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THE EXPERIMENTAL PROGRAMME FOR THE S.G.H.W. CORES

IN DIMPLE

By

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ABSTRACT

The objects of the S.G.H.W. experiment in DIMPLE are reviewed and related to current uncertainties in the design of the power reactor. The use of the driver lattice technique is shown to be a consequence of fuel supply limitations. The proposed experimental programme is described and a brief discussion of the methods of theoretical analysis is presented.

A.E.E.,
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OBJECTS OF THE EXPERIMENT

1. The experiments being carried out in DIMPLE were initially intended to achieve the following objectives.

- (i) a confirmation of the correctness of the present calculated enrichments in the proposed S.G.H.W. power reactor.
- (ii) an examination of the effect of coolant voidage on reactivity and, if possible, on stability.
- (iii) the determination of such other parameters of the system as can be compared with the refined calculations which will be developed for the S.G.H.W. reactor.

The time available for the experiments in DIMPLE is limited and the experiments have been designed to use fuel originally intended for the A.G.R. experimental physics programme. Furthermore it is not possible to generate steam water mixtures in DIMPLE. The results of the experiments in DIMPLE will therefore require careful interpretation before they can be applied to a full scale power reactor. In some respects it may not prove possible to fulfil the objectives listed above because of differences between the experiment in DIMPLE and the conditions in a large, irradiated, hot reactor with two phase coolant flow.

Three types of experiment are proposed for the S.G.H.W. experiments:-

- (i) measurements of the macroscopic flux distributions in a region of the reactor core having a similar composition to the power reactor. From these measurements the material buckling of the given lattice may be deduced, provided that it can be shown that an equilibrium neutron spectrum exists over a sufficiently large region.
- (ii) measurement of the small changes in reactivity brought about by local variation of the composition or density of coolant in the core region;
- (iii) measurements of the fine and hyperfine flux distribution (at different neutron energies) and measurements of the ratios of different reaction rates. From the measurements it is hoped that the fast fission factor, resonance escape probability, conversion factor and thermal spectrum hardening may be deduced.

2. DESIGN OF THE DIMPLE CORE

The experiment in DIMPLE was designed with the object of reproducing, as far as possible, the nuclear characteristics of the core proposed for the boiling region of the S.G.H.W. power reactor design study. The parameters of the power reactor core and those of the DIMPLE core are compared in detail in A.E.E.W. Memo. 64 (H. R. McK. Hyder 1960).

The design of the DIMPLE core was restricted by the need to make use of UO₂ fuel designed for the A.G.R. experiments. Three different types of UO₂ fuel were available and their relative merits and disadvantages are compared below. All the different type of fuel were available in the form of cylindrical pellets clad in aluminium to form rods of 14" overall length with a contained fuel length of

about 12.5". The three types of fuel and the chief factors affecting their use are as follows:-

(i) 1.28 C₀ UO₂ pellets: 0.5" diameter.

The fuel is approximately the same diameter as the fuel rods proposed for the power reactor, which would render measurements of the resonance escape probability easy to interpret. The amount of cladding also corresponds closely to that in the proposed power reactor. Calculations showed however that a lattice of 1.28 C₀ fuel would be very unreactive and that in order to achieve criticality a driver zone of very high reactivity would be required. This would detract from the accuracy of material buckling measurements and render perturbation measurements difficult to interpret. The use of 1.28 C₀ fuel in a two zone critical experiment was therefore rejected. An exponential experiment using 1.28 C₀ fuel was also considered but was rejected for the same reasons as the 1.6 C₀ exponential which is, in any case, a better simulant of the power reactor.

(ii) 1.6 C₀ UO₂ pellets: 0.3" diameter.

This was a marginal case but it seemed likely that a lattice of this fuel would not be critical at dimensions which could be accommodated in the DIMPLE tank. There were thus two possibilities remaining:

(a) a set of exponential experiments could be carried out at different pitches. From the measured relationship between pitch and material buckling it would then be possible to deduce values for η eff and the effective resonance integral. This technique has been extensively used on low enrichment, graphite, gas-cooled lattices (see e.g., A.E.R.E. R/R 2771) but it is unsuitable for the S.G.H.W. core design where the presence of H₂O coolant results in the lattice parameters being relatively insensitive to changes in pitch.

Exponential experiments would also involve major engineering difficulties in modifying DIMPLE to eliminate the D₂O and graphite reflectors. Furthermore each change of pitch would involve the manufacture of a new set of lattice plates and a delay of 3-4 weeks to cover the drying out of the DIMPLE tank and the total dismantling and re-assembly of the core. Finally, the choice of a sub-critical system, with its necessarily limited flux, renders resonance escape probability, fast fission factor and certain thermal neutron spectrum measurements impossible.

(b) a two zone critical reactor experiment could be constructed, the inner zone consisting of 1.6 C₀ fuel arranged to simulate the power reactor, while the outer zone would contain high enrichment fuel arranged, so far as possible, to match the spectrum in the inner zone. This arrangement offered a number of advantages. It was believed that the inner zone could be made large enough to obtain an equilibrium neutron spectrum which was independent of the characteristics of the driver region. The use of a critical system would permit small changes in reactivity to be measured and the high flux levels obtainable would permit a wide range of activation measurements to be undertaken.

(iii) 2.5 C₀ UO₂ pellets. 0.3" diameter.

Calculations on a lattice of this fuel suggested that it was too reactive for the present experiments. It was estimated that a core of only 20 or 30 channels would be critical. Such a core was considered to be too small to permit useful measurements to be made in a region of constant neutron spectrum and experiences with the DIDO mock up in DIMPLE suggested that the determination of leakage would be difficult.

The inadequacy of cores loaded with 1.28 C₀ or 2.5 C₀ fuel and the disadvantages of sub-critical experiments in an exponential stack led to the choice of a two region critical core, the inner region containing 1.6 C₀ oxide fuel disposed so as to simulate the conditions in the power reactor, the outer (driver) region being made up of high enrichment fuel in pressure and calandria tubes identical to those in the inner region.

Although it would assist interpretation of the experiments to maintain the composition of the outer driver zone unchanged as the inner zone is varied, the overriding requirement is to keep the critical D₂O height within safe limits. Consequently it might be necessary to alter the amount of fuel in the driver zone as the inner zone is changed.

Having chosen the enrichment and diameter of the fuel the core parameters which could be varied in order to reproduce the nuclear characteristics of the power reactor were:-

- (i) lattice pitch and type of lattice (square or hexagonal)
- (ii) number and arrangement of UO₂ fuel rods in each fuel cluster.
- (iii) total 'coolant' volume
- (iv) height of core
- (v) nature and composition of coolant.

Because the size of the calandria tubes was fixed, the amount of D₂O per lattice cell was determined solely by (i). The coolant volume is determined only by (ii) and the detailed design of the fuel supports, since the pressure tube size was also fixed.

The objective in designing the inner region of the core was to reproduce the neutron spectrum of the power reactor by achieving similar values of the ratios,

$$R_H = \frac{\text{barns of equivalent } \frac{1}{v} \text{ absorption}}{\text{coolant atom}}$$

and
$$R_D = \frac{\text{barns of equivalent } \frac{1}{v} \text{ absorption}}{\text{moderator atom}}$$

in the two systems. For a given fuel loading and fuel channel design R_D depends only on pitch. With the 1.6 C_0 0.3" diameter rods available for the DIMPLE experiment it was necessary to reduce the pitch to 8.375" in order to reproduce the value of R_D corresponding to a power reactor with a 9.5" pitch. For this arrangement it did not prove possible to design the fuel channel so that R_H for the two systems was equal with both channels full of water of $\rho = 1 \text{ gm/cm}^3$. However the range of R_H to be expected in the power reactor could be reproduced in DIMPLE by appropriate mixtures of D_2O and H_2O in the pressure tubes, treating the D_2O as H_2O of reduced moderating power in the calculation of R_H and ignoring it in calculating R_D .

The height of core was eventually chosen to be 7', corresponding to six clusters of UO_2 rods in each channel. The height was determined by the amount of fuel available, the highest point to which the safety rod banks could be raised and the desire to have as tall a core as possible in order to improve the accuracy of axial buckling measurements.

The choice of coolant simulant was determined principally by cost and the short time available to develop methods of displacing water from the coolant region in the pressure tubes. The possibility of using a foamed plastic of low density to simulate steam voids was considered but laboratory tests showed that it would be impossible to mould the only practicable material (partially expanded polystyrene) around the 90 rod clusters of UO_2 fuel. The estimated cost of fabricating matrices of polystyrene was prohibitive if up to five sets of different densities were required. Other objections to the use of polystyrene at intermediate densities were its unknown water absorption and thermal neutron scattering properties.

It was eventually decided to use mixtures of H_2O and D_2O in the pressure tubes. It was calculated that the thermal utilization and resonance escape probability to be expected in steam/water mixtures could be closely reproduced by appropriate volume ratios of D_2O and H_2O . The availability of D_2O and the comparatively low cost of recovering degraded material were contributory factors in the choice. It was recognized that the presence of the D_2O would render certain measurements in DIMPLE difficult to interpret and properties of the power reactor which depend on the molecular scattering properties of the coolant, on its temperature and on its spatial distribution and on the voidage in the pressure tube will not be reproduced in DIMPLE.

The design of core inner region finally chosen has the following parameters:-

- (i) square lattice of pitch 8.375"
- (ii) 90 rods of 1.6 C_0 oxide per cluster
- (iii) all the volume of the pressure tube not occupied by fuel rods or aluminium end plates (designed to reduce axial fine structure) is flooded with coolant.
- (iv) height - 6 fuel clusters (7')

- (v) coolant - Air, D₂O, 75% D₂O - 25% H₂O, 50% D₂O - 50% H₂O, 25% D₂O - 75% H₂O, H₂O.

The inner region contains 52 fuel channels and is surrounded by 24 channels containing highly enriched 'spikes' of U/Al. alloy (93V, 20% U wt.). The spikes are 28" long and are mounted in special frames. Three such frames one above the other are the same height as the inner region core. The number of spikes in each frame is adjusted to maintain criticality at nearly constant D₂O height as the composition of the inner zone varies.

The arrangement of the core and of the oxide fuel clusters and 'spike' frames are shown in figures 1-8. The location of safety rods and search tubes is shown in figure 3 and the sealing arrangements, by means of which the pressure tubes may be loaded and unloaded without opening the D₂O vessel, in figure 8.

3. MEASUREMENTS OF REACTIVITY

DIMPLE is designed for control by variation of heavy water height. Two depth probes (accurate to 0.1 mm) enable small changes in D₂O height to be measured and reproduced while the control pump, which is a reversing variable speed, positive displacement device, enables the operator to produce small changes in reactivity at will. Once a calibration scale relating reactivity to D₂O height has been measured for a particular core, the reactivity worth of safety rods, fuel or absorbers can be determined simply by observing the change in D₂O height required to maintain criticality.

The calibration of the reactivity/D₂O height scale will be obtained initially by doubling time measurements. These measurements require knowledge of the effective delayed and prompt neutron fractions. The latter is affected by the irradiation history of the fuel and it will be necessary to ensure that the fuel is relatively inactive before the calibrations are performed. It is also affected by the amount of D₂O in the core and it may be expected to vary appreciably as the composition of the coolant simulant in the pressure tubes changes. The effectiveness of the delayed neutrons in the S.G.H.W. core is likely to be greater than in a one region core of the same overall dimensions because of the increased leakage of fast neutrons from the spiked region near the core boundary. An attempt will be made to calculate this change in effectiveness. Reactivity calibration by doubling time measurement is limited to a range of reactivity corresponding to a doubling time of 20 secs. (i.e. about 0.2%). Thus it will be possible to calibrate over a range of height of 3-10 cms. in one stage before perturbing the core to alter the critical height.

It is possible in principle to measure sub-critical values of reactivity by the source jerk technique. This method depends critically for accuracy on a low inherent source strength in the reactor and may not prove to be practical in the S.G.H.W. experiment because of the photoneutron source which will develop in operation. If possible it will be used to supplement the results of the doubling time calibrations.

It is usual to operate DIMPLE without top reflector since changes in D₂O height can then be more easily interpreted in terms of variation of axial leakage without the complication of an unknown and varying reflector saving. In S.G.H.W.

cores however the presence of moderator in the pressure tube will make it necessary to estimate the reflector savings in all the experiments except the first, in which the pressure tubes do not contain moderator.

The oxide fuel rods in their cluster frames are 14" long overall; however the fuel extends only over 12.6" and it is expected that increasing the D₂O height over the fuel-free region at the end of an oxide rod will result in a small change of reactivity which will be difficult to calculate. For this reason it is proposed to make all the reactivity measurements by varying the D₂O over the range of height corresponding to the fuel-bearing region of the top fuel element. The loading of U²³⁵ spikes in the outer region will be adjusted if necessary to ensure that the critical height remains within this range.

Because it is impracticable to vary the moderator height in the pressure tubes with change of moderator height in the tank it has been decided to fill the pressure tubes with fluid to the top of the top fuel packs. This will enable the fluid height to be checked visually without unloading fuel.

It is proposed that the following changes of reactivity should be measured for each of the six cores, after calibration of the D₂O height/reactivity relationship.

- (i) Initial critical height as a function of number of U²³⁵ spikes in outer region.
- (ii) Reactivity worth of control rods and cadmium tubes throughout the UO₂ fuelled region:
- (iii) Statistical weighting of various fuel rods.
- (iv) Reactivity worth of a single fuel cluster.
- (v) Reactivity change resulting from non-uniform coolant distribution in a single channel (the coolant density decreases as height increases in a boiling channel).
- (vi) Reactivity change resulting from non-uniform radial coolant distribution in a single UO₂ fuel cluster.

4. MEASUREMENTS OF FLUX DISTRIBUTION

The measurements of flux distributions and reaction rates are expected to yield:

- (i) a value of the material buckling for the inner core region.
- (ii) details of the fine structure and thermal utilisation.
- (iii) estimates of spectrum hardening in the D₂O and fuel cluster.
- (iv) measurements of fast fission factor and resonance escape probability.

The majority of flux measurements will be made by foil activation but some preliminary macroscopic distributions will be determined by the use of U²³⁵ fission

chambers, and the thermal spectrum hardening will be measured at one characteristic point by means of U²³⁵, Pu²³⁹ and U²³³ fission chambers.

Since the foil activation measurements will yield macroscopic and fine structure distributions simultaneously the techniques are described in terms of the foils employed rather than the results to be deduced.

(a) Au/Mn/Ni alloy foils.

The thermal and epithermal flux distributions will be deduced from the manganese and gold activities induced in these foils. As the coolant composition in the centre zone is varied from air to D₂O and then to H₂O the relative importance of thermal and epithermal activation will vary in both the gold and manganese. In the air filled core about 20% of the manganese activity and 80% of the gold activity will be due to epithermal neutrons, compared with 5% and 50% respectively in the H₂O moderated case.

The foils are absolutely calibrated so that the total flux can be derived from either the gold or manganese activity. The ratio of epithermal to thermal flux (r on the Westcott convention) will be deduced from the measured ratio of the two activities. This procedure will avoid the use of cadmium. The foils are 0.25" dia., sufficiently small to cause little flux perturbation and to permit detailed examination of fine structure.

(b) Mn/Ni alloy foils.

Hyperfine thermal flux variation (between adjacent UO₂ rods) will be measured with thin ribbons of Mn/Ni alloy, which will subsequently be cut up and γ counted.

(c) Lu/Mn/MgO cermet pellets.

The thermal spectrum hardening will be characterized by the use of these pellets. The foils, which have been calibrated in a known spectrum, exhibit a ratio of Lu¹⁷⁷ (6.7 day activity) to Mn⁵⁶ activity which depends sharply on the effective neutron "temperature" of the thermal group of neutrons. For near-Maxwellian spectra the ratio of activities varies by 1% for 3°C change in effective temperature. Assuming a spectrum model for the thermal group (initially the Wigner Wilkins spectrum will be chosen), it will be possible to characterize the spectrum as a function of position in the lattice cell to a reasonable accuracy.

(d) Rh foils.

The fast (>0.2 MeV) flux will be measured by irradiating pure Rh metal foils and counting the K X-rays resulting from the decay of the 57 min isomeric state excited by inelastic scattering of fast neutrons, using the measured cross section for the Rh¹⁰³ (n, n') Rh^{103m} reaction.

(e) 1.6 C₀ uranium and depleted uranium foils.

Measurements will be made, in the D₂O filled core, of the U²³⁸/U²³⁵ fission ratio and the U²³⁸ capture/U²³⁵ fission ratio in a number of UO₂ rods in a 90

rod cluster. The measurements will be made by the Industrial Power Reactor Division. The measurements will lead to values of the fast fission factor and conversion factor and it is hoped that it will be possible to extend the measurements to include irradiations under Cd from which the resonance escape probability may be obtained.

It is intended subsequently to use these techniques, or modifications of them, to measure the same quantities in the following cores.

From the irradiations described in §§ a - e it will be possible to obtain:

- (i) the axial and radial limits of the region of constant neutron spectrum in the core inner region.
- (ii) the macroscopic axial and radial flux distribution in this region, from which the material buckling can be derived by one group analysis.
- (iii) the axial and radial fine structure of thermal, epithermal and fast flux.
- (iv) the hyperfine thermal flux structure in the fuel cluster.
- (v) the U^{238} resonance capture distribution in the fuel.
- (vi) the radial variation of thermal spectrum hardening across a lattice cell.
- (vii) some limited information about flux and spectrum in the outer core region and in the region of non-equilibrium spectrum from which it may be possible to correlate the multi-group multi-region criticality calculations.

5. INTERPRETATION OF THE RESULTS

In view of the differences between the Dimple experiments and current proposals for the S.G.H.W. itself, few, if any, of the results will be directly applicable to the power reactor. The value of the experiment lies in the fact that it provides an opportunity to test methods of calculation under conditions approximating to the power reactor.

It is not possible at this stage to see the degree of elaboration which will finally be required in the reactor physics calculations and trial calculations are in progress with a number of schemes. The situation in the S.G.H.W. is evidently more complex than in systems previously encountered due to the presence of two different moderators at different physical temperatures. In the Dimple experiment the two moderators are at the same physical temperatures: thus, since the thermal neutron spectrum depends on the physical temperatures and since flux peaking depends on the spectrum, there are factors influencing power reactor design which cannot be studied in the present series of Dimple experiments.

It is nevertheless essential to have available some theoretical scheme for the immediate correlation and guidance of the experimental measurements. It is therefore proposed that this shall be attempted on the following basis:-

- (i) Lattice calculations will be made on the basis of few (2-4) group diffusion models. These will enable the reaction rates of the various activation materials discussed above to be calculated as a function of position in the lattice cell. Comparisons of experimental results and predictions based on models containing various amounts of detail should show which features it is necessary to include in the calculations.
- (ii) Slowing down and thermal diffusion calculations will also be made. When these are combined with the lattice parameters it will be possible to estimate the material buckling. This can be compared with the measured buckling derived from the macroscopic flux distribution. In particular the changes in buckling with effective coolant simulant density can be compared with theory. This will show how well the proposed theoretical methods can be expected to estimate changes in reactivity with coolant density in a power reactor.
- (iii) A theoretical investigation of the dependence of the thermal neutron spectrum on position within the lattice cell using the free gas model for neutron moderation is in progress. This will throw light on the effects of the difference in temperatures between D₂O moderator and H₂O coolant in the power reactor and should show how these are likely to influence flux peaking factors. It is evident that this method must be shown to account for the Dimple flux peaking and neutron spectrum measurements before confidence can be placed in its use in power reactor calculations.

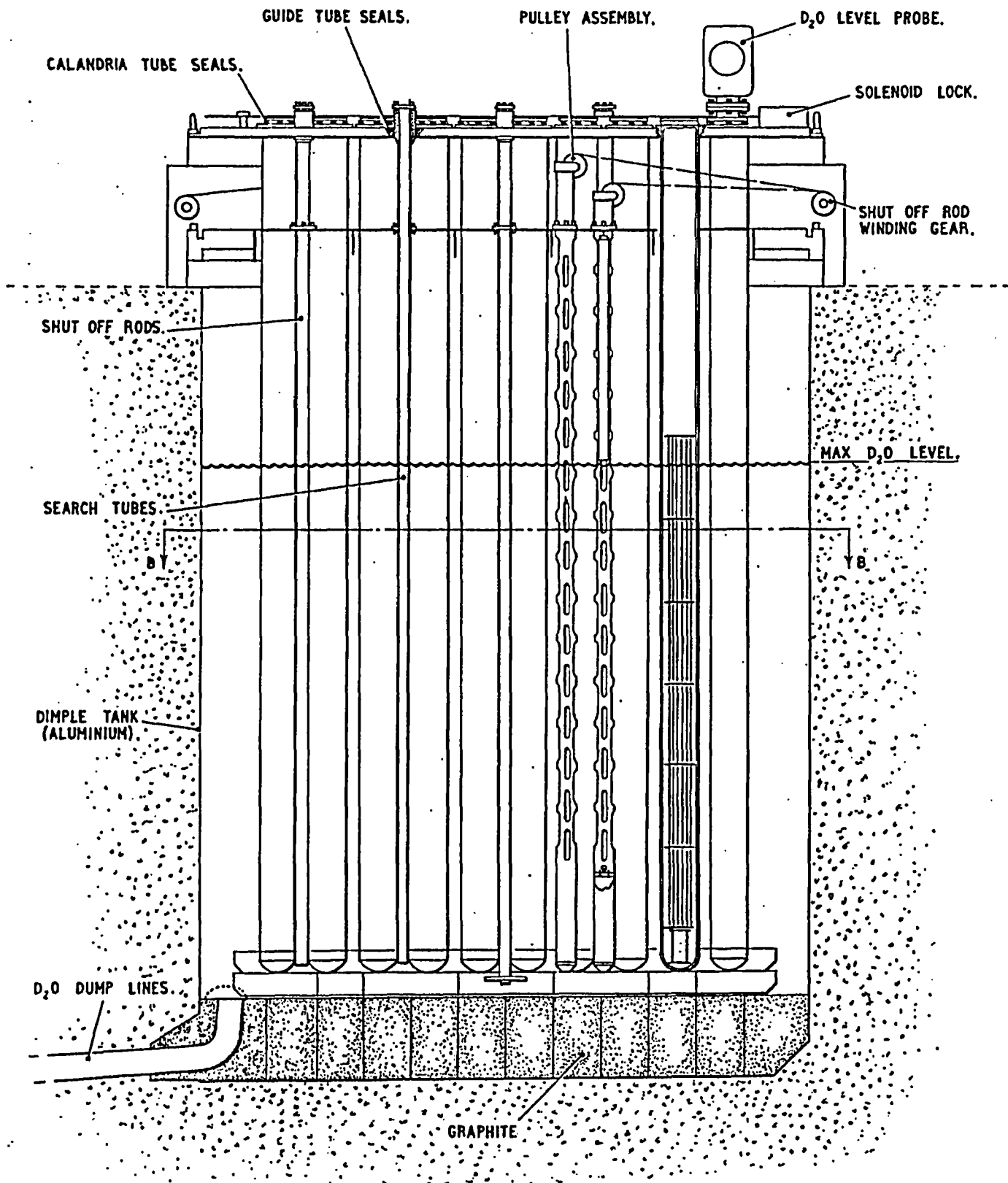


FIG.1. GENERAL ARRANGEMENT OF DIMPLE TANK.

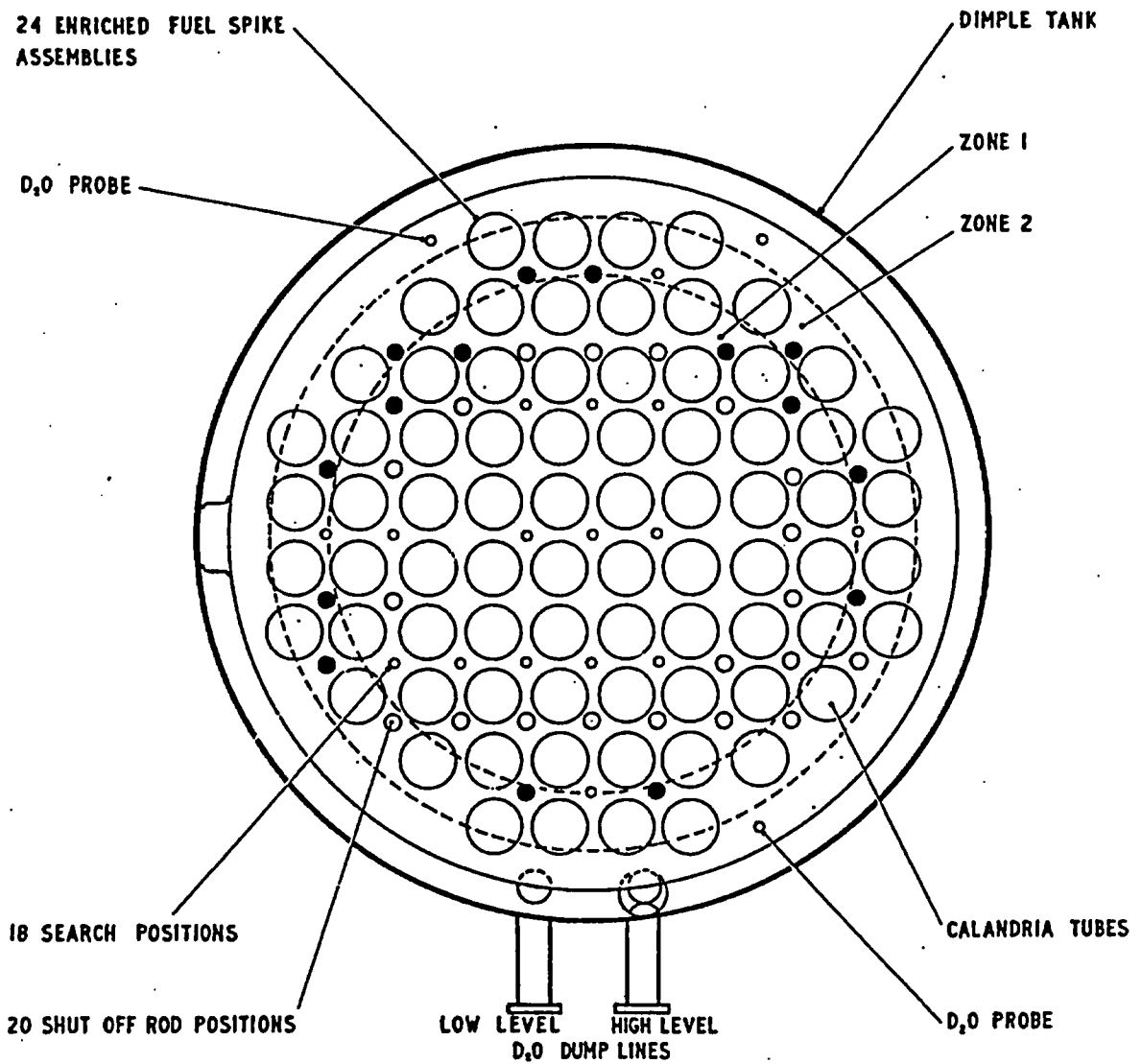
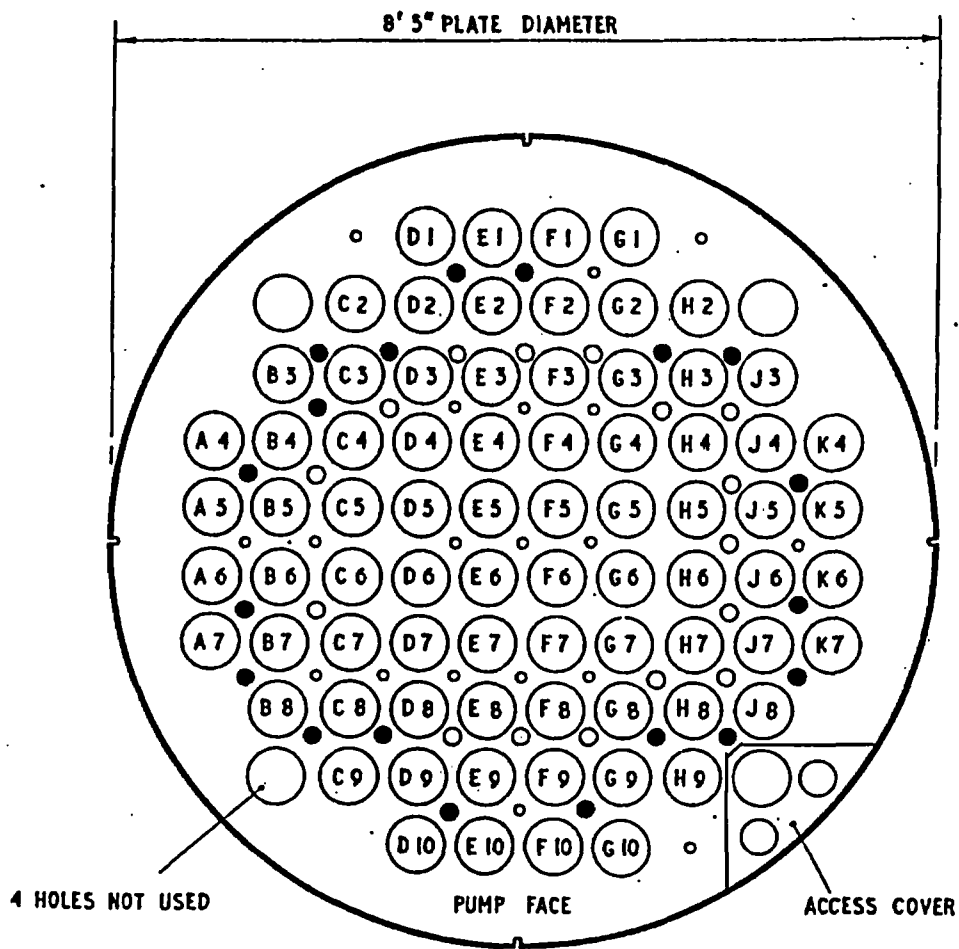


FIG 2. G.A. DIMPLE TANK HORIZONTAL SECTION B-B



- CONTROL ROD POSITION (INITIAL USE)
 - CONTROL ROD POSITION
 - SEARCH TUBE
- LATTICE PITCH 8.375"

FIG3. UPPER LATTICE PLATE

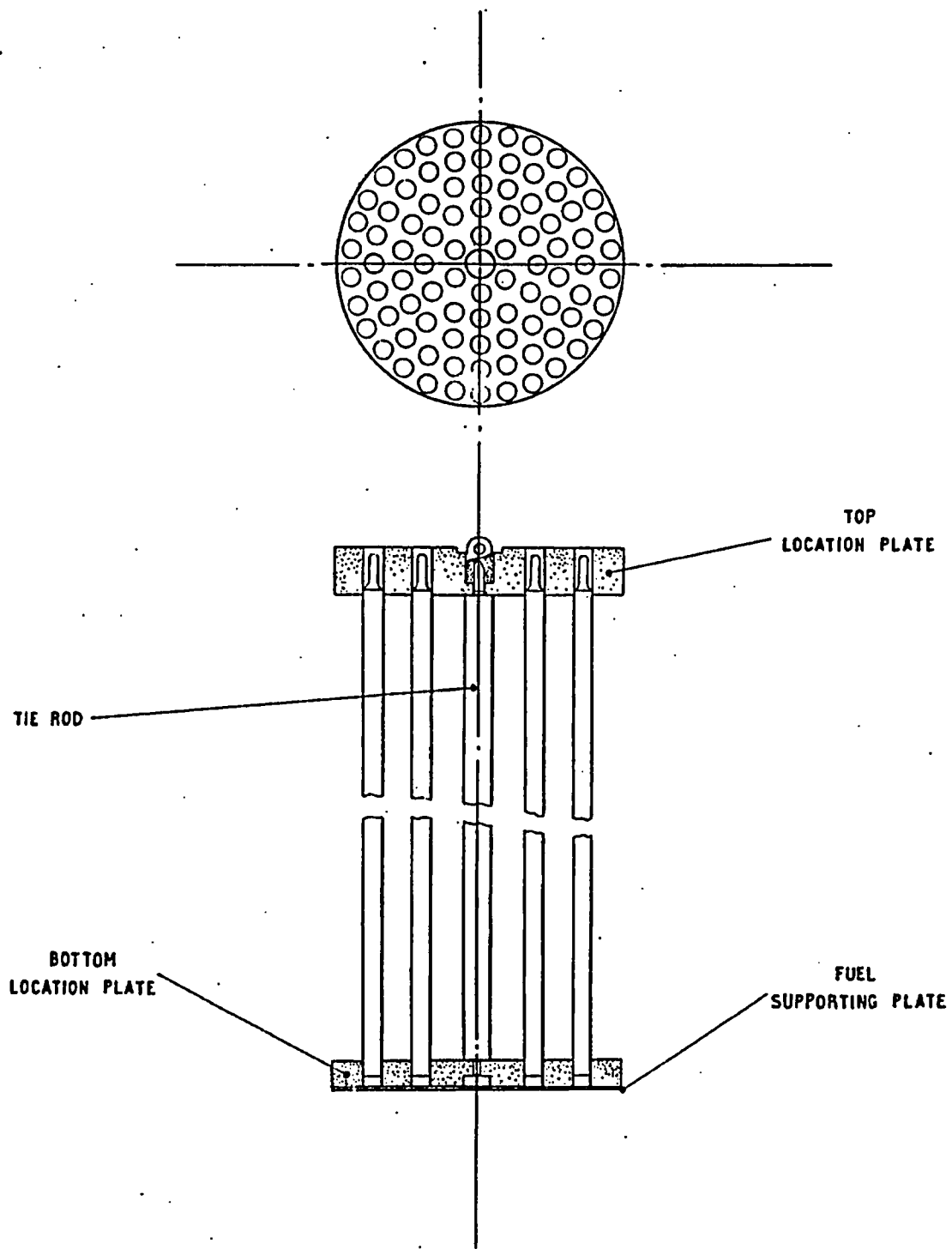
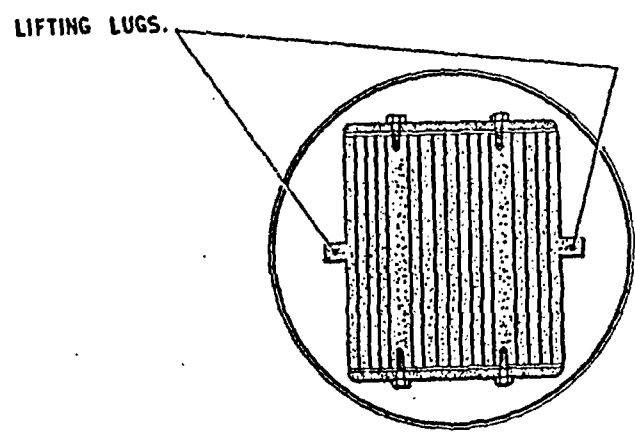
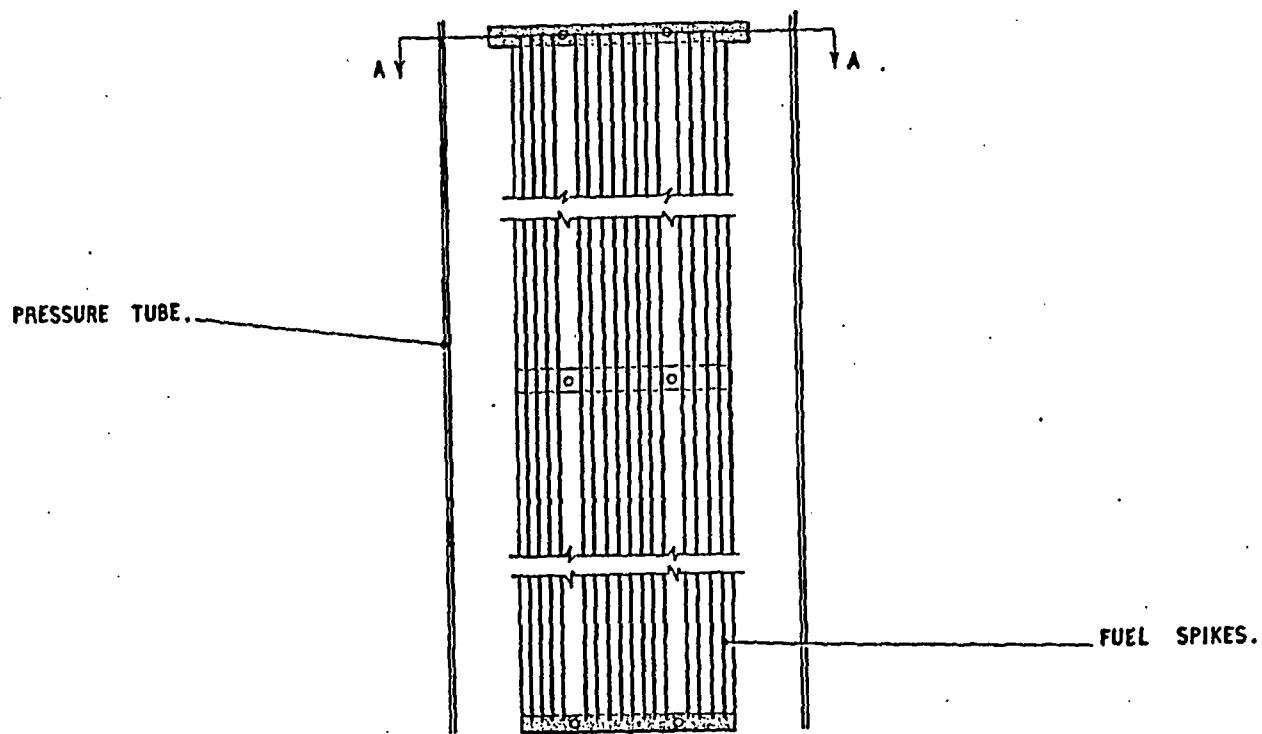


FIG 4. G.A. OF FUEL ROD ASSEMBLY.



VIA A-A.

FIG 5. SPIKE ASSEMBLY.

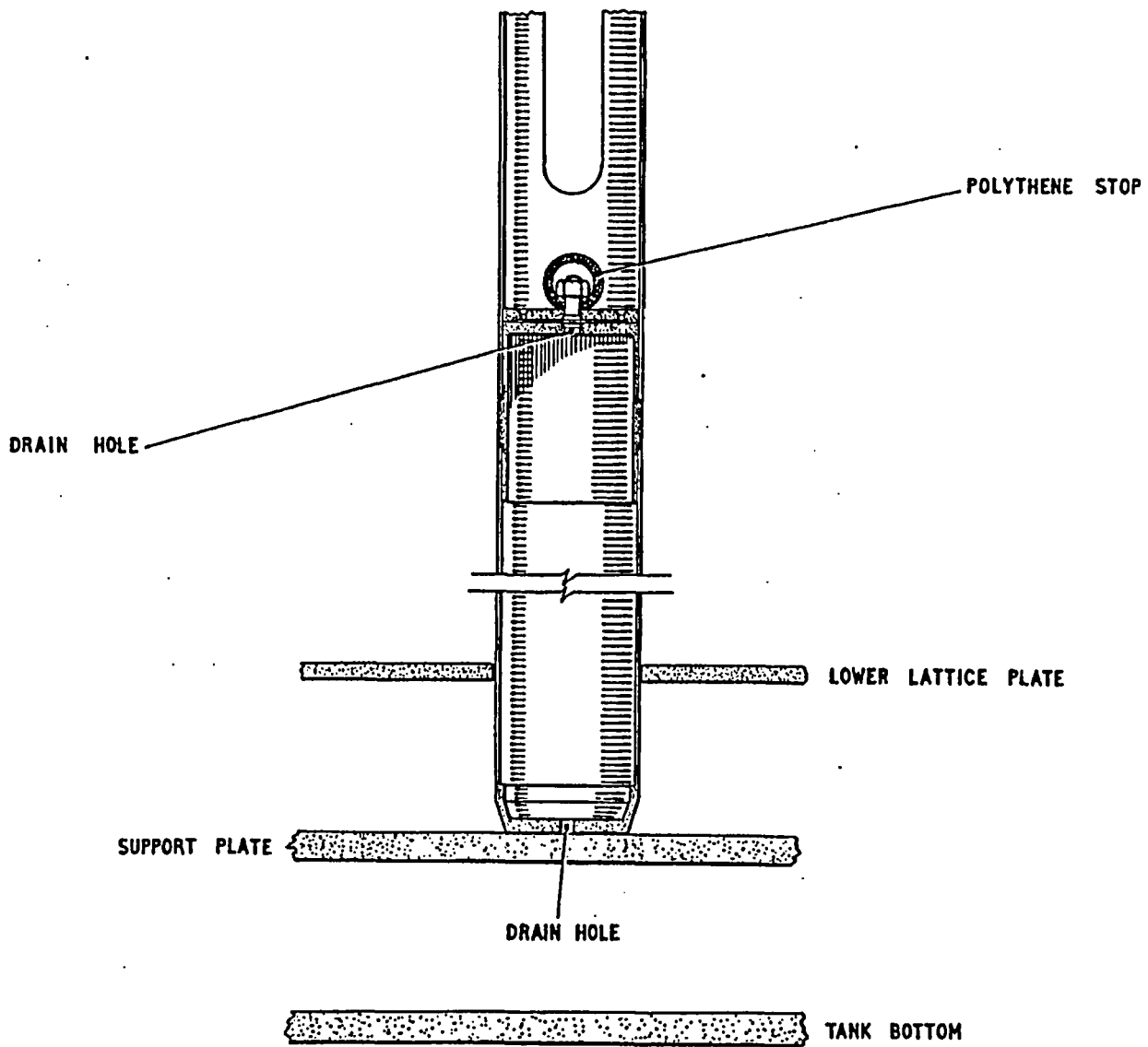
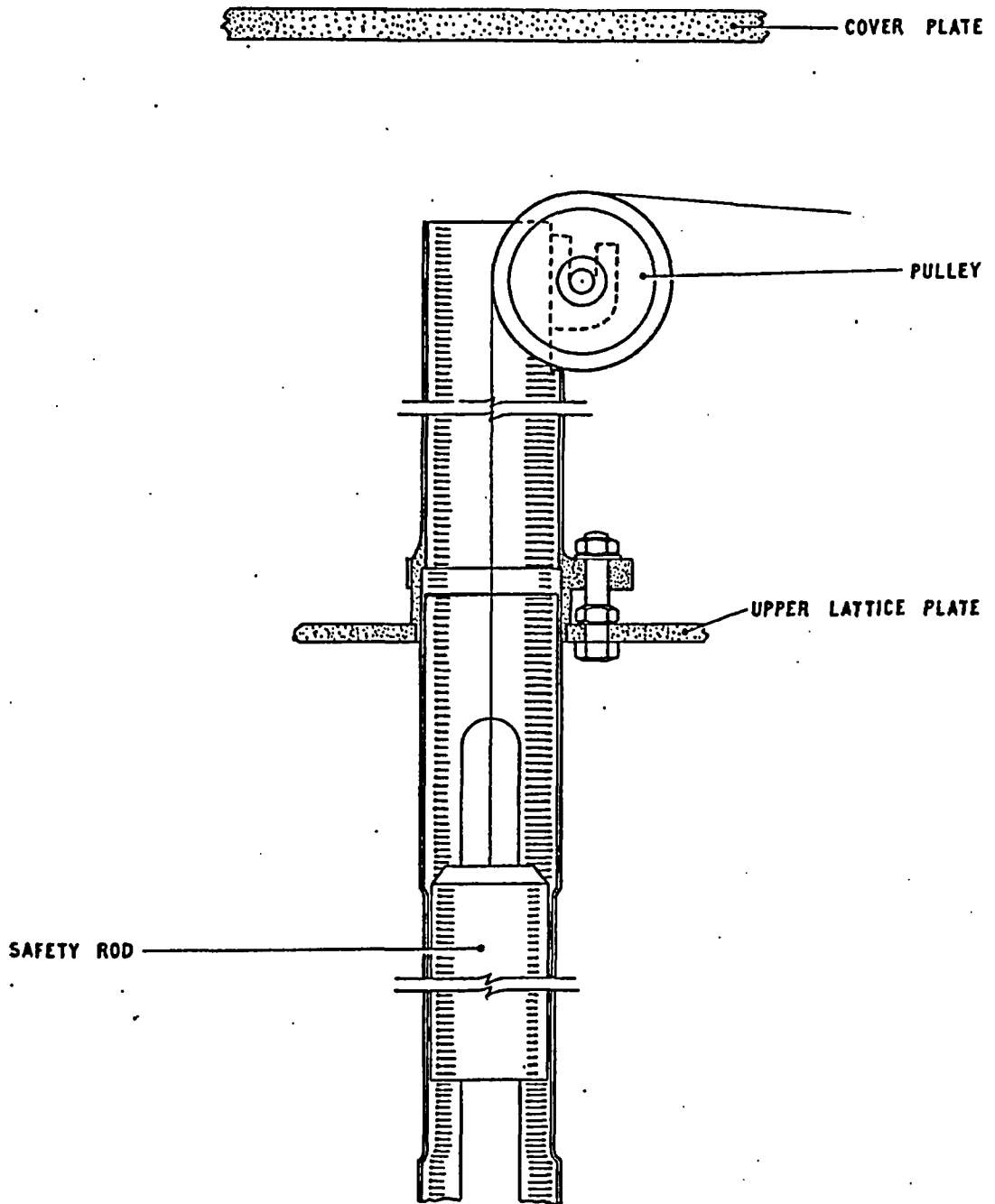


FIG.6. SAFETY ROD SLEEVE AND STOP.



0 1 2 3
INCHES

FIG 7. SAFETY ROD, STANCHION, AND PULLEY.

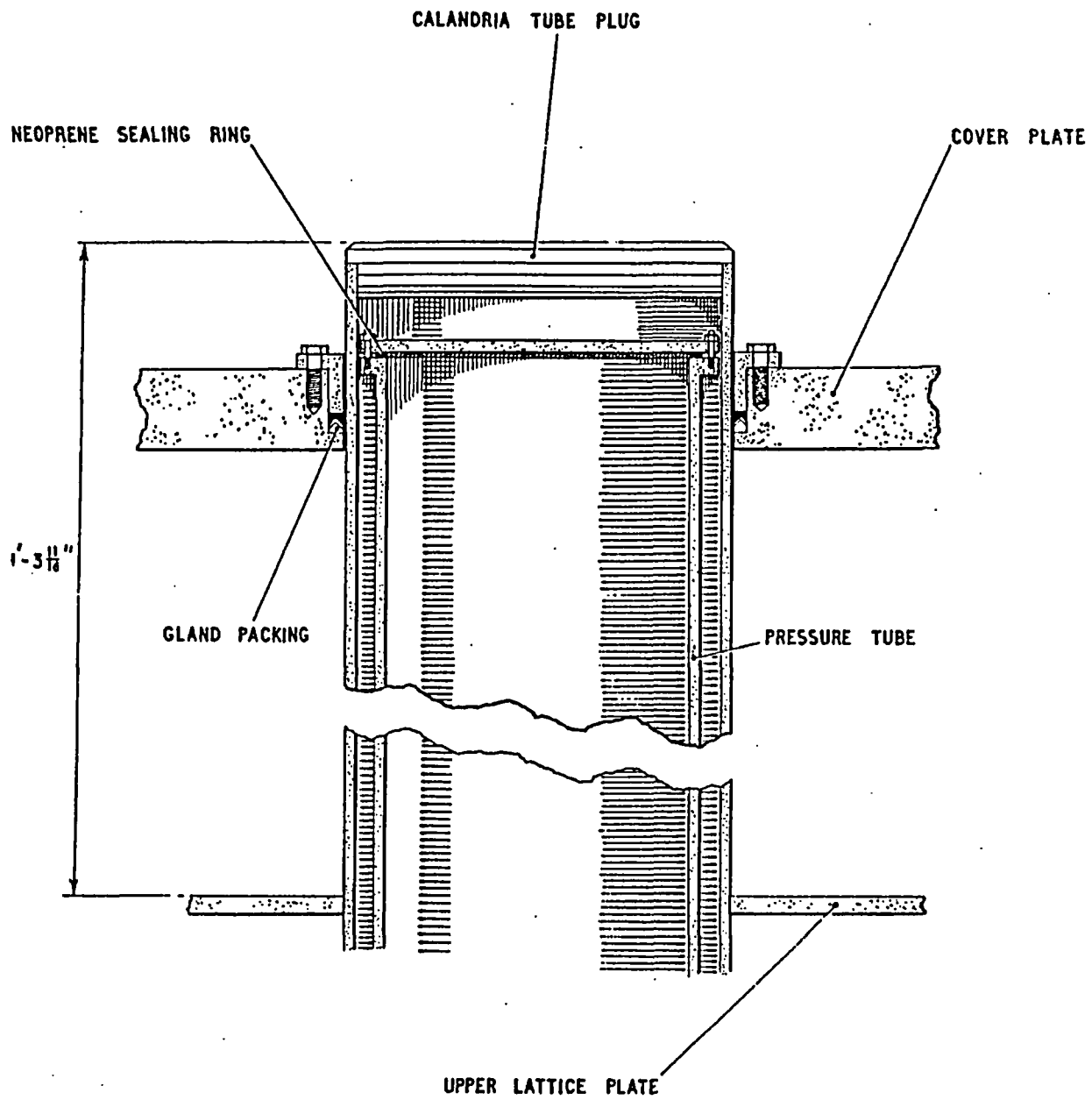


FIG 8. CALANDRIA AND PRESSURE TUBE SEALS.

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