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THE PHYSICS PROBLEMS OF REACTOR SHIELDING

**The Design and Performance of an Open-Well Shield
for High-Power Propulsion Reactor Testing**

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Abstract

The 1500 megawatt Phoebus 1B reactor and its 81-inch-diameter open-well shield constitute a unique combination of a high-power reactor source coaxial with a respectably attenuating bulk shield system possessing a single large end opening permitting free radiation leakage. This system is nuclearly rather simple, and is fairly easy to describe analytically, so the results of calculations and measurements made on it may be of interest to a large segment of the shielding community.

This paper first mentions the kinds of reactors tested in the nuclear propulsion program, and then discusses one basic reactor type, presenting for it, for the first time, a nuclear model with enough detail to permit valid calculations on its external radiation environment. It then touches on testing to show why a facility shield is needed and what it should do, and goes into particulars of the shield itself.

The approach to the nuclear design of this 19-inch-thick borated-water shield is sketched carefully, not only to show the reasoning leading to its final configuration, but also because this process lends further insight into the special nature of propulsion reactor testing. Engineering design is not covered, but a careful description is given of the shield's features as built.

Measurements relating to the shield system's performance are given. These include its reactivity effect, its thermal performance (including that of its borated-water coolant system), its lateral attenuation, and its effect on neutron fluxes in the area.

1. Introduction

The reactors now being developed in the U.S.A. for space propulsion applications are in concept rather simple heat exchange systems in which it is important to obtain high specific impulse and good reliability with minimum weight.

Specific impulse is a rocket engine performance parameter

roughly proportional to $\sqrt{T/MW}$, where T is the absolute temperature of the propellant gas, and MW its mean molecular weight. Specific impulse enters exponentially in rocket performance equations. Nuclear rocket engines offer to double it compared to chemical engines, which would provide gains very significantly affecting the cost, difficulty, and even reliability of advanced missions.

Since many of their design features are imposed by their applications, these reactors have quite unusual characteristics compared to their stationary counterparts. They are hydrogen cooled. They have high powers and very high power densities. Their components are exposed to extremely high neutron and gamma flux densities, in thermal environments which range from under 100°R to over 4000°R. They demand minimized weight in items not producing energy, such as reflectors or intrinsic shields, and, as a result, they exhibit high radiation leakage. They are designed to operate essentially unshielded in the emptiness of space. They, and any attendant external shield systems, must be ducted to permit the flow and escape of large quantities of hydrogen propellant. Their operating times are measured in minutes.

Because little data on these systems have been previously released, this report spends more time in bringing the reader up-to-date and less on calculations and results than the author would otherwise desire. But in order to show how some of these characteristics pertained to, and influenced, the design of the shield employed around the Phoebus 1B reactor during its testing in February, 1967, one must show something about how testing is done, for the test cell itself is a previous response to these characteristics and influences and is the facility which the shield is trying to protect.

2. Brief History of LASL Reactor Testing

The Los Alamos Scientific Laboratory has been involved in the nuclear rocket propulsion program, or Rover Program, [1] since its inception. It is not my purpose here to review the overall history of this program at Los Alamos nor to attempt to describe past or present alternatives, but solely to touch upon

the major propulsion reactor experiments conducted by LASL, which are summarized in Table 1.[2] These all concern themselves with uranium-graphite fueled reactors employing hydrogen as reactor coolant and as expellant to provide thrust.

The early Kiwi-A reactors had small fuel regions and thick, heavy reflectors. They used gaseous hydrogen, operated at power levels of ~100 MW, and had as primary test objectives the demonstration of the feasibility of the heat exchange concept and of the reasonableness of the fuel material choice.

The next reactors were of the Kiwi-B-1 class. Their relatively short runs explored reactor operations with liquid hydrogen as a stored and pumped propellant. They operated at higher power (1000 MW) and higher power density, assisted in developing fuel elements, and investigated various reactor design concepts.

Still at the 1000 MW power level, the later Kiwi-B-4's proved the basic feasibility of the present internal configuration, demonstrated restart capabilities, and showed ability to operate at sustained power for periods of several minutes without excessive fuel element corrosion from hot hydrogen. This reactor concept has undergone only minor changes in Los Alamos reactors of this general power level since the Kiwi-B-4D, so that we may think of the Kiwi-B-4D, the Kiwi-B-4E, the Phoebus 1A, and the Phoebus 1B as reactors so similar in nuclear characteristics that they possess practically identical external radiation environments. The main differences between Kiwi-B-4 and Phoebus 1 reactors are in capabilities; the Phoebus 1 reactors are evolutionary steps toward higher power density, better fuel element performance, etc. Some typical Phoebus 1B properties, obtained from experiment, are:[2] power, 1460 MW; hydrogen flow rate, 94.4 lb/sec; nozzle chamber temperature, 4040°R; nozzle chamber pressure, 736 psia; and fuel element exit temperature, 4400°R.

Improvements resulting from Phoebus 1 experience are being incorporated into Phoebus 2 reactors, which will exhibit a culmination in power levels for LASL reactor testing. The Phoebus 2 is to operate at 5000 MW thermal power with corresponding internal flux densities greater than 5×10^{16} nv. Other Phoebus 2A design parameters are:[2] hydrogen flow rate, 285 lb/sec; nozzle chamber

<u>REACTOR/EXPERIMENT DESIGNATION</u>	<u>DATES</u>	<u>~ POWER MEGAWATTS</u>	<u>OBJECTIVES</u>
Kiwi A	July 1959	100	Demonstration of feasibility and material choice.
Kiwi A' through A3	July 1960 - Oct. 1960	100	Investigate basic fuel element, control, and design problems. Used gaseous hydrogen. Core structural problems.
Kiwi B1A and B1B	Dec. 1961 + Aug. 1962	1000	Operation on liquid hydrogen. Higher power. Further fuel development.
Kiwi B4A	Nov. 1962 - Aug. 1964	1000	Resolution of present fuel-core configuration. Restart capability and 10-minute duration.
Phoebus 1A and 1B	June 1965 - Feb. 1967	1000 1500	Details of core fuel corrosion, high power density, and 30-minute duration demonstration.

TABLE 1. SEQUENCE OF MAJOR LASL NUCLEAR PROPULSION REACTOR EXPERIMENTS

temperature, 4500°R, nozzle chamber pressure, 625 psia; and fuel element exit gas temperature, 4550°R.

The Phoebus 1B had the largest energy release ($\sim 2.7 \times 10^6$ MW-sec) of the reactors to date, thanks to improvements in hydrogen storage capability. As we shall see later, this capability also brought about a requirement for a reactor shield.

3. The Phoebus 1B Reactor

In this paper we will be concentrating our attention on the Kiwi-B-4 and Phoebus 1 reactors. Figure 1 shows the general features of a Phoebus 1 reactor as they would appear in an axial slice and in a cross section, both taken through the fueled region. The cross section illustrates the disposition of the poison vanes on the 12 rotating control rods. The axial section, which includes some dimensions in inches, shows that the uranium-graphite fueled region is about 35 inches in diameter by 52-inches long. It is surrounded by a graphite reflector annulus about 2-inches thick which is inside the 4-1/2-inch-thick beryllium reflector annulus containing the control rods. The whole reactor and reflector system is encased in a near-cylindrical aluminum pressure shell, which is about 7/8-inch thick and just over 51 inches in outside diameter in the region alongside the core.

The thin, flat, unfueled structure at the upper (nozzle) end of the core and the thicker, flat, unfueled structure at the lower end of the core primarily serve core-support purposes. A thermal neutron absorber (not shown) is located between the core support plate and the fueled region, in order to minimize effects from the return of neutrons moderated in the high-density hydrogen about to enter the core.

When this epithermal reactor is operating at its design thermal power of 1500 MW, its internal flux density is near 4×10^{16} neutrons/sq cm/sec, and its leakage neutron flux densities in the vicinity of the pressure shell exceed 10^{14} neutrons/sq cm/sec.

The entering flow of liquid hydrogen first cools the nozzle, and the coolant then successively passes through the reflector,

Fig. 1. Main Features of Phoebus 1B Reactor.

core-support plate, and fueled region, warming at every step to finally emerge as a very hot gas.

The next figure (Fig. 2) shows a mathematical reduction simulating this reactor for calculational purposes. The axial power profile to be used for the fission-originated radiation sources in this model is of cosinusoidal shape, though shifted so that it has higher intensity toward the core inlet end than the nozzle end. This profile is adequately described by the function $1.36 \cos 0.01986 (z - 60.0)$, where z is distance in centimeters measured from the core inlet end. The radial source function is flat -- i.e., uniform -- in recognition of the fact that these reactors are designed to provide the same (maximum) energy uptake to the gas flowing in every core coolant channel regardless of its radial location. To complete the calculational base, note that Fig. 2 is scaled and its material regions numbered; Table 2 gives the compositions and densities ascribed to each of these regions, plus some data applicable to the reactor shield system.

4. Testing at Test Cell "C"

All power testing of propulsion reactors presently takes place at the Nuclear Rocket Development Station, which is operated by the joint AEC-NASA Space Nuclear Propulsion Office in an isolated region of the Nevada desert adjoining the Nevada Test Site. Various facilities exist at NRDS[1], including two complexes for engine or reactor assembly and disassembly, and three different test cells with their associated hardware and control rooms, but we will here concentrate on Test Cell "C", at which all tests of concern to this paper have taken place. An overall view of this test cell, including its control area some two miles away, is shown in Fig. 3.

As Fig. 4 may imply, a reactor to be tested at this test cell is mounted on its own test car in an assembly building, transported to the test cell on that car, and plugged into the side of the test cell building. Its prime mover is then withdrawn (Fig. 5). The car's stepped shield plug mates with an

Fig. 2. Calculational Model of Phoebus 1B.

Material	Elemental Densities (g/cm ³)										Total Density (g/cm ³)
	U	C	Be	Al	Nb	Fe	H	O	Na	B	
1	.214	1.234	-	-	.136	.037	-	-	-	-	1.621
2	-	1.488	-	-	-	-	.002	-	-	-	1.490
3	-	-	1.656	-	-	-	.011	-	-	-	1.667
4	-	-	-	2.700	-	-	-	-	-	-	2.700
5	-	-	-	-	-	-	.008	-	-	-	0.008
6	-	-	-	1.626	-	.333	.003	-	-	-	1.962
7	-	1.091	-	-	.776	.036	-	-	-	-	1.903
8	-	-	-	-	-	4.000	-	-	-	-	4.000
9	-	-	-	-	-	1.180	-	-	-	-	1.180
Shield borated water:					-	-	.107	.907	.010	.020	1.044
Privy roof borated water and steel balls:					-	4.800	.043	.363	.004	.008	5.218

TABLE 2. ELEMENTAL DENSITIES AND COMPOSITIONS FOR THE REACTOR MOCKUP

Fig. 3. Overall View of Test Cell "C".

Fig. 4. Phobos 1B Entering the Test Area.

Fig. 5. Phoebe 1A in Test Position.

opening in the 5-1/2-foot-thick test cell face, and most reactor services pass through this plug, including fluids (such as coolants, purge gases, and hydrogen propellant), reactor control signals and control power, and instrumentation calibration and output signals.

After test, the reactor is too radioactive to be approached, so it is coupled to its prime mover and taken from the area by remote control.

For convenience, reactors are tested at this site in an up-firing position -- i.e., the nozzle is oriented upward and the resultant thrust is downward on the test car. The reactor center line stands 13 feet 2 inches from the test cell face. The reactor's control rod actuators are located inside a test car room, known colloquially as the "privy", lightly shielded from the reactor by its roof. This privy roof shield -- 9 feet square and 19-inches thick, employs aluminum plates, water, and steel balls.

The importance of a test facility lies largely in the equipment it provides. The dominant feature of Test Cell C (shown in closer view in Fig. 6) is its enormous liquid hydrogen capacity, for it possesses two fifty-five-foot-diameter dewars installed in 1965, each capable of storing over 250,000 gallons, and two "small" dewars each capable of storing 55,000 gallons, for a total storage capacity of over 600,000 gallons. (This is over 360,000 pounds of liquid hydrogen, which can provide for more than an hour's operation of a 1500 MW propulsion reactor consuming almost 100 pounds of hydrogen per second.) Test Cell C has a novel hot water to liquid hydrogen heat exchanger to generate high pressure gaseous hydrogen for driving the turbines of the propellant pumps, and it possesses various large investments in such things as storage for other cryogens and gases, in instrumentation and control systems, and in its liquid hydrogen pumps.

The expense of these facilities discourages their duplication and economy demands their fullest possible use by all appropriate organizations. Consequently, a decision was made in 1965 to move some future NERVA testing to Test Cell C, in addition to the Los Alamos Scientific Laboratory testing already planned for there. (NERVA stands for Nuclear Engine for Rocket Vehicle

Fig. 6. Dewars and Other Facilities at Test Cell "C".

Applications, and is a program contracted to an industrial team of Aerojet-General and Westinghouse to incorporate reactor developments into flight-type engine configurations.[1])

This joint usage policy meant that tests would be more frequent, and the longer reactor operations permitted with the newly installed hydrogen capacity would involve larger fission energy releases, with proportionately larger problems from postoperating activation. Thus, in the late summer of 1965, it was also decided that shields should be provided around all propulsion reactors to be tested at Test Cell C. The primary purpose of these shields would be to suppress area activation from the forthcoming runs of large energy release. The shields' secondary purposes would be to reduce test cell face heating, to promote more convenient personnel access during periods when the once-operated reactors were still present, etc.

Figure 7 shows the energy-normalized postshutdown activation measured on the test pad after the Kiwi-B-4 runs.[3] The overall energy release of the unshielded Kiwi-B-4E was $\sim 7 \times 10^5$ MW-sec, yet its residual activation had permitted relatively unrestricted pad operations within about two months. A design objective for the new shields was to hold pad activation from future runs, insofar as possible, to levels as low or lower than those resulting from the Kiwi-B-4E.

5. Shield Reactivity Considerations

Once the goal of a shield was decided upon, it was necessary to get some quick feeling for what it might accomplish and what its effects upon a reactor might be.

It was most important that the neutronic interaction between reactor and shield should be very small. The reactor fuel elements are operated near the limit of their allowable power generation and cooling capability in order to obtain the highest possible performance for propulsion purposes, and any shield interaction causing substantial increases in local fission rates by perturbing the overall fission distribution within the fuel, or causing gross changes in the reactivity worth of the control rods

Fig. 7. Decay Characteristics of Test Cell "C" Pad Activation.

would likely be so undesirable as to preclude that shield's use. Of course, some interaction had to occur; the heat generated in the shield would require water cooling -- one could expect several megawatts, with local material heating rates of ~ 3 W/g -- and the reactor would certainly be sensitive to any external moderator or reflector.

(This was truly a novel design situation. Reactors and shields are usually designed together. Yet here a shield was needed for a reactor system whose ultimate use would incorporate no such shield, and whose performance possibilities for that ultimate use must be carefully measured, while the reactor's construction practically precluded its ignoring the undesired and extraneous shield environment. Ideally, a shield was wanted which would provide its own isolation from the reactor, absorbing neutrons as if it were the void of space.)

For a quick investigation, approximately twenty one-dimensional DTF neutron transport k-calculations were run in order to arrive at a feeling for the neutronic effects of hydrogenous material outside the pressure shell. One datum was available for normalizing such reflector calculations. That was the empirical 1.8\$ reactivity worth of a 2-inch thickness of polyethylene 3 feet high arranged in a close-fitting sheath around a Kiwi-B reactor. Such a sheet in a 52-inch (core-sized) length would be expected to be worth over 2\$, with small further gains with increasing length.

These survey problems first mocked up either a bare reactor or one with two inches of close-fitting polyethylene around it, and adjusted the relative boron concentration in a cylindrical poison sheath inside the reactor's reflector, to find a concentration for which the calculated reactivity addition attributable to the polyethylene would be correct. (In these one-dimensional problems, one could have adjusted either the density or the position of the sheath simulating the control rods; we chose density.) The reactivity increments obtained in this way appear in the first line of Table 3.[4] Based on the polyethylene data, the 0.002 boron concentration column warranted the most attention in this table and was considered as most likely to (conservatively)

Shield No.	Shield Description	* N _B :	Reactivity Addition, \$				
			0	0.002	0.005	0.01	0.014
1	2-in. polyethylene, close fitting	**	6.8 (1.33)	2.6 (1.14)	1.41 (1.08)	0.94 (1.05)	0.72 (1.04)
2	0.5-in. Al plus 4-in. H ₂ O and 2-in. Al, homogenized, close fitting		6.1 (1.29)	2.6 (1.13)			0.84 (1.04)
3	Shield 2 with 1/4-in. "thick" B between reactor and shield		0.98 (1.03)	0.73 (1.03)			0.55 (1.02)
4	Shield 2 with 15-in. gap between reactor and shield		4.5	1.8 (1.09)	0.94 (1.06)	0.73 (1.04)	0.59 (1.03)
5	Shield 4 with 1 w/o B added to Al in H ₂ O-Al region		1.6 (1.06)	0.83 (1.04)			0.44 (1.02)
6	Shield 5 with B concentration doubled		1.3 (1.05)	0.73 (1.03)			0.42 (1.00)

* N_B = Boron concentration in the reflector sheath in units of 10²⁴ atoms/cc.

** Values in parentheses = $\frac{\text{Ratio of edge-to-central power with shield}}{\text{Ratio of edge-to-central power without shield}}$

TABLE 3. ONE-DIMENSIONAL CALCULATIONS OF REACTIVITY AND POWER PERTURBATION FOR VARIOUS SHIELD GEOMETRIES AND REFLECTOR POISON CONCENTRATIONS

predict shield interactions.

Other reflectors (or rough "shields") consisting of 6-1/2-inch (effectively "infinite") thicknesses of water and aluminum were then studied. Some of these calculations placed a half-inch of aluminum backed by a six-inch (2-inch aluminum-4-inch water) homogenization directly around the same reactor model. Other calculations interposed a "black" sheet of boron between the reactor and this shield. The effect of introducing a gap in such one-dimensional calculations was studied by pulling the shields 15 inches away from the reactor. Lastly, the effects of these shields at 15 inches from the reactor were studied for one and two weight percent boron added to the water of the aluminum-water mix region.

The results, given in Table 3, showed that shield poisoning was necessary, but they were not particularly sensitive to the form of this poison, whether sheets or 1 or 2% solution. The one and two percent boron solutions differed little. Both held the calculated reactivity effect of the shield down to a possibly tolerable 75 or 80 cents and held the fuel-edge fission increase down to just a few percent. (Little advantage would be likely from further poisoning; the small difference in reactivity between the 1 and 2% solutions seemed to imply that most of the residual reactivity increment came from essentially unpoisonable effects associated with fast-to-intermediate neutron reflection.) Other experience* indicated that such one-dimensional calculations on gap shields could overestimate reactivity contributions, perhaps due to inadequate treatment of neutron leakage, and we expected that might also be the result here. Thus, these early studies intimated that the reactivity effects of a well-poisoned shield would be inside our allowable control margins and that any small edge-peaking in fissions associated with the shield could be accommodated by internal reactor adjustments. Though it was deemed necessary to obtain final information through use of a shield mockup in critical assembly studies, prior to any such

* For example, we were acquainted with some Westinghouse tests where a predicted 21 cents reactivity from a one-dimensional calculation on a 135° partial shield turned out to be an empirical 9.4 cents reactivity when the shield was built.[5]

experimental confirmation, it was decided to employ an aluminum shield cooled by flowing borated water for this basic shielding job.

The basic considerations in choosing borated water for the form of the poisons in the side shield dealt with cost and convenience. That reasoning is omitted here, though two arguments pertinent to shielding may be mentioned. Borated water simplifies the problem of heat removal, since the large power generated from the (n,α) reaction heat goes directly into the fluid rather than having to be removed from the walls by conduction. Borated water also avoids other problems that might have been posed by solid poison-bearing materials. It was suspected that at discontinuities -- such as penetrations for shield tie members, etc. -- there could be extra thermalization of neutrons which might have gotten into the shield through unpoisoned spaces, thus causing anomalously high local absorption in the solids in the very regions where the local flow rates might be least.

6. Sizing the Shield

A rather arbitrary side clearance of 15 inches was chosen to afford some working volume between the reactor and shield, thus establishing the shield's internal diameter at approximately 81 inches.

Previous studies on a Phoebus 2 geometry had shown that a shield overlapping the exit end of the reactor and fitting tightly around the nozzle throat would produce intolerably large reactivity and core-end fission effects; though such shields would have offered minimum neutron escape, a cylindrical well shield was chosen instead.

The greater the shield's height, the less the solid angle for neutron escape, and the better the shield would be. However, an upper limit was fixed by the fact that, for the shield's own sake, it should not intrude too close to the nozzle flame. We wished to allow this shield to serve for the testing of the NRX-A (NERVA) reactors, which stand about three feet higher in their testing configuration than ours (though they are very similar

reactors internally). A shield height which was near the value generally regarded as the greatest permitted by the Phoebus 1 nozzle exit plume was chosen, which would conveniently allow the NRX-A's propellant feed line to come over the top of the shield rather than through it. This overall height was 150 inches, measured from the privy roof. Such a shield would extend from ~18.5 inches below the Phoebus 1 core (the privy roof location) to ~80 inches above it, its lower interior dimensions with respect to the Phoebus reactor are depicted by the coordinate lines drawn on Fig. 2.

7. Fixing Neutron Attenuation Objectives

To estimate the effectiveness of a shield of this height for the two reactors around which it might be employed, we treated it as a black body and studied the fast neutron leakage through its end opening. One case positioned the top of the shield 78 inches above the core end (Phoebus 1 test configuration) and one placed the shield only 42 inches above the core end (NRX-A configuration). These studies used neutron moments data incorporated into the QAD point kernel program [6] by D. M. Peterson. The results, shown in Fig. 8, indicated that approximately 17% of the fast neutron leakage from an unshielded reactor would escape the "low" shield, and approximately 10% of the unshielded leakage would escape the "high" shield. Both reactors emit most of their thermal neutrons to the side, and these would be expected to experience great attenuation, but such neutron emission would not be expected to be half the total. Thus it was probable that this "highest practical" shield necessarily would allow a neutron escape near 5 - 10% of that from a bare reactor.

If 50 - 100 out of every 1000 neutrons were going to get out of our shield regardless of what was done at the reactor's side, it hardly made sense to strive for a lateral attenuation so high as to capture every one of the other neutrons -- i.e., we could let perhaps 5 to 10 of those 900 - 950 neutrons escape via shield transmission without much changing the overall activation level in the reactor environs. Therefore, it was decided to aim for a

Fig. 8. Relative Fast Neutron Leakage as a Function of Shield Height

lateral attenuation factor of at least 100 for fast neutrons, but not to be concerned over much more. This led to the use of about 15 inches of water, based on literature values for the survival of fission neutrons in a water medium. In picking this thickness, the additional neutron attenuation to be expected from the aluminum to be used in the shield was deliberately overlooked.

8. Establishing a Gamma Ray Attenuation

The shield's gamma ray attenuation objective was decided primarily through study of empirical heating data from the Phoebus 1A run, which showed that the temperature approximately 1/3 of an inch into the concrete test cell face had risen 167° during a run whose total fission energy release was about 7×10^5 megawatt-seconds.[7] The possibility of runs of 4 - 5 times this overall energy release was contemplated and, if the previous increase was all due to radiation heating, test cell face temperatures over 900°F might be expected from these projected operations if the heating was not attenuated. Such temperatures would be intolerable for concrete, whose water of hydration would likely be expelled at such high heat.

When the 167° temperature rise associated with Phoebus 1A was examined, it was felt that at least 135° of it was easily explicable as nuclear radiation heating. The other 30° might be associated with the flame heating,* or with nuclear radiation heating that had not been taken into precise account. The conservative approach was to assume this came from the flame for, if that had caused the 30° rise, a run liberating five times as much energy release -- and therefore consuming five times as much hydrogen propellant -- might possibly develop a 150°R temperature increase on the test cell face through its flame alone. One had to consider the possibility that any nuclear radiation heating permitted would have to be added to a basic, unattenuatable 150°

* The hot propellant emerging from the reactor burns during ground tests, with a greatly increased thermal radiation energy release which can add significantly to the heating of components near the reactor.

temperature increase. However, only a factor-of-10 attenuation on the "known" gamma ray heating of the test cell face was needed to limit that to about a 70° rise in such future runs. Even with an initial (ambient) temperature of say 70°F, and a 150° temperature rise from the flame, a further temperature rise of ~70° from nuclear radiation heating would still result in temperatures less than 300°F. Since such a temperature would be safe for concrete, it was decided that an approximate factor-of-10 attenuation for gammas would be satisfactory.

With this in mind, we ran a brief series of point kernel calculations to study the gamma ray transmission of various aluminum-water mixes. The mass attenuation characteristics of these materials are so similar that the net attenuation produced for any mass is insensitive to its distribution between aluminum and water. Approximately 66 grams per square centimeter were found to be required for a factor-of-10 effect straight away from the reactor side. Since 15 inches of water was to be used for neutron attenuation (providing 38 grams per square centimeter), only 4 inches of aluminum (27 grams per square centimeter) was needed in addition, and it appeared feasible to build a shield satisfying all reasonable attenuation objectives within an overall thickness of 19 inches.

(The radiation heating at the test cell face was studied for plain water and aluminum in this shield, ignoring the adverse reactivity effect this would have on the reactor. We wondered if avoiding the exothermic (n, α) reaction in the boron of the water might permit us to lower the power in the shield, which otherwise appeared to be about 10 megawatts per thousand megawatts of reactor power. This study demonstrated another good reason for having borated water, for, without it, the 2.2 MeV gammas from neutron capture in the water's hydrogen increased the dose outside the shield to nearly 1/3 that from an unshielded reactor at the same distance.)

9. Nuclear Heating Calculations on the Shield

Space does not permit a full description here of the studies

which went into determining the shield's nuclear heating. These were performed by people in LASL Group N-2, including C. W. Watson, J. R. Streetman, D. M. Peterson, R. E. Malenfant, and G. A. Graves. A few of them will be published elsewhere. The results of some are collected in a 22-page LASL internal memorandum.[8]

The neutron heating analysis proceeded through one-dimensional DTF and two-dimensional DDK neutron transport calculations to Monte Carlo studies. The DDK calculations were run in the fixed source (inhomogeneous mode). The problems were so large that machine storage would not accommodate the usual flux times cross-section integrals, or "activity tables", so it was necessary to perform auxiliary processing of tape outputs of the multigroup fluxes to obtain the DDK predictions of kinetic heating and neutron capture. Side shield studies were performed without the privy roof, and vice-versa. The DDK fluxes at deep penetration distances showed little tendency to converge. The total kinetic heating and neutron absorption in gross shield regions was ultimately found by Monte Carlo methods, while its distribution within these regions was presumed to be that shown by the DDK problems. Completely trustworthy neutron spectra throughout the side shield probably do not yet exist, though some promising comparisons have been reported [9] between Monte Carlo results and the fissions observed in enriched uranium wires placed in and about the shield during the reactor runs.

All shield gamma ray heating calculations used the QAD point kernel code [6] with water buildup factors. The core gamma source is described by a 19-line spectrum containing -- when we are not concerned with its time dependence -- 86 MW per 1000 MW of reactor power. (For regions in and near these reactors, frequent comparisons have been made between core source heating predictions from the QAD point kernel code and those from sophisticated Monte Carlos. Good agreement -- within 10 to 20% -- is usually shown, with the evidence of the discrepancies tending to suggest that QAD usually overpredicts the heating slightly.) Several other sources of non-core capture gamma rays, with intensities and distributions derived from neutron studies, were also considered. Plots of QAD values for the gamma ray heating at points within a region

were integrated to obtain the total energy deposition over a region.

Table 4 gives values for the predicted energy deposition for the side shield, separated into the heating in the aluminum plates and in the water. The total predicted was about 9.8 MW per 1000 MW of reactor power. Similar predictions for the privy roof shield, made without allowance for any protection afforded by the side shield, gave values of about 2.4 MW per 1000 MW of reactor power.

10. The Final Shield Arrangement

The shield is a 150-inch-high annular aluminum structure 19-inches thick and 81 inches in interior diameter. It consists of 8 full-height segments, which overlap along their sides to prevent unattenuated radiation leakage between segments (cf. Fig. 9).

The shield's surface plates and curved interior flow baffles (Fig. 10) have a total aluminum thickness of four inches ($3/4$, $1/2$, $1/2$, $3/4$, $3/4$, and $3/4$ inch -- going from inside to outside), which divide it into five spaces (1, 3- $1/2$, 3- $1/2$, 3- $1/2$, and 3- $1/2$ inches) for series flow of the borated cooling water. The thin (1 inch) flow space closest to the reactor keeps fluid speeds high for good heat transfer. Water enters this first space from a small chamber which carries the flow inward along the bottom of each segment; it exits from the last flow space into a plenum at the top of the segment exterior. The shield's areal mass density, in the normal direction through it, is about 67 g/cm^2 , of which $\sim 27 \text{ g/cm}^2$ is due to the 4 inches of aluminum and $\sim 40 \text{ g/cm}^2$ is due to the 15 inches of borated water (whose density is $\sim 1.05 \text{ g/cm}^3$).

The shield was originally intended to have a total flow of 2000 gpm, which will adequately remove its heat, but, for reasons of shield hydraulics, it was operated in the Phoebus 1B full-power test at a flow rate of ~ 4500 gpm. (This high flow rate prevented obtaining usable data on the water temperature increase per flow pass, as had been desired, leaving only integral data.)

Total Deposition (in MW per 1000 MW Reactor Power)

<u>Region</u>	<u>Reactor Gamma</u>	<u>Non-Reactor Gamma</u>	<u>Neutron Kinetic</u>	<u>Boron(n,α)</u>	<u>Total</u>
1st aluminum plate (3/4-in.)	.77	.12	.03	-	.92
1st water space (1-in.)	.36	.06	.34	3.00	3.76
2nd aluminum plate (1/2-in.)	.37	.07	.01	-	.45
2nd water space (3-1/2-in.)	.85	.18	.34	1.39	2.76
3rd aluminum plate (1/2-in.)	.17	.04	-	-	.21
3rd water space (3-1/2-in.)	.49	.10	.08	.11	.78
4th aluminum plate (3/4-in.)	.18	.04	-	-	.22
4th water space (3-1/2-in.)	.28	.05	.03	.02	.38
5th aluminum plate (3/4-in.)	.10	.02	-	-	.12
5th water space (3-1/2-in.)	.14	.02	.01	-	.17
6th aluminum plate (3/4-in.)	.05	.01	-	-	.06
Total from each source	3.76	.71	.84	4.52	9.83

TABLE 4. DISTRIBUTION OF PREDICTED HEATING AMONG SIDE SHIELD REGIONS

Fig. 9. View of the Basic Shield Annulus.

Fig. 10. Internal Construction of a Shield Segment.

A lip on a horizontal bracket welded onto the upper outside surface of each shield segment permits hooking the segment over the top edge of a support girdle. Four segments, making up a shield half, hang on a girdle, which is a 1-inch steel plate five feet high conforming to the shield's outer radius. With respect to the reactor, the girdle is located with its lower edge 24.6 inches above the reactor core inlet end. The girdles are each reinforced with external circumferential ribs near top and bottom, and are welded to heavy steel I-beam and plate supporting members. (The shield is not "black" for gamma rays, as it practically is for neutrons, and the irregular distribution of this external steel mass complicates the interpretation of field gamma ray measurements.) Each of these two welded structures holding a shield half is provided with rollers thus constituting a sort of carriage which sets on stationary steel supports (Fig. 11). By a chain drive, each shield half can be moved to the side a distance of about five feet, so that the reactor on its test car can be positioned or retrieved without interference.

If the need arises -- e.g., as a result of activation, contamination, or simply as a desire for more working space -- each of these carriages, with its mounted shield half, can be picked up as a unit and placed on its own railroad car for removal from the area.

A conical "impedance collar" inside the shield extended downward from its top to near the reactor nozzle during the Phoebus 1B run. Its upper surface was fine mesh stainless steel below which there was about an inch of refractory fiber material retained by a coarse lower mesh. The cone was divided into halves, each attached to a shield half by light supports. The impedance collar served three purposes. It kept thermal radiation out of the cavity. It sought to suppress acoustic coupling between the shield cavity and the nozzle exhaust, which is an extremely intense source of noise distributed over a wide frequency spectrum and which is quite capable of inducing resonances and producing damage. It was also an impedance to gas flow, thereby permitting a nitrogen atmosphere to be maintained within the shield cavity. Nitrogen was a coolant for some instrumentation within the shield but enough additional flow was introduced to prevent hydrogen

Fig. 11. Shield Halves in Closed Position on their Support Structure.

and oxygen from entering the shield cavity and producing an explosion.

11. The Borated Water System [10]

Borated water for the shield is supplied at $\sim 80 - 90^{\circ}\text{F}$ from an insulated 250,000-gallon run tank. The water is withdrawn from the bottom of the tank and, after its circuit through the shield, is returned to the top of the tank through a distributor ring designed to minimize the velocity with which the heated, returning fluid impinges on the cool, stored fluid. Tests have shown that the hotter liquid can remain stratified above the cooler liquid and effectively unmixed with it throughout the time required to circulate most of the tank's original 250,000 gallons, thus allowing use of the full storage tank capacity with a recirculating system. The tank, of course, always remains full.

The makeup of the borated water solution was aimed at high solubility and low corrosion rates to the metals (mainly steel and aluminum) in the system. The final mix, whose pH is ~ 6.70 , was: 13.35 grams of 10-mol borax ($\text{Na}_2\text{B}_4\text{O}_7 \cdot 10\text{H}_2\text{O}$) plus 17.76 grams of boric acid in 200 grams (cc) of water. The actual mixing was done on a larger scale; 100-pound bags of chemically pure salts were dumped into a mix tank containing 20,000 gallons of 130°F water. The full 250,000 gallons of solution required $\sim 127,000$ pounds of 10-mol borax, $\sim 168,000$ pounds of boric acid, and $\sim 1,900,000$ pounds of water.

The run tank can be heated or cooled as needed. Its storage temperature of $80 - 90^{\circ}\text{F}$ was chosen to avoid precipitation of solids, which was thought likely to occur near $\sim 68^{\circ}\text{F}$. Further LASL work indicates that this solution will not undergo pronounced precipitation until cooled below $\sim 45^{\circ}\text{F}$, and then re-solution is rapid at temperatures above 60°F . Precipitation has given no difficulty.

Corrosion has also been no problem. Elevated temperatures exist only for brief periods and corrosion rates at ambient temperatures are very low. Laboratory tests at LASL indicate

corrosion rates of only 0.019 inches/year to 6061 aluminum alloy at solution boiling temperature. Run tank borated water samples taken three months apart have shown no aluminum concentrations greater than 1 ppm and no iron concentrations over 10 ppm. During this time, the fluid was circulated for some 480 hours, 150 hours being at 90°F, and the rest at temperatures between 90 and 120°F.

The solution mixed last November, with makeup added as required, will be used both in the Phoebus 1B shield for the NRX-A6 test and in the new and larger shield to be employed around the Phoebus 2 reactor during its tests in late 1967 or early 1968.

12. Reactivity Effects from an Actual Shield

The Pajarito Research Kiwi Assembly (PARKA) employed for these measurements is a reactor very similar in nuclear characteristics to the Phoebus 1. The mockup shield placed about the PARKA was essentially a 7-inch-thick cylindrical annulus of 81-inches inside diameter and 12-feet height. It was composed of six curved segments, each extending 60° around the reactor. Each segment had 1/2-inch-thick aluminum walls and provided a clear radial space of 6 inches for filling with water. The shield was positioned so that it extended about 1-1/2 feet below the bottom of the PARKA core and about 6 feet above it. There was no horizontal shield mockup.

The shield reactivity worths reported here were inferred from reactor control rod settings at different critical configurations, using previously established values of control rod effectiveness; however, the control rod effectiveness may have changed slightly with the mockup shield in place.

<u>Condition</u>	<u>Reactivity, \$</u>
shield empty	.15
shield filled with water	.90
shield filled with 2 w/o borated water	.30

In conformity with our qualitative expectations, the actual reactivity change produced by the water-filled shield geometries was only about half that predicted by the one-dimensional survey

problems summarized in Table 3.

A fission distribution was measured with the borated-water-filled shield in place, to determine the core edge fission peaking produced by the shield. It was found that evidence of fission boosting at large radii was barely detectable, and it was decided that it would not be necessary to have the shield in place during future fission measurements planned for the Phoebe 1 during its critical checkout at Los Alamos.

13. Nuclear Energy Deposition in the Shield

A good experimental determination of the total energy deposition in the shield requires precise knowledge of flow rates and temperature changes. The shield system field measurements do not warrant perfect confidence, so the results [11] given here probably have a 10% uncertainty, and perhaps more. They imply quite adequate agreement between theory and calculations.

The indicated flow rate through the side shield prior to the full power run was ~4675 gpm. At the onset of an intermediate power hold of ~600 MW, this indication became ~4400 gpm and remained at that throughout the 1460 MW full power run. It returned to its initial value after shutdown. This change may have come from some radiation effect on the instrumentation, or from a difference in fluid properties and impedances under nuclear heating. The inlet water temperature reading was 91°F, while the two outlet temperatures for the shield halves were 115°F and 116°F. These temperatures may easily be uncertain by 2°. Assuming 4400 gpm of water was heated about 24-1/2°F, the side shield total power would have been ~15.6 megawatts. The predicted value for this reactor power was ~14.4 megawatts. One should not draw any serious inference from this comparison, but perhaps the side shield heating was underestimated.

The horizontal privy roof shield water flow was 700 gpm indicated. Its inlet water temperature was 92°F (same source as that above, so both should have read the same) and its exit temperature, after cooling the reactor support legs, was 126°F. This would correspond to ~3.47 megawatts deposited, versus around

3.5 megawatts predicted for a slightly different total set of components.

14. Test Cell Face Heating

The temperature channel on the front face of Test Cell "C" which recorded a 167°R rise during the Phoebus 1A run (7.29×10^5 MW-sec) recorded only a 40°R rise during the four times larger Phoebus 1B run (2.73×10^6 MW-sec). For reasons previously mentioned concerning flame heating, etc., these data do not permit a direct inference of shielding worth. However, they do show that the shield did its job quite well, and they imply that the amount of flame heating of the test cell face may not have been as large as the possibility suggested in Section 8.

The following data [12] compares the heating between the two Phoebus runs:

<u>Distance from Test Cell Face</u>	<u>Temperature Rise, $^{\circ}\text{R}$</u>	
	<u>Phoebus 1A</u>	<u>Phoebus 1B</u>
1/3 inch	~ 167	40
3 inch, on reinforcing rod	165	20
9 inch	40	~ 4
15 inch	20	~ 1

The shield's effectiveness in reducing test cell heating increases with depth, perhaps due to its suppressing neutron captures in the test cell wall.

Temperature rises in some lead bricks, formerly serving a shielding purpose in a "window" of the test cell wall, averaged $\sim 174^{\circ}\text{R}$ from a Kiwi-B-4E run of 5×10^5 MW-sec and $\sim 188^{\circ}\text{R}$ from a Phoebus run of 7×10^5 MW-sec, but they were only $\sim 25^{\circ}\text{R}$ from the full power Phoebus 1B run.[13]

15. Attenuation Within the Shield

One shield half had six thin-walled (1/16-inch) tubes welded into it to provide horizontal holes, or ports, which could accommodate other thin-walled ($\leq 1/16$ -inch) tubes loaded with dosimeters for shield depth-dose measurements. These were located

22 inches above the reactor core inlet end. Three holes, of 3/8-inch i.d., employed small sulfur pellets and $^6\text{LiF-Pb}$ cans containing glass gamma dosimeters. One went all the way through the shield, one penetrated 15-3/8 inches and the other 7-1/8 inches from the exterior shield surface. Neutron dosimeters (^{235}U -loaded wires; ^{238}U , ^{237}Np , Au, and Cd-covered Au foils) appropriately segregated to prevent interference were used in three similar 7/8-inch i.d. ports. Two of these extended all the way through the shield and one extended 11-3/8 inches into the shield from outside. The shorter ports were to see if the dosimeter tubes themselves perturbed their measurements. Space not used for dosimetry inside any tube was loaded to conform to the adjacent shield properties as closely as possible; for this, aluminum spacers were used at shield plate positions and 2% borated polyethylene spacers were used next to the borated water regions.

The data collected within these tubes is to be included in a field measurements summary report by John Ahlquist [14], who has kindly given permission to use some of his data in advance of publication.

Figure 12 uses gamma ray dose rates measured within the shield on two early EP's ("experimental plans", or reactor operations) of 10.5 and 393 MW-sec. All data from both runs were scaled to 1000 MW-sec of reactor energy release and converted to rads aluminum, so as to be directly comparable to the heating calculations, also shown. (Two "calculated" points were actually inferred from scaling nearby results.) The overall agreement is quite good, but we may have underestimated the heating near the first two flow passes in the shield at this axial location, probably by inadequately accounting for the soft gamma rays from boron capture.

Figure 13 shows similar data for the high and low energy neutron fluences within the shield, again scaled to 1000 MW-sec energy release. The EP-1 sulfur data had very poor counting statistics, because of the low energy release, and are not used here to any depth. (Qualitatively, their response dropped off more rapidly with depth than that of the sulfurs on EP-2.) The EP-2 sulfur data have high internal consistency, and imply a shield

Fig. 12. Gamma Ray Dose Rates Through the Shield.

Fig. 13. Neutron Fluence Through the Shield.

attenuation factor of ~ 200 for neutrons of energy ≥ 2.5 MeV, indicating the shield performed as well as intended. (The overall reduction of the epi-sulfur fluence exceeds 400, but purely geometrical effects account for about a factor-of-two.)

The thermal neutron data in Figure 13, at least after the first three points, must be regarded as purely qualitative and the dashed curve is only a guess as to how things may go. Results from both EP's did scale to the same initial fluence (1.1×10^{14} nvt), both dropped under 10^{10} nvt, and both gave about $5 - 6 \times 10^9$ nvt for the thermal fluence on the external surface (which might be an air-scattering effect). However, these data are derived by subtracting the response of a cadmium-covered gold foil in one stringer from the response of a bare gold foil in another stringer, the assumption being that both foils are at the same depth in the shield. Because of the rapid decrease in low energy neutron fluence with depth, small positional differences become quite important. These showed up as factors of 2 - 4 between EP-1 and EP-2 inferred fluence at 6 - 10-inch depths, and even led to negative values (from Cd-covered Au $>$ bare gold at a "comparable" location) in some cases where data from the short neutron port was a member of the comparison pair.

Neutrons of intermediate energy generally behaved between these two extremes, though some of the data suffered from similar problems of counting statistics and positional uncertainties. The most important implication of the neutron data is that the shield was so black that most of the neutrons producing effects in the area probably escaped the end of the shield and underwent air scatter.

16. External Shield Attenuation Effects

The apparent value of the shield at larger distances in reducing the dose from a unit of reactor energy release is summarized in Figure 14.[14] From 60 feet out, the data compared are from identical locations with respect to the Kiwi-B-4E and and the Phoebus 1B reactor tests. If the Phoebus 1B had been unshielded, all the ratios derived should have been unity, though

Fig. 14. Apparent Shield Worth Versus Distance.

some additions since the B-4E test (such as a concrete wing wall and two dewars near the 292° line) might have shown an influence in the 60 - 300-feet range.

The gamma attenuation factor, though variable, seems to be equal to or greater than 15 everywhere. Part of its changing value close-in is probably due to vagaries of "view", as local shield support structures influence how detectors "see" the reactor. The gradual rise at large distances (800 to 2000 feet) may be from a similar cause, or it may result from something more fundamental. It is still to be resolved.

The neutron shielding effect is greatest near the reactor and least at large distances, as expected, for at large distances the arriving neutrons have usually suffered several collisions and the flux density is as dependent on the overall strength of the emission as on its initial direction. The apparent shielding value of 10 for neutrons at distances near 2000 feet probably implies approximately a factor-of-ten reduction in overall neutron escape. The near parallel between the thermal neutron data and that for "other neutrons" (boron-shielded Pu response) probably just means that the thermal neutrons result from the local fast source. There should be little original thermal neutron escape.

General estimates of the neutron shielding effects likely as a function of distance were derived before the Phoebus 1B run by Monte Carlo calculations,[15] but they were only exploratory in nature and not expected to tell precisely how things would go. Two of these compared flux-weighted neutron numbers arriving at the ground plane with and without a shield. They were simplified both geometrically and nuclearly. They allowed air scattering and air capture, but had no ground albedo, and a neutron history terminated immediately on contact with the ground. For the model mentioned, these calculations indicated approximate number attenuation factors of 40 at 45 feet, 22 at 75 feet, 16 at 150 feet, 8.5 at 300 feet, and ~4 at 500 feet. These data are not directly comparable to those on Figure 14, and do not necessarily conflict with them.

Figure 14 implies that the pad activation might have been down by a factor of ~100 at 30 feet. We know that the reduction

was striking, but during the lengthy reactor cooldown, some contamination was blown onto the pad and we were unable to get the direct activation measurement we had hoped for upon reactor removal.

17. The Phoebus 2 Shield

This same general shield concept will be employed around the Phoebus 2 reactor, whose core is ~55 inches in diameter and whose pressure vessel o.d. is ~81 inches. That aluminum borated-water shield will be 29-1/2-inches thick (divided into 6-inches of aluminum and 23-1/2-inches of water), 107 inches i.d., and 174 inches high. It will be provided with a few more ports for depth-dose measurements. The privy roof on the Phoebus 2 test car will be 32-inches thick and 14 feet square. Solid angle factors will probably preclude the overall (long-distance) neutron attenuation of this shield being any better than that of the Phoebus 1B shield, but it may be more effective in the immediate vicinity of the test car.

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