAWER TUBE	0.0762	0.030
C = CLADDING Na CAN	0.0381	0.015
D = Na	1.2700	0.500
E = CLADDING Na CAN	0.0381	0.015
F = DEPLETED URANIUM	0.3175	0.125
G = CLADDING FUEL ALLOY	0.0381	0.015
H = Pu-U-Mo ALLOY	0.6350	0.250
I = CLADDING FUEL ALLOY	0.0381	0.015
J = CLADDING Na CAN	0.0381	0.015
K = Na	1.2700	0.500
L = CLADDING Na CAN	0.0381	0.015
M = DEPLETED URANIUM	0.3175	0.125
N = DRAWER TUBE	0.0762	0.030
O = MATRIX TUBE *	0.1524	0.060

14 REGIONS, 4 MATERIALS, 6 PROPERTY RANGES

* POSITION O REPRESENTS THE BEGINNING OF NEXT CELL

Fig. XII-2. Regions of Cells

Figure XII-3A shows the nodal point arrangement within a given material or region. The corresponding temperature equation used in the calculations is as follows:

$$K_{12}(\theta_1 - \theta_2)\,\Delta t + K_{32}(\theta_3 - \theta_2)\,\Delta t = C_2(\theta_2^1 - \theta_2)$$

where

$$K_{12} = K_{32} = \frac{k}{\Delta\chi}$$

$C_2 \equiv \rho\sigma\Delta\chi$

$k \equiv$ thermal conductivity of material

$\Delta\chi \equiv$ distance between nodal points

*A zone is comprised of a number of identical cells which contain various materials in locations called regions.

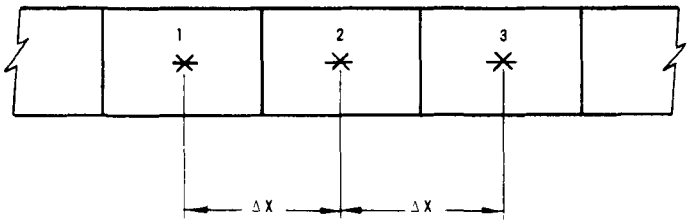
$\rho \equiv$ material density

$\sigma \equiv$ material heat capacity

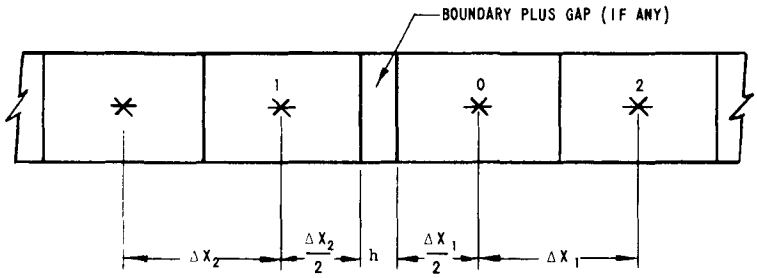
$\theta_n \equiv$ temperature of nodal point "n" at time t

$\theta_n^1 \equiv$ temperature of nodal point "n" at time t + Δt.

Unit dimensions are assumed in the directions transverse to that of the calculations.



3A



3B

Fig. XII-3. Nodal Diagram Used in Design Basis Accident (DBA) Calculation

Figure XII-3B shows the nodal point arrangement used in the vicinity of contact between two materials. This arrangement is used to allow the introduction of large contact conductances without requiring impracticable computer running times to achieve "time stability," i.e., to avert an implied violation of the second law of thermodynamics during the calculation. The corresponding temperature equation is as follows. Again unit dimensions in the transverse directions are assumed.

$$[H_{10}(\theta_1 - \theta_0) + K_{20}(\theta_2 - \theta_0)]\,\Delta t = C_0(\theta_0^1 - \theta_0)$$

where

$$H_{10} \equiv \frac{1}{\left(\frac{\Delta\chi_2}{2}\right)\left(\frac{1}{k_2}\right) + \frac{1}{h} + \left(\frac{\Delta\chi_1}{2}\right)\left(\frac{1}{k_1}\right)}$$

$$K \equiv \frac{k_1}{\Delta\chi_1}$$

$$C_0 \equiv \rho_1 \sigma_1 \Delta\chi_1$$

$$\frac{\Delta\chi_2}{2} \equiv \text{distance between nodal point "0" and the boundary}$$

$$k_2 \equiv \text{thermal conductivity of material 2}$$

$$\frac{\Delta\chi_2}{2} \equiv \text{distance between nodal point 1 and the boundary}$$

$$k_1 \equiv \text{thermal conductivity of material 1}$$

$$h \equiv \text{boundary conductance between materials}$$

$$\rho_1 \equiv \text{density of material 1}$$

$$\sigma_1 \equiv \text{heat capacity of material 1}$$

$$\Delta\chi_1 \equiv \text{distance between nodal points in material 1}$$

The maximum allowable time step in the calculation is given by the following if Fig. 3A is dominant.

$$\Delta t \leq \frac{\rho\sigma\Delta\chi^2}{2k}.$$

If Fig. XII-3B is dominant, the maximum time step in the calculation is given by

$$\Delta t \leq \frac{1}{\dfrac{1}{\rho_1 C_1 \Delta\chi_1\left(\dfrac{\Delta\chi_1}{2k_1} + \dfrac{1}{h} + \dfrac{\Delta\chi_2}{2k_2}\right)} + \dfrac{k_1}{\rho_1 C_1 \Delta\chi_1^2}}.$$

Table XII-1 contains material constants used as input data to the codes. A number of material temperature ranges have been selected to represent the material parameters with constants within each temperature range. A temperature range terminating after a fuel temperature rise of 48°C was introduced to allow the fuel alloy to maintain an expansion reactivity coefficient equal to zero until the maximum possible gap between the fuel and its container is filled. A dummy range was added to allow the calculation to continue beyond sodium boiling, if necessary.

The calculation assumes fuel to be expelled from the reactor 0.1 sec after the fuel temperature reaches the melting point of stainless steel. In actuality the fuel will probably begin flowing out shortly after it is melted and shutdown would begin sooner. A metal fire is assumed to start when the fuel starts to flow and is assumed to generate heat at a rate of 2.3×10^5 calories sec^{-1}. In addition, sodium vapor is assumed to be generated

at a rate proportional to the heat flow into the sodium after the sodium boiling temperature is reached. The sodium vapor is assumed to burn instantaneously.

Table XII-1

SELECTED THERMAL DATA FOR REACTOR MATERIALS

Range	Upper Temp, °K	Heat Capacity, cal/gm °C	Thermal Conductivity, Calorie Units	Latent Heat, cal/gm
		Fuel Alloy		
1	348	0.030	0.064	-
2	850	0.036	0.078	-
3	1230	0.0395	0.100	-
4	1230	10^7	0.130	15
5	3700	0.047	0.233	-
6	3700	10^7	0.300	450
7*	3700	10^7	0.300	
		Sodium		
1	370	0.338	0.212	-
2	370	10^7	0.207	27.05
3	650	0.318	0.192	-
4	850	0.305	0.163	-
5	1155	0.302	0.133	-
6	1155	10^7	0.115	1005
7*	4200	0.302	0.01	-
		Stainless Steel		
1	500	0.121	0.040	-
2	900	0.138	0.048	-
3	1670	0.168	0.066	-
4	1670	10^7	0.079	50
5	2500	0.250	0.080	-
6	2500	10^7	0.080	2100
7*	4200	0.250	0.080	-
		Depleted Uranium		
1	1400	0.0362	0.084	-
2	1400	10^7	0.120	20.0
3	1900	0.0440	0.160	-
4	2800	0.0460	0.240	-
5	4200	0.0480	0.300	-
6	4200	10^7	0.300	450.0
7*	4200	0.048	0.300	-

*Dummy Range: See text.

The heat generated by burning vapor and liquid is assumed to immediately heat the cell air to raise its temperature and the cell pressure. The calculation assumes no heat transfer to the cell walls or contents, i.e., an adiabatic situation is calculated. A flow rate of 2400 cfm/psig is taken for the cell air exhaust rate through the sand filters. This is somewhat conservative compared to the 3200 cfm per psig measured for a scaled mockup using the same type of sand that will be used in the full size filter. Temperature and pressure calculations are incremented each time a transfer is effected to the heat transfer subroutine since this subroutine in turn calls a subroutine which calculates the energy released due to burning fuel and sodium which in turn calls the pressure subroutine. The energy derived is comprised of a part due to burning liquid fuel and sodium and a part due to burning vaporized sodium.

$$\Delta Q = 1.52 \times 10^4 \, \Delta M_s + 920 \, \Delta t$$

where

$\Delta Q \equiv$ energy released during the time interval Δt

$\Delta M_s \equiv$ quantity of sodium vapor produced during the time interval Δt

$\Delta t \equiv$ time interval between successive calculations.

The pressure is calculated as indicated in the following sequence:

1. A pressure increase due to heat input ΔQ to the cell air is calculated. It is assumed that a negligible fraction of the cell air escaped from the cell during the time increment Δt; i.e., a constant volume process is initially assumed.

$$\therefore \Delta P = \alpha \frac{\Delta Q}{V}$$

where

$\alpha \equiv$ constant

$\Delta P \equiv$ increase in pressure in time Δt due to heat input

$V \equiv$ cell volume.

2. A new (intermediate) cell pressure P_2 is calculated and an average cell pressure during time Δt is calculated.

$$P_2 = P_1 + \Delta P$$

$$\overline{P} = P_1 + 0.5 \, \Delta P$$

where

$P_1 \equiv$ cell pressure at beginning of time increment Δt.

3. The change in cell air temperature associated with the ΔP increase is now calculated

$$\Delta T = \frac{T_1}{P_1} \Delta P$$

where

$T_1 \equiv$ cell temperature at the start of interval Δt.

4. A new temperature and an average temperature is calculated.

$$T_2 = T_1 + \Delta T$$

$$\overline{T} = T_1 + 0.5\ \Delta T.$$

5. Next the quantity of air which flows out of the cell during Δt is calculated as is an average specific volume.

$$\Delta V = B(\overline{P} - P_0)\ \Delta t$$

$$\overline{v} = \gamma \frac{\overline{T}}{\overline{P}}$$

where

$P_0 \equiv$ atmospheric pressure

$V \equiv$ constant

$\gamma \equiv$ constant.

6. The change in the cell air weight and a new cell air weight is calculated.

$$\Delta W = \Delta V / \overline{v}$$

$$W_3 = W_1 - \Delta W$$

where

$W_1 \equiv$ weight of cell air at beginning of time Δt.

7. Finally a new pressure which accounts for air flow out of the cell is calculated.

$$P_3 = P_2(W_3/W_1).$$

The subroutine then resets the initial values of the problem as follows and repeats the entire procedure on each pass.

$$P_1^1 = P_3$$

$$W_1^1 = W_3$$

$$T_1^1 = T_2.$$

In excess of one hundred passes through the pressure code are made subsequent to the initiation of fuel flow (shutdown) or sodium vaporization. Δt's are of the order of tens of milliseconds. Thus, although approximate, the calculation should adequately describe the situation.

Table XII-2 summarizes the input used in the heat generation and pressure calculations.

Table XII-2

HEAT AND PRESSURE CALCULATION INPUT DATA

Specific Heat Air (Constant Volume)	0.174 Btu/lb-°F
Cell Volume	36,500 ft^3
Initial Cell Pressure	14.7 psia
Atmosphere Pressure	14.7 psia
Flow Rate through Sand Filters (B)	40 ft^3/psig-sec
Initial Cell Air Temperature	530°R
Initial Weight of Cell Air	2,700 lbs
α (See text)	2.13
γ (See text)	0.37
Heat Addition Rate due to Burning Fuel	210 Btu/sec
Heat Addition Rate due to Burning Liquid Na	710 Btu/sec
Heat Addition Rate due to Burning Vaporized Na	1.52×10^4 Btu/kg

C. Parameters Influencing DBA Results

Up to the point where fuel first starts to melt the calculation of the course of the excursion is relatively unambiguous. To that point the excursion will essentially proceed as shown in Fig. XI-3. However, from that point the exact course of the excursion depends on the interplay of several interrelated processes:

(a) The expansion characteristics of the drawer materials.

(b) The time required for the fuel to penetrate to clad.

(c) The temperature at which the sodium cans fail.

(d) The heat transfer versus time characteristics for materials in a matrix drawer.

(e) The flow phenomena for molten material in a drawer.

It is desirable to digress and discuss each of these points in some detail.

1. Expansion Characteristics

A desirable characteristic of the usual cores in the ZPR systems is that they are, in general, very dense. That is, the various material pieces--uranium, sodium, carbon, etc.--take up all but a small fraction of the matrix volume. Typically, a volume fraction of about 6 vol % is left void. The 6 vol % is distributed as the clearance between the drawer and the matrix walls, and clearance between the matrix tubes. The expansion characteristics of the drawer materials, and in particular of sodium, are such that on heating they tend to fill the available void space at relatively low temperatures.

Table XII-3 lists the average volumetric expansion coefficients of the core materials over the range 20 to 900°C, including phase changes where relevant. The volumetric expansion of each constituent as well as the total volumetric expansion for the 3500-liter carbide core are also shown. The net decrease in void volume at any temperature will be the difference between the total volumetric expansion of the constituents and the volumetric expansion of the matrix tube containing them. In the actual situation the steel matrix temperatures may or may not lag the drawer constituent temperatures, depending on the particular array in any given drawer, but there will be no tendency for the steel matrix to be at a higher temperature than

Table XII-3

EXPANSION PROPERTIES OF CORE MATERIALS
(3500 liter Carbide Core)

Average Volumetric Expansion Coefficient 20-900°C x 10^{+6}				
U/Pu Fuel	Uranium	Stainless	Carbon	Sodium
79	61 2	54	5	293

Change in Volume $\frac{\Delta V}{V}$ % at Temperature T, °C								
T, °C	Fuel	U	SS	C	Na	All Materials	Matrix Steel	Net Void Decrease
20	-	-	-	-	-	-	-	-
300	2.2	1.7	1.5	0 1	8 2	4 3	-1 5	2.8
500	3.8	2 9	2 6	0.2	14 1	7.4	-2 6	4.8
700	5.4	4 2	3 7	0 3	20 0	10.6	-3.7	6 9
900	7.0	5 4	4 8	0 4	25 8	13.7	-4 8	8.9

the average of the constituents. Assumption of steel matrix temperatures equal to the average temperature gives a reasonable but conservative estimate, of the net decrease in void at any temperature. This is also listed in Table XII-3.

2. Penetration Rates of Iron and Steel by Hot Fuel

Good data exist on the penetration rates of iron and steel by uranium and uranium-fissium alloys. The ZPR fuel is a uranium-28 wt % plutonium-2.5 wt % molybdenum alloy which forms eutectics with iron analogously to uranium but with the melting points of both the pure fuel and fuel-iron eutectic lowered somewhat by the presence of plutonium. (Actually the fuel alloy does not have a single well-defined melting point. Its solidus is estimated to be 880°C and its liquidus 980°C. It is felt that the properties of this alloy are well enough known that the estimate is quite exact.) The melting points of uranium and the fuel alloy are 1125°C and 880-980°C, respectively, and the iron-eutectic melting points are 725 and 625°C, respectively.

Walter and Kelman[3] have studied the rate of penetration of Type 304 stainless steel by molten uranium and uranium-fissium alloy. Molten uranium exhibited penetration rates of 15 mil/sec at 1150°C, about 25°C above its melting point, dropping to 4 mil/sec at 1187°C and increasing again to 6.5 mil/sec at 1350°C. The reversal in penetration rate with temperature was attributed to the formation of an intermetallic compound UFe_2, which retards the reaction. At 1235°C this compound melts, allowing the reaction rate to increase with further temperature increase. The penetration rates of stainless steel by uranium below the uranium melting point have been studied by McIntosh and Bagley.[4] They found that a 20-mil wall of stainless steel was penetrated in less than one minute in the temperature range 850-950°C, about 200°C below the melting point of uranium alone.

Thus the evidence is that above the eutectic melting point, but considerably below the melting point of the fuel alone, times of the order of tens of seconds are necessary to penetrate the 15-mil clad. As the fuel melting point is reached the penetration rates increase to times of the order of a second. A further temperature increase causes a temporary decrease in penetration rate and then the rate increases again until the steel itself melts at about 1400°C.

More recent data by Kelman[5] and Savage[6] on eutectic penetration, shown in Fig. XII-4, substantiate the trends of the previous work, and show the faster penetration rates of iron as opposed to stainless steel. The eutectic is shown to rapidly increase in its penetration rate some 60°C or so below the melting point of uranium alone, rising to a maximum at about 1100°C.

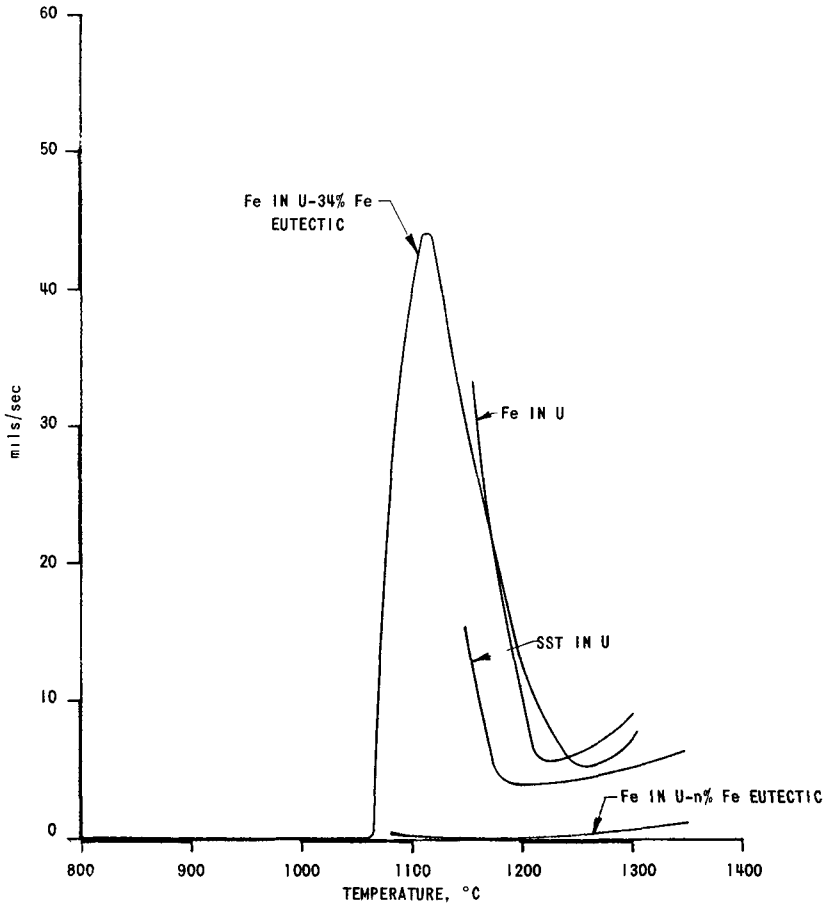


Fig. XII-4. Penetration Rate of Uranium and U-Fe Eutectics through Iron

The penetration rates were linear in time: twice the steel thickness required just twice the time at any given temperature.

3. Sodium Can Failure

A number of tests have been done to give information on the temperatures at which typical ZPR sodium cans will fail and release their contents. The cans are of two thicknesses, 1/4 in. and 1/2 in. by 2 in. high and are of various lengths, but mostly 7 or 8 in. The walls are 15-mil Type 304 stainless steel.

In the first test, a 1/4 in. sodium-filled can was heated without constraint in a furnace. The can eventually failed somewhat above 1200°C, but only after the sides had ballooned out so that the can had become, in effect, a 1.45 in. dia cylinder. Failure occurred at the ends. Calculations indicate that failure is to be expected at 1240°C.

In the second test a 1/4 in. can was totally constrained top and bottom and on both sides by a 1/2 in. steel plate, with only the ends under no constraint. Failure occurred at about 830°C, both ends failing, with

molten sodium squirting in both directions, causing an immediate fire. After cooling, it was found that only about 2 g of the original 46 g of contained sodium had been released.

In the third test a complete matrix position was mocked up, with a loaded drawer in a matrix tube, and the assembly was heated. (Brass was used as a substitute for fuel, since it has about the same melting point.) Four 1/4 in. cans of sodium were present. The matrix tube provided the only constraint. The sodium cans failed at 885°C, causing an immediate fire inside the matrix tube and splitting the matrix tube longitudinally on one of the top corners. The brass melted apparently simultaneously and ran out both ends. Examination after cooling showed some ballooning of the sodium cans to take up the space left by the brass.

In three additional tests of materials heated to 700°C in a simulated matrix loading, involving a total of 14 more sodium cans, no failures occurred.

It is concluded from the tests completed to date that:

(a) the cans will not fail below 700°C unless attacked by fuel; and

(b) the failure point is dependent on the degree of constraint.

With the degree of constraint present in a matrix position, none of the four cans present in that test failed below 885°C. For subsequent discussion the thermal failure point is taken as 885°C.

4. Heat Transfer between Drawer Materials

The precise temperature distribution in any given drawer at any point in time during an excursion depends on the array within the drawer, and the rate at which the fuel temperature increases.

The thermal properties of the drawer materials are given in Table XII-4. The effects of various arrays and contact resistances on the time constant for heat transfer to the sodium are shown in Table XII-5.

Table XII-4

THERMAL PROPERTIES OF DRAWER MATERIALS

	ρ, lb/ft^3 (gm/cm^3)	k, $\frac{\text{Btu}}{\text{hr ft °F}}$ (cal/sec cm °C)	C_p, $\frac{\text{Btu}}{\text{lb °F}}$ or cal/gm °C
Fuel	1140 (18.4)	17.9 (0.07)	0.037
Na	53.5 (0.97)	50 (0.21)	0.306
SS	500 (7.8)	9.2 (0.04)	0.120
U	1190 (18.7)	16.5 (0.07)	0.030
C	110 (1.8)	80 (0.33)	0.20

Table XII-5

TIME CONSTANT (SECONDS) FROM FUEL TO SODIUM

Intervening Material	1/4 in. Na Can				1/2 in. Na Can			
Heat Transfer Coefficient h =	∞	11,000	1,500	500	∞	11,000	1,500	500
None	0.55	0.68	1.4	3.2	2.2	2.5	4.0	7.5
0.030 in SS	0.95	1.1	1.8	3.6	3.0	3.3	4.8	8.3
1/8 in U, 0.030 in SS	1.8	2.1	3.6	7.1	4.7	5.2	8.2	15
1/8 in C, 0.030 in SS	1.0	1.3	2.8	6.3	3.3	3.8	6.8	14
1/8 in C, 1/8 in U, 0.030 in SS	2.0	2.3	4.5	9.9	5.0	5.8	10	21

A range of heat transfer coefficients, corresponding to various contact resistances are shown. MacAdams[7] quotes experimental determinations of contact conductances for aluminum blocks which gave a value of h = 500 for zero pressure to h = 1100 Btu/hr-ft^2-°F for a pressure of 3800 psi. These values are shown to represent extreme limits of the range for solids. Wetting of the boundary by molten fuel or sodium increases the conductance to 100,000 Btu/hr-ft^2-°F.

Barzelay, Tong, and Holloway[8] have measured the contact conductance of stainless steel interfaces for various surface finishes as a function of pressure. They obtain conductances for a 30 microinch rms roughness of 1200 to 1700 Btu/hr-ft^2-°F when a minimum surface contact pressure of 5 psi was applied. The conductances increase to values of 1700 to 2200 with contact pressures of 425 psi. The stainless steel sodium cans available for use with ZPR-6 and -9 have a 2B finish, which corresponds to a 15-30 microinch rms surface roughness. If some allowance is made for minor surface irregularities and warping of the cans, the 30 microinch rms surface roughness would seem applicable. Thus a choice of h = 1500 is probably the most reasonable value to use to obtain thermal time constants.

5. Flow Phenomena of Melt within a Drawer

The viscosities of molten metals in general are not much greater than water. In fact, the kinematic viscosity (which is in some ways a better comparative index of fluid behavior) is smaller for liquid metals than for water. Thus from the point of view of fluid flow the molten metals behave much like water and can be poured, agitated, and flowed through orifices and ducts in much the same way as water.[9] For example, the viscosity of plutonium-9.6% iron alloy at 808°C has been measured as 6.14 cm,[10] corresponding to a kinematic viscosity of about 0.35 cs. This compares to a kinematic viscosity of about 1 cs for water at room temperature, and about 0.23 cs for sodium at 900°C.

Molten fuel and sodium both would tend to run out or be squirted out very rapidly if their temperature remains above the liquidus. As both

will ignite on exposure to air, they are very unlikely to solidify until they contact some very substantial cool thermal mass, which is by definition well away from the positions of high flux and importance.

To provide a frame of reference in which possible flow phenomena can be judged, the following cases are examined:

(a) Total blockage of the channel so no flow is possible.

(b) Expansion of materials such that flow is very constricted.

(c) Fuel clad destroyed so a relatively large channel exists.

6. Total Blockage

If a channel is momentarily blocked, with heat being transferred continuously to the sodium, the sodium cans would rupture somewhere around 885°C, the boiling point of sodium at atmospheric pressure. Only enough sodium would vaporize to maintain a pressure in the available void volume equal to the vapor pressure of the liquid at each temperature. That is, for the no-flow case, each increment of heat added to the liquid results in enough additional vapor generation to maintain pressure equilibrium at the new temperature. In a ZPR channel the liquid occupies a substantial fraction of the total volume. The specific volume of the vapor is so much greater than that of the liquid that only a small amount of vapor is generated by the addition of heat: The majority of the heat goes into increasing the temperature (and pressure) of the superheated liquid.

Thus as the temperature increases above the atmospheric boiling point of sodium, the pressure in the blocked channel (assuming some void remains) simply follows up the saturated vapor pressure versus temperature curve. The exponential nature of this curve causes the pressure in a blocked channel to rise rapidly, tending to open the channel at relatively low temperatures, as heat flows into the sodium. (If no void remains the pressures rise even more sharply to open the channel.)

Consider a ZPR channel without flow. At 885°C sodium is at its boiling point at atmospheric pressure, while at 1000°C for example, the saturated vapor pressure is 2.615 atm. Assuming perfect-gas laws, the specific volume of saturated sodium vapor is 4130 cc/g at 885°C and 1870 cc/g at 1000°C. The specific volume of liquid sodium at 885°C is 1.35 cc/g at 885°C and some small amount more at 1000°C, but for our purposes can be taken as constant at 1.35 cc/g. In a typical ZPR channel with 45 vol % sodium and 6 wt % void, the amount of void volume per gram of sodium liquid is $1.35 \times 0.06/0.45 = 0.18$ cc/g of liquid. Thus at 885°C a maximum of (0.18 cc/g of liquid)/(4130 cc/g of vapor) = 4.35×10^{-5} g of vapor per gram of liquid could be present. At 1000°C, $0.18/1870 = 9.65 \times 10^{-5}$ is the maximum possible amount. Thus in raising the pressure of 2.615 atm (23.8 psig) only 5.3×10^{-5} g of vapor would be generated per gram of liquid

present. Even this is an overestimate because most of the 6 vol % void would be taken up by thermal expansion. The liquid would be superheated (1000-885) = 115°C and if it was assumed that at this pressure the blockage was opened to release the liquid to atmosphere a fraction $(C_p\Delta T/h_v)$ = (0.31 x 115)/895 = 0.04 would vaporize. (C_p = the specific heat, h_v = heat of vaporization, cal/g, ΔT = amount of superheat.)

Thus in situations where rapid flow does not start immediately on melting of the constituents, substantial pressures are generated to expel materials to allow flow and a net vapor generation of only a few percent of the total sodium in the melting zone results.

7. Constricted Flow

In a channel where one postulates that one or more of the sodium cans have released before the fuel has destroyed its clad and opened a clear channel for rapid flow, the amount of flow possible is rather limited. Postulation that at least one sodium can has failed infers that it at least is at 885°C. Assuming that the fuel has not reached a temperature where it has destroyed its clad implies a rather slow excursion, for as will be discussed later in any reasonably rapid excursion the sodium temperature lags the fuel temperature sufficiently that there is ample time for the molten fuel to escape before the sodium cans fail. A slow excursion implies time for the drawer materials to heat rather uniformly. Therefore, as Table XII-2 shows, the expansions of materials is greater than the void available for their expansion. The release of the sodium can would relieve the consequent pressure but the can itself would still be in place and there would be little net area for flow.

Assume an arbitrary 1% of the cross sectional area of the channel as a possible flow area. In this constricted channel the friction loss will be the controlling factor for flow, and the Darcy equation for friction loss can be written as:

$$\Delta P = f\left(\frac{L}{D_e}\right)\left(\frac{V^2}{2g}\right)\rho$$

where

ΔP = pressure drop

f = friction factor

L = channel length

D_e = equivalent diameter = 4 x (hydraulic radius)

v = velocity of fluid

g = gravitational constant

For a smooth-walled channel, Bonilla[11] gives a relationship for f over the whole turbulent range:

$$f = 0.0056 + 0.5(Re)^{-0.32}$$

where

$Re = D_e v\rho/\mu$ = Reynolds number

D_e = equivalent diameter

v = velocity of fluid

ρ = density of fluid

μ = viscosity of fluid

In discussing the flow of liquid-vapor mixtures, Bonilla[11] remarks that although higher vapor than liquid velocities, or "slip flow," are generally indicated, an assumption of equal velocities for both components is simpler and gives results that agree well with experimental data. For our purposes this assumption of a single "equivalent fluid" is certainly good enough.

To estimate a relevant Re, assume that the 1% flow area to be distributed as a rectangular slot at the top of the drawer 2 in. wide and 0.020 in. high. The actual area would have a very irregular shape and this assumption probably underestimates the resistance to flow. The equivalent diameter of this slot is 4 x (Area/Perimeter) = $4(2 \times 0.020/4) = 0.040$ in. Taking a density of 0.65 g/cc for the liquid-vapor mixture, and a viscosity of 0.167 cP (at 900°C), then for an average velocity of 10 ft/sec, the Reynolds number is 1.2×10^4, corresponding to f = 0.030, and a pressure drop of 4 psi/ft of channel. The mass flow rate corresponding to this pressure drop is 50 g of sodium per second. This compares to about 350 g of sodium that is normally present per foot of matrix tube.

To totally void one foot of channel in a second would require a pressure drop of 125 psi/ft. This would drive out axial blanket drawers, for example, decreasing reactivity.

8. Fuel Can Destroyed

Once the fuel clad is destroyed the molten fuel mixture will very rapidly flow out. Some idea of the rapidity of this process can be gained from the following considerations: The fuel is 1/4 in. thick. If thermal expansion of the surrounding materials has not exerted pressure on the liquid column, the full 1/4 in. width is available for flow. If on the other hand, there is some ballooning of the sodium cans, the flow area will be reduced. However the decrease in flow area is just due to fuel being displaced axially from it, which is precisely what is desired. One cannot have it both ways: Either there is a full 1/4 in. width of flow path, or the fuel has already been displaced in reducing the flow area.

Assume no displacement has taken place. A fuel melt velocity of 1 ft/sec gives a Reynolds number of 1.82×10^4, for the 1/4 in. wide channel, a friction factor of 0.0273 and a pressure drop of 0.05 psi/ft. For a velocity of 10 ft/sec, Re = 1.83×10^5, f = 0.0160 and the pressure drop is 2.9 psi/ft. This says that a foot of channel would be cleared of fuel in one second if a pressure of only 0.05 psi/ft was built up, and in 1/10 of a second if the pressure drop was 2.9 psi/ft. There will, of course, be exit losses in an orifice effect at the matrix face. For a velocity of 1 ft/sec along the 1/4 in. wide channel, the pressure drop through a 1/4 in. diameter orifice at the end would be about 0.33 psi. There will be some acceleration loss as well, but this shows the sort of speeds one might expect in emptying the channel of molten fuel once the fuel was released with any sodium vapor driving it.

The high density of a free-standing molten column of fuel causes significant driving pressures in comparison to the above just due to its own weight.

$$\Delta p = \rho(Z_1 - Z_2) \approx 18 \times (21/12)(62.4/144) = 1.3 \text{ psi}$$

That is there is a hydrostatic pressure of 1.3 psi on the melt at the bottom of the drawer, causing it to run out rapidly just due to this. About 300 g/sec, for example, would discharge initially through a 1/4 in. orifice due to this hydrostatic head.

D. Metal Fires

A review of current knowledge of uranium, plutonium and sodium fires is given in Appendix E of Ref. 12. It is concluded that although the subject is complex, a considerable body of experimental data has been accumulated in recent years so that a reasonably quantitative understanding of the relevant processes has been achieved. While it is still not possible to trace the course of a postulated fire in detail, a conservative handling of the data given in Appendix E of Ref. 12 will put reliable upper bounds on the consequences of fires.

A significant uranium or fuel-alloy fire is considered incredible, for the following reasons:

(a) The conditions under which ignition can take place can be predicted with reasonable confidence. Ignition of plates of the sizes used for fuel in these assemblies is extremely unlikely at room temperatures. The ignition temperature in these sizes is about 600°C for the fuel alloy and about 700°C for uranium. Only very severe nuclear excursions in the incredible range, that is without operative controls, would raise the fuel temperature to this point.

(b) For the fuel alloy, in addition to reaching the ignition temperature, the 15 mil stainless steel clad must also be violated.

(c) Even if ignition did take place, it is by no means certain that the fire would sustain itself. Rather specific conditions of heat transfer and exposed surface must be met to sustain burning. Environmental temperatures near normal room temperatures make sustained burning unlikely.

(d) For materials in place in the matrix, fires would be oxygen limited.

Thus the conditions for ignition, or for sustained burning after ignition, only exist in a Design Basis Accident. The latter is considered incredible so it follows that a significant rapid burning uranium or fuel-alloy fire must also be considered incredible.

Sodium also will not ignite spontaneously in room temperature air (unless it is sprayed into the air in the form of a vapor mist). The 15 mil stainless steel clad must be violated before any exposure to air is possible. Violation of the clad and vapor spray are again only possible in a Design Basis Accident, so significant rapid burning sodium fires as well are considered incredible.

E. Design Basis Accident

The situation regarding fires, however, is different for the case of the Design Basis Accident where it is postulated that considerable quantities of molten fuel would be expelled from the matrix and onto the reactor bed. The exact quantity of molten fuel on the bed is, of course, difficult to estimate; however, one can take an extreme case which will delineate an upper limit, with assurance that the actual burning rates will be considerably less. Although a variety of factors influence burning rates of uranium or uranium-plutonium metals, the two main factors are the temperature of the surrounding air and the surface-to-mass ratio of the burning material. Whether or not the material will ignite, and the burning be sustained is dependent on the balance between the oxidation rate (and this is in turn dependent on the amount of surface area exposed for oxidation) and the heat removal processes occurring in any particular situation. Thus, the dependence on surface-to-mass ratio of the burning material is very relevant to the situation under discussion here. Where the molten fuel has flowed onto the cold reactor bed, the heat conduction to the cold bed will tend to rapidly cool the molten fuel, in effect lowering the effective surface-to-mass ratio, and decreasing the likelihood of sustained or rapid burning. Thus ignoring the heat conduction to the bed will tend to overestimate the combustion rate. Secondly, the ambient atmosphere will be approximately at normal room temperatures, initially at least, and taking the burning rates for an ambient temperature above the ignition temperature will again overestimate the

magnitude of the combustion rate. Finally, it is difficult to estimate the exact surface area of the exposed molten fuel, for the halves would be at some intermediate point in their closure, but an overestimate should result from an assumption that the halves are separated by the full 1.5 m and the molten uranium-plutonium fuel flows onto the bed fully covering the surface. This gives a surface area 1.5 m wide by 2.5 m long, or an exposed surface area of 3.6×10^4 cm^2. Reference to Fig. E-1 of Ref. 12 shows that with ambient air temperatures above the ignition point the burning rate is 13 mg of oxygen/min/cm^2 of the exposed surface. Taking the most likely combustion products as being PuO_2 and U_3O_8 gives an average heat release in combustion of 223 kcal/mole of uranium-plutonium alloy. A burning rate of 13 mg O_2/cm^2 · min) for 3.6×10^4 cm^2 of surface gives a total burning rate of 14.6 moles of oxygen/min. Thus the total heat release in the combustion process is $14.6 \times 223 = 3.25 \times 10^3$ kcal/min. The plutonium burning rate is $0.3 \times 14.6 = 4.37$ moles/min or 1.05 kg/min. The uranium burning rate is $0.7 \times 14.6 = 0.75$ moles/min or 1.9 kg/min.

The total weight of air in the cell is 1.29×10^3 kg. Taking the specific heat of air at constant volume as 0.17 gives a value for the thermal mass of the cell air of 0.219×10^3 kcal/°C. Thus for a heat release in the combustion process of 3.25×10^3 kcal/min, the resultant air temperature increase is 15°C/min, and the resulting pressure increase in the cell is 0.75 psi/min. It will be shown in the next section that this magnitude of pressure rise is handled very easily by the emergency venting equipment.

Furthermore, if it is postulated that the sodium cans lose their integrity and release sodium between the halves as well, the rate at which this material will burn can also be estimated. The average rate of combustion of sodium pool burning was found to be about 0.07 to 0.15 g/min/cm^2 of exposed surface,[12] depending on the amount of exposed surface and the depth of the sodium pool. Taking a mean value of 0.11 g/min/cm^2 of exposed surface, and assuming that sufficient sodium cans have been destroyed that the sodium could also completely cover the reactor bed, the resulting combustion rate is 4.0 kg of sodium per minute. With an average heat release on combustion for sodium of 2.7×10^3 kcal/kg, this results in a heat release rate due to burning sodium of 10.8×10^3 kcal/min. The sum of the energy release rates due to burning fuel and sodium liquids is 2.3×10^5 calories sec^{-1} and is the quantity used in the DBA calculation.

The accident calculation is started assuming the reactor is just critical at time zero and table motion is adding reactivity. Figure XII-2 depicts the material arrangement in a typical cell and Table II-1 contains material constants used as input data. Tables XII-6 and XII-7 contain other pertinent input data.

Table XII-6

DATA FOR DESIGN BASIS ACCIDENT

Number of Zone Reactor Divided into	5
Reactivity Addition Rate	7.9×10^{-4} $\Delta k/k$ sec^{-1}
Number of Delayed Neutron Groups	18
Effective Delayed Neutron Fraction	3.266×10^{-3}
Shutdown Mechanism	Fuel Flow Out of Reactor
Material Arrangement	As Shown in Fig. 2
Initial Gap between Materials	5×10^{-4} cm
Gap (Boundary) Conductivity	$\{(1.79 \times 10^{-7}\theta + 5.73 \times 10^{-5})/\text{Gap}\}$ cal/cm^{-2} sec^{-1}(°C)$^{-1}$
Maximum Boundary Conductivity between Solids	0.2 cal/cm^{-2} sec^{-1}(°C)$^{-1}$
Maximum Boundary Conductivity between Liquid and Material	14 cal/cm^{-2} sec^{-1}(°C)$^{-1}$
Initial Neutron Density	10^{2} neutrons cm^{-3}
Initial Material Temperature	300°K
Initial Rate of Fuel Flow from Zone 1 of Reactor	45 kg sec^{-1}

Table XII-7

ZONAL PARAMETER VALUES

Zone	1	2	3	4	5
Na Mass[a] kg	23.2	51.9	64.7	121	183
^{238}U Mass[a] kg	112	250	311	581	878
Fuel Mass kg	220	492	613	1144	1730
Fuel Worth, $\frac{\Delta k}{k}$ kg^{-1}	-2.73×10^{-4}	2.24×10^{-4}	1.83×10^{-4}	-8.17×10^{-5}	-3.05×10^{-5}
^{238}U Doppler Coefficient for Region 6[b]	-3.6×10^{-4}	-5.0×10^{-4}	-5.2×10^{-4}	-4.3×10^{-4}	-2.5×10^{-4}
^{238}U Doppler Coefficient for Region 13[b]	-3.6×10^{-4}	-5.0×10^{-4}	-5.2×10^{-4}	-4.3×10^{-4}	-2.5×10^{-4}
Fuel Doppler Coefficient[b]	-4.2×10^{-4}	-5.7×10^{-4}	-6.1×10^{-4}	-5.1×10^{-4}	-2.9×10^{-4}
Expansion Coefficient	-6.5×10^{-6}	-9.0×10^{-6}	-9.4×10^{-6}	-7.9×10^{-6}	-4.5×10^{-6}

[a]Material mass in each region of zone occupied by material
[b]Input to code is listed number divided by 300°C

As seen from Fig. XII-5 which shows the average reactor neutron density, integrated average neutron density, and excess reactivity as a function of time, the average neutron density increases slowly at first. At 4.13 seconds the reactor reaches prompt critical, and the neutron density rapidly increases from its value of 3.87×10^{3} neutrons cm^{-3} to the first maximum of 1.38×10^{7} neutrons cm^{-3}. This burst results in sensible heating and the resulting Doppler reactivity feedback in the fuel subsequently "turns" the excursion. At 4.21 seconds the fuel temperatures in zones 1 through 5 are respectively 321, 317, 310, 306, and 302°K. Negligible heat transfer has occurred to this time. Figure XII-6 shows a plot of zonal fuel temperatures as a function of time. Figures XII-7, XII-8, and XII-9 show the two sodium region (regions 11 and 14) temperatures and the hottest depleted uranium temperature (region 6) in each zone as functions of time.

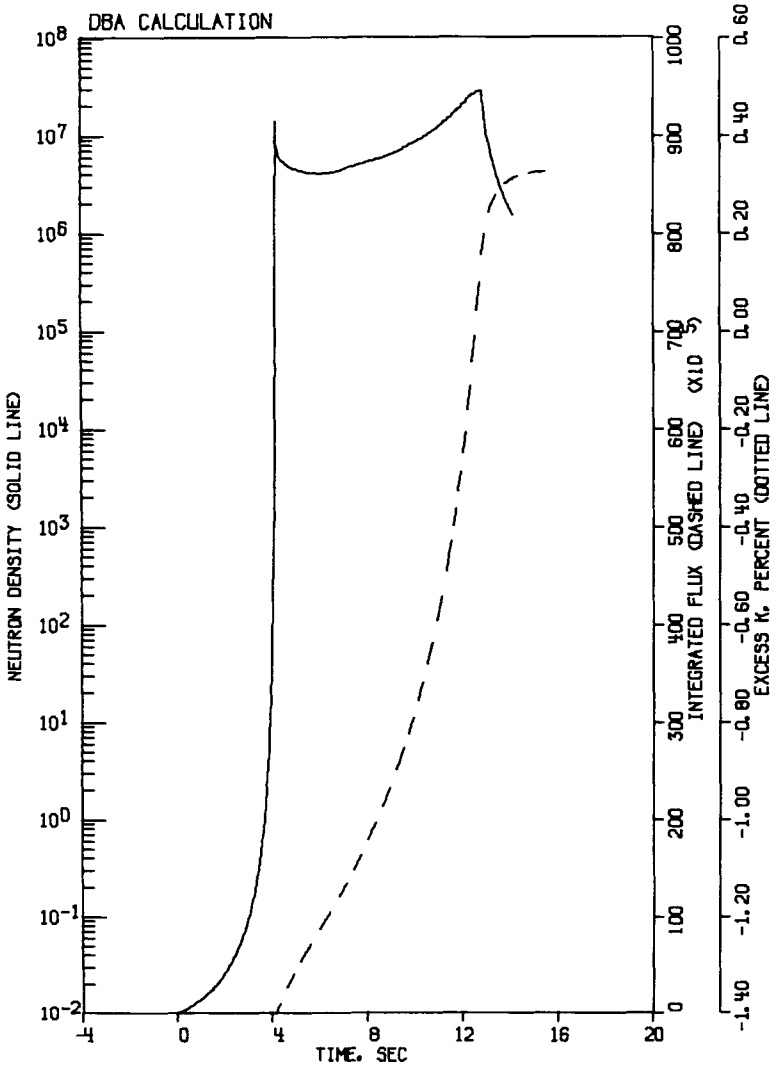


Fig. XII-5. Design Basis Accident Analysis for 3500 liter Core

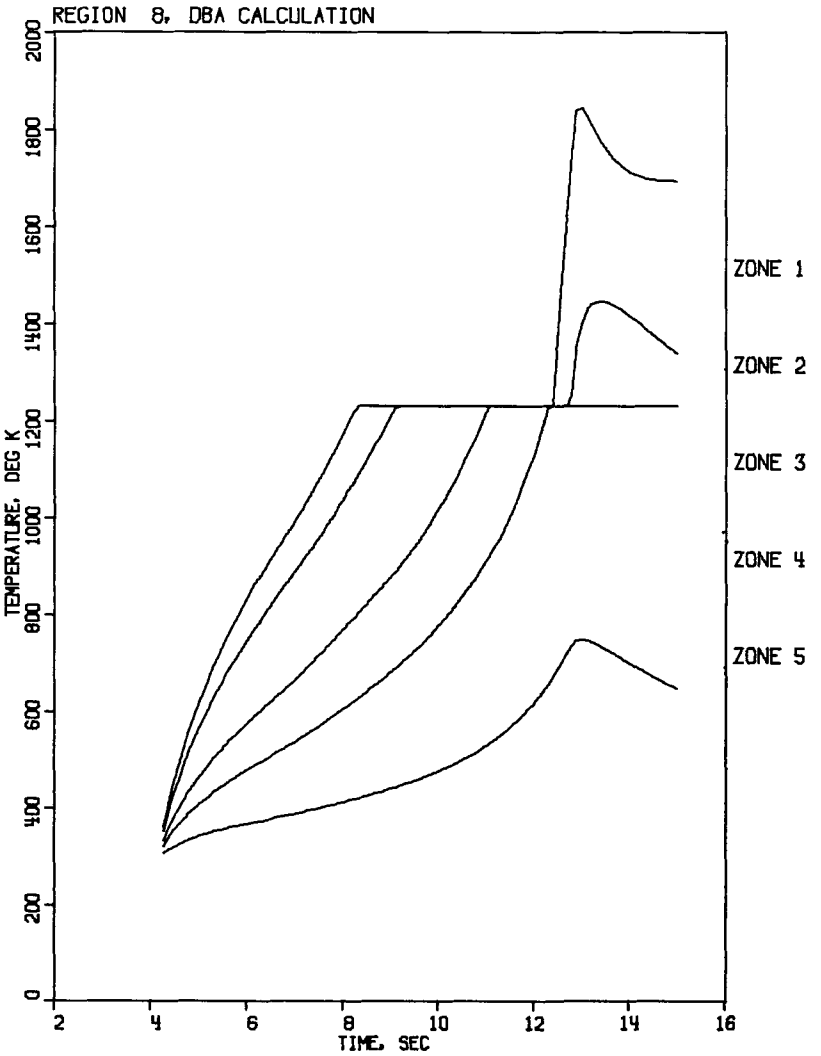


Fig. XII-6. Fuel Temperatures for DBA

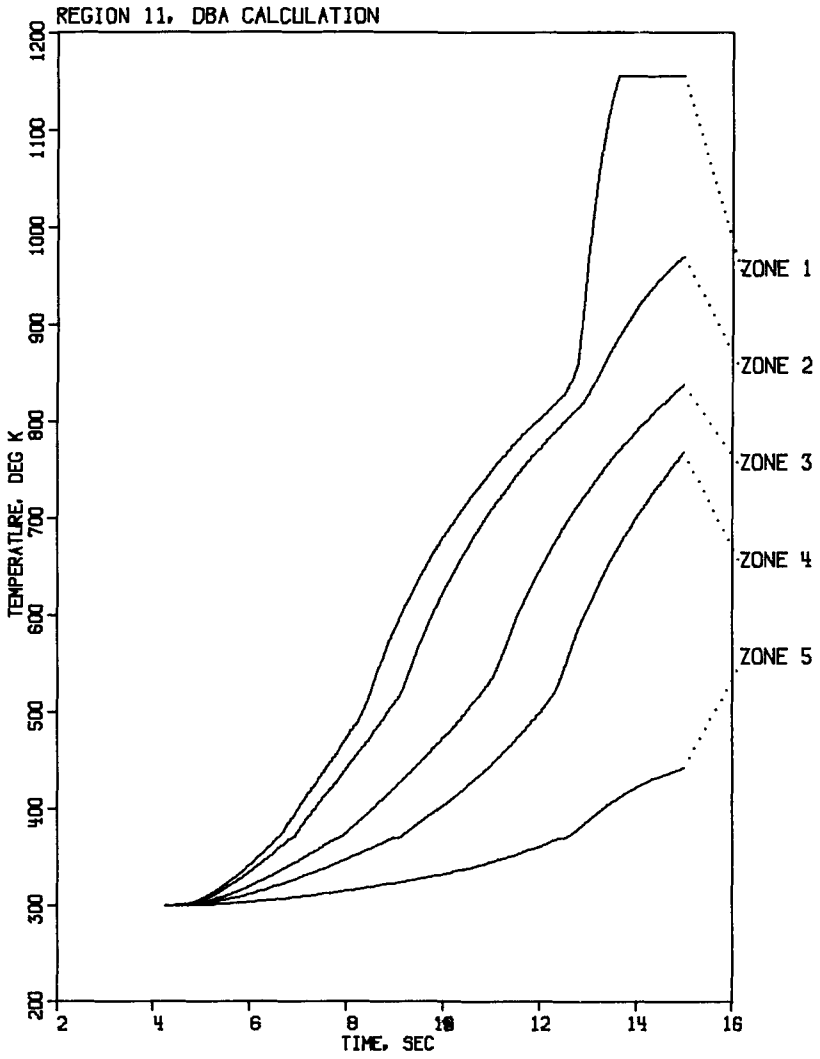


Fig. XII-7. Sodium Temperatures in Region 11 for DBA

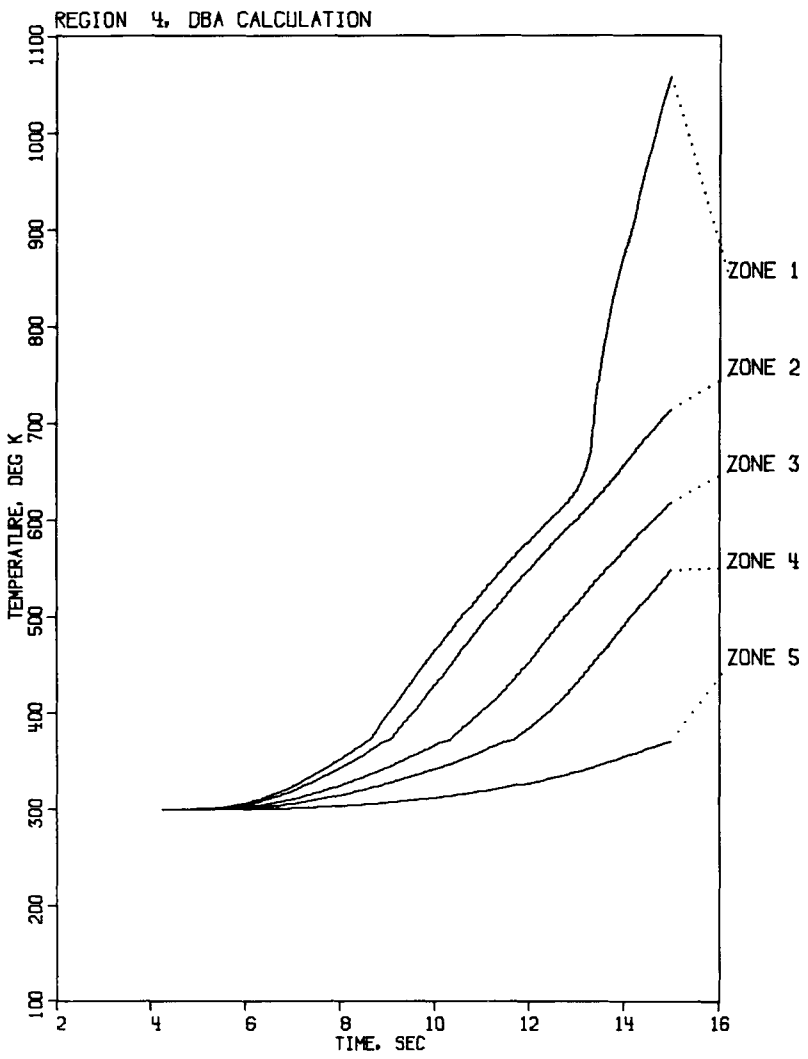


Fig. XII-8. Sodium Temperatures in Region 4 for DBA

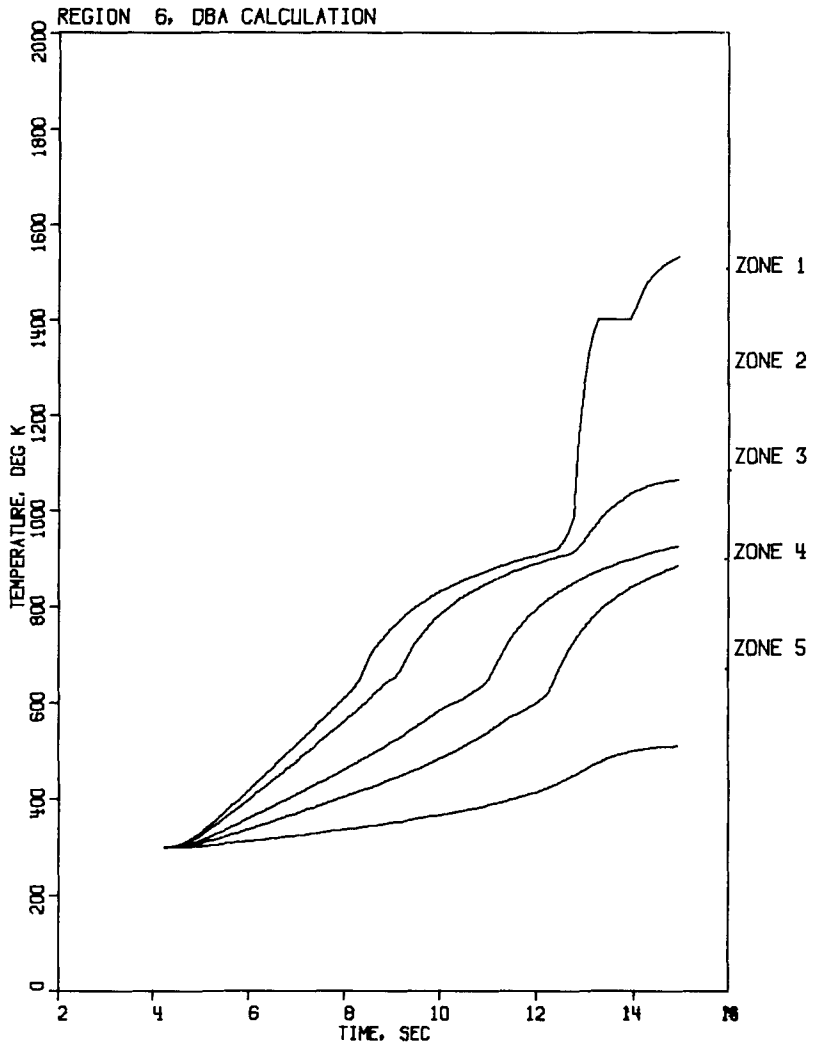


Fig. XII-9. Depleted Uranium Temperatures for Region 6 for DBA

After the excursion has been turned and the reactivity has been reduced below prompt critical, the Doppler and expansion reactivity feedback continue to decrease the k_{ex}. At 6.61 seconds, the sodium adjacent to the fuel (region 11) in zone 1 begins melting; the zone 1 fuel temperature is 930°K and the excess reactivity has been reduced to 1.602×10^{-3} $\Delta k/k$ The sodium adjacent to fuel in zone 2 begins to melt at 6 88 seconds and the fuel in zone 1 begins to melt after 8.29 seconds. The course of the excursion is graphically depicted in the aforementioned curves. When all of the fuel in zone 1 is melted at 12.39 seconds, its temperature rapidly rises to 1670°K at 12.73 seconds. A 0.1 second delay is assumed before fuel begins flowing out of the reactor.

With the commencement of fuel flow, and the resulting loss in reactivity, the flux is turned after reaching its second maximum of 2.81×10^{7} neutrons cm^{-3}. The reactor becomes subcritical at 13.00 seconds and the k_{ex} continues to decrease with increasing time. Burning of molten

metal begins at the time of fuel flow. Figure XII-10 shows the cell pressure as a function of time. It is noted that the rate of pressure rise due to burning liquid metal is small and the equilibrium pressure (where the pressure decrease due to venting just balances the pressure increase due to combustion) is <3 psig. The rate of heat input to the cell air is 9.24×10^2 Btu sec^{-1} (2.3×10^5 cal/sec^{-1}).

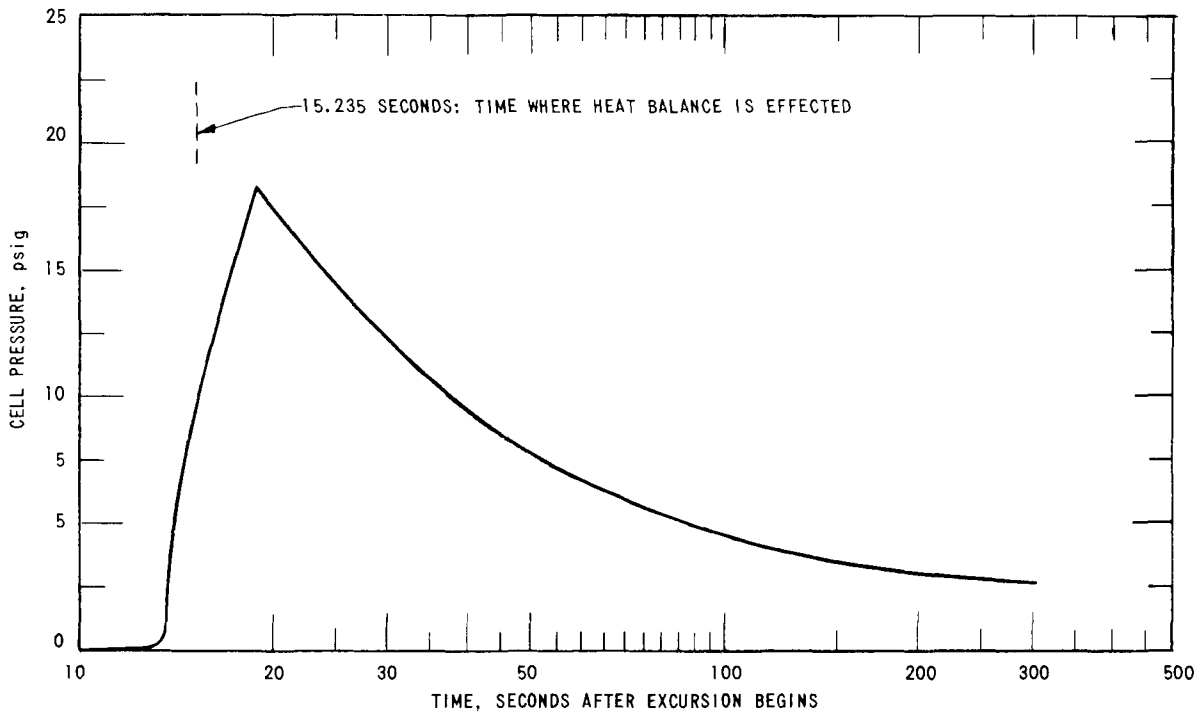


Fig. XII-10. Cell Pressure Time History

Boiling commences in the zone 1, region 11 sodium at 13.63 seconds and the rate of heat input to the cell air increases to 1.8×10^5 Btu sec^{-1} (4.6×10^7 cal/sec^{-1}). Boiling continues until 15.23 seconds at which time the cell pressure has risen to 22.6 psia (7.9 psig); the reactor is 2.3% subcritical and the excursion is over. Boiling is continuing in zone 1 and about to begin in zone 2. A total of 2.7×10^{20} fissions have occurred.

Because of the impracticably long computer running time required to achieve final equilibrium temperatures within the reactor zones by continuing the computer calculation, this approach was not attempted. Instead a heat balance was used to terminate the accident. To do this the heat energy from the fuel and depleted uranium in zone 1 is shared with the zone 1 sodium resulting in these material temperatures dropping to just below the boiling point of sodium and the production of sodium vapor. In zone 1 the quantity of sodium boiled off is independent of the relative heat transfer rate to materials on either side of the fuel because of the relatively high depleted uranium temperature. However in the other zones the relative heat transfer rates must be obtained and the fraction of excess

fuel heat going to the materials on each side of the fuel calculated. In one case a heat balance between the proper fraction of fuel heat, the depleted uranium in region 6 and the sodium in region 4 is calculated and the resulting sodium vapor determined. In the other case a heat balance between the appropriate fraction of fuel and sodium in region 11 is calculated and the resulting sodium vapor determined. The result is the maximum amount of sodium that can be vaporized by the after-heat in each zone.

The general equation used to compute equilibrium temperatures and the resulting sodium vapor produced is as follows.

$$f_i M_f (T_f - T_{sB})\, C_f + M_f \overline{LH} g + M_u (T_u - T_{sB})\, C_u + M_u \overline{LH}_u h - M_s (T_{sB} - T_s)\, C_s = \overline{LH}_s W_s$$

where

M_f = mass of fuel

M_u = mass of ^{238}U

M_s = mass of sodium

W_s = mass of sodium vaporized

f_i = fraction of fuel heat flowing to sodium region under consideration

T_f = fuel temperature before balance

T_u = ^{238}U temperature before balance

T_s = sodium temperature before balance

T_{sB} = temperature at which sodium boils

$\overline{LH}_f$ = latent heat of fuel

$\overline{LH}_u$ = latent heat of ^{238}U

$\overline{LH}_s$ = latent heat of sodium

C_f = specific heat of fuel alloy

C_u = specific heat of ^{238}U

C_s = specific heat of sodium

g = fraction of fuel in molten state

h = fraction of ^{238}U in molten state

This approach is conservative since it does not consider the fact that some of the other lower temperature materials within the reactor will actually acquire a substantial portion of the residual after-heat. It also does not

consider the fact that the release of molten fuel from the reactor decreases the thermal conductivity to the sensitive material, sodium, and that fuel is continuing to flow out of the reactor carrying its excess heat with it.

The following conditions exist at the end of the computer calculation and are pertinent to the heat balance calculation. Fuel has flowed from zone 1 of the reactor for 2.52 seconds at an average rate of about 38 kg sec^{-1} Negligible amounts of ^{238}U have flowed out of the reactor. The zone 1 fuel is at a temperature of 1692°K and the ^{238}U in region 6 of zone 1 is at 1421°K. The ^{238}U in region 13 is 937°K, over 200° below the boiling point of sodium, and therefore is ignored. Thus there remains about 125 kg of fuel in zone 1 and 120 kg of ^{238}U in region 6 of zone 1 to transfer excess heat to the sodium in zone 1. The sodium in region 11 of zone 1 is boiling and that in region 4 of zone 1 is at 1081°K. A heat balance results in an additional 8.5 kg of sodium being vaporized from zone 1. It is emphasized that this is the maximum amount of sodium which could be boiled from zone 1; the actual amount would be less.

In zone 2 it is assumed that no fuel has flowed out of the reactor because the trigger temperature of 1670°K, the melting temperature of the stainless steel clad, has not been reached. This is conservative since the fuel in zone 2 began melting at 9.10 seconds and would have undoubtedly have penetrated the clad, resulting in the loss of fuel by the time 15 seconds have elapsed. This fuel would not actually be available to transfer excess heat to the sodium. However we ignore that fact and assume that all of the fuel remains in zone 2. The rate of heat transfer to the sodium of region 4 is 39% of the total heat flow from the fuel and that to region 11 is 61%. The fuel temperature in zone 2 is 1327°K, the region 11 sodium temperature is 975°K, the ^{238}U in region 6 is 830°K and the sodium in region 4 is at 722°K. Consequently no sodium can be boiled from region 4. With 61% of the excess fuel heat flowing to region 11, 4.4 kg of sodium is boiled off. No sodium can be boiled from zones 3, 4, and 5.

Thus the after-heat is capable of boiling no more than 13 kg of sodium The burning of 13 kg of sodium will raise the cell pressure by less than another 12 psig. Adding this to the 22.6 psia pressure which exists at 15.2 seconds results in a cell pressure of less than 35 psia (20 psig). As shown in Fig XII-10, this pressure rapidly decreases with time even as molten burning continues. (The after-heat is assumed to boil the sodium off at the same rate as at 15.3 seconds until the 12.9 kg of sodium, calculated to be vaporized by means of the heat balance equation, is consumed. This results in a higher than actual heat addition rate to the cell and the abrupt break in the pressure curve and thus is again conservative.) As the pressure decreases to lower values, argon gas can be injected into the cell to extinguish the fire. It is noted that a cell pressure less than 3 psig results in sufficient flow through the sand filter to cause a decreasing pressure in the cell despite the energy added by burning molten metal.

While an accident of this magnitude results in an extremely large energy release, the cell pressure rating is not exceeded and the cell remains intact--containing the accident.

It should be noted that this accident was calculated using the intermediate table drive speed to achieve a relatively high reactivity addition rate. A reactivity addition rate of 24¢ per second was assumed for the calculation whereas measurements on Assembly 5 of ZPR-6 indicate a gap worth for a similar size core which would result in a reactivity addition rate of only about half this value. Thus the reactivity addition rate used in the calculation is highly conservative. Figure XII-11 shows the gap worth of the 2600 liter UC core, Assembly 5 of ZPR-6.

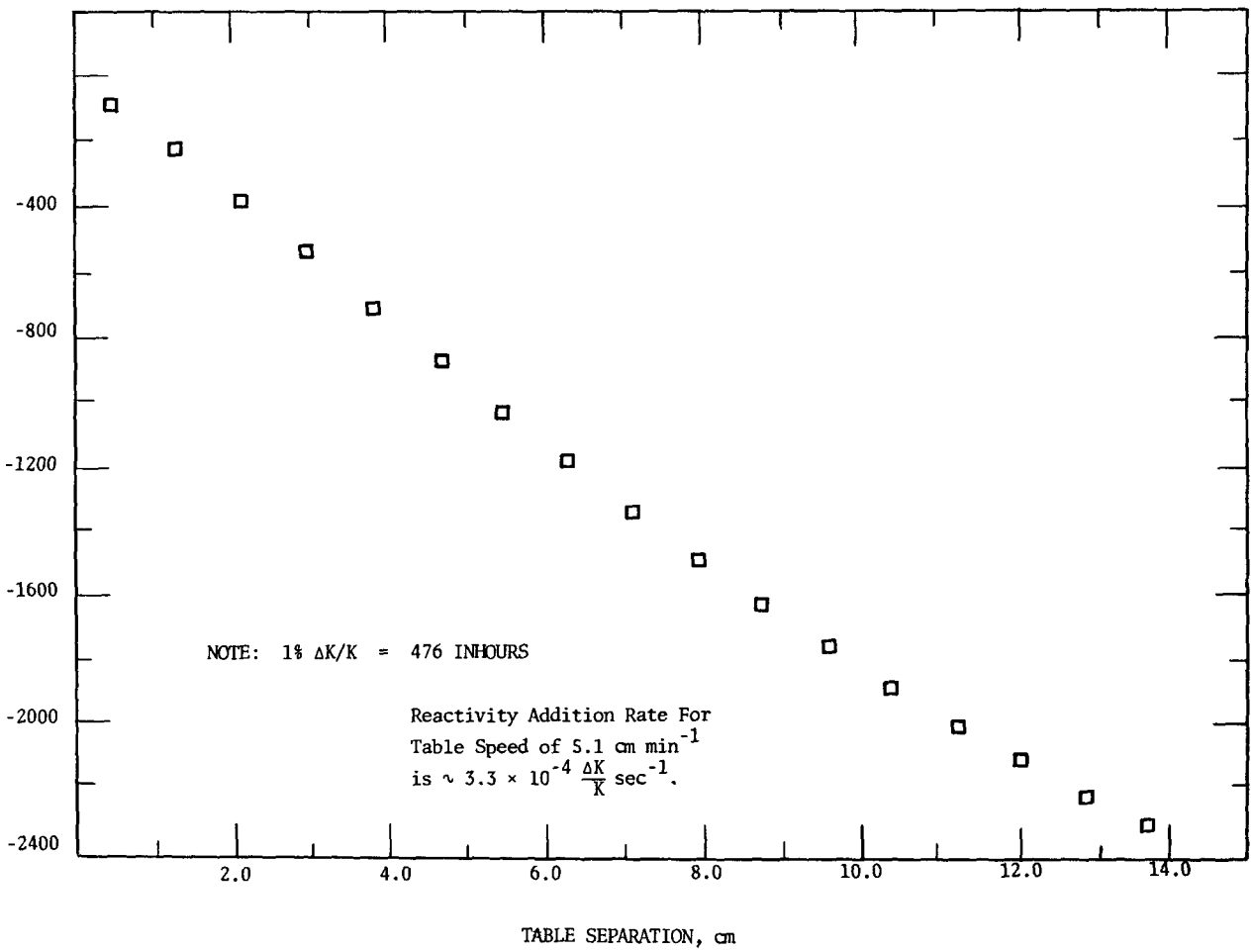


Fig. XII-11. Gap Worth Curve (Measured) for Assembly 5 of ZPR-6

An accident resulting from the low table drive speed, other factors remaining the same, would of course result in a smaller effect. In addition it must be remembered that the overload required to achieve this accident is very large; beyond the limit of credibility. Finally, waiting 0.1 second after the fuel has reached the melting point of its stainless steel clad before

assuming fuel flow out of the reactor certainly increases the accident yield over that which would result if clad penetration calculations were used to determine the time of fuel flow and shutdown.

To estimate the total amount of plutonium that may burn, consider the following: Argon can be introduced into the cell at an initial rate of 36,000 cfm (average rate over entire discharge time of 6000 cfm). It is therefore possible to reduce the oxygen content in the room to 4% (the level below which it has been found that sodium will no longer burn) in about 3 min, and to reduce the oxygen concentration to 1% in about 20 minutes. However to be conservative we assume that the duration of the fire would be about 1 hour. As the oxygen concentration is reduced and more argon is introduced into the cell, the burning rates would decrease. As shown in Appendix E of Ref. 12, there is considerable evidence that the uranium burning rates are diffusion limited, and as the oxygen content decreases and the argon content increases, the oxygen will find it progressively more difficult to reach the flame front. However, a conservative estimate will certainly result from assuming that the initial burning rates hold during the full one-hour period. In one hour approximately 60 kg of plutonium will therefore have burned.

As previously mentioned the total number of fissions in this excursion is 2.7×10^{20}. This is a very substantial energy release, but as it takes place over a time scale of 10 seconds it cannot be compared directly to excursions where the release occurs in a few milliseconds: There will not be the same tendency to generate strong explosive forces.

F. Radiological Hazards Assessment

The amounts of radiologically hazardous material released to the atmosphere by the hypothetical DBA is now considered. There are two possible paths for atmospheric release: (a) through the emergency exhaust system; and (b) leakage through the cell walls into the confinement shell volume, through the HEPA filters, and up the stack. Consider first the particulate attenuations available in these two routes.

1. The emergency exhaust system has a 30-in. deep sand filter plus two sets of HEPA filters in series. The sand filter will be field tested and shown to have an attenuation of at least 10^3 for diotylphthalate (DOP) aerosol at a flow velocity of about 15 fpm. This may be compared with attenuations of 10^3 obtained for 30-in. thickness of the same type of sand in the laboratory scale tests using DOP aerosol at a test velocity of about 15 fpm. The HEPA filters in series will be field tested to show an attenuation of at least 10^4 using DOP aerosols at flows of 2400 cfm. Cheever et al.,[13] has shown that the combination of 30 in. of the same type sand and one HEPA filter in series results in total attenuations of 10^6 at 1, 5, 13, and 28 fpm flow velocities for plutonium fume. Although laboratory tests have not been

done on a 30 in. sand filter followed by two HEPA filters, it is estimated that the second HEPA filter would introduce a factor between 10-100 in additional attenuation. Thus, the total attenuation is estimated to be between 1×10^7 and 10^8 for plutonium fume. Although Cheever et al.,[13] has measured an attenuation of 10^6 for plutonium fume for a 30 in. sand filter followed by one bank of HEPA filters, we will use a more conservative value of 10^5 for the total attenuation for the emergency exhaust system until additional experimental evidence becomes available to justify a higher value.

2. The confinement shell and workroom and control room exhaust systems each have two HEPA filters in series. Kessie[14] has shown experimentally that two HEPA filters (each having attenuation of 10^4) in series result in attenuations of 10^6 to 10^9 for fine particle size (~0.1 μ) plutonium aerosols. The HEPA filters in these systems had been field tested and shown to have an attenuation of 10^4 with DOP aerosol. For the purposes of this analysis we believe it appropriate to use a conservative value of 10^5 attenuation for the two HEPA filters in these systems.

In the DBA it was hypothesized that approximately 60 kg of plutonium (220 kg of fuel alloy) melted and burned over the period of an hour. Carter, Foy, and Stewart[15] have shown that only about 0.05% of the total mass of any plutonium burned actually becomes airborne.

Since there is some uncertainty on the dependence of the airborne fraction on the fuel burning temperature we will use a conservative value of 1% in our analysis here. On this basis, only approximately 600 g of the burned plutonium becomes actually airborne and available for release to the atmosphere via either of the two routes previously discussed.

Using the flow rate of 3000 cfm/psig through the emergency exhaust system, 6 cfm/10 psig or 0.6 cfm/psig through the concrete cell walls (this corresponds to a leakage of 24% of the volume per 24 hr at 10 psig), the pressure time history curve of Fig. XII-10, and 600 g of plutonium homogeneously mixed in the cell air, we calculate that only about 0.12 g can leak through the cell wall and the rest (~600 g) goes into the emergency exhaust system.

Using a value of 10^5 attenuation for the emergency exhaust system filters only about $600/10^5$ or 6 mgs of plutonium are released to the atmosphere through this route.

The 4 ft thick concrete walls do act as a filter. Tests on samples of 4 ft thick concrete at ANL have resulted in a measured attenuation of 10^3 for uranine aerosol at flow rates of 0.01 cfm/ft^2 of concrete. However, again being conservative, we shall assume that the cell walls have a 100% transmission for this analysis and 120 mg of plutonium is distributed into the confinement shell and workroom and control room areas. These areas are

exhausted through two sets of HEPA filters whose attenuation is conservatively considered to be 10^5. This results in an additional 120 mg/10^5 or 1.2 μg released to the atmosphere through this route. A total of about 6 mgs of plutonium is thus released up the stack to the atmosphere using this conservative analytical approach for the hypothesized DBA.

The radiological hazards associated with the release of 1 mg of various types of plutonium (different isotopic composition) have been analyzed in Appendix A. Table A-7 shows that the release of 1 mg of plutonium-Type B (12% ^{240}Pu) results in 1.33×10^{-4} mpbb and 1.23×10^{-3} mpbb for 1 mg of plutonium-Type C (30% ^{240}Pu) inhaled by an observer at 225 m (location of the Plutonium Fuel Fabrication Laboratory, the nearest building to the Applied Physics Laboratory) and 2.2×10^{-4} mpbb for these plutonium compositions respectively at 1300 m (the nearest site boundary)

In Appendix A the meteorological Pasquill stability condition F under fumigation conditions was assumed in order to obtain dispersion coefficients and to calculate ground-level air concentrations of plutonium. The X/Q at 225 m and 1300 m are respectively 7.8×10^{-4} and 1.3×10^{-4} for fumigation conditions If instead we consider the average worst meteorological conditions excluding fumigation as the meteorological criteria for the DBA we should use Pasquill stability condition A for an observer at 225 m and condition D for an observer at 1300 m for wind speeds of 1 m/sec. Under these assumptions the X/Q factor would be 8×10^{-5} for 225 m and 4.2×10^{-5} for 1300 m. The release of 6 mgs of either Type B or C plutonium to the atmosphere under the average worst meteorological conditions results in inhalation hazards as shown in Table XII-8

Table XII-8

FRACTION OF MAXIMUM PERMISSIBLE BODY BURDENS INHALED FOR PLUTONIUM AT 225 m AND 1300 m FOR 6 mg RELEASE

	225 m (Pasquill Type A)	1300 m (Pasquill Type D)
Pu-A (4% ^{240}Pu)	8×10^{-5}	4.2×10^{-5}
Pu-B (12% ^{240}Pu)	8×10^{-5}	4.3×10^{-5}
Pu-C (30% ^{240}Pu)	7.4×10^{-4}	4×10^{-4}

The results of the analysis as shown in Table XII-8 indicates that the inhalation dose for an observer at 225 m for the plutonium C sample (the plutonium with the highest specific activity) is 7.4×10^{-4} mpbb. This is approximately a factor of 150 less than the 0.1 mpbb, the AEC criterion for the maximum permissible inhalation hazards under the hypothesized DBA circumstances for fast plutonium fueled ZPR's.

The foregoing analysis represents the best current estimate of the inhalation hazards associated with burning of 60 kg of plutonium in the cell as a consequence of the hypothesized DBA. This assumes that the negative pressure of the confinement shell system exists and that the air from the confinement shell is exhausted through the two HEPA filters exhaust system.

If no credit is taken for the negative pressure of the confinement shell (i.e., both exhaust fans are inoperative) the analysis would have to be modified as discussed below.

It has been shown previously that no greater than 120 mg of plutonium would be distributed to the confinement shell, workroom, control rooms, mechanical rooms, and other negative pressure areas. The amount of plutonium which might escape into the confinement shell volume alone is no greater than 60 mg. Since that which escapes into the control room, basement, or penthouse mechanical rooms encounters two barriers before reaching the atmosphere. None of the walls of these areas see the direct outside wind velocities (i.e., there is a buffer region between the -0.1 in. water pressure areas and the outside). Only three of the six walls of the reactor cell face directly into the confinement shell volume.

The limiting criterion for the confinement shell leakage rate is 1000 cfm at -2.5 in. of water pressure. If there is a wind condition which creates a negative pressure outside one face of the shell, it is conceivable that some shell air might leak out of the shell. Assuming a wind speed of 1 m/sec it is possible to produce a differential pressure of a maximum of about 0.024 in. of water. The amount of leakage would be no greater than about 30 cfm. Assuming a uniform mixture of the plutonium in the shell volume, the maximum amount of plutonium which might leak out in half an hour's time is about 0.5% of the 60 mg or about 0.3 mg. The 0.3 mg of plutonium would, however, be released close to ground level and would have a different attenuation from that released at 46 m height at distances close to the point of release. The value of X/Q at 225 m for wind conditions of 1 m/sec and Type A (Pasquill categories) with a 46 m height release is 8×10^{-5}. For a ground release, X/Q for an observer at 225 m would be about 5×10^{-4}. The observer at 225 m would see about $5 \times 10^{-4}/8 \times 10^{-5}$ or about 6 times the amount released at 46 m height. The total amount of plutonium that an observer at 225 m would inhale would correspond to the equivalent of 6 mg + 6 x 0.3 mg or about 8 mg released from the 46 m stack. This corresponds to 1×10^{-3} mpbb for Type C plutonium. Thus, even if one postulates the failure of operation of the confinement shell exhaust system, the amount of plutonium which an observer might inhale is still a factor of about 100 below the 0.1 mpbb criterion for the maximum permissible body burden limit for inhalation doses under accident conditions.

It should be pointed out that the foregoing analysis is highly conservative for the following reasons:

a. The assumption was made that all of the 60 kg of plutonium melted, burned and 1% became airborne instantaneously at t = 13 secs. Since the cell pressure peaks at about 18 secs due to the vaporization and combustion of sodium, if one takes even rather high burning rates such as 1 kg/min of plutonium, the amount of plutonium in oxide particulate form available to leave the cell via the emergency exhaust system is small when there is a large flow of air out of the cell at elevated pressures.

b. In the case of the amount of plutonium escaping to the atmosphere via the confinement shell route, we have not taken any credit for the filter action of the concrete cell walls. We have also assumed that the leakage from the confinement shell was directly through an opening to the atmosphere rather than through sand or dirt around the base of the confinement shell.

The fission products from the hypothesized DBA require a separate consideration. A total of 2.7×10^{20} fissions was generated during the course of the DBA. The DBA results in the melting and the assumed burning of approximately 60 kg of plutonium (220 kg of fuel alloy) out of the total critical mass of 1354 kg of plutonium. The rest of the plutonium remains within its stainless steel jacketing. The fraction of fissions generated in the 60 kg of plutonium, taking into consideration the power distribution, is 20% of the total fissions. The products of only 5.4×10^{19} fissions are, thus, released to the cell. If we make the same assumptions as in the case of plutonium, approximately $5.4 \times 10^{19}/10^5$ or 5.4×10^{14} nongaseous fission products might be released through the emergency exhaust system and up the stack. On the other hand, we shall postulate that all of the gaseous fission products and the volatile radioiodine of the 5.4×10^{19} fissions would be released out the emergency exhaust system to the atmosphere. The noble gas fission products and the radioiodine constitute the principal radiological hazard of the fission products because there is no assumed attenuation of these products by the filter system. The radioactivity released from the stack after the hypothesized DBA is summarized in Table XII-9.

Table XII-9

SUMMARY OF DBA

Critical Mass	1354 kg Pu	Pu Released to Atmosphere	
Number of Fissions in DBA	2.7×10^{20}	a. From Emergency Exhaust System	6 mg
Quantity of Fuel Melted and Burned	220 kg	b Through Cell Walls	0.1 mg
Quantity of Pu Melted and Burned	60 kg	c Total	6 mg
Pu Airborne in Cell	600 g	Fission Products Released from Stack	
Number of Fissions in Melted Fuel	5.4×10^{19}	a Nongaseous Products from	5.4×10^{14} fissions
Particulate Attenuation		b Gaseous Products from	5.4×10^{19} fissions
a 30 in. Sand Filter + 2 HEPA	10^5		
b. 2 HEPA + Concrete Wall	10^5		

The radiological hazards associated with the release of the gaseous fission products (noble gas and radioiodine) from 5.4×10^{19} fissions are investigated in Appendix A. The radiological hazards associated with the release of the gaseous fission products from the 5.4×10^{19} fissions are summarized in Table XII-10 for the average worst meteorological condition.

Table XII-10

DOSES TO THYROID AND TOTAL BODY IN Rem

Conditions
Wind Speed: 1 m/sec
Stack Height: 46 m
Breathing Rate: 3.47×10^{-4} m^3/sec
Fissions: 5.4×10^{19}

Doses	225 m (Pasquill Type A) No Evacuation	1300 m (Pasquill Type D)
Thyroid (Radioiodine)	44	20
Total Body (Noble Gases)	0.61 (γ) 4 (β)	0.44 (γ) 0.2 (β)

It is seen from the table that an observer at 225 m, the closest laboratory facility (Plutonium Fuel Fabrication Laboratory) would receive a dose to the thyroid of about 44 Rem which is substantially below the 300-Rem criterion for once in a lifetime emergency dose for radiation workers. This assumes that the observer stands in the cloud, breathing the radioiodine for as long as the activity exists.

REFERENCES

1. D. Meneghetti, "Introductory Fast Reactor Analysis," ANL-6809 (December 1963).
2. D. A. Meneley, L. C. Kvitek, and D. M. O'Shea, "MACH1 A One-dimensional Diffusion-theory Package," ANL-7223 (June 1966).
3. C. M. Walter and L. R. Kelman, "Penetration Rate Studies of SS by Molten Uranium and Uranium-Fissium Alloy," Nucl. Mater. 6, 3, 281-290 (1962).
4. A. B. McIntosh and K. Q. Bagley, "British Nuclear Energy Conf. Journal" 3 (January 1958).
5. L. Kelman (private communication).
6. H. Savage, "Penetration of Jacket Materials by Molten U-Pu-Zr and U-Pu-Ti Alloys," Annual Progress Report for 1966, Metallurgy Division, ANL-7299, p. 115.
7. W. H. MacAdams, "Heat Transmission," McGraw-Hill Book Co., New York (1942).

8. M. E. Barzelay, K. N. Tong, and G. F. Holloway, "Effect of Pressure on Thermal Conductance of Contact Joints," NACA-TN-3295 (1955).

9. Reinhart Shuman, Jr., "Metallurgical Engineering," Addison-Wesley, Inc. Cambridge (1952).

10. E. Grison, W. Lord, and R. Fowler, "Plutonium 1960," Cleaver-Hulme Press, London (1961).

11. C. F. Bonilla, "Fluid Flow in Reactor Systems," Nuclear Engineering Handbook, McGraw-Hill Book Co., New York (1958).

12. W. Y. Kato et al., "Preliminary Safety Analysis Report on the Use of Plutonium and ^{233}U in ZPR-6 and -9 [Addendum No. 3 to Safety Analysis Report, Argonne Fast Critical Facility (ZPR-6), ANL-6271]," ANL-7211 (to be published).

13. C. L. Cheever, D. R. McFee, J. Sedlet, and T. L. Duffy, "ZPPR Roof Sand Filtration of Uranium, Plutonium and Uranine Aerosols," Proc. 9th AEC Air Cleaning Conf., Boston, Sept. 13-16, 1966. USAEC Report CONF-660904 (January 1967), Vol. 2, pp. 942-964.

14. R. W. Kessie and D. Ramaswami, "Removal of Plutonium Hexafluoride from Cell Exhaust Air by Hydrolysis and Filtration," ANL-7066 (December 1965).

15. R. F. Carter et al., "The Particulate Material Formed by the Oxidation of Plutonium," AERO CONF./8 (1960).

Chapter XIII

CONCLUSIONS

1. The basic scope of this Final Safety Analysis Report is the same as that given by ANL-6271, the safety analysis report and addenda outlining the program and hazards associated with the use of ^{235}U fuel on ZPR-6 and -9 but altered to include the use of plutonium as fuel on these facilities.

2. The ZPR-6 and -9 facilities employ the following safeguards against a nuclear accident and subsequent release of radioactivity to the atmosphere:

a. Employment of highly trained, skillful personnel for judgement and operational functions. The operating experience of ZPR-6 and -9 on uranium-fueled large dilute fast cores since 1963 and 1964, respectively, has provided trained and experienced personnel for the safe operation of fast criticals at the Illinois Site.

b. Strict administrative procedures for controlling loading changes and operation with periodic review and inspection of the operation.

c. The use of well-designed, reliable fail-safe mechanical, and electrical equipment, and interlocks with frequent maintenance and testing procedures.

d. Construction of assemblies which have only inherent calculated negative power coefficients.

e. The use of the following systems for maintaining containment:

(1) Stainless steel jacketing for plutonium and sodium.

(2) Reinforced concrete cells for blast resistance and containment.

(3) Steel confinement shell as a backup to confine any particular leakage from the cells.

(4) Emergency venting system consisting of sand filters, HEPA filters and 46-m stack to aid in maintaining cell integrity in the event of pressure buildup in cells.

(5) Argon gas purge system for quenching large metal fires.

3. A maximum credible accident which postulates a prompt critical excursion due to criticality on intermediate table speed with only the high-level trips operating results in a 200°C increase in fuel temperature

and 2.3×10^{17} fissions total energy release for the 50-liter core and 58°C and 4.6×10^{17} fissions for the nominal 3500 liter core, but there is no release of activity to the atmosphere since the fuel elements remain intact.

4. The MCA will be used as a guide (as was the case in ANL-6271) to determine the type of assemblies which may be constructed on ZPR-6 and -9. Only those cores which result in an MCA with maximum fuel temperatures less than 200°C will be constructed.

5. As an aid in establishing the adequacy of the containment, and facility design a class of accidents (designated Design Basis Accidents) which are considered incredible and virtually impossible is hypothesized and analyzed. The DBA in essence shows the type of accident which may be handled by the reactor containment. In addition to the highly improbable events postulated for an MCA, the complete failure of all nuclear instrumentation and safety circuits, total operator disregard for the status of the reactor, and continuous addition of reactivity is hypothesized for the DBA. The DBA results in 2.7×10^{20} fissions, the burning of 60 kg of plutonium, and the vaporization and rapid oxidation of about 22 kg of sodium. The reinforced concrete cell, confinement shell, and emergency exhaust system withstand this type of accident with only minimal release (6 mg) of plutonium to the atmosphere. The release of 6 mg of plutonium results in inhalation doses no greater than 7.4×10^{-4} times mpbb (maximum permissible body burdens) for Type C plutonium (30% ^{240}Pu) to an observer at 225 m and 4×10^{-4} mpbb at 1300 m (site boundary) using pessimistic atmospheric conditions of 1 m/sec wind velocity and assuming the average worst meteorological conditions. These calculated body burdens are about a factor of 150 less than the 0.1 mpbb criterion for maximum permissible body burdens under accident conditions for fast plutonium fueled ZPR's. To calculate the hazards associated with the release of the gaseous fission products to the atmosphere, it is assumed that only the fission products in the portion of the core which has melted and penetrated the stainless steel jacketing are released to the cell and to the atmosphere. Under the hypothesized DBA conditions 60 kg of plutonium (220 kg of fuel alloy) melt and penetrate their stainless steel jacketing. The fraction of total fissions in the 60 kg of plutonium considering the power distribution is 20%. Thus the products from $0.20 \times 2.7 \times 10^{20}$ or 5.4×10^{19} fissions are considered released to the atmosphere. This results in doses to the thyroid of no greater than 44 Rem to an observer at 225 m and 20 Rem to an observer at 1300 m. The gamma dose to the total body due to the noble gas fission products would be 0.61 Rem at 225 m and 0.44 Rem at 1300 m.

6. On the basis of the foregoing arguments, it is concluded that the ZPR-6 and -9 facilities may be operated utilizing plutonium fuel at the Argonne, Illinois Site of Argonne National Laboratory without undue risk to the health and welfare of the general public in the vicinity of the Laboratory and to Laboratory personnel.

APPENDIX A

Radiological Hazards of Plutonium and Gaseous Fission Products

1. Introduction

The hazards associated with the release of plutonium and gaseous fission products as a consequence of a nuclear accident are analyzed. It is assumed for this analysis that 1 mg of plutonium is released from the exhaust stack and that the gaseous fission products from 5.4×10^{19} fissions are available for release from the stack. Calculations of the radiological hazards due to inhalation and ingestion of plutonium, inhalation of radioiodine, and the external dose from noble gases are presented.

2. Meteorological Attenuation Factors

If a nuclear accident were to occur releasing radioactivity in the cell, particles and gases from the burning fuel would be exhausted from the reactor cell, through the emergency exhaust system, and out the 46 meter high stack. For simplicity, the following assumptions are made for the radiological hazards analysis.

1. The activity emanates from a point source at an elevation of 46 m above the base of the stack.

2. The people downwind remain in fixed positions from the start of the release of activity until the last trace of effluent has passed by them.

3. The meteorological conditions remain unchanged during the period of exposure.

4. There is no depletion of active material in the cloud by ground deposition.

The time-integral of the ground-level air concentration of activity, χ (curie-sec/m^3), at a distance D meters downwind and y meters crosswind from a point source which emits a total of Q curies of activity at an elevation of h meters can be estimated by[1]

$$\chi = \left[Q/\pi\bar{u}\sigma_y\sigma_z\right]\left[\exp(-y^2/2\sigma_y^2 + h^2/2\sigma_z^2)\right] \qquad (1)$$

where $\bar{u}$ is the average windspeed and σ_y and σ_z are functions of D. At zero crosswind distance (y = 0), differentiating Eq. (1) under the assumption that $\sigma_y = a\sigma_z$, gives the distance of maximum concentration at the point downwind for which $\sigma_z = h/\sqrt{2}$. The maximum concentration is then given by

$$\chi_{max} = Q\sqrt{2}/\pi\sigma_y\bar{u}eh \qquad (2)$$

where σ_y is evaluated at the distance D_{max} for which $\sigma_z = h/\sqrt{2}$. The quantity $\chi_{max}\,\bar{u}/Q$ was evaluated for Pasquill's six standard atmospheric stability conditions using values for σ_z and σ_y given in Figs. A-1 and A-2. The results are shown in Table A-1.

Equation (1) describes the ground level concentration for the release of activity over a long term (greater than a few minutes). For short term or instantaneous releases, the ground level concentration will be greater than that predicted by Eq. (1). Instantaneous release concentrations can be obtained by replacing σ_y and σ_z in Eqs. (1) and (2) by σ_{yI} and σ_{zI}, dispersion coefficients for instantaneous releases. Reference 1 has summarized recent experiments for instantaneous releases and suggests the use of the values for σ_{yI} and σ_{zI} given in Table A-2. The maximum ground level concentrations due to instantaneous release for stability conditions A and D have been evaluated and are presented in Table A-1.

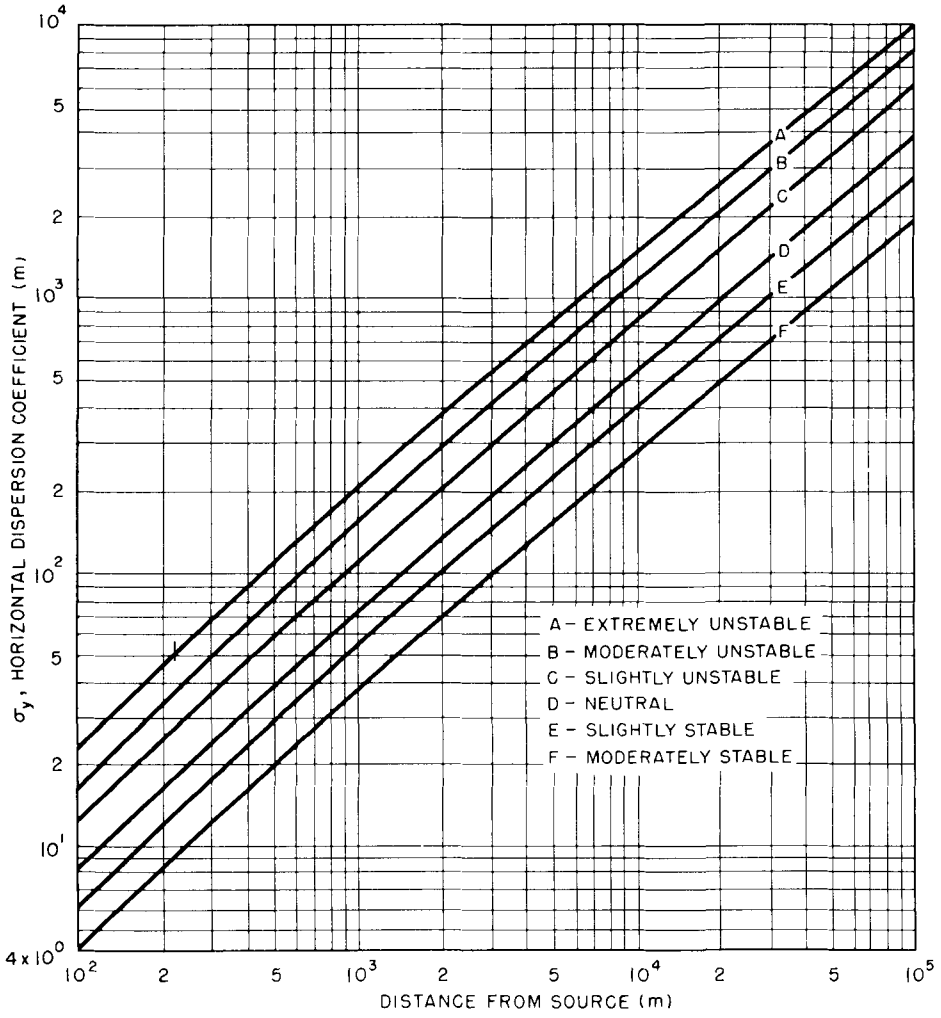


Fig. A-1. Lateral Diffusion, σ_y, vs Downwind Distance from Source for Pasquill's Turbulence Types. (From Ref. 1.)

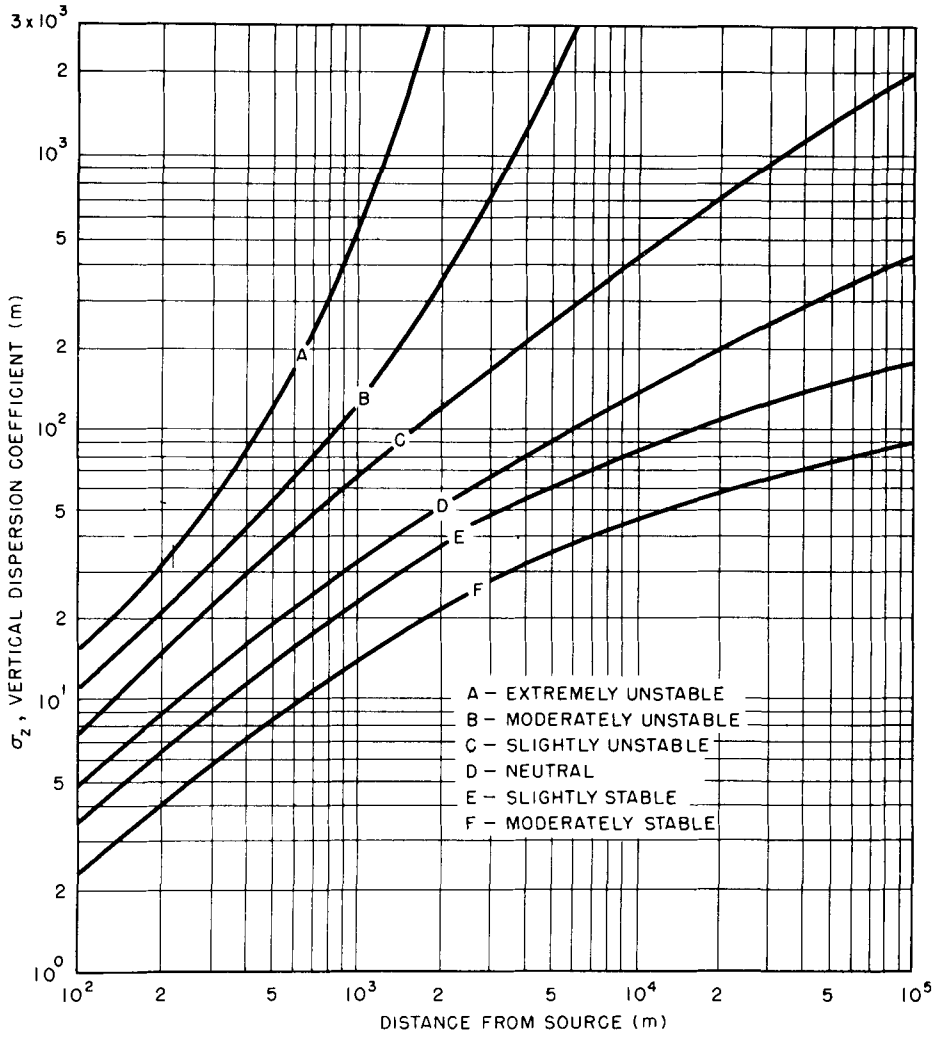


Fig. A-2. Vertical Diffusion, σ_z, vs Downwind Distance from Source for Pasquill's Turbulence Types. (From Ref. 1.)

Table A-1

METEOROLOGICAL ATTENUATION FACTORS

Pasquill Stability Class	Distance Downwind (m) to Maximum Concentration		$\chi_{max}\,\bar{u}/Q$ (m^{-2})	
	Continuous Release	Instantaneous Release	Continuous Release	Instantaneous Release
A	2.3×10^2	2.8×10^2	8.0×10^{-5}	1.44×10^{-4}
B	3.1×10^2		7.5×10^{-5}	
C	4.5×10^2		6.9×10^{-5}	
D	9.8×10^2	2.1×10^3	5.0×10^{-5}	5.15×10^{-5}
E	1.6×10^3		4.4×10^{-5}	
F	4.0×10^3		2.9×10^{-5}	

Fumigation: $D = 2.3 \times 10^2$ m; $\chi\bar{u}/Q = 7.8 \times 10^{-4}$ (m^{-2})
$D = 1.3 \times 10^3$ m; $\chi\bar{u}/Q = 1.3 \times 10^{-4}$ (m^{-2})

Table A-2

SUGGESTED ESTIMATES FOR σ_{yI} AND σ_{zI}

Parameter	Conditions	D = 100 m	D = 4000 m	Approximate Power Function for 100 m ≤ D ≤ 4000 m
σ_{yI} (m)	Unstable	10.0	300.0	$0.14[D]^{0.92}$
	Neutral	4.0	120.0	$0.06[D]^{0.92}$
	Very Stable	1.3	35.0	$0.02[D]^{0.89}$
σ_{zI} (m)	Unstable	15.0	220.0	$0.53[D]^{0.73}$
	Neutral	3.8	50.0	$0.15[D]^{0.70}$
	Very Stable	0.75	7.0	$0.05[D]^{0.61}$

Fumigation conditions occur where the nocturnal temperature inversion is rapidly dissipated by warming due to solar heating after dawn. Under such conditions, material which had accumulated aloft during the inversion condition is mixed downward uniformly from the ground up to the base of the inversion. The ground-level air concentration may be estimated by[1]

$$\chi_F = Q/\sqrt{2\pi}\ \bar{u}\sigma_y h' \qquad (3)$$

where h' is the height of the inversion layer. The height h' can be estimated[2] as $h + \Delta h + 2\sigma_z$. (In this formula, Δh is the height above the stack to which the effluent rises due to pressure in the exhaust system. The value of $\Delta h = 0$ will be assumed here.) The stability condition F was used to obtain values for the dispersion coefficients for use in Eq. (3). At a downwind distance of 225 m, a value of 7.8×10^{-4} m^{-2} was computed for $\chi_F \bar{u}/Q$.

Nonmeteorological factors which tend to reduce the effective stack height[1,6] and thereby increase ground-level air concentrations have been considered. Downwash, due to eddies in the lee of an isolated stack, does not occur to any appreciable extent (i.e., more than one stack diameter) as long as the wind velocity is less than the stack draft velocity (16 m/sec). In the present case, the minimum stack draft velocity corresponds to above average windspeeds so that any downwash effects would be mitigated by improved atmospheric dispersion. Turbulence generated by obstacles will not cause downwash if the stack is at least $2\frac{1}{2}$ times the height of any structure located within 20 stack lengths. In the present case, the highest obstacle within 20 stack lengths is the steel containment shell around the fast critical assembly cells and the top of this shell is about 14 m above ground level.

Argonne data[4,5] taken at an elevation of 46 m indicates that the average wind speed there is ~5.5 m/sec and that the wind speed is greater than ~1.5 m/sec about 96% of the time. Calms (windspeeds <0.5 m/sec)

occurred only 0.7% of the time; calms of longer than one hour duration occurred only 0.1% of the time; calms of longer than two hours duration occurred only 0.02% of the time. Thus, a windspeed of 1 m/sec represents a conservative value for use with ground level air concentration calculations.

Activity released through the stack may be brought to the ground by rain. The maximum total ground deposition due to washout, W_{max} (curies/m^2), at a distance D meters downwind and zero meters crosswind from a point source which emits a total of Q curies of activity can be estimated by[1]

$$W_{max} = Q/\sqrt{2\pi}\, e\sigma_y D \qquad (4)$$

Ground deposition also occurs under nonprecipitating conditions, however, the maximum dry deposition is roughly an order of magnitude less than the maximum washout deposition.[3] Figure A-3 gives W_{max}/Q as a function of distance downwind for the six Pasquill conditions. It can be seen in Fig. A-3 that $W_{max}/Q < 2.5 \times 10^{-5}\ m^{-2}$ at all distances beyond the site boundaries (~1300 m) for a range of meteorological conditions including precipitating and nonprecipitating weather.

In order to make an estimate of the maximum water contamination hazard, it is assumed that:[4,5]

i) W_{max}/Q is $10^{-5}\ m^{-2}$.

ii) It would take one to two months for this deposited activity to seep into poorly constructed, shallow wells.

iii) It would be diluted by 0.1 m^3 of water per square meter of ground deposition.

The resulting concentration of activity in the ground water is 10^{-4} Q curies/m^3.

3. Plutonium and Uranium Hazards

To estimate the hazards associated with the release of plutonium or uranium to the atmosphere, it is assumed that 1 mg of plutonium or uranium escapes from the exhaust stack and travels downwind. The radiological hazards are estimated for ingestion and inhalation of these contaminants.

Pertinent physical properties of some uranium and plutonium isotopes are presented in Table A-3. Three fuel samples (A, B, and C) of plutonium with different isotropic abundances are considered. Column 6 gives the amount of activity inhaled for each isotope computed with the relation:

$$\text{Amount Inhaled} = [\text{activity/mg}] \times \chi/Q \times BR$$

where activity/mg is given in column 5, χ/Q (sec/m^3) was evaluated for the fumigation condition at 225 m downwind, at a windspeed of 1 m/sec, and BR is the active breathing rate given as 3.47×10^{-4} m^3/sec. Column 7 gives the concentration of each isotope in the ground water for the deposition conditions discussed in Section 2.

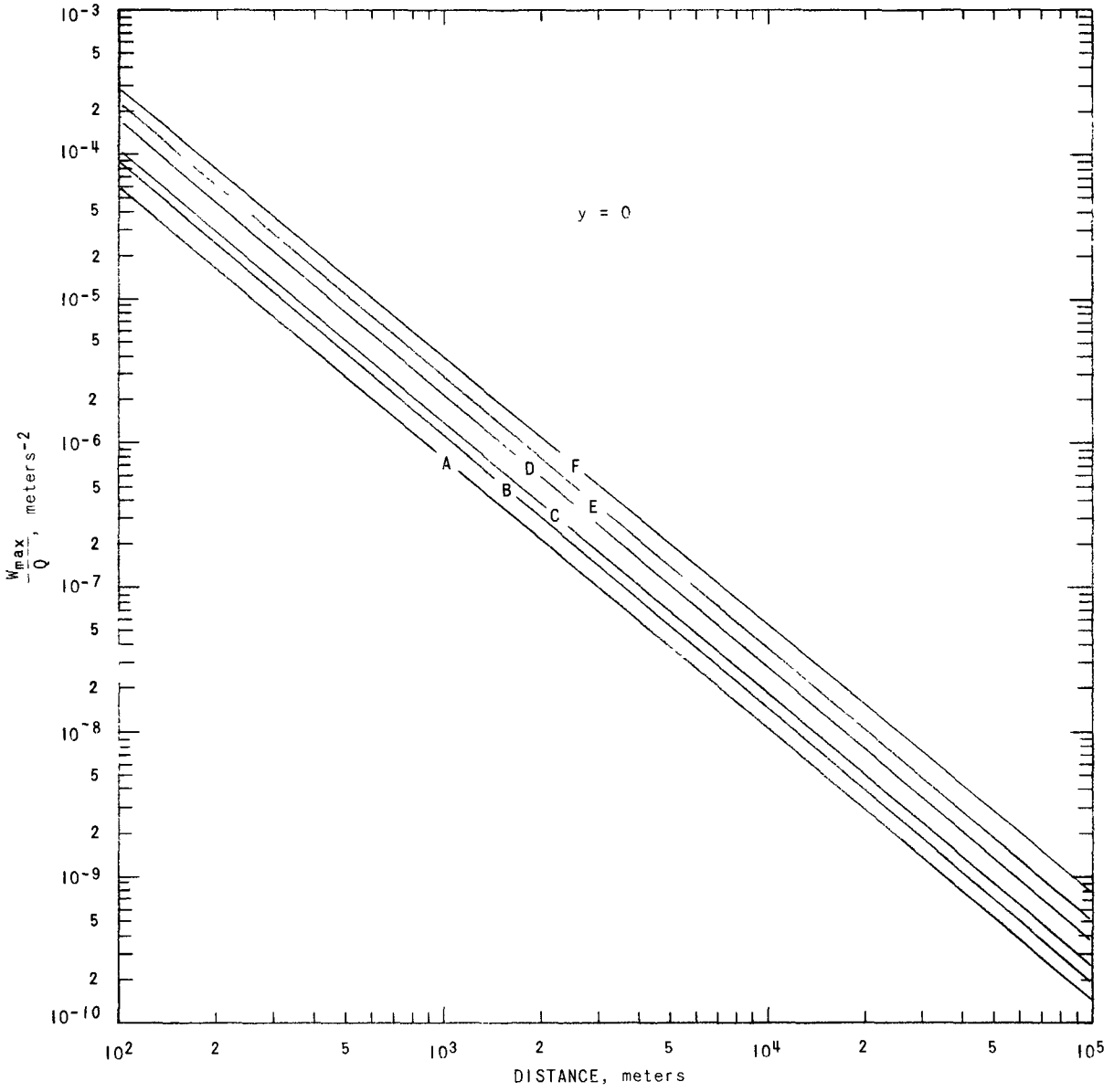


Fig. A-3. Maximum Deposition per Unit Source Strength (in meters^{-2}) from Washout at the Optimum Scavenging Rate as a Function of Distance Downwind (in meters). The crosswind distance is zero.

Table A-3

PHYSICAL CHARACTERISTICS AND CONTAMINATING CONCENTRATIONS FOR A 1 mg RELEASE OF URANIUM AND PLUTONIUM SAMPLES

Isotope	Sample	Half-Life, yr	Specific Activity, Ci/g	Isotopic Composition of Sample, wt %	Activity for 1 mg of Sample, μC	Amount of Isotope Inhaled at 225 m for $\chi/Q = 7.8 \times 10^{-4}$, μC	Concentration of Isotope in Ground Water at 1.3 km for $W/Q = 10^{-5}$, μC/cc
^{232}U		74	20.8	0.004	8.32×10^{-1}	2.25×10^{-7}	8.32×10^{-11}
^{233}U		1.62×10^{5}	9.49×10^{-3}	98.33	9.33×10^{0}	2.53×10^{-6}	9.33×10^{-10}
^{234}U		2.48×10^{5}	6.16×10^{-3}	0.127	7.82×10^{-3}	2.12×10^{-9}	7.82×10^{-13}
^{235}U		7.1×10^{8}	2.15×10^{-6}	0.0102	2.19×10^{-7}	5.92×10^{-14}	2.19×10^{-17}
^{238}U		4.51×10^{9}	3.35×10^{-7}	1.53	5.13×10^{-6}	1.38×10^{-12}	5.13×10^{-16}
^{238}Pu		89	16.9				
	A			-	-	-	-
	B			0.05	8.44×10^{0}	2.29×10^{-6}	8.44×10^{-10}
	C			1.0	1.69×10^{2}	4.57×10^{-5}	1.69×10^{-8}
^{239}Pu		2.44×10^{4}	6.14×10^{-2}				
	A			95.0	5.83×10^{1}	1.58×10^{-5}	5.83×10^{-9}
	B			86.0	5.28×10^{1}	1.43×10^{-5}	5.28×10^{-9}
	C			54.0	3.32×10^{1}	8.97×10^{-6}	3.32×10^{-9}
^{240}Pu		6.76×10^{3}	2.2×10^{-1}				
	A			4.5	9.92×10^{0}	2.68×10^{-6}	9.92×10^{-10}
	B			12.0	2.64×10^{1}	7.13×10^{-6}	2.64×10^{-9}
	C			30.0	6.61×10^{1}	1.78×10^{-5}	6.61×10^{-9}
^{241}Pu		13	1.14×10^{2}				
	A			0.5	5.69×10^{2}	1.54×10^{-4}	5.69×10^{-8}
	B			2.0	2.28×10^{3}	6.16×10^{-4}	2.28×10^{-7}
	C			12.0	1.37×10^{4}	3.71×10^{-3}	1.37×10^{-6}
^{242}Pu		3.79×10^{5}	3.9×10^{-3}				
	A			-	-	-	-
	B			-	-	-	-
	C			3.0	3.40×10^{-8}	1.09×10^{-11}	2.34×10^{-11}

The ingestion hazard can be conservatively estimated if it is considered as resulting from the intake of plutonium or uranium and assuming that they are in the soluble form for which the bone is the critical organ. However, for the inhalation hazard, the insoluble form of uranium, for which the lung is the critical organ, must be considered in addition.

Hazards are evaluated in terms of the maximum permissible body burden, mpbb, and the maximum permissible concentration in water $(MPC)_W$, and in air $(MPC)_a$. The mpbb is defined as that amount of the radionuclide which is deposited in the total body and produces the maximum permissible RBE dose rate to the body organ of reference (bone in our case).[7] The $(MPC)_W$ or $(MPC)_a$ is defined as that concentration of the radionuclide in water (or in air) which if ingested by the standard man as a consequence of occupational exposure for a period of 50 years, would result in the accumulation of the mpbb.[7] The upper limit for single exposure under accident conditions, has been arbitrarily taken as one-tenth of the mpbb.

a. Ingestion Hazard

To determine the extent of the ingestion hazard, the concentrations in the ground water given in Table A-3 are compared in Table A-4

with the $(MPC)_W$ values[7] for the most harmful isotopes (^{234}U, ^{235}U, ^{238}U and ^{242}Pu contribute negligible amounts compared to the other isotopes). The estimated maximum concentration of ^{241}Pu (Sample C) which might occur in the ground water is $6.35 \times 10^{-4} \times (MPC)_W$. The $(MPC)_W$ concentration is based on the exclusive continuous ingestion of water (1000 cc/day in food and 1200 cc/day in fluids) containing the maximum permissible concentration in water of the radioisotope under consideration. Assuming that the water intake in the form of food is not contaminated, a person drinking from a contaminated well would ingest ^{241}Pu at the rate:

$$6.35 \times 10^{-4} \times (MPC)_W \times 1200/2200 = 3.11 \times 10^{-4} \times (MPC)_W$$

or 0.03% of the maximum permissible rate for continuous exposure. Summing over the isotopes of each sample gives the total ingestion rate in terms of the $(MPC)_W$ shown in Table A-5. The plutonium Sample C is the worst case with an ingestion rate of about 0.1% of the $(MPC)_W$. One would have to ingest water with a concentration of $1.1 \times (MPC)_W$ for 50 years to receive 10% more than the maximum permissible body burden. Thus, the ingestion hazard is small.

Table A-4

INGESTED ACTIVITY FROM GROUND WATER AT 1.3 km

Isotope	Sample	$(MPC)_W$ for Critical Organ for 168 hr/wk Exposure, $\mu C/cc$	Concentration in Ground Water ÷ $(MPC)_W$
^{232}U		8×10^{-6}	1.04×10^{-5}
^{233}U		4×10^{-5}	2.33×10^{-5}
^{238}Pu		5×10^{-5}	-
	A	-	-
	B	-	1.69×10^{-5}
	C	-	3.38×10^{-4}
^{239}Pu		5×10^{-5}	
	A	-	1.17×10^{-4}
	B	-	1.06×10^{-4}
	C	-	6.69×10^{-5}
^{240}Pu		5×10^{-5}	
	A	-	1.98×10^{-5}
	B	-	5.28×10^{-5}
	C	-	1.32×10^{-4}
^{241}Pu		2×10^{-3}	
	A	-	2.85×10^{-5}
	B	-	1.14×10^{-4}
	C	-	6.85×10^{-4}

Table A-5

INGESTION RATE OF ACTIVITY FROM GROUND WATER AT 1.3 km

Sample	Ingestion Rate	Sample	Ingestion Rate
U	$3.37 \times 10^{-5} \times (MPC)_W$	Pu-B	$2.90 \times 10^{-4} \times (MPC)_W$
Pu-A	$1.66 \times 10^{-4} \times (MPC)_W$	Pu-C	$1.22 \times 10^{-3} \times (MPC)_W$

The ingestion hazard in reality would be less than predicted above. The one or two months it takes for the water to seep into wells far exceeds the time in which the surrounding communities would be warned of the hazard. Drinking water could be supplied to these communities from outside the accident area. In addition, if people were subjected to contaminated ground water for a long term, one must include a diminishing concentration with time in the above calculations due to continuous dilution by water seeping into the wells.

b. Inhalation Hazard

The hazard resulting from inhalation of plutonium or uranium assumed to be in the soluble form can be determined by computing the activity accumulated in the bone which is given by:

$$A(\mu C) = Q \times \chi/Q \times BR \times f_a = [\text{Amount Inhaled}] \times f_a$$

where [Amount Inhaled] is given in Table A-3 and f_a is the fraction of the isotope reaching the organ of reference by inhalation. Values of f_a (Ref. 8) and A are given in Table A-6 along with values of the mpbb. Dividing A by the mpbb for each isotope and summing over all the isotopes of a sample gives the fraction of the total body burden inhaled given in Table A-7. In the worst case, plutonium Sample C, 0.12% of the mpbb is accumulated in the bone. This is to be compared to the upper limit of 10% of the mpbb set for accident conditions.

Table A-6

INHALED ACTIVITY DUE TO A 1 mg RELEASE OF PLUTONIUM AND URANIUM AT 225 m

Isotope	Sample	f_a Fraction of Isotope Reaching Organ of Reference by Inhalation	mpbb Maximum Permissible Burden in Total Body, μC	A Activity Accumulated in Organ of Reference, μC	(A/mpbb)
^{232}U		0.028	0.01	6.30×10^{-9}	0.63×10^{-6}
^{233}U		0.028	0.05	7.08×10^{-8}	1.41×10^{-6}
^{238}Pu		0.2	0.04	-	-
	A			-	-
	B			4.57×10^{-7}	0.11×10^{-4}
	C			9.14×10^{-6}	2.71×10^{-4}
^{239}Pu		0.2	0.04	-	-
	A			3.16×10^{-6}	0.79×10^{-4}
	B			2.86×10^{-6}	0.72×10^{-4}
	C			1.79×10^{-6}	0.46×10^{-4}
^{240}Pu		0.2	0.04	-	-
	A			5.37×10^{-7}	0.18×10^{-4}
	B			1.43×10^{-6}	0.36×10^{-4}
	C			3.57×10^{-6}	0.89×10^{-4}
^{241}Pu		0.2	0.9	-	-
	A			3.08×10^{-5}	0.34×10^{-4}
	B			1.24×10^{-5}	0.14×10^{-4}
	C			7.41×10^{-4}	8.24×10^{-4}

Table A-7

FRACTION OF THE TOTAL BODY BURDEN INHALED FOR URANIUM AND PLUTONIUM AT 225 m

Sample	Fractional mpbb Inhaled	Sample	Fractional mpbb Inhaled
U	2.04×10^{-6}	Pu-B	1.33×10^{-4}
Pu-A	1.31×10^{-4}	Pu-C	1.23×10^{-3}

In the case of uranium, the insoluble form of the sample and its effect on the lung must be considered. It is assumed[8] that 25% of the insoluble compound is immediately exhaled, $62\frac{1}{2}\%$ is deposited in the lungs and upper respiratory passages and subsequently swallowed in the first 24 hours and the remaining $12\frac{1}{2}\%$ is retained in the lungs with a half-life of 120 days. The amount retained in the lungs can be compared with the equilibrium amount of radioisotope (E) accumulated in the lungs after a continuous inhalation of air containing the $(MPC)^{Lung}$ of the radioisotope. After continuous inhalation of contaminated air for more than about seven effective half-lives (840 days) the rate at which a radioisotope is retained in the lungs will be equal to the rate at which the retained radioisotope leaves the lungs, i.e.,

$$0.693 \times E/120 = 0.125 \times 1 \times 10^{7} \times (MPC)_a^{Lung}.$$

The factor 1×10^{7} represents the volume of air breathed in one day (cc/day) for eight hours at the active breathing rate. The $(MPC)_a^{Lung}$ is selected for 40 hours per week. It is found that:

$$E = 2.17 \times 10^{8} \times (MPC)_a^{Lung}.$$

The amount of activity retained in the lungs due to inhalation (0.125 times the amount inhaled in Table A-3), and the equilibrium amount (E) due to inhalation of the $(MPC)_a^{Lung}$ is given in Table A-8 along with the ratio of these two quantities. It is noted that the total amount accumulated from the 1 mg release is 0.002% of the equilibrium amount, a small hazard compared to the release of the soluble form of the material.

Table A-8

ACTIVITY INHALED DUE TO A 1 mg RELEASE OF URANIUM (INSOLUBLE FORM) AT 225 m

Isotope	X Amount of Activity Retained Lungs, μC	$(MPC)_a$ for Lung Based on 40 hr/wk, μC/cc	E Equilibrium Activity Accumulated in Lung due to Inhalation of Air Containing the $(MPC)_a$, μC	X/E
^{232}U	2.90×10^{-8}	3×10^{-11}	6.51×10^{-3}	4.29×10^{-6}
^{233}U	3.16×10^{-7}	1×10^{-10}	2.17×10^{-2}	1.46×10^{-5}
			Total	1.89×10^{-5}

The inhalation hazard outside the Laboratory Site can be evaluated by using the meteorological attenuation factor, $\chi\bar{u}/Q$, for the fumigation condition at the nearest site boundary, a distance of 1300 m from the stack. The value of $\chi\bar{u}/Q$ is 1.3×10^{-4} at this point. The amount of contaminants inhaled can be obtained by multiplying the previous results by a factor of 0.167 (ratio of meteorological attenuation factors, $1.3 \times 10^{-4}/7.8 \times 10^{-4}$). Thus, for the soluble form of the contaminants, 0.02% of the mpbb is accumulated in the bone for the worst case, plutonium Sample C.

4. Gaseous Fission Products

The hazards associated with the inhalation of radioiodine and the external body dose from noble gases are analyzed for the release of fission products from 5.4×10^{19} fissions. The model for the release of the gaseous fission products from the stack as a function of time is based on the DBA analysis (Chapter 12) of the pressure buildup in the reactor cell and gas release from the cell during an accident.

a. Gaseous Fission Product Release

In conjunction with the DBA analysis, the following assumptions have been made concerning the fission products:

i) All fissions occur at time six seconds (accident time scale when the reactor goes prompt critical).

ii) Fission products from 5.4×10^{19} fissions are released to the cell instantly at time 13 seconds (when the fuel melts and the cladding fails).

iii) The fission products are spread uniformly throughout the cell upon release.

iv) Only the transient time from cell to stack is considered.

v) There is no filtering action by the sand filter because we deal with only gaseous fission products.

vi) The receptor remains in a fixed position downwind until all gases are exhausted from the stack. Thus, only the total number of gaseous fission product atoms released from the stack is of concern. (Rates of release from the stack are not considered.)

The time scale in the DBA analysis is divided into small time intervals for which pressures, gas volumes, and temperatures in the cell are computed. The present analysis makes use of the following parameters:

$\Delta W(t_i)/W(t_i) \equiv F_i$, the fraction of cell air released from the cell during time interval t_i.

$\overline{P_i}$ — the average pressure (psig) in the cell during time interval t_i.

$T_i = 140.4/\overline{P_i}$ — the time spent by the fission products in the piping and filter section before being released from the stack.

$^{89}Se\ (t_i)$ — the average number of ^{89}Se atoms during time interval t_i after instantaneous fission at time 6 seconds. (A similar definition holds for all fission product atom numbers.) Values were derived from data given in Ref. 9.

As an example, consider the ^{89}Kr fission chain (decay half-lives in parentheses).

$$^{89}Se\ (2.0\ s) \longrightarrow {}^{89}Br\ (4.4\ s) \begin{cases} \xrightarrow{11.8\%} {}^{88}Kr\ (2.8\ h) \\ \xrightarrow{88.2\%} {}^{89}Kr\ (3.2\ m) \end{cases}$$

The calculation proceeds in the following manner:[9]

1. At the end of the first time interval t_1, the fraction of air released from the cell during the time t_1 is given by F_1. The number of atoms in the ^{89}Kr chain released from the cell is:

$$F_1[^{89}Se(t_1) + {}^{89}Br(t_1) + {}^{89}Kr(t_1)].$$

The number remaining in the cell is:

$$[1 - F_1][^{89}Se(t_1) + {}^{89}Br(t_1) + {}^{89}Kr(t_1)].$$

2. The number of atoms of ^{89}Kr released from the stack is given by:

$$F_1 \times 0.882 \times [^{89}Se(t_1 + T_1) + {}^{89}Br(t_1 + T_1)] + F_1[^{89}Kr(t_1 + F_1)].$$

Here it is assumed that the noble gas precursors retained in the particulate filters are released from the stack when they decay to ^{89}Kr and that the gaseous precursors become ^{89}Kr as they leave the stack. These assumptions lead to a somewhat conservative estimate of the total ^{89}Kr atoms released from the stack.

3. At the end of the second time interval, the number of atoms in the ^{89}Kr chain released from the cell is:

$$F_2[1 - F_1][^{89}Se(t_2) + {}^{89}Br(t_2) + {}^{89}Kr(t_2)]$$

The number remaining in the cell is:

$$[1 - F_2][1 - F_1][^{89}Se(t_2) + {}^{89}Br(t_2) + {}^{89}Kr(t_2)]$$

4. The number of atoms of ^{89}Kr released from the stack is given by:

$$F_2[1 - F_1] \times 0.882 \times [^{89}Se(t_2 + T_2) + {}^{89}Br(t_2 + T_2)] + F_2[1 - F_1][^{89}Kr(t_2 + T_2].$$

The calculation proceeds in the manner described above for the first two intervals and continues for time intervals until time six minutes. The total number of atoms of ^{89}Kr released from the stack is given by the sum

$$\sum_{i=1}^{6\ \text{min}} F_i \left[\prod_{j=1}^{i-1} (1 - F_j)\right] \left[0.882 \times \left(^{89}Se(t_i + T_i) + {}^{89}Br(t_i + T_i)\right) + {}^{89}Kr(t_i + T_i)\right].$$

About 10% of the original number of atoms in the chain still remain in the cell after six minutes. The pressure in the cell has reached an asymptotic low value and only 3% of the cell air is being exhausted per 12 sec time interval. Thus, for simplicity, these remaining number of atoms are added to the six minute sum to give the total number of atoms of ^{89}Kr released from the stack for all time following the accident. (This procedure conservatively overestimates the total number of atoms released by a few percent.) The release of radioiodine is treated in the same manner as the noble gas release. The results are given in Table A-9.

Table A-9

GASEOUS FISSION PRODUCTS RELEASED FROM THE STACK DURING AN ACCIDENT[a]

Isotope	Number of Atoms Released from Stack (5.4×10^{19} Fissions)	Isotope	Number of Atoms Released from Stack (5.4×10^{19} Fissions)
^{83}Kr	2.938×10^{17}	^{137}Xe	2.051×10^{18}
^{85m}Kr	7.031×10^{17}	^{138}Xe	2.060×10^{18}
^{87}Kr	1.386×10^{18}	^{139}Xe	5.703×10^{17}
^{88}Kr	1.922×10^{18}	^{140}Xe	1.543×10^{17}
^{89}Kr	1.549×10^{18}	^{131}I	1.582×10^{18}
^{90}Kr	6.169×10^{17}	^{132}I	2.363×10^{18}
^{91}Kr	1.357×10^{17}	^{133}I	3.535×10^{18}
$^{135}Xe_I$	2.650×10^{18}	^{134}I	4.151×10^{18}
$^{135}Xe_{II}$	1.577×10^{18}	^{135}I	2.978×10^{18}

[a] These atoms numbers are a result of the gaseous fission product release model described in Section a.

b. External Dose from Noble Gases

The external dose from noble gases is evaluated for the fumigation condition at distances 225 and 1300 m downwind from the stack and at a windspeed of 1 m/sec. At close distances to the stack, it is necessary to consider the dose due to β emission and γ emission separately. The range of β particles in air is approximately 2 m. Thus, the dose from β particles can be computed assuming the receptor submerged in a semi-infinite cloud of noble gases with a cloud concentration given by the ground level concentration for the fumigation condition. However, the range of γ rays in air is large compared to the dimensions of the cloud at 225 and 1300 m. It is necessary in both cases to consider the dose received from the entire cloud.*

The dose from β emission, ${}_{\beta}D_{\infty}$ (Rem), can be computed with the following formula,[1]

$$ {}_{\beta}D_{\infty} = 0.23 \times \overline{E}_{\beta} \times \psi $$

where $\overline{E}_{\beta}$ is the effective energy of the β particle (obtained from Ref. 10), ψ is the ground level noble gas concentration, $\psi = [\lambda N_0/3.7 \times 10^{10}] \times e^{-\lambda x/\bar{u}} \times [\chi/Q]$ curie-sec/m^3, N_0 is the number of noble gas atoms released from the stack, Table A-9, and λ is the noble gas atom decay constant.

The doses received from each noble gas isotope are presented in Table A-10 for distances, x, of 225 and 1300 m.

Table A-10

EXTERNAL DOSE FROM NOBLE GAS β EMITTERS[a]

Isotope	Half-life	$\overline{E}_{\beta}$	${}_{\beta}D_{\infty}$ (Rem) 225 m	${}_{\beta}D_{\infty}$ (Rem) 1300 m
^{83}Kr	1.9 h	0.07	9.80×10^{-3}	1.47×10^{-3}
^{85m}Kr	4.4 h	0.24	3.47×10^{-2}	5.55×10^{-3}
^{87}Kr	78.0 m	0.95	9.05×10^{-1}	1.27×10^{-1}
^{88}Kr	2.8 h	0.34	2.10×10^{-1}	3.36×10^{-2}
^{89}Kr	3.2 m	1.33	1.58×10^{1}	4.74×10^{-2}
^{90}Kr	33.0 s	1.07	6.00×10^{-1}	-
^{91}Kr	10.0 s	1.20	0.97×10^{-5}	-
^{135}Xe$_{I}$	15.3 m	0.08	5.87×10^{-1}	4.11×10^{-2}
^{135}Xe$_{II}$	9.2 h	0.30	4.80×10^{-2}	7.6×10^{-3}
^{137}Xe	3.9 m	1.17	1.73×10^{1}	1.21×10^{-1}
^{138}Xe	17.0 m	0.80	4.49×10^{0}	3.14×10^{-1}
^{139}Xe	41.0 s	1.33	1.32×10^{0}	-
^{140}Xe	16.0 s	1.33	2.46×10^{-3}	-
		Total Dose	41.3 (Rem)	0.7 (Rem)

[a]These dose numbers are a result of the gaseous fission product release model described in Section a.

*The range of γ rays in air is considered large compared to the dimensions of the cloud if σ, the cloud dispersion coefficient ($= \sqrt{\sigma_x \sigma_y}$ for isotropic cloud) is greater than $1/\mu$ where μ is the total absorption coefficient in air.

The γ dose received by a receptor at ground level from a finite, isotropic ($\sigma_y = \sigma_z$) cloud centered at h meters above the ground, is given by[1]

$$D_\gamma(x,y,0) = 0.1616\ \mu\mu_a\ \overline{E}_\gamma Q_0(x)(I_1 + kI_2)/\overline{u}\ \text{(Rem)}$$

where

μ_a is the energy absorption coefficient for air (1/m)

μ is the total absorption coefficient for air (1/m)

$k = (\mu - \mu_a)/\mu_a$

$\overline{E}_\gamma$ is the average gamma energy (MeV)

$Q_0(x)$ is the curies of activity at distance x from the stack given by $Q_0(x) = \lambda N_0 e^{-\lambda x/\overline{u}}/3.7 \times 10^{10}$

N_0 is the number of noble gas atoms released from the stack (Table A-9)

$\overline{u}$ is the average windspeed, and

I_1 and I_2 are functions of h and the cloud dispersion coefficient for an isotropic cloud ($\sigma = (\sigma_y \sigma_z)^{1/2}$).

Values of μ_a, μ, k, I_1 and I_2 were taken from the graphs of Figs. A-4, A-5 and A-6. A tabulation of these values and the doses from the γ emitting isotopes is presented in Table A-11.

The external β dose is 41 Rem at 225 m and about 1 Rem at 1300 m. The penetration of low energy β radiation is not great and this dose can be considered as being absorbed by the skin. The maximum permissible dose to the skin is 30 Rem per year or 1500 Rem for 50 years of occupational exposure.[8] If the accident dose limit is taken as 1/10 the lifetime limit (150 Rem), the calculated value of 41 Rem is relatively safe. The actual dose to the skin will be less than 41 Rem due to attenuation by the clothing of the receptor. Using energy dependent factors discussed in Ref. 1 and a clothing thickness of 25 mg/cm^2 (including a 7 mg/cm^2 layer of dead skin), the external β dose is reduced to 31 Rem.

The external γ dose is 0.6 Rem at 225 m and 0.44 Rem at 1300 m. These calculated values are to be compared to the accident dose limit of 25 Rem for the whole body dose.

Table A-11

EXTERNAL DOSE FROM NOBLE GAS γ EMITTERS

Isotope	Average γ Energy $\bar{E}_\gamma$, MeV	Energy Absorption Coefficient for Air μ_a, m^{-1}	Total Absorption Coefficient for Air μ, m^{-1}	$(\mu - \mu_a)/\mu_a$, k	x = 225 m: Descriptive Functions for Cloud I_1	x = 225 m: Descriptive Functions for Cloud I_2	x = 225 m: Dose $D_\gamma(x\,0\,0)$ Rem	x = 1300 m: Descriptive Functions for Cloud I_1	x = 1300 m: Descriptive Functions for Cloud I_2	x = 1300 m: Dose $D_\gamma(x,0,0)$, Rem
^{85m}Kr	0.16	3.3×10^{-3}	1.7×10^{-2}	4.1	0.27	0.27	1.64×10^{-3}	0.40	0.28	2.28×10^{-3}
^{87}Kr	1.19	3.5×10^{-3}	7.5×10^{-3}	1 2	1 10	0.60	$4\,90 \times 10^{-2}$	1.40	0.58	$5\,35 \times 10^{-2}$
^{88}Kr	1.95	3.1×10^{-3}	5.8×10^{-3}	0.9	1.60	0.70	4.38×10^{-2}	2.00	0.68	5.17×10^{-2}
$^{135}Xe_I$	0.45	3.8×10^{-3}	1.2×10^{-2}	2.1	0.50	0.38	1.99×10^{-1}	0.70	0.38	$1\,21 \times 10^{-1}$
$^{135}Xe_{II}$	0.26	3.7×10^{-3}	$1\,4 \times 10^{-2}$	2.9	0.40	0.33	$2\,65 \times 10^{-3}$	0.55	0.32	3.59×10^{-3}
^{138}Xe	1.50	3.3×10^{-3}	6.7×10^{-3}	1.0	1.20	0.62	3.17×10^{-1}	1.60	0.60	2.03×10^{-1}
							Total Dose 0.61 Rem			Total Dose 0.44 Rem

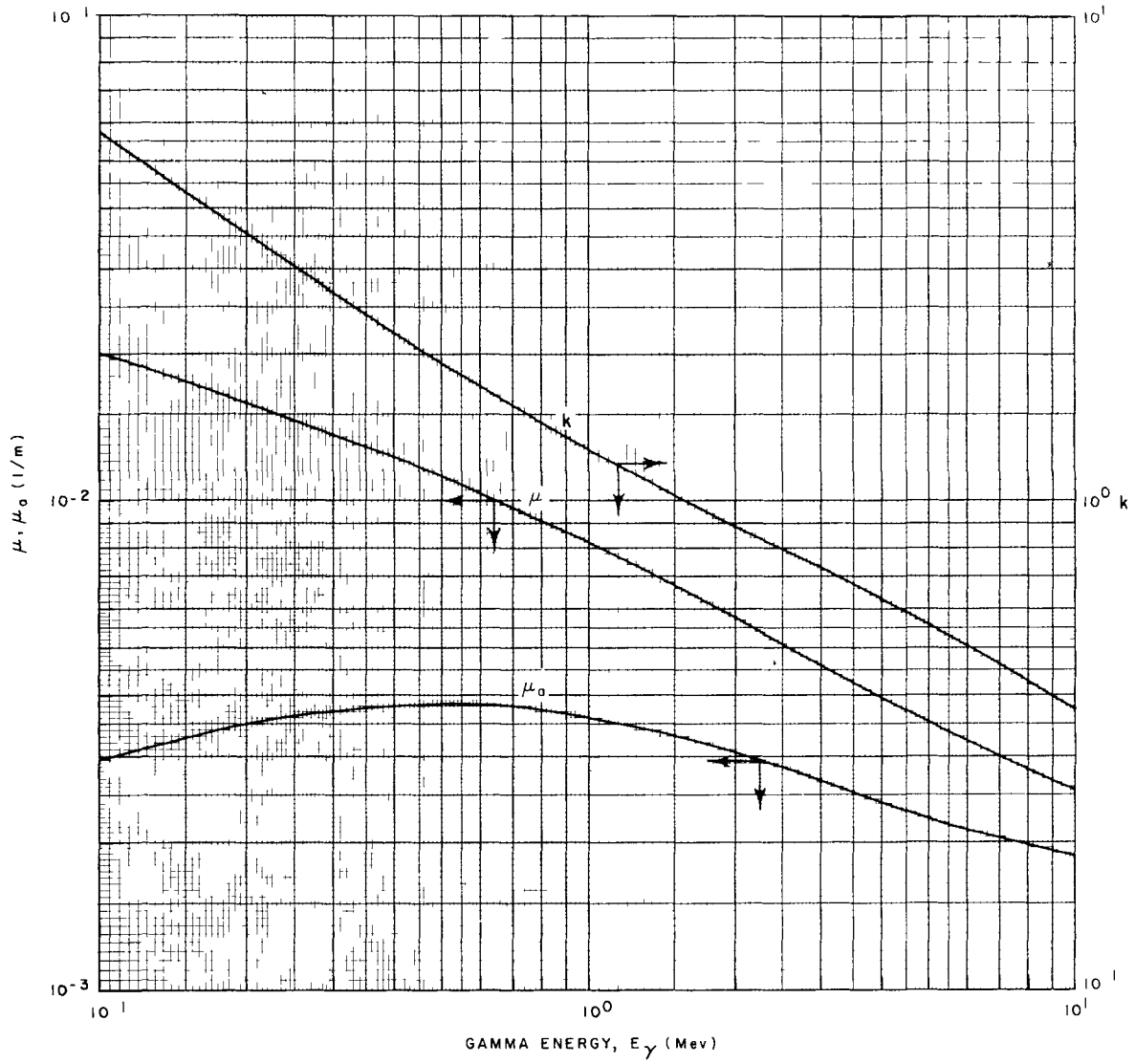


Fig. A-4. Absorption Coefficients and Values of the Buildup Constant for Air at S.T.P. (from Ref. 1)

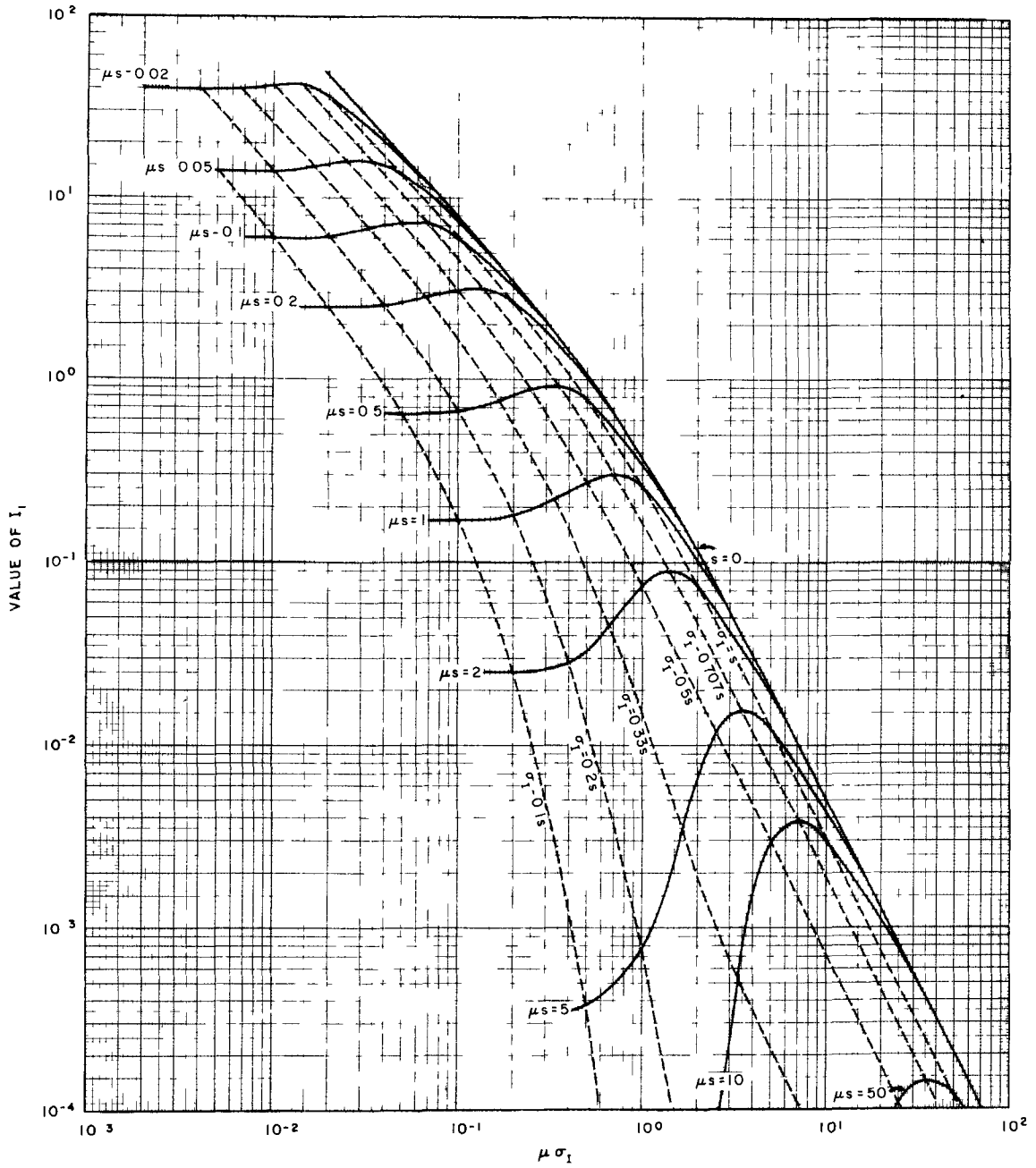


Fig. A-5. Values of the I_1 Integral (from Ref. 1)

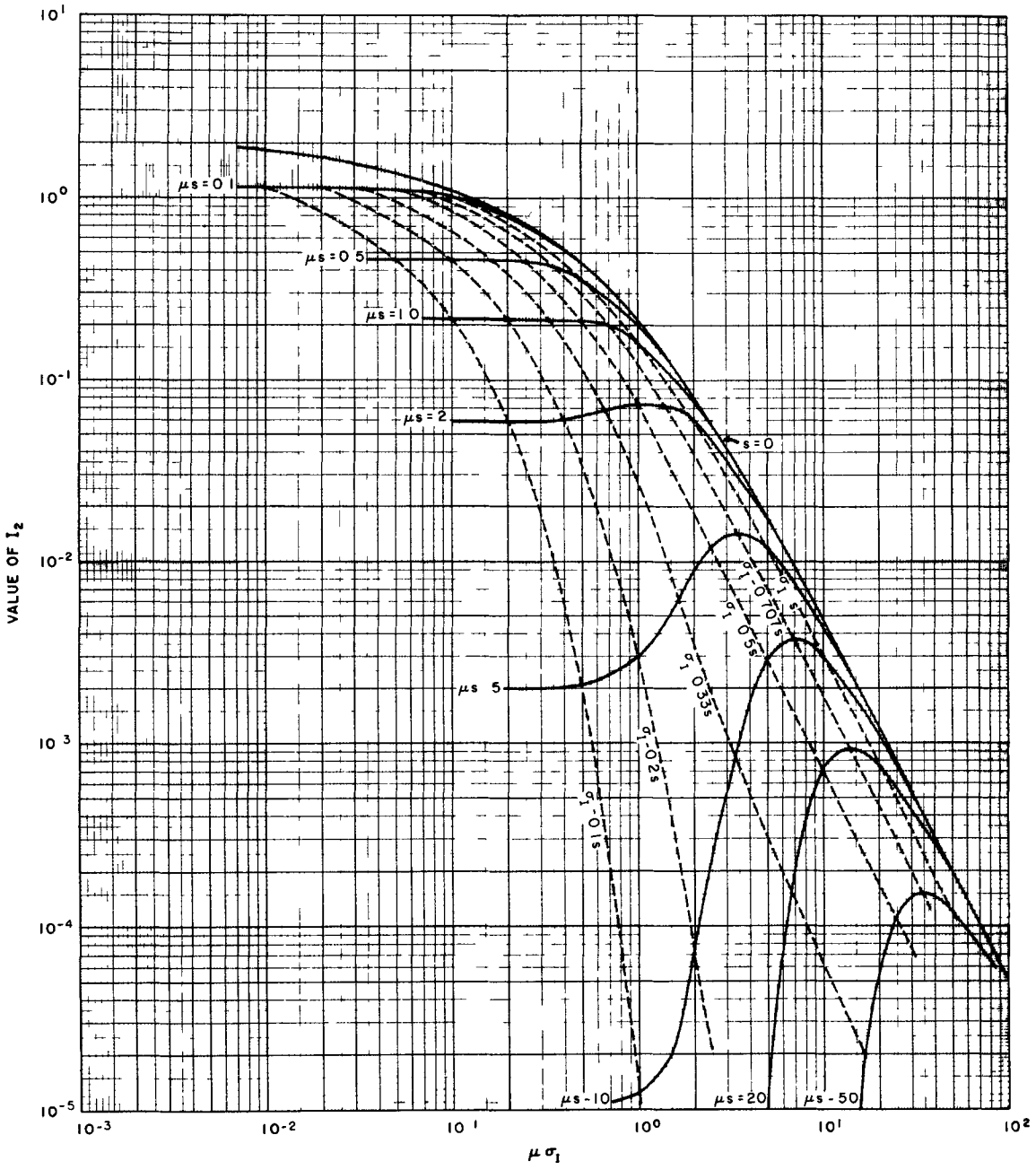


Fig. A-6. Values of the I_2 Integral (from Ref. 1)

c. Inhalation of Radioiodine

The hazard due to inhalation of radioiodine is evaluated for the fumigation condition at 225 and 1300 m downwind from the stack and for a windspeed of 1 m/sec. Under these conditions, the activity, I, inhaled and taken up by the organ of reference (thyroid) is given by

$$I = \lambda N_0 e^{-\lambda x/\bar{u}} \times [\chi/Q] \times BR \times f_a,$$

where

N_0 is the number of iodine atoms released from the stack (Table A-9),

BR is the active breathing rate (3.47×10^{-4} m^3/sec), and

f_a is the fraction of the iodine inhaled which is taken up by the thyroid.

The activity in the thyroid at time t after inhalation is

$$I \times e^{-\lambda_{eff} t}$$

where

$\lambda_{eff} = \lambda + \lambda_b$, the effective decay constant for iodine in the thyroid, and

λ_b = biological decay constant for iodine in the thyroid (138-day half-life).[7]

The dose rate (Rem/sec) to the thyroid is then given by

$$D'(\text{Rem/sec}) = [I \times e^{-\lambda_{eff} t} \text{ dis/sec}] \times \epsilon \text{ (MeV/dis)} \times 1.6 \times 10^{-6} \text{ (ergs/MeV)} \times 10^{-2} \text{ (g-rads/ergs)} \times 1/m \text{ } (g^{-1}),$$

where m is the mass of the thyroid (20 g).

The total dose to the thyroid is obtained by integrating the dose rate over the life span of the receptor (~50 yr). The final expression for the total dose to the thyroid due to inhalation of radioiodine is now

$$D(\text{Rem}) = I \times \epsilon \times 1.6 \times 10^{-8}/\lambda_{eff} m.$$

Using values of ϵ from Ref. 7, the doses to the thyroid were computed for several iodine isotopes and are presented in Table A-12.

Table A-12

DOSES TO THE THYROID FROM INHALATION OF RADIOIODINE

Isotope	Half-life	Energy per Disintegration, ε, MeV/sec	Exposure for Entire Time of Accident (Release Model of Section a)			Exposure for 15 min after Accident (Release Model of Section d)	
			Number of Atoms Released	Dose, Rem, at 225 m	Dose, Rem, at 1300 m	Number of Atoms Released	Dose, Rem,[a] at 225 m
^{131}I	8.05 d	0.23	1.58×10^{18}	1.72×10^{1}	2.87×10^{0}	1.13×10^{17}	1.23
^{132}I	2.3 h	0.65	2.36×10^{18}	7.54×10^{1}	1.13×10^{1}	9.10×10^{16}	2.90
^{133}I	20.8 h	0.54	3.54×10^{18}	9.24×10^{1}	1.48×10^{1}	1.69×10^{18}	44.09
^{134}I	53.0 m	0.82	4.15×10^{18}	16.30×10^{1}	2.12×10^{1}	1.71×10^{18}	67.19
^{135}I	6.7 h	0.52	2.98×10^{18}	7.78×10^{1}	1.24×10^{1}	2.89×10^{18}	75.62
				Total Dose 436 Rem	63 Rem		191 Rem

[a]Evacuation within 15 minutes is only considered possible within the site. Thus, calculations for this release model were not obtained at a distance of 1300 m.

The total dose at 225 m, is 436 Rem. This is higher than the limit of 300 Rem; however, the dose calculated here represents a conservative estimate of the hazard.

d. Doses Received for a Limited Time of Exposure

The model for the release of gaseous fission products discussed in Section a included the assumption that the receptor remained in a fixed position downwind from the stack until all effluent had passed by. Thus, holdup time for the fission product precursors in the particulate filters was not considered. In order to calculate the radioiodine dose received for a limited time of exposure (evacuation from the immediate area is now assumed possible), precursor holdup times in the filters were programmed into the computational procedure of Section a. For each time interval, the number of precursor atoms entering the filter from the cell and the number leaving the filter as radioiodine gas due to decay was computed.

As an example, consider the ^{134}I fission chain:

$$^{134}Sb(48.0\ s) \rightarrow {}^{134}Te(42.0\ m) \rightarrow {}^{134}I(53.0\ m).$$

Using quantities defined in Section a, the calculation proceeds as follows:[9]

1. At the end of the first time interval t_1, the number of atoms in the ^{134}I chain released from the cell is:

$$F_1[{}^{134}Sb(t_1) + {}^{134}Te(t_1) + {}^{134}I(t_1)].$$

The number remaining in the cell is

$$[1 - F_1][{}^{134}Sb(t_1) + {}^{134}Te(t_1) + {}^{134}I(t_1)].$$

2. The number of ^{134}I atoms released from the stack is

$$F_1[{}^{134}I(t_1 + T_1)].$$

3. The number of ^{134}I precursors released to the filter is

$$F_1[^{134}Sb(t_1 + T_1) + ^{134}Te(t_1 + T_1)].$$

4. At the end of the second time interval t_2, the number of atoms in the ^{134}I chain released from the cell is

$$F_2[1 - F_1][^{134}Sb(t_2) + ^{134}Te(t_2) + ^{134}I(t_2)].$$

The number remaining in the cell is

$$[1 - F_2][1 - F_1][^{134}Sb(t_2) + ^{134}Te(t_2) + ^{134}I(t_2)].$$

5. The number of ^{134}I atoms released from the stack is

$$F_2[1 - F_1][^{134}I(t_2 + T_2)] + P_2[\text{number of precursor atoms in the filter during } t_2]$$

where P_2 is the fraction of the precursors decaying to ^{134}I during t_2.

The total number of ^{134}I atoms released from the stack during the first fifteen minutes of the accident is given by

$$\sum_{i}^{15\ \text{min}} \left[F_i \left[\prod_{j=1}^{i-1} (1 - F_j) \right] \left[^{134}I(t_1 + T_1) \right] + P_i \left[\begin{array}{c} \text{number of precursor atoms} \\ \text{in the filter during } t_i \end{array} \right] \right].$$

The results of these calculations and the radioiodine dose received during 15 minutes after the start of the accident are shown in Table A-12. The dose from all iodine isotopes is 191 Rem.

Considering holdup times in the filters for noble gas precursors would not decrease the external dose appreciably since the precursors have short half-lives compared to the 15 minute exposure interval (essentially all the ^{89}Kr and ^{137}Xe is released from the stack within 15 minutes).

e. Doses Received for a Noninstantaneous Fuel Fire

The model for the release of gaseous fission products discussed in Section a included the assumption that the fission products are released to the cell instantly at 13 seconds (time at which molten fuel is first assumed to be exposed to air and burning starts). This assumption is equivalent to describing the burning of fuel and subsequent release of fission products as resulting from an instantaneous fuel fire. If this assumption is relaxed in favor of a continuous burning rate, the doses received from fission product gases are considerably decreased.

In addition to precursor holdup times in the filters (described in Section d) a uniform burning rate for a period of one hour was programmed into the computational procedure of Section a. Using the ^{89}Kr chain as an example and the quantities described in Section a, the amount of ^{89}Kr chain released to the cell by the fire during time interval Δt_j is given by

$$[\Delta t_j/3600] \times [^{89}Se(t_j) + {}^{89}Br(t_j) + {}^{89}Kr(t_j)].$$

The amount of ^{89}Kr chain available for release from the cell during t_j is now given by

$$\sum_{i=1}^{j} \Delta t_i/3600 \left[\prod_{k=i}^{j-1} [1 - F_k]\right] \left[^{89}Se(t_j) + {}^{89}Br(t_j) + {}^{89}Kr(t_j)\right].$$

For each time interval, the precursor and gas atom inventory was computed for atoms in the cell and filters, and atoms released from the stack. The calculational procedure is essentially the same as described in Section d. The total noble gas and radioiodine gas atoms released from the stack during the 15 minutes after the start of the accident are given in Tables A-13 and A-14 along with doses for each isotope.

Table A-13

EXTERNAL DOSE FROM NOBLE GASES FOR A ONE HOUR FUEL BURNING RATE AND 15 min EXPOSURE TIME (RECEPTOR AT 225 m DOWNWIND OF STACK)

Isotope	Number of Atoms Released from Stack during 15 min	βD_∞ Dose, Rem
^{83}Kr	1.36×10^{15}	4.51×10^{-5}
^{85m}Kr	1.11×10^{17}	5.60×10^{-3}
^{87}Kr	2.19×10^{17}	1.45×10^{-1}
^{88}Kr	3.13×10^{17}	3.40×10^{-2}
^{89}Kr	7.88×10^{16}	8.10×10^{-1}
^{90}Kr	7.60×10^{15}	7.20×10^{-3}
^{91}Kr	1.07×10^{15}	7.70×10^{-8}
$^{135}Xe_I$	2.96×10^{16}	6.50×10^{-3}
$^{135}Xe_{II}$	6.43×10^{16}	1.97×10^{-4}
^{137}Xe	1.25×10^{17}	1.06×10^{0}
^{138}Xe	2.62×10^{17}	5.80×10^{-1}
^{139}Xe	7.92×10^{15}	1.80×10^{-2}
^{140}Xe	1.40×10^{15}	2.24×10^{-5}
		Total Dose 2.7 Rem

Table A-14

RADIOIODINE DOSE FOR A ONE HOUR FUEL BURNING RATE AND 15 min EXPOSURE TIME (RECEPTOR AT 225 m DOWNWIND OF STACK)

Isotope	Number of Atoms Released from Stack during 15 min	Dose, Rem, to Thyroid
^{131}I	1.99×10^{16}	0.22
^{132}I	1.51×10^{16}	0.48
^{133}I	2.92×10^{17}	7.67
^{134}I	2.79×10^{17}	10.92
^{135}I	4.94×10^{17}	13.23
		Total Dose 32.5 Rem

The external β dose from the noble gas atoms is 2.7 Rem. The dose would not increase significantly for exposure times greater than 15 minutes since almost all of the ^{89}Kr and ^{137}Xe chains have decayed during the first 15 minutes following the accident. The radioiodine dose to the thyroid is now 32.5 Rem, a factor of 10 below the accident limit.

REFERENCES

1. Meteorology and Atomic Energy 1968, USAEC, Division of Technical Information (1968).
2. H. Moses (private communication).
3. F. A. Gifford, Jr., "Atmospheric Dispersion Calculations Using the Generalized Gaussian Plume Model," Nuclear Safety 2(2), 56-58 (December 1960).
4. H. Moses and J. H. Willett, "Five-Year Climatological Summary, July 1949-June 1954," ANL-5592.
5. W. Y. Kato, G. J. Fischer, and L. R. Dates, "Safety Analysis Report, Argonne Fast Critical Facility (ZPR-VI)," ANL-6271 (December 1963).
6. H. Moses, G. H. Strom, and J. E. Carson, "Effects of Meteorological and Engineering Factors on Stack Plume Rise," Nuclear Safety 6(1), 1-19 (1964).
7. "Recommendation of the ICRP; Report of Committee II on Permissible Dose for Internal Radiation," Pergamon Press (1964).
8. "Report of ICRP Committee II on Permissible Dose for Internal Radiation," Health Physics 3 (1960).

9. C. D. Swanson and E. M. Bohn, "GASOUT--The Code Used to Calculate Gaseous Fission Product Release for a ZPR-6 and -9 Design Basis Accident," ANL-7534 (January 1970).

10. Radiological Health Handbook, PB-121784R, U.S. Department of Health, Education, and Welfare (1960).

APPENDIX B

Sand Filter Efficiency Tests

Aerosol filtration tests have been carried out on a test column as shown in Fig. B-1 loaded with 30 in. of 20 to 50-mesh Idaho sand, 21 in. of 4 to 16-mesh aggregate, and completion of the nominal 15-ft depth with larger aggregate.

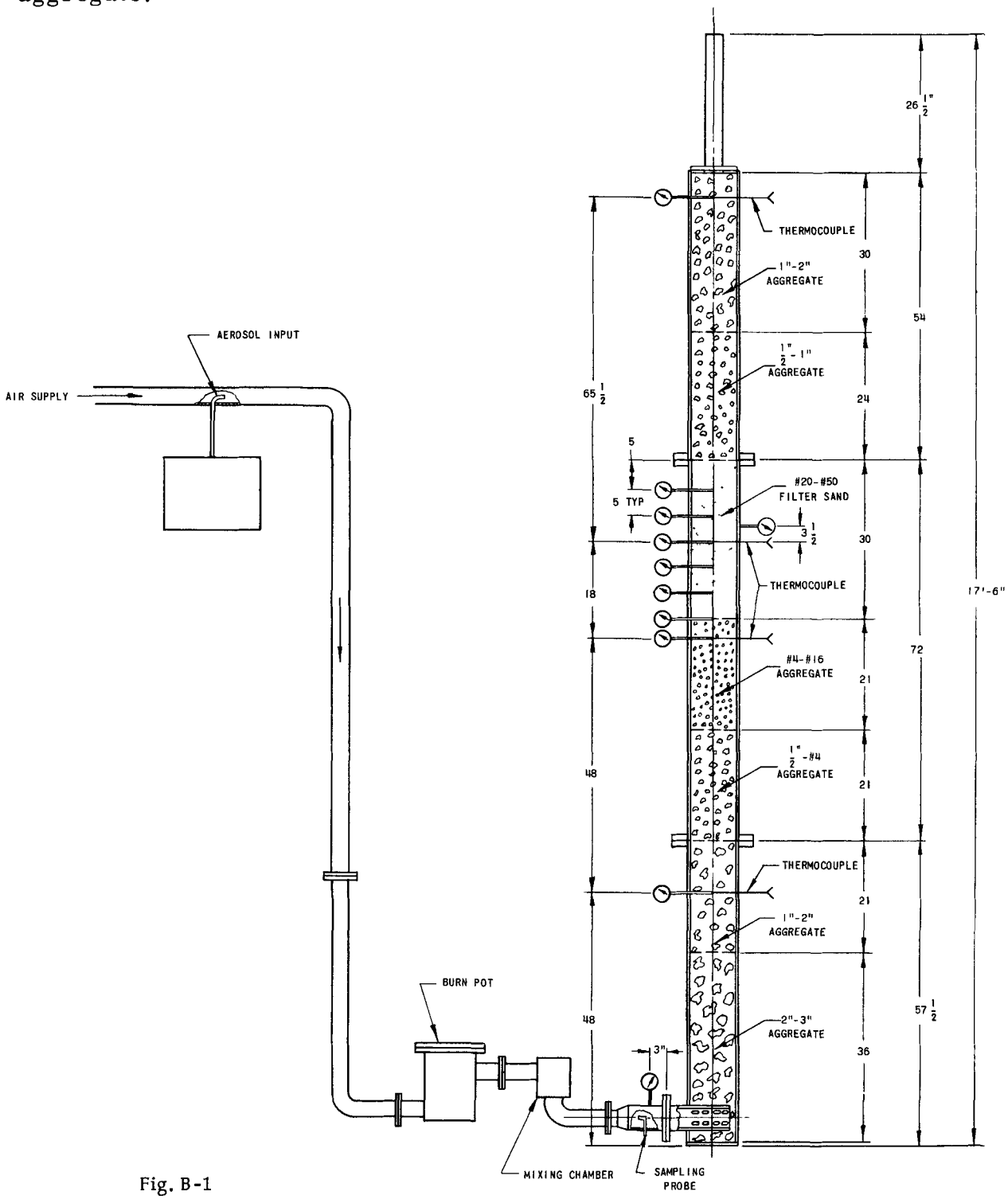


Fig. B-1
Sand Filter Test Column

1. Fluorescein Aerosol Penetrations Tests

Loading No. 1 of Batch No. 2 of Idaho sand was received August 4, 1967. Connecticut 4 to 16-mesh material was used. Pressure versus flow tests for this loading indicate 1210 cfm/psi per 12-ft diameter sand filter.

Fluorescein aerosol was generated from a 0.5% solution of fluorescein-free acid in ethanol. A standard collision atomizer nozzle at a generating pressure of 30 psig followed by an impactor with five impaction holes, each 1.5 mm in diameter, was used. This provided a solid submicron-size test aerosol with a lower detection limit by fluorescence analysis of about 0.0001 μg. Electron microscope photographs of the aerosol were also made. The rate of input of fluorescein aerosol at the upstream sampling point varied (with two exceptions) within a factor of two of 700 μg/min. Samples were drawn simultaneously from upstream and downstream sampling points using Type AA membrane filters in disposable plastic holders. Samples of upstream aerosol were obtained by electrostatic precipitation for electron microscope observations and photographs.

The results of these tests are as shown in Table B-1 and plotted in Fig. B-2.

In these tests the sand filter initially showed penetration of 0.0024, 0.0039, and 0.0014%. However, after more and more tests were run, ending up with a total of twenty-two tests, the penetration of this aerosol at a nominal velocity of 1 fpm decreased to as low as 0.000052%. The penetration at higher velocities goes up to 0.1%. The sand filter should show a penetration of less than 0.01% for this aerosol at test velocities of 1 or 10 fpm.

Control runs at 1 and 10 fpm showed negative or insignificant amounts of fluorescein as compared to samples taken during test runs. A problem

Table B-1

RESULTS OF FLUORESCEIN AEROSOL TESTS
(Loading No. 1, Batch No. 2, Idaho Sand)

Test No.[a]	Column Velocity, fpm	Percent Penetration	Test No.[a]	Column Velocity, fpm	Percent Penetration
1	0.855	2.4×10^{-3}	14	62.0	5.43×10^{-2}
2	0.912	3.9×10^{-3}	15	148.0	6.32×10^{-2}
5	1.03	1.37×10^{-3}	16	1.02	1.64×10^{-4}
7	19.8	1.08×10^{-2}	17	1.025	2.12×10^{-4}
8	2.0	9.18×10^{-5}	18	2.02	2.17×10^{-5}
10	103.2	1.04×10^{-1}	19	191.5	9.39×10^{-2}
11	5.07	1.38×10^{-4}	20	1.025	5.19×10^{-5}
12	40.4	5.08×10^{-2}	21	0.52	4.05×10^{-5}
13	11.2	2.6×10^{-3}			

[a]Tests 6 and 9 were void because of sample filter missing and leakage, respectively. Results of Tests Nos. 3, 4A, and 4B are omitted since they were run with a different experimental generator nozzle.

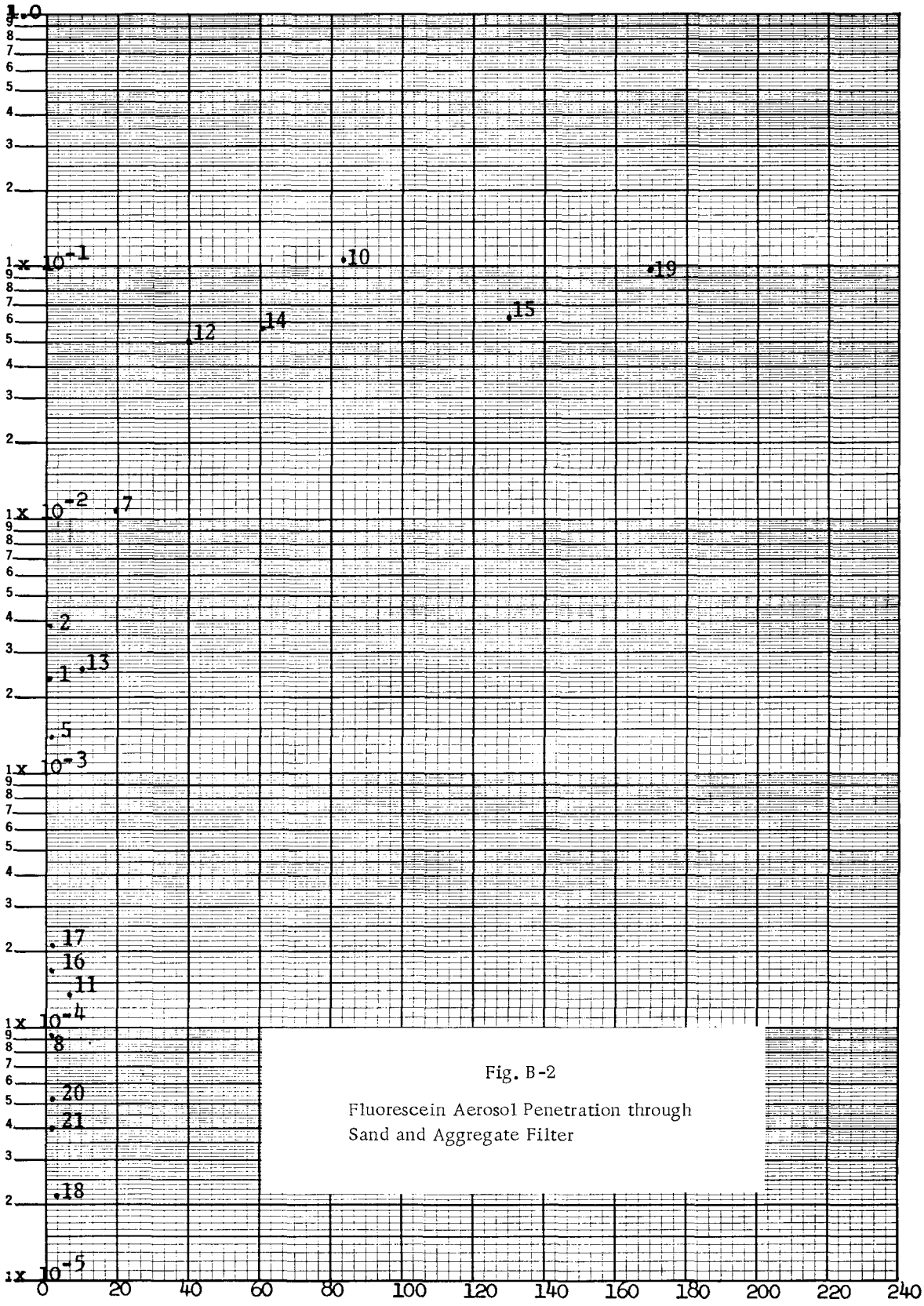


Fig. B-2

Fluorescein Aerosol Penetration through Sand and Aggregate Filter

which was encountered was partial clogging of the orifices in the collision atomizer nozzle, probably due to drying of fluorescein solution which splashed up into the orifice. This resulted in a lower total output of aerosol but with larger particles passing through the impactor since the impaction velocity is lowered. This problem may be minimized by frequent checking of the flow rate and by cleaning the orifices.

2. Uranium Aerosol Penetration Tests

Loading No. 2 of Batch 2 of Idaho sand was received August 4, 1967. Eau Claire 4 to 16-mesh material was used. Pressure versus flow tests for this loading indicated 1590 cfm/psi per 12-ft diameter sand filter.

Uranium turnings were burned to produce the test aerosol. The turnings were loaded into a holder containing a Nichrome wire coil for remote ignition by electrical heating of the wire. This holder was placed inside a steel chamber and a gasketed cover was bolted in place to form a leak-tight inlet air system for the test column. Again, in these tests, simultaneous samples of the upstream and downstream air were drawn through Type AA membrane filters. The analysis of the uranium samples was by the standard fluorometric technique with the lower confidence limit of 0.02 μg of uranium. Also, samples of upstream uranium aerosol were obtained for characterization of the aerosol using the electron microscope. A few tests with fluorescein aerosol were also run on this column for comparison to results from the previous loading.

The results of the uranium aerosol tests are shown in Table B-2 and plotted in Fig. B-3.

This column loading has more air flow than the previous loading and is was anticipated that it might therefore allow relatively more aerosol penetration. The highest results of any test run was $1.77 \times 10^{-1}\%$ at 149 fpm. Another run at nearly the same velocity, 152 fpm, gave $3.50 \times 10^{-2}\%$ penetration. It is concluded that the sand filter should provide a

Table B-2

RESULTS OF URANIUM AEROSOL TESTS
(Loading No. 2, Batch No. 2, Idaho Sand)

Run No.	Velocity, fpm	Uranium Turnings, g	Percent Penetration	Run No	Velocity, fpm	Uranium Turnings, g	Percent Penetration
25	1.01	22.6	1.36×10^{-2}	35	108.8	120.84	1.18×10^{-1}
26	0 495	26.7	1.08×10^{-2}	36	20.35	258.73	2.64×10^{-2}
27	0.239	21 8	$<1\ 64 \times 10^{-2}$	37	60.0	228.30	1.20×10^{-1}
28	4 875	33 7	$<4.5 \times 10^{-2}$	38	39.2	488.89	1.825×10^{-2}
29	20.7	34.6	<2.02	39	152.0	504.79	3.50×10^{-2}
30	2.03	51.79	$<7.27 \times 10^{-3}$	40	200.0	587.21	4.48×10^{-2}
32	20.5	80.30	$<3.5 \times 10^{-1}$	42	110.0	539.98	2.94×10^{-2}
33	4.875	69.95	6.05×10^{-3}	43	1.01	501.65	1.26×10^{-2}
34	11.31	95.04	$7\ 04 \times 10^{-3}$	49	149.0	398.66	1.77×10^{-1}

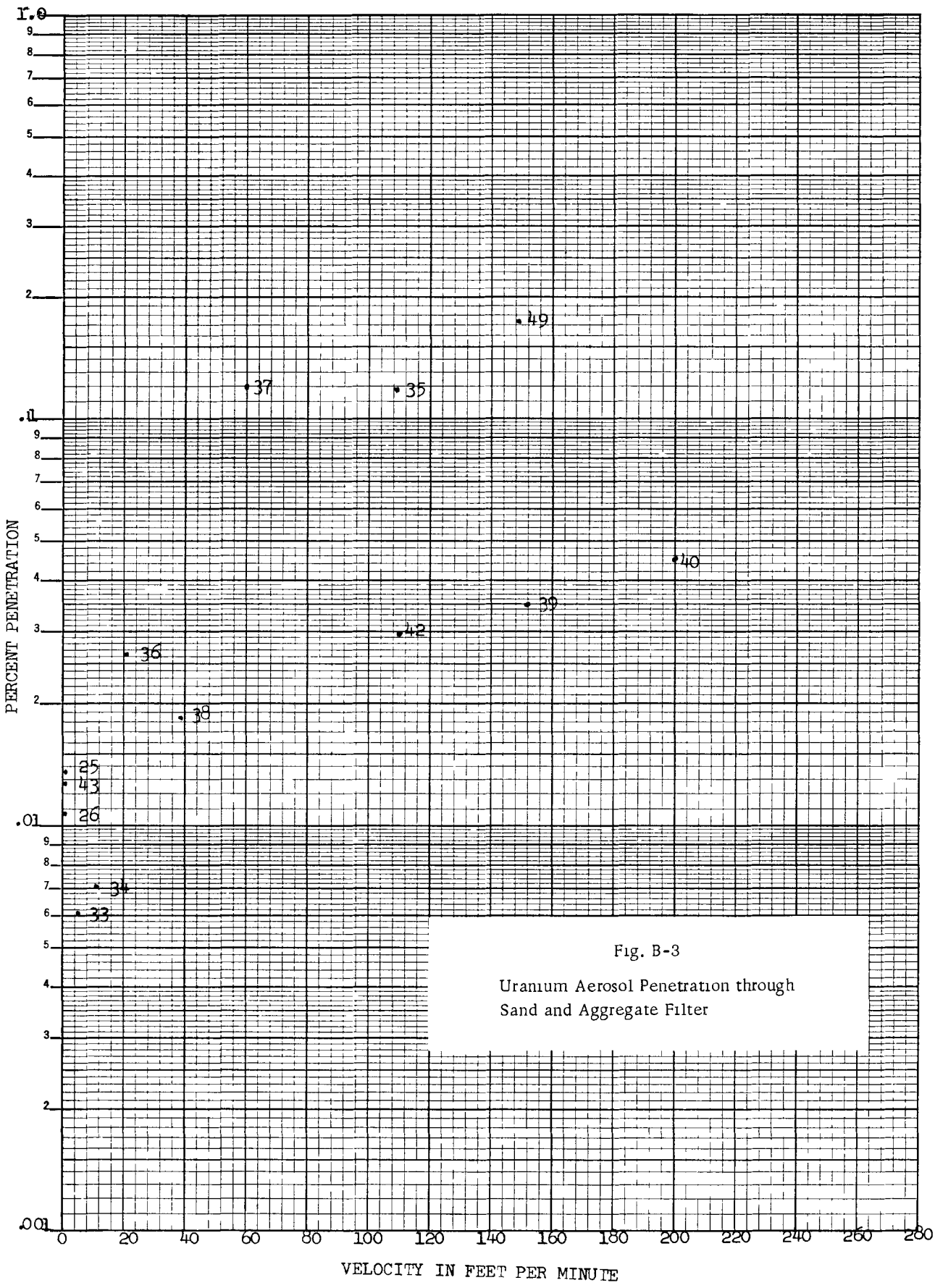


Fig. B-3

Uranium Aerosol Penetration through Sand and Aggregate Filter

decontamination factor of 10^3 or greater for this aerosol which had been shown in previous testing to give results similar to those obtained by tests with plasma torch generation of plutonium aerosol. The high flow rates were sustained relatively much longer during testing than they could be during a reactor cell accident even with complete discharge of the cell atmosphere. A number of control runs were made which indicated no significant inteference with tests at the low velocities. However, after the series of high velocity tests, a high velocity control run showed results similar to an actual run. This was due to the control run being carried out with uranium oxide powder still in the burn chamber with air turbulence generation of dust. On subsequent high velocity control runs after cleaning the burn chamber, there was still an appreciable amount of uranium resuspended into the air stream from material deposited on interior surfaces in previous runs.

The weight of sand and aggregate above the bottom of the sand is about 105 pcf average (7.5 ft by 0.492 ft^2) and is 388 lb. The pressure versus flow tests for this column indicated a flow of about 14 fpm/psig with most of the pressure drop occurring in the sand. At a flow rate of about 150 fpm, the lifting pressure at the bottom of the sand will be about the same as the weight of the material above the bottom of the sand. In the 200-fpm test some fluidization may occur to the extent permitted by the screen and tightness of packing of material in the column.

Results plotted in Fig. B-3, should be interpreted as being tests with different aerosol size composition and with increasing amounts of aerosol particles in the column as the tests progressed. For this reason the individual tests results are plotted without drawing a curve of penetration versus velocity.

Fluorescein aerosol penetration tests for comparison purposes were as shown in Table B-3.

Table B-3

RESULTS OF FLUORESCEIN AEROSOL PENETRATION TESTS
(Loading No. 2, Batch No. 2, Idaho Sand)

Run No.	Velocity, fpm	Percent Penetration	Run No.	Velocity, fpm	Percent Penetration
23	1.01	3.61×10^{-5}	56A	1.03	2.015×10^{-4}
24	1.01	2.38×10^{-5}	56B	1.03	1.81×10^{-4}
53	1.00	3.74×10^{-2}			

The reason why the penetration results in Runs No. 23 and 24 were much lower than for the previous column loading and lower than results from test Run No. 56 at the end of the test series is not known.

ACKNOWLEDGMENTS

The authors would like to acknowledge the contributions of the following individuals in the preparation of material for this report. These contributions range from reactor physics analysis to design of the modification to the ZPR-6 and -9 facility and cells.

R. A. Banike
R. C. Doerner
J. Haugsnes
W. P. McDowell
R. D. Moore
A. E. Reich
R. S. Reid
L. J. Kelly
C. A. Jachimiec