NYO-9066
AEC R & D Report
UC-81 Power Reactors

THE PEBBLE BED REACTOR PROGRAM

Quarterly Progress Report
July 1, 1959 through September 30, 1959.

2775

Work Performed Under AEC Contract AT(30-1)-2207

SANDERSON & PORTER
NEW YORK, N.Y.

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Pebble Bed Reactor Program

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1.0 Evaluation of Experimental Facilities

Several different types of experimental facilities have been evaluated to determine the most suitable and economical means of demonstrating the technical feasibility of the Pebble Bed Reactor concept. This evaluation is presented in detail in S & P 1965-3, issued August 31, 1959.

The criteria for demonstrating technical feasibility are: 1) Bulk irradiation of fuel elements in order to obtain statistical data on performance, 2) Maintenance of system activity at specified levels, and 3) Feasible maintenance procedures. An additional criterion, which is required to achieve breeding in a full scale plant is - 4) Demonstrate the mechanics of continuous refueling under load.

The use of three different types of facilities was investigated: a) in-pile loops, b) driven subcritical assemblies, and c) a self-sustaining reactor experiment.

In evaluating the worth of these different facilities, each involved the testing of a successively larger number of fuel elements and included further work in the related areas of feasibility. An increasing cost is associated with each step. Although each step offered some increment of information, only the final step, a self-sustaining reactor experiment, is capable of complete demonstration of the technical feasibility of the concept.

Summarizing the advantages of the experimental reactor, the quantity of fuel elements is such that some statistical significance can be placed in the results of irradiations of different fuel element types. The gas flow is of a magnitude that off-gas cleanup systems are prototype, rather than laboratory size. System maintenance procedures as developed and demonstrated would be directly translatable to a prototype or power reactor. Reloading equipment as used on such a reactor would be identical to that required for a larger system. By initiating the design and construction of an experimental reactor at this time, it is possible to plan on the design and construction of a thorium-uranium breeding prototype in the 1962-1965 periods, followed by a central station power plant within an elapsed ten year period.

It is recommended that a) the development of unit fuel elements continue with the present capsule program, b) the problems of equilibrium system activity, gas stream and system decontamination be explored in the presently planned circulating loop, and c) we proceed with the preliminary design and nuclear engineering analysis of a low-powered reactor experiment in order to establish refined estimates of cost, fabrication schedule and general feasibility.

2.0 Primary Loop Decontamination

The objective of this work is to develop means of controlling, reducing, or removing fission product activity in the primary loop to the point where practical maintenance procedures are possible.

The principal activity in this area during the past quarter has been: 1) the analysis of systems for concentrating the gaseous fission products in the primary loop and 2) the analysis of fission product removal systems and their effectiveness in holding down the primary loop activity.

2.1 Fission Product Concentration

The use of the ultracentrifuge as a means of concentrating the fission products, released to the primary gas stream, into a relatively small by-pass stream has been studied under subcontract by the Research Laboratories for the Engineering Sciences of the University of Virginia. Results of this study will be published as a report entitled "On Centrifugal Enrichment of the By-Pass Stream of the Pebble Bed Reactor" the general results presented in this report may be applied to a wide range of similar separation or purification problems.

The analysis begins with the application of isotope separation theory to derive formulae for the total length of centrifuge needed to attain a given enrichment of a given by-pass flow. Then certain of the formulae for equilibrium activity are shown written in terms of the separation theory parameters. The combined formulae allow the calculation of the length of centrifuge vs. the activity reduction for different reactors and by-pass flows.

The conclusions reached are that the length of centrifuge required is much greater than had been expected and the approximate annual operating costs would be much higher than anticipated. The general conclusion therefore is that centrifugal enrichment is feasible only as a last resort.

In view of this adverse finding, a brief study was made of the possibility of purifying the primary stream by adsorption of the contaminants on a suitable solid suspended in the helium stream and subsequently removed by a centrifuge. Removal of the solid by a centrifuge is entirely feasible. However, the possibility of adsorption on a suspended solid requires investigation.

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2.2 Gas Purification

Three schemes have been examined for removing fission products from the gas stream, namely adsorption, absorption and filtration.

The continuous adsorption of volatile fission products from a gas stream has been discussed with Arthur D. Little, Battelle Memorial Institute, The Babcock & Wilcox Company and Oak Ridge National Laboratory, all of whom have had experience in the experimental study of this operation. A search of the project and open literature for pertinent data has been made. No evidence has been found to indicate that charcoal adsorber beds have ever been used for the adsorption of radionuclides under conditions similar to those which prevail in the PBR. The nearest case was found in the HRT charcoal adsorbers. However, these are operated under conditions sufficiently unlike those of the PBR as to introduce uncertainty concerning the fundamental adsorption mechanism involved in the two cases.

On the assumption that a charcoal adsorber for the PBR can be analyzed by the same techniques used to analyze the HRT adsorber beds, the decontamination factor obtainable with a reasonably large size bed was determined. Since significantly higher flow rates would be required, recourse was made to certain unpublished data of W.E. Browning on the effect of superficial velocity through the bed on the number of equilibrium stages per foot of bed depth. The overall results of this analysis are shown in Figure 2.1. It can be seen that the decontamination factor obtainable with such a bed is indeed insignificant, and is probably not worth the added cost and problems associated with its use. These conclusions must remain tentative until such time as the fundamental adsorption mechanism is determined.

Theoretical work on adsorbtion fundamentals is continuing and it is anticipated that during the next quarter, work will be started on an experimental unit to simulate adsorber operation under PBR conditions.

Absorption as a means of removing fission products from a gas stream has been studied by R. Manowitz and Co-Workers at BNL. For the most part, organic hydrocarbon type solvents have been investigated and gas solubility data has been presented. Insofar as PBR application is concerned, the very stringent purity requirements placed on the primary helium stream mitigate against the use of absorbers since the problem of purifying the gas stream of the hydrocarbon solvent which it would pick up on passing through

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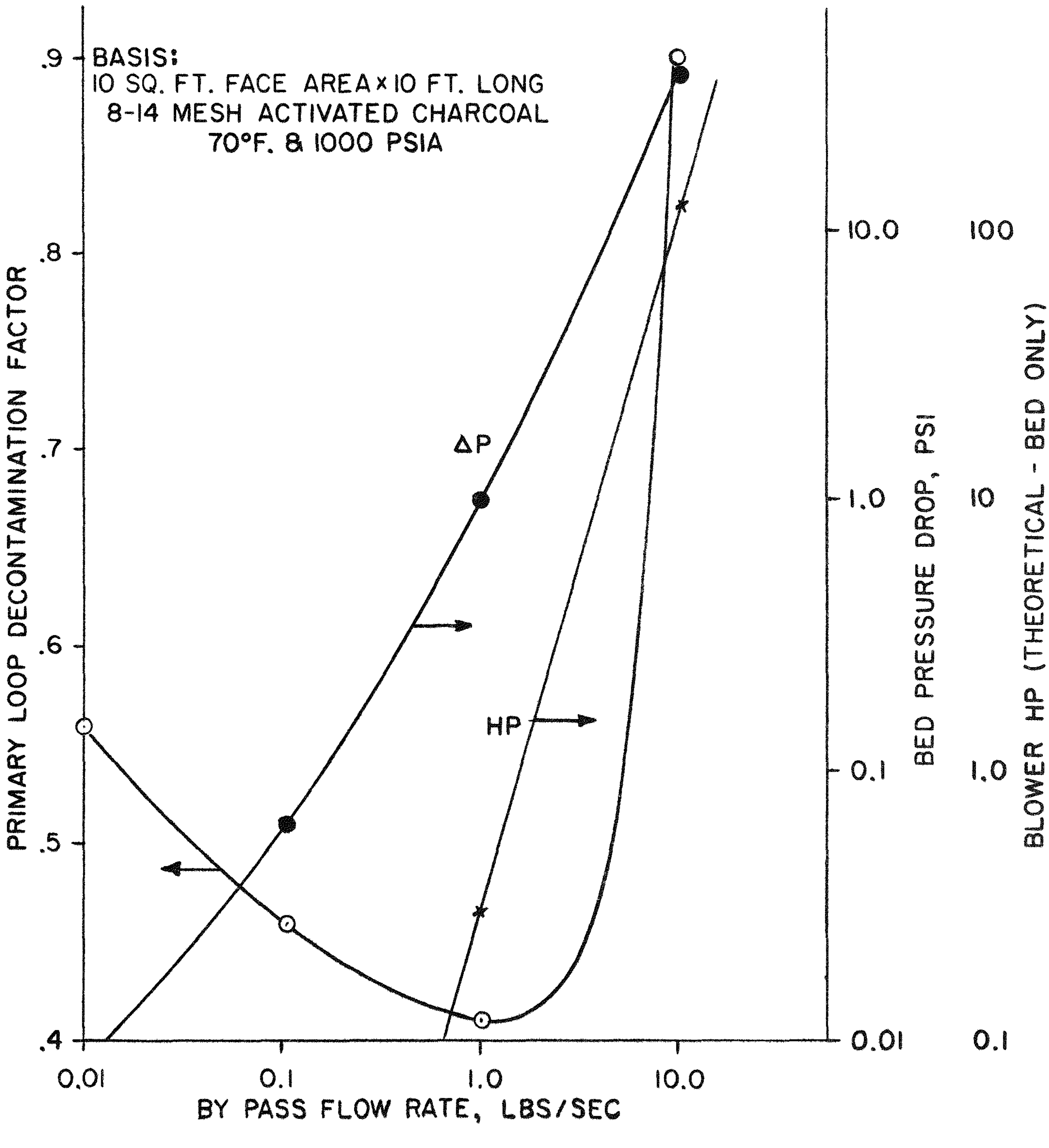

PLOT OF PRIMARY LOOP DECONTAMINATION FACTOR
& ADSORBER BED ΔP AND BLOWER POWER

FIG. 2-1

the absorber is indeed a problem of comparable magnitude to the initial problem of removing the fission products. For this reason no further work will be done on the use of absorption in connection with the PBR.

The study of removal of fission product gases by diffusion barrier separation (1) has been recently renewed, because of actual fabrication and performance data received from Corning Glass (2).

Corning have built and tested laboratory units of 30 inch length containing 0.2 pounds of .010 OD x .00165 wall tubing. These have been tested and the diffusion characteristics determined. These results show that Vicor glass will pass 125% as much helium as Silica glass, instead of the 178% reported by McAfee in (1).

Corning doubt that any tube spacing need be designed into a bundle for an application such as that contemplated where the impurities in the feed are very small. Tubes tend to bow during fabrication and some consideration has been given to deliberate bowing during manufacture. Corning believe tube bundles capable of operation up to 500°F can be made with resin tube sheets and hope to exceed this temperature with water-cooled tube sheets. They believe 15 feet to be a practical length.

A diffusion cell has been designed using the following data:

Tubing (Corning recommendations)

OD - 0.010 in.
Wall - 0.00165 in
Length - 15 ft.

Operating Conditions

Inlet temp.	500°F
Inlet press.	2000 psia
Outlet press.	1000 psia

(1) Techniques of Diffusion Separation, K. B. McAfee, Jr. Bell Telephone Laboratories, prepared for publication in a Supplement to the Encyclopedia of Chemical Technology in 1960

(2) Private Communication, John R. Blizard, Corning Glass.

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Characteristics

Weight of glass	30,000 lbs.
Volume	450 Ft^3
Face Area	38.3 Ft^2
Isothermal Power	350 KW

The 125 eMW PBR has been designed with three loops. Each loop would incorporate a diffusion cell, its dimensions being 4.5 ft. in diameter and 15 feet long. A sketch of a diffusion cell is given in Figure 2.2.

This size is reasonable enough to justify a more detailed study of the design and consider the establishment of an experimental program to determine the effectiveness of this cell in removing fission products.

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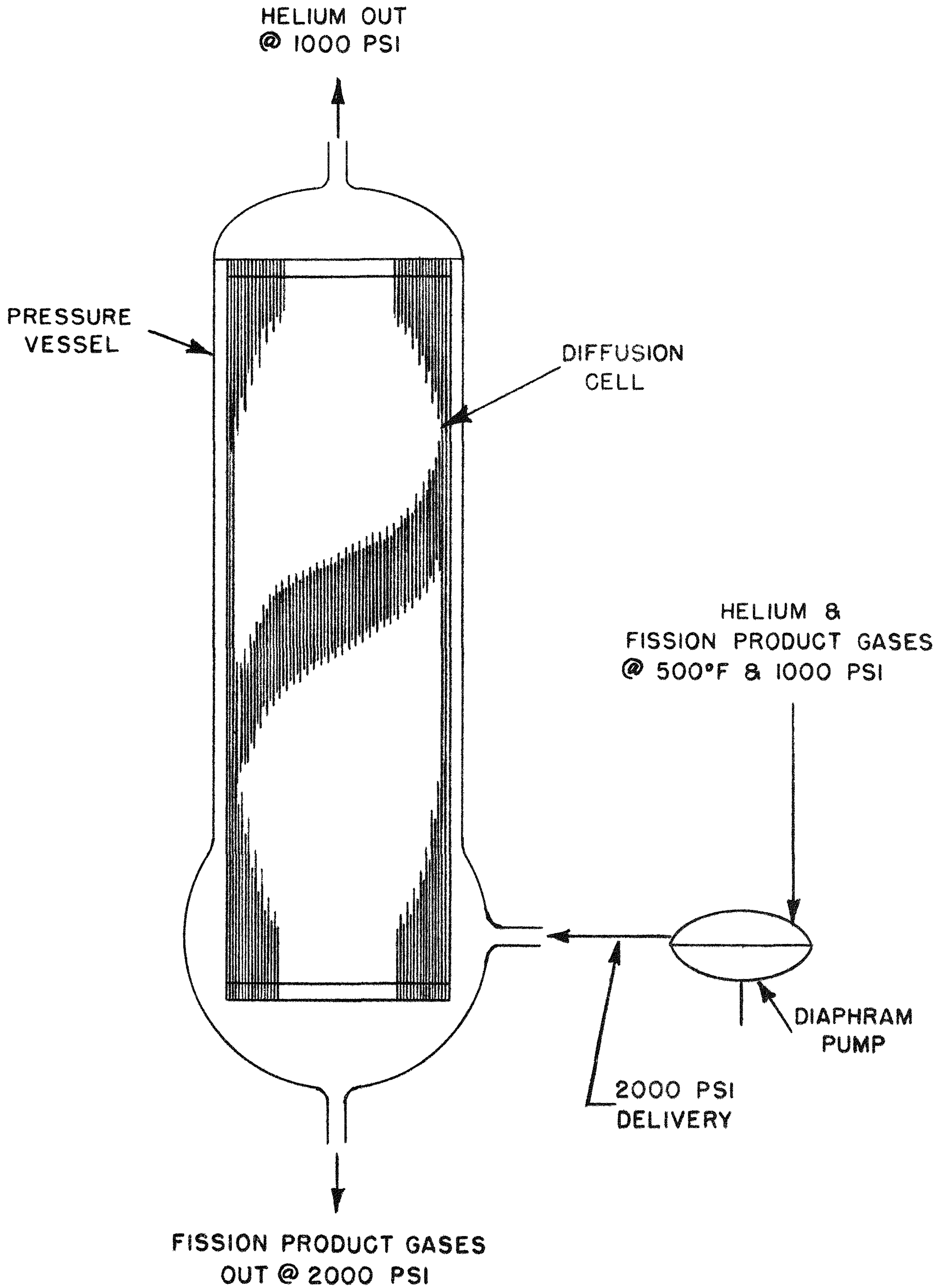

GLASS DIFFUSION CELL

FIG. 2-2

3.0 PBR Nuclear Model (1)

A nuclear model for the PBR has been constructed in order to have a means of rapidly determining the effect of different design variables on reactor characteristics. As a first approach the model was set up for a small core, 5 to 10 tMW output, as would be used in an experimental reactor.

The usual fast diffusion equation from two group theory is inadequate to describe the interaction within the core of the PBR for most of the cases examined to date because of the unusually large number of fast adsorptions. The high fast adsorption and the relatively weak moderation of the PBR make use of age theory and the usual expressions for resonance escape probability, inappropriate. Accordingly, a multigroup form of the fast diffusion equation was used.

3.1 Core Equations

The fast neutron balance equation for the core is:

$$D_1\nabla^2\phi_1 - \Sigma_{1s}\phi_1 - \Sigma_{1a}\phi_1 + \nu_1\Sigma_{1f}\phi_1 + \nu_2\Sigma_{2f}\phi_2 = 0 \qquad (1)$$

where the terms, going from left to right, represent, fast leakage, fast removal by scattering, fast removal by adsorption, production from fission in the fast group, production from fission in the thermal group.

The thermal neutron balance equation for the core is:

$$D_2\nabla^2\phi_2 - \Sigma_{2a}\phi_2 + \Sigma_{1s}\phi_1 = 0 \qquad (2)$$

where the terms again represent thermal leakage, thermal removal by adsorption and production by slowing down. The nomenclature is given in Table 3.1.

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(1) Preliminary Draft Two Group Calculations, Dr. J. L. Meem, University of Virginia.

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Table 3.1

Nomenclature

D_1 , D_2	Fast and thermal diffusion coefficients in core
D_{1r} , D_{2r}	Fast and thermal diffusion coefficients in reflector
ϕ_1 , ϕ_2	Fast and thermal neutron fluxes in core
ϕ_{1r} , ϕ_{2r}	Fast and thermal neutron fluxes in reflector
Σ_{1s}	Slowing down cross-section
Σ_{1a} , Σ_{2a}	Fast and thermal adsorption cross-sections
Σ_{1F} , Σ_{2F}	Fast and thermal fission cross-sections
ν_1 , ν_2	Neutrons per fast and thermal fission
$-\nu^2$	Root of buckling
ξ	Logarythmic energy decrement
$\bar{\Sigma}_s$	Average fast scattering cross-section
E_0	Upper energy limit of fast group
E_{th}	Lower energy limit of fast group
τ_r	Age in reflector
L_r	Diffusion length in reflector
T	Reflector or blanket thickness

In the above equations, Σ_{1s} is the slowing down cross-section for fast neutrons as defined by Glasstone and Edlund,

$$\Sigma_{1s} = \bar{\Sigma}_s \Big/ \left[(1/\xi) \ln (E_0/E_{th}) \right]$$

where $\bar{\Sigma}_s$ is the average scattering cross-section from fission energy to thermal energy. This is essentially constant in graphite.

Assuming that all fast adsorptions are in U-235,

$$\Sigma_{1a} = N_{25} \int \sigma_{a_{25}} \phi_1(E)\, dE \Big/ \int \phi_1(E)\, dE$$

Assuming ϕ_1 (E) is proportional to $\frac{1}{E}$,

$$\Sigma_{1a} = N_{25} \int (\sigma_{a_{25}}/E)\, dE \Big/ \ln (E_0/E_{th}) = N_{25}\, \sigma_a(\text{res}) \Big/ \ln (E_0/E_{th})$$

where σ_a(res) is the resonance adsorption integral for U-235.

By identical reasoning:

$$\Sigma_{1F} = N_{25}\, \sigma_F(\text{res}) \Big/ \ln (E_0/E_{th})$$

where σ_F(res) is the resonance fission integral for U-235.

Recent values for the infinite dilution resonance integrals I_o, are given by Stoughton and Halperin (2).

3.2 Reflector Equations

For the case where there is no highly absorbing material in the reflector, the diffusion equations are conventional.

The fast equation is:

$$D_{1r}\nabla^2\phi_{1r} - \Sigma_{1r}\phi_{1r} = 0 \tag{3a}$$

and the thermal equation is:

$$D_{2r}\nabla^2\phi_{2r} - \Sigma_{2r}\phi_{2r} + \Sigma_{1r}\phi_{1r} = 0 \tag{3b}$$

where the leakage and removal terms retain their usual definitions. Rearranging equations 3a and 3b gives:

$$\nabla^2\phi_{1r} - (1/\tau_r)\,\phi_{1r} = 0 \tag{4a}$$

and

$$\nabla^2\phi_{2r} - (1/L_r^2)\phi_{2r} + [D_{1r}/(D_{2r}\tau_r)]\,\phi_{1r} \tag{4b}$$

where $L_r^2 = D_{2r}/\Sigma_{2r}$ $\qquad \tau_r = \Sigma_{1r}/D_{1r}$.

The reflector coupling coefficient is conventionally given by

$$S_3 = (D_{1r}/D_{2r})\Big/[(\tau_r/L_r^2) - 1] \tag{5}$$

3.3 Solution for Core Buckling and Coupling Coefficients

Assuming that the geometrical dependence of the flux obeys:

$$\nabla^2\phi + B^2\phi = 0 \tag{6}$$

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(2) Stoughton and Halperin, Reactor Cross Sections, NUCLEAR Science and Engineering, 6, 100, (August, 1959).

the core equations become:

$$-D_1 B^2 \phi_1 - \Sigma_{1s}\phi_1 - \Sigma_{1a}\phi_1 + \nu_1 \Sigma_{1f}\phi_1 + \nu_2 \Sigma_{2f}\phi_2 = 0 \tag{7a}$$

$$-D_2 B^2 \phi_2 - \Sigma_{2a}\phi_2 + \Sigma_{1s}\phi_1 = 0 \tag{7b}$$

or

$$\left(D_1 B^2 + \Sigma_{1s} + \Sigma_{1a} - \nu_1 \Sigma_{1f}\right)\phi_1 - \nu_2 \Sigma_{2f}\phi_2 = 0 \tag{8a}$$

$$-\Sigma_{1s}\phi_1 + \left(D_2 B^2 + \Sigma_{2a}\right)\phi_2 = 0 \tag{8b}$$

For mathematical convenience, let:

$$\Sigma_1 = \nu_1 \Sigma_{1f} - (\Sigma_{1s} + \Sigma_{1a}) \tag{9}$$

Then:

$$\left[B^2 - (\Sigma_1/D_1)\right]\phi_1 - (\nu_2 \Sigma_{2f}/D_1)\phi_2 = 0 \tag{10}$$

and

$$-(\Sigma_{1s}/D_2)\phi_1 + \left[B^2 + (\Sigma_{2a}/D_2)\right]\phi_2 = 0 \tag{11}$$

Eliminating ϕ_1 and ϕ_2 :

$$\left[B^2 - (\Sigma_1/D_1)\right]\left[B^2 + (\Sigma_{2a}/D_2)\right] = \nu_2 \Sigma_{2f}\Sigma_{1s}/D_1 D_2 \tag{12}$$

or

$$(B^2)^2 + \left[(\Sigma_{2a}/D_2) - (\Sigma_1/D_1)\right]B^2 - (\Sigma_1 \Sigma_{2a} + \nu_2 \Sigma_{2f}\Sigma_{1s})/D_1 D_2 = 0 \tag{13}$$

Note that Σ_{2a}/D_2 may be identified as $1/L^2$ where L^2 is the square of the thermal diffusion length in the core. The term Σ_1/D_1 has the same role as the term $1/\tau$ in conventional two group theory but by no means has the same physical interpretation.

For cases where there are highly absorbing materials in the reflector, as would be the case when using a breeder blanket, the same model will be used with the absorption cross-sections suitably modified to account for these absorptions.

The two roots for B^2 are:

$$\mu^2 = \frac{1}{2}\left[-\left(\frac{\Sigma_{2a}}{D_2} - \frac{\Sigma_1}{D_1}\right) + \sqrt{\left(\frac{\Sigma_{2a}}{D_2} - \frac{\Sigma_1}{D_1}\right)^2 + 4\left(\frac{\Sigma_1\Sigma_{2a} + \nu_2\Sigma_{2f}\Sigma_{1s}}{D_1 D_2}\right)}\,\right] \qquad (14)$$

$$-\nu^2 = \frac{1}{2}\left[-\left(\frac{\Sigma_{2a}}{D_2} - \frac{\Sigma_1}{D_1}\right) - \sqrt{\left(\frac{\Sigma_{2a}}{D_2} - \frac{\Sigma_1}{D_1}\right)^2 + 4\left(\frac{\Sigma_1\Sigma_{2a} + \nu_2\Sigma_{2f}\Sigma_{1s}}{D_1 D_2}\right)}\,\right] \qquad (15)$$

These are best determined by expanding the radical.

Let $b = (\Sigma_2/D_2) - (\Sigma_1/D_1)$ and $C = (\Sigma_1\Sigma_{2a} + \nu_2\Sigma_{2f}\Sigma_{1s})/D_1D_2$.

$\mu^2 = (C/b) - (C^2/b^3)$ + negligible terms

$\nu^2 = \mu^2 + b$

The core coupling coefficients are found as usual from the thermal equation.

$$-D_2B^2\Phi_2 - \Sigma_{2a}\Phi_2 + \Sigma_{1s}\Phi_1 = 0 \qquad (16)$$

Let $\Phi_1 = AX$ and $B^2 = \mu^2$

$\Phi_2 = S_1 AX$

$S_1 = \Sigma_{1s}/(\Sigma_{2a} + D_2\mu^2)$

Let $\Phi_1 = CY$ and $B^2 = -\nu^2$

$\Phi_2 = S_1 CY$

$S_2 = \Sigma_{1s}/(\Sigma_{2a} - D_2\nu^2)$

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3.4 Critical Size, Average Flux and Conversion Ratio Calculation

From the root for the buckling given by equation (14) one may calculate the critical size of a bare spherical core to be π/μ. From the roots given by equations (14) and (15) one may calculate by an iterative procedure the radius of a reflected spherical core to be the value of the radius R which satisfies the following relation:

$$\mu \cot \mu R - (1/R) = [\rho_2 \delta C_1 + \rho_1 \gamma C_2 + \beta C_3] / [C_1 + C_2 + C_3] \qquad (17)$$

where

$$\rho_2 = D_{2r} / D_2$$

$$\rho_1 = D_{1r} / D_1$$

$$\delta = -(1/L_r)\coth(T/L_r) - (1/R)$$

$$\gamma = -(1/\sqrt{\tau_r})\coth(T/\sqrt{\tau_r}) - (1/R)$$

$$\beta = \nu \coth(\nu R) - (1/R)$$

$$C_1 = S_1(\rho_1 \gamma - \beta)$$

$$C_2 = S_2(\beta - \rho_2 \delta)$$

$$C_3 = S_3 \rho_2 (\delta - \gamma)$$

T = reflector or blanket thickness

Average fast and thermal fluxes for the core and reflector (blanket) regions are then computed by integrating (using Simpson's rule) the expressions for the flux over the volume of each region and dividing by the volume.

The conversion ratio can then be computed for each region of the core by totalling the fast and slow absorption in fissile and fertile material in each region.

4.0 Bed Characteristics

In order to achieve a uniform burnup in a continuously loaded Pebble Bed Reactor, it is essential to match the radial ball flow rate through the core with the radial flux so that the nvt per fuel element is constant.

There are a number of variables in the core design which influence the radial ball flow rate. These are: bottom grate angle, ratio of hood to core diameter, ratio of hood diameter to height and the shape of the hood. A model of a reactor core with unloading valve has been constructed to study the effect of these variables.

A schematic drawing of the model core is shown in Figure 4.1 while the completed model as used for study of ball flow is shown in Figure 4.2. The core is a lucite cylinder, eight inches inside diameter. Balls are presently being used are three eighths inch colored lucite which makes the model one quarter scale of a 32 inch core. At a later date, quarter inch balls may be used, making the model one sixth scale of a 48 inch core.

The bottom grate, illustrated in Figure 4.3, is turned of hard wood. Grates of various bottom angles have been made to check the influence of this variable. It is also possible to vary details such as edge radius, outlet valve hole size or shape, if such changes are deemed to be desirable.

A number of sizes and shapes of hoods have been constructed, as illustrated in Figure 4.3, which mount on the bottom grate and are supported in a manner similar to that which would be used in a full scale reactor. These hoods are turned of hard wood and supported on the grate with brass posts, threaded into the hood.

The unloading valve used with this model differs from the bulk unloading valve previously developed and described in NYO 2373 in that it unloads one ball at a time. The valve, illustrated in Figure 4.4, is a simple, single plate star feeder. It incorporates a shut-off plate above the star wheel, making it possible, in a full scale device, to shut off the flow of balls from the bed and remove the valve for repair or replacement. The star wheel of this valve is driven at constant speed by a synchronous motor through gearing. The speed is such that the rate of unloading is 435 balls per minute. The bed holds 7,887 3/8 inch balls in a 7-1/2 inch height and can be unloaded in 27.15 minutes.

In addition to design and construction of the model a considerable amount of time has been spent in shakedown operation and experimenting with different methods of determining the radial variation in flow.

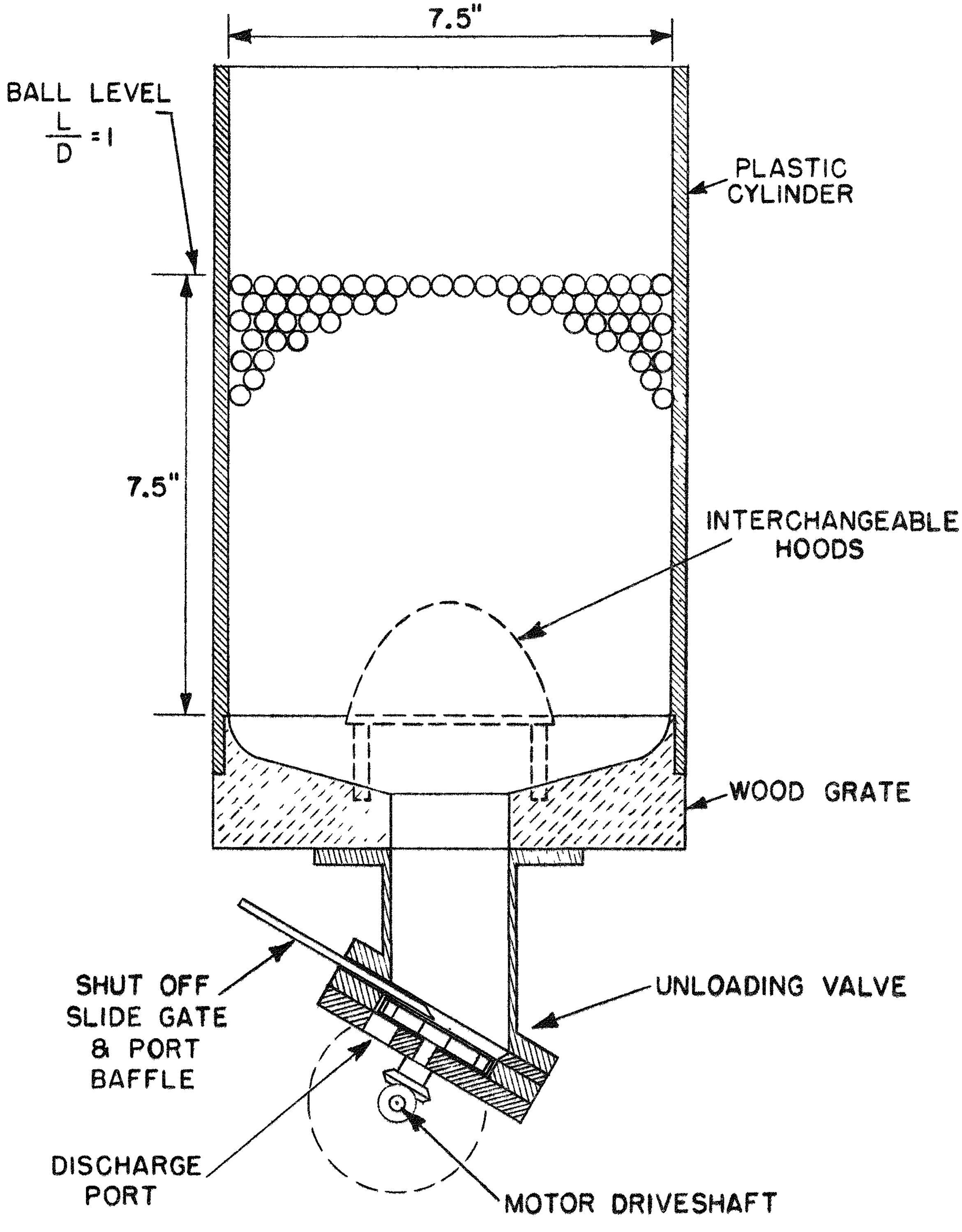


FIG. 3-1 MODEL CORE SCHEMATIC

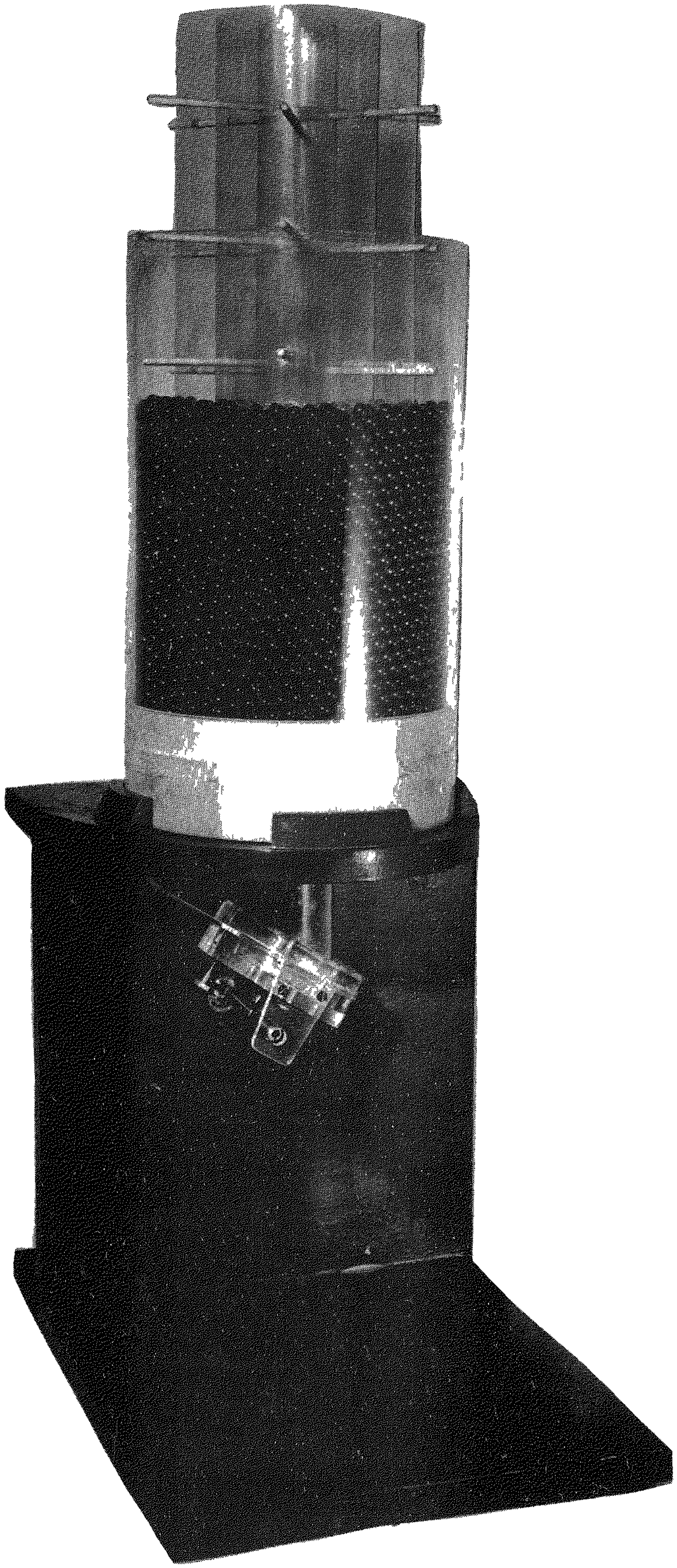


FIG. 3-2 MODEL CORE

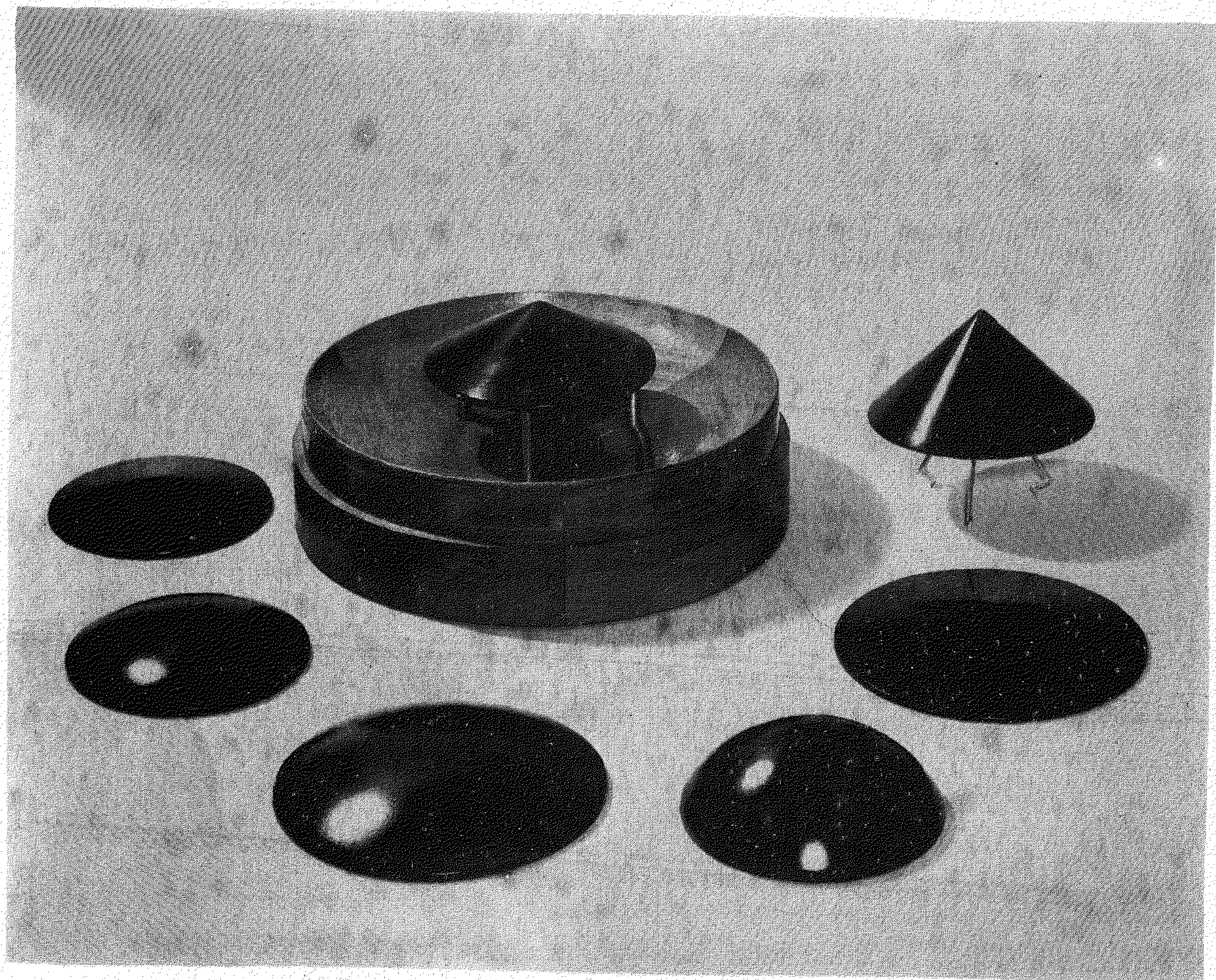
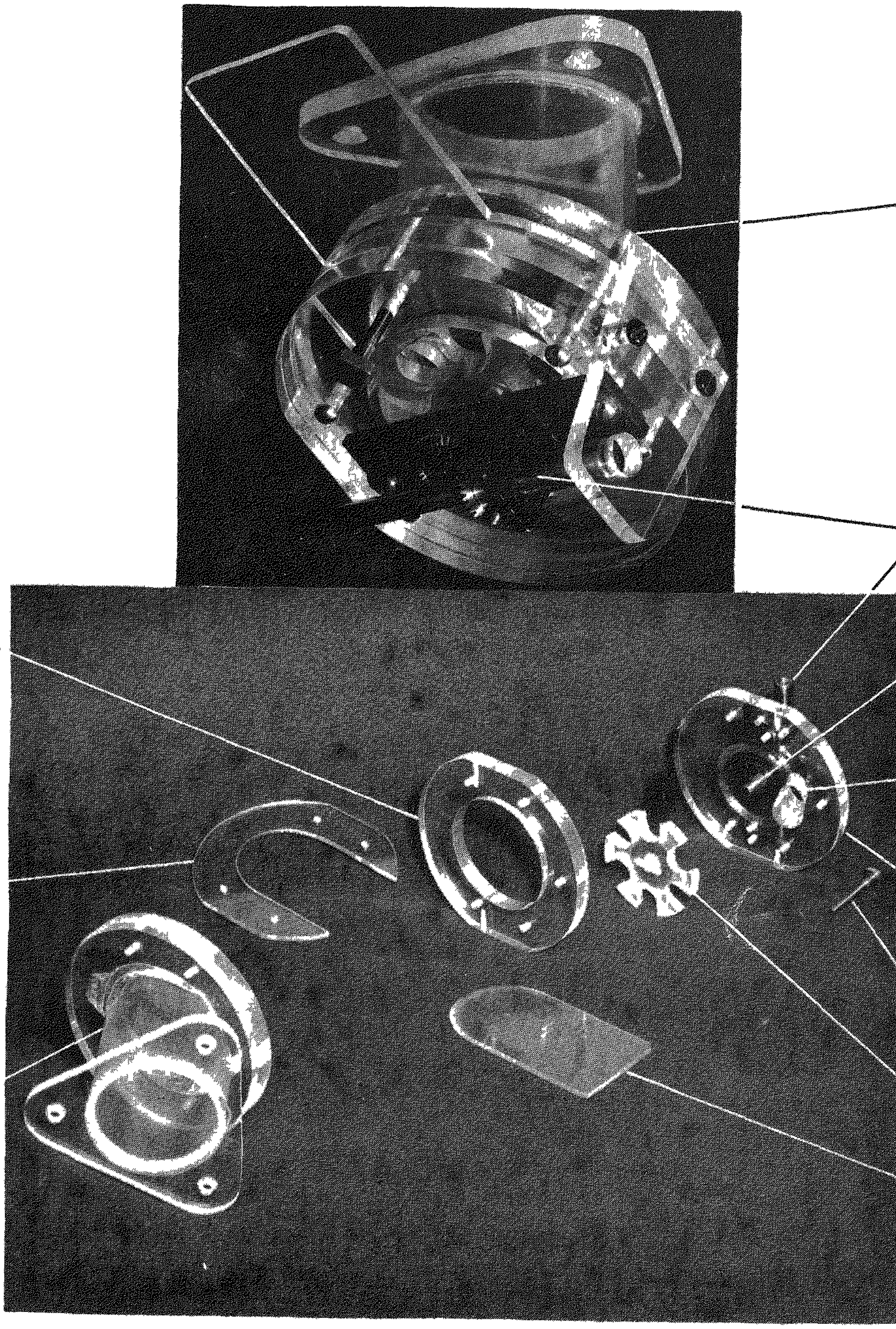


FIG. 3-3 BOTTOM GRATE & HOODS



EXPLODED VIEW OF PARTS

FIG 3-4 STARWHEEL UNLOADING VALVE