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WANL-TME-059

APRIL 30, 1962

**SUMMARY OF NUCLEAR INFORMATION
FOR PRELIMINARY B-1 FLIGHT DESIGN**

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SUMMARY OF NUCLEAR INFORMATION FOR
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1.0 INTRODUCTION

This report summarizes the information currently available, concerning the NERVA B-1 and the KIWI B-1A. In addition, certain of the results presented are discussed in detail in order to clarify their significance.

The purpose has been to compile in convenient form nuclear characteristics of the B-1 type reactor which may be used as a basis for further work. As more complete or better data becomes available, it will be added by addendum.

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2.0 SUMMARY AND CONCLUSIONS

Presented in this report are the best available values for reactivity worths, control parameters, radial and axial power distributions, and heating rates for the NERVA B-1 rocket reactor.

The NERVA B-1 core is the same as the core of the KIWI B-1B. There are differences in the graphite reflector between these two designs but these are minor from a nuclear standpoint. The differences in the core loading between the KIWI B-1B and the KIWI B-1A have a slight effect on reactivity. The maximum power peaking at the core-carbon interface is also less in the KIWI B-1B than it is in the KIWI B-1A. The NERVA B-1 geometry includes a shield worth 0.30\$ whereas the KIWI geometry does not include a shield.

Certain important nuclear parameters are summarized in Table 2.1. Reactivities measured at various stages of the LASL test program on the KIWI B-1A are listed in Table 2.2. The anomalous reactivity behavior of the KIWI B-1A as indicated in Table 2.2 has not been completely resolved to date; however, the larger cold reactivity observed at NTS appears to be the result of smaller spacing between modules than was the case at Pajarito. The irreversible loss of reactivity observed at NTS appears to be partly the result of the loss of hydrogenous cement when the reactor first became hot. During these tests, all reactor shim voids in the reflector were filled with beryllium. In order to operate hot critical with the vanes further in, it appears necessary to increase the reactivity of the system.

TABLE 2.1

"BEST" ESTIMATE, NERVA B-1 NUCLEAR PARAMETERS*

Total Excess Reactivity	2.50\$**
Total Worth of All Vanes	
Hot	(11.5 \$)
Cold	8.8 \$
Temperature Coefficient	
Near Operating Temperatures	2.6 (10^{-6}) $\Delta k/^{\circ}F$
Cold	2.34 (10^{-5}) $\Delta k/^{\circ}F$
Critical Vane Positions (LASL)	
Cold Measured	123 $^{\circ}$
Hot Predicted	(137 $^{\circ}$)
Hot Measured	165 $^{\circ}$
Prompt Neutron Lifetime (cold)	43 $\pm$ 2 μ S
Cold Worths in Core; \$/% Change In Density of Material in Question	
Carbon	0.58
Oy	0.27
Worth of Reflector, Δk_{eff}	
Drums 0 $^{\circ}$	0.10
Drums 180 $^{\circ}$	0.17
Worth of Reflector Shims Δk_{eff}	0.02
Worth of Hydrogen $\Delta k_{eff}/kg H_2$	
Core	0.071
Reflector	0.002

*Values in parenthesis are calculated.

**Includes 0.15\$ from nozzle plus (0.30\$) from shield.

TABLE 2.2
REACTIVITIES MEASURED AT VARIOUS STAGES OF LASL TEST PROGRAM
(Chronological Order)

	<u>Excess Reactivity Dollars</u>
Cold Critical, Pajarito	2.16
Initial Cold Critical NTS	2.96
Initial Warm Critical	1.38
Cold Critical (after low power run)	1.57
Hot Critical, (3600°R)	0.14
Warm Critical (after hot run)	1.41

Table 2.1 shows that the experimental hot critical vane position was significantly different than the LASL prediction. This behavior appears to be primarily the result of the unexpectedly large initial core expansion observed during the power runs at NTS. In addition, the calculational model used may be partly at fault. The LASL calculations were performed using the one dimensional DSN code normalized to the cold experimental power distribution through the use of an artificial poison region in the reflector. A two dimensional code may be necessary to predict vane worth in an accurate and consistent manner. WANL is currently using the CURE two dimensional diffusion theory code to predict the worth of the vanes. The LASL predicted power distribution near the core-carbon interface is low. This may be a result of the normalization used.

The use of a nuclear model which represents, with sufficient accuracy, the neutronic conditions within the reactor is to be preferred with respect to a procedure which requires artificial normalization. This is true because perturbations, such as temperature changes and changes in the positions of control vanes, may destroy the relation between the physical and nuclear models. The nuclear representation of the control vanes is one of the most important nuclear problems at the present time because of the intimate coupling between the vane positions and reactivity, power peaking, and control system requirements. Work is in progress at WANL to obtain a more accurate nuclear representation for the vanes based upon a two dimensional approach.

Another area in which substantial methods improvement is required is in the calculation of temperature and power coefficients.

The LASL calculated cold to hot reactivity changes are listed in Table 2.3. The reactivity change resulting from the fuel Doppler coefficient has not been determined.

TABLE 2.3
COLD TO HOT REACTIVITY CHANGES

	<u>Reactivity Contribution</u>	
	4100°R	3600°R
Longitudinal Expansion	- 1.0\$	- 0.86\$
Radial Expansion	- 0.5	- 0.43
Thermal Base Shift	- 0.3	- 0.26
H ₂ in the Core	+ 0.3	+ 0.34
	<hr/>	<hr/>
	- 1.5\$	- 1.2 \$

Table 2.4 provides an indication of the accuracy with which the radial power distribution can be calculated at the present time. The measured hot power distribution used is not that corresponding to the full power temperature profile, although this represents the best power distribution currently available. The calculated axial power distributions, with and without a shield, are not very different except near the inlet where a 24% increase in power density occurs as a result of shield backscatter and neutron moderation. Table 2.5 shows that agreement is good between LASL and WANL axial hot power distributions.

TABLE 2.4

COMPARISON OF MEASURED AND PREDICTED RADIAL POWER DISTRIBUTIONS

<u>Radius, Inches</u>	Maximum % Error* of LASL and WANL Predicted Power Distribution	
	<u>LASL</u>	<u>WANL</u>
0 - 13	5%	5%
13 - 15.7	3%	15% (low)
Near Core-Carbon Interface	20% (low)	5%

TABLE 2.5

COMPARISON OF WANL AND LASL CALCULATED** AXIAL HOT POWER DISTRIBUTIONS

<u>Radius cm</u>	<u>WANL Shielded</u>	<u>WANL Unshielded</u>	<u>LASL Unshielded</u>
0	0.239	0.203	0.205
20	0.613	0.608	0.65
40	0.897	0.897	0.915
60	1.000	1.009	1.000
80	0.910	0.914	0.872
100	0.662	0.662	0.600
120	0.284	0.284	0.215
128	0.090	0.090	0.03

*Based on LASL hot critical measurement.

**The LASL distribution was obtained by applying cold/hot and control vane correction factors to a LASL cold measured power distribution. The WANL calculations include hydrogen whereas the LASL calculations do not appear to; thus, the WANL and LASL calculations are not directly comparable.

Little fuel burnup is experienced because the integrated flux time during the operating life of a rocket reactor is relatively small. The same considerations apply to xenon poisoning. The per cent loss in effective multiplication factor for the KIWI B-1A reactor at peak xenon concentration after operating 20 minutes at full power is calculated to be 0.161.

The total power from fissioning, gamma, and beta decay of fission products as a function of time resulting from a step input of -3.5 per cent in reactivity has been calculated, and results are presented in Section 7.3. Heating rates in reactor components as a function of time have been computed for normal operating conditions, and for various startup and shutdown programs. Results are summarized in Tables 2.6, 2.7 and 2.8. The total decay power versus time after shutdown is shown in Table 2.9.

TABLE 2.6
PER CENT OF REACTOR POWER ABSORBED AT STARTUP

Component	1% of full power at 30 sec.		100% of full power at 60 sec.	
	Min.	Max.	Min.	Max.
Core	90-92	90-93	90-92	90-93
Graphite Sleeve	0.20	0.23	0.20	0.25
Graphite Reflector	0.40	0.46	0.40	0.49
Be Reflector	0.60	0.71	0.60	0.72
Al Pressure Vessel	0.064	0.074	0.064	0.079
Al Support Plate	0.14	0.17	0.14	0.19
Li H - SS Shield	0.19	0.20	0.19	0.22

TABLE 2.7
PER CENT OF REACTOR POWER ABSORBED
FOR 20 MINUTES OPERATION AT FULL POWER

Component	Operation	Time After Shutdown (sec.)		
		10	10^3	10^4
Core	91-93	91-93	78-88	78-89
Graphite Sleeve	0.28	0.66	1.34	1.26
Graphite Reflector	0.55	1.36	2.62	2.33
Be Reflector	0.83	1.88	3.46	2.98
Al Pressure Vessel	0.092	0.27	0.51	0.43
Al Support Plate	0.20	0.62	1.28	1.16
Li H - SS Shield	0.28	0.48	0.81	0.68

TABLE 2.8
PER CENT OF REACTOR POWER ABSORBED AFTER PROGRAMMED SHUTDOWN

Component	Operation	15 Min. Full-Power 10 Min. Half-Power	15 Min. Full-Power 20 Min. Quarter-Power	
		10^3 sec.	Operation	10^3 sec.
Core	91-93	77-88	91-96	79-89
Graphite Sleeve	0.29	1.34	0.29	1.39
Graphite Reflector	0.56	2.89	0.59	2.78
Be Reflector	0.84	3.74	0.88	3.73
Al Pressure Vessel	0.094	0.61	0.10	0.59
Li H - SS Shield	0.23	0.92	0.24	0.90

TABLE 2.9
PER CENT OPERATING POWER VERSUS TIME AFTER SHUTDOWN
FOR FULL POWER OPERATING TIME OF 1200 SECONDS

<u>Time After Shutdown, sec.</u>	<u>Per Cent of Operating Power</u>
10^2	1.2
10^3	2.6 (10^{-1})
10^4	3 (10^{-2})
10^5	2 (10^{-3})
10^6	9 (10^{-5})

3.0 REACTOR DESCRIPTION

Essentially all of the WANL nuclear calculations for the NERVA B-1 rocket reactor have been performed using a nuclear model based on the KIWI B-1A because the core of the KIWI B-1A was the first considered. Subsequently, the core of the KIWI B-1B was specified for the NERVA B-1. Calculations indicate that practically all results obtained for the KIWI B-1A are, within the accuracy associated with current calculations, also representative of the KIWI B-1B core. This matter is considered in greater detail in Section 3.4.

Although reference is made herein to the KIWI B-2 and the KIWI B-4 reactors, only the KIWI B-1A and the KIWI B-1B are described in this report. Subsequent NERVA reports will include descriptions of the KIWI B-2 and the KIWI B-4.

The Los Alamos Scientific Laboratory has conducted experiments to demonstrate the feasibility of a nuclear reactor as a heat source for a rocket engine. The reactor consists of a cylindrical nuclear core, a concentric reflector containing the control drums, and a control system. LASL has conceived three reactor designs, KIWI B-1, B-2 and B-4, all of which utilize the same reflector and control-system design. However, the three core designs differ. The B-1 core uses structural graphite in tension to contain and support the fuel rods. The B-2 core uses a monolithic block of structural graphite at the exhaust end of the core such that the prismatic graphite elements and the fuel rods are in compression. The B-4 uses steel tie rods in tension to support prismatic graphite fuel elements at the exhaust end of the reactor, placing the fueled graphite in compression.

The KIWI reactors are designed for test firing with the nozzle upward. Hydrogen from the turbopump enters the nozzle tubes and flows around and through the beryllium and graphite reflectors which encircle the reactor core. The hydrogen flow is thus utilized to cool these components before entry into the core inlet plenum. From this plenum the hydrogen flows through the core, where heat is added and then exhausts through the nozzle.

3.1 KIWI B-1A

A cutaway of KIWI B-1A, Figure 3.1, shows the relationship of the major components forming the reactor assembly.

The beryllium reflector consists of 12 segments, each of which is individually supported by two standoffs pinned to the pressure vessel. Each segment has a 4-in.-dia. axial hole to receive a control drum, and numerous small-diameter holes to provide coolant passages. Hydrogen also flows through and around the control drums.

The control drums are supported by ball bearings. These are located in housings fastened to the reflector standoffs which serve to locate the reflector segment radially and tangentially. Axial retention depends upon gravity and pressure drop across the reflector. These forces provide a positive load to seat the standoffs in the pressure-vessel head. There is no provision for negative loads.

The core consists of 163 graphite modules arranged to form a cylinder. Each module of structural graphite has a modified hexagonal cross section and supports cylindrical fuel rods. Each module is individually supported from an aluminum support plate by a stainless steel support tube threaded into the graphite. The pressure drop

across the fuel rods causes tensile loads in the structural graphite. The support plate is radially positioned by a flow baffle attached to the head of the pressure vessel. Axial positioning of the core is accomplished by spring-loading the support plate against the cylindrical graphite reflector, which transfers the core pressure-drop load to the exhaust nozzle.

The modules are assembled with a 0.037-in. nominal clearance to accommodate radial thermal expansion. During operation the clearance varies from essentially 0.037 in. at the support plate to zero at the plane of maximum temperature. Because of this assembly clearance, careful handling is required to prevent excessive transverse displacement of the modules and the accompanying high bending stresses.

The inner reflector consists of concentric cylinders of structural graphite and stainless steel. The inner cylinder of 5/8-in. -thick graphite has an inside covering of 1/8-in. -thick pyrolytic graphite tiles which form a thermal barrier between the core and the reflector. At the point of maximum temperature the temperature drop across the tiles is about 2500°F.

A stainless steel seal barrel separates the inner 5/8-in. -thick cylinder from a 1-1/2-in. outer cylinder of structural graphite. This steel barrel provides a seal between the high-pressure hydrogen in the reflector region and the lower-pressure hydrogen in the exhaust plenum. The outer cylinder also provides axial support of the core during operation.

3.2 KIWI B-1B

There are two differences in the core loading between the KIWI B-1A

and the KIWI B-1B:

1. There are 12 less fuel slugs in the lightly loaded region of the KIWI B-1B than there are in the KIWI B-1A.

2. There is an uninterrupted circle of fuel elements immediately adjacent to the core-carbon interface in the KIWI B-1B. In the KIWI B-1A (see Figure 3.1) the loading in this circle is irregular.

3.3 NERVA B-1

In keeping with program objectives, only the KIWI B-1 core support has been modified for the NERVA B-1 design, Figure 3.2. The core supports must be capable of carrying the loads imposed by the environmental conditions and still be sufficiently flexible to allow the differential thermal expansions without overstressing the core modules.

Core lateral support consists of felt strips which serve as distributed springs, radially loading graphite T-strips lined with pyrolytic graphite tiles in such a way that the tiles are trapped against the core. The graphite-felt springs bundle the modules so that they touch at the core exit, causing the modules to act as a unit under the influence of lateral loads. Lateral loads and core thermal expansions cause radial compression of the felt springs to varying degrees along the axis.

A graphite reflector barrel surrounds the core and is axially spring-loaded against the nozzle tubes to seal at the exhaust plenum. A metal support barrel fits over this graphite reflector to act as a lateral support and as a seal at the inlet end of the core. This steel cylinder is connected to the core support plate by a quick-disconnect clamp for remote handling.

Slots machined in the inside of the graphite barrel retain the felt springs and guide the graphite T-strips. The core outlet end of each T-strip rides on a spring-loaded graphite block which seals the felt from hydrogen flow. Coolant flows through axial holes drilled the full length of the barrel and T-strips. Finger springs located at the core inlet position the graphite barrel axially and take up the differences in thermal expansion. A support ring, attached to the core exit end of the steel cylinder with a quick-disconnect clamp, supports the graphite reflector during assembly. Spacers between the beryllium reflector and the steel cylinder center the core and maintain flow clearances.

The beryllium reflector designed for NERVA, Figure 3.3, consists of 12 segments arranged to form a hollow cylinder. These are clamped between two aluminum support rings by tie bolts, and are doweled to fit into only one position. Partial cooling of the beryllium reflector is provided by axial holes in the segments and support rings. Additional cooling is provided at the inside and outside diameters. The beryllium reflector contains 12 axial holes for the reactor control drums located at the midpoint of each segment. Each drum is supported by two bearings at the cylinder end rings.

The control drum consists of a beryllium cylinder, boron plate, and aluminum enclosure, Figure 3.4. It is cooled by hydrogen flow through and around the assembly. For core assembly a torsion spring attached to the lower bearing housing is used to keep the control drums in the closed position, against a stop.

A longitudinal view of the NERVA B-1 core assembly is shown in Figure 3.2; the NERVA control drum assembly is shown in Figure 3.4. The shield is cooled using the heated bleed flow pattern shown in Figure 3.5. The KIWI-B reflector sector is shown in

Figure 3.6. The NERVA B-1 reflector is essentially the same as the KIWI-B reflector; however, the number of heat transfer holes required in the reflector of the NERVA B-1 has not been firmly established.

3.4 DIFFERENCES BETWEEN NERVA B-1 AND KIWI B-1 SERIES REACTORS

3.4.1 PHYSICAL DIFFERENCES

The NERVA B-1 core is the same as the KIWI B-1B core except that there are no graphite plungers in the graphite reflector; carbon "felt" being used instead, and there is a thin steel barrel in the carbon reflector of the NERVA B-1 similar to the one installed in the KIWI B-1A. The barrel in the KIWI B-1A is 0.025 inch thick whereas the barrel in the NERVA B-1 is 0.050 inch.

The NERVA B-1 will have a shaped support plate with a thickness varying from approximately 3 inches at the core edge to about 7 inches near the core axis. The KIWI B-1A has a support plate with a constant thickness of 6 inches.

The current concept of the NERVA B-1 includes an aluminum pressure vessel with a wall thickness ranging between 3/4 to 1 inch. Consideration is being given to a titanium pressure vessel with a wall thickness of approximately 0.3 inch.

3.4.2 NUCLEAR CONSEQUENCES

From a nuclear standpoint, the most significant difference between the KIWI B-1B and the KIWI B-1A is the change in the fuel loading in the lightly loaded zone. The uninterrupted circle of fuel elements immediately adjacent to the core-carbon interface in the KIWI B-1B reduces the maximum power peaking with respect to that generated with the interrupted pattern in the KIWI B-1A.

The power shapes were calculated, using the AIM-5 one dimensional diffusion theory code, with the different loadings homogenized over the lightly loaded region. The dimensions of the lightly loaded region were held constant. The eigenvalues calculated for the two problems were found to be essentially the same indicating that the carbon added had practically the same importance as the fuel removed.

With respect to the NERVA B-1, the change in reactivity and power distribution resulting from the specification of the titanium pressure vessel will also have a small nuclear effect, the primary consequence being a change in the heating rate in the pressure vessel itself. The same comment applies to the differences in the core support plate designs. The addition of a shield increased the reactivity of the NERVA B-1 series reactors. The thicker stainless steel barrel (0.050 inch vs 0.025 inch) in the NERVA B-1 reduces reactivity by approximately 90 cents with respect to the barrel installed in the KIWI B-1A.

4.0 AVAILABLE MEASUREMENTS

4.1 COLD MEASUREMENTS

All cold measurements referred to in this report were performed by LASL and are reported in LASL Progress Reports N-2-6052 of June 21 - July 20, 1961; N-2-6082 of July 21 - August 20, 1961; N-2-7048 of August 21 - September 20, 1961; and N-2-7000 of September 18, 1961.

When the reactor was set up at NTS for the run at design power, another cold critical measurement was made prior to the hot run. At that time an apparent gain in cold critical reactivity of 80 cents was observed at NTS in excess of that observed during the previous (KIVA) measurements. The results obtained are discussed in greater detail in Section 4.2.

It is important to note that the power peaking at the core-carbon interface is dependent upon the critical vane positions. Figure 5.8 indicates this fact, although this figure refers to the hot case. As a result, the cold critical power distribution obtained at NTS may have been significantly different than that observed during the KIVA checkout.

4.1.1 POWER DISTRIBUTION

The cold radial fission distribution is shown in Figure 4.1 which is a plot of relative fissions per gram of Oy versus cold-packed radial location in the core. This distribution was measured with 0.006" diameter Oy wires.

4.1.2 CONTROL VANE WORTH

During the KIVA checkout, delayed critical was found with all twelve

control vanes set at 123° from the "IN" position. The total excess reactivity was found to be 2.20\$ which included a 0.15\$ contribution from the mockup nozzle. The integral and differential curves of reactivity versus control vane position for a typical vane based on data obtained during the KIVA checkout are shown in Figures 4.2 and 4.3 respectively.

As indicated in Section 4.1.1, the cold reactor was observed to be more reactive at NTS than was the case at KIVA. This suggests that the fraction of neutrons leaking into the reflector was different for each measurement. The worth of the vanes is a function of the core leakage; thus, the worth of the vanes as a function of position (at NTS) may have been different than that observed during the KIVA checkout.

4.1.3 PROMPT NEUTRON LIFETIME

Neutron chain decay measurements on the cold KIWI B-1A near critical gave a value of $43 \pm 2 \mu S$ for the prompt neutron mean-life. The experimental data used are reproduced from LASL Progress Report N-2-6052 and are shown in Figures 4.4 and 4.5.

4.1.4 MATERIAL WORTHS AND COEFFICIENTS

In LASL Progress Report N-2-7048, material worths are reported for the KIWI B-4. This information is reproduced below because LASL indicated that the combined worth of Oy and graphite in the core of the KIWI B-4 ZEPO is the same as that measured in the core of the KIWI B-1A.

GRAPHITE

Fifty $1/4$ " graphite rods were added in a 60° sector of the core. The rods were uniformly distributed in the sector and occupied 0.33% of the mass of the core. The measured worth of the graphite rods was 19.07 cents which results in a carbon reactivity coefficient of 58 cents/% change in carbon density.

ORALLOY

A reactivity worth of 27 cents/% change in Oy density was found for Oy in the KIWI B-4 ZEPO.

The combination of 58 cents/% change in carbon density plus 27 cents/% change in Oy density yields an overall core density reactivity coefficient of 85 cents/% change in core density. This value is the same as that observed for the KIWI B-1A.

WANL calculations indicate that hydrogen in the core is worth 0.071 $\Delta k/kg H_2$ whereas hydrogen in the reflector is worth 0.002 $\Delta k/kg$.

A plot of reactor reactivity versus reflector coolant volume is shown in Figure 4.6.

4.2 HOT MEASUREMENTS

4.2.1 REACTIVITIES AT VARIOUS STAGES

The reactivities measured at various stages of the LASL test program for the KIWI B-1A are summarized below in Table 4.1.

TABLE 4.1

REACTIVITIES MEASURED AT VARIOUS STAGES OF LASL TEST PROGRAM

	Excess Reactivity, Dollars
Cold Critical, Pajarito	2.16
Initial Cold Critical NTS	2.96
Initial Warm Critical	1.38
Cold Critical (after low power run)	1.57
Hot Critical, (3600°R)	0.14
Warm Critical (after hot run)	1.41

As shown in Table 4.1, an apparent gain in cold critical reactivity of 80 cents was observed at NTS in excess of that observed at Pajarito. The core was more completely compressed at NTS than it was at Pajarito; LASL estimates that this accounts for 50 cents of the difference. The remaining 30 cents is attributed to increased back-scatter from the environment at NTS.

Table 4.1 also indicates an irreversible loss of 140 cents as a result of high temperature operation. This becomes evident when the excess reactivity during the "initial cold critical NTS" is compared with "cold critical (after low power run)". This behavior has not been completely resolved at the present time; however, it appears to be due in part to the burning out of hydrogenous contact cement used to glue graphite filler strips to fuel elements.

In Table 4.1 the full power reactivity (3600°R) is given as 0.14\$. This means that each vane was holding down 1.17 cents of reactivity. The cold experimental vane worth curve indicates that the hot critical vane positions were 165° if each vane was worth 1.17 cents.

It should be noted that the hot critical test was performed with all shim voids in the reflector filled with beryllium. In order to have the reactor critical hot with the drums further in, it may be necessary to increase the reactivity of the system.

4.2.2 RADIAL POWER DISTRIBUTION

Results of the hot critical run were reported in LASL Progress Reports N-2-7225 and N-2-7166. The hot critical power distribution is shown in Figure 4.7. The temperature profile actually achieved during the hot run was used by LASL to

obtain a calculated curve which is also shown in Figure 4.7. With respect to the calculated curve, LASL indicated that the correction for perturbation of fissioning flux due to control rod withdrawal is not known accurately and may be responsible for much of the discrepancy between the measured and predicted curves. The results of calculation and experiment are compared in Section 5.0.

4.2.3 TEMPERATURE COEFFICIENTS

A high and apparently reproducible initial temperature coefficient of $0.3\$/100^{\circ}\text{F} = 2.34 (10^{-5}) \Delta k/^{\circ}\text{F}$ was observed. It appears that the large magnitude may be due to some peculiar initial module expansion or warping resulting in extreme radial expansion.

The LASL calculated temperature dependent reactivity contributions reported are reproduced in Table 5.2 where it is indicated that the positive contribution of the hydrogen coolant is 34 cents. The reversible reactivity loss observed when the reactor temperature was increased to 3600°R was 1.40\$. Subtracting the calculated value of +0.34\$ for the worth of hydrogen given in Table 5.2, the temperature coefficient reduced the reactivity by 1.06\$ when the core temperature was raised to approximately 3600°R . This corresponds to a temperature coefficient of $2.6 (10^{-6}) \Delta k/^{\circ}\text{R}$.

5.0 COMPARISON OF CURRENT CALCULATION AND EXPERIMENT

The purpose of comparing current calculations with current experiments is to indicate the accuracy which can be expected from calculations at the present time.

5.1 COLD CALCULATIONS

5.1.1 LASL ANALYTICAL MODEL KIWI B-1A

Infinite cylinder 24 group DSN calculations were performed in which axial leakage was accounted for through the use of an effective leakage cross section; " DB^2 ". The control vanes were represented by a boron bearing cylindrical sheath centered in the Be reflector. The geometry of the calculations is shown in Figure 5.1. The compositions of the regions are given in Table 5.1 which was reproduced from LASL Progress Report N-2-7166. The gross reactor characteristics are given in Section 3.0.

5.1.2 CALCULATIONS

LASL duplicated the cold experimental power distribution, using the DSN code, by placing an artificial poison region in the reflector. The thermal group was divided into several energy groups in which scattering both up and down in energy was allowed. The resulting match is shown in Figure 5.2. The eigenvalue predicted by the DSN calculation, in which the power shape was matched, was not reported.

A primary purpose of the calculations described above was to provide an experimentally normalized base case which could then be perturbed to calculate the power distribution for the hot critical condition. Apparently it was assumed that the required position and nuclear properties of the poison region could be held constant for the hot calculations.

TABLE 5.1

COMPOSITIONS OF THE MATERIALS USED IN THE LASL
ANALYTICAL MODEL

Material	Densities (gm/cm ³)					
	U	C	Nb	Be	B	Al
I	.1260	1.468	.0866			
II	.0756	1.468	.0866			
III		1.818				
IV		1.400				
V				1.719		
VI				1.719	.090	
VII				1.719		
VIII						2.43

Although an adequate match was achieved with the cold experimental power distribution, it is not clear that the same artificial region will provide the correct match when the temperatures are changed. The ultimate justification for the use of such a procedure is the success realized in predicting the hot power distribution. This problem is discussed in the next section.

5.2 HOT CALCULATIONS

5.2.1 LASL CALCULATIONS

5.2.1.1 CONTROL VANE POSITIONS

In Section 5.1.2, it was indicated that LASL duplicated the cold experimental power distribution, using the DSN code, by placing an artificial poison region in the reflector. LASL then estimated the hot critical vane position to be approximately 14° greater than the cold critical angle; i.e., 137° vs 123° using calculated results for the reactivity loss resulting from temperature increase. LASL then applied calculated cold to hot correction factors to predict the power distribution of the hot critical reactor. These factors are shown in Figure 5.3. The experimental hot critical control vane position was found to be (see Section 4.2) about 165° instead of 137° estimated by LASL. This uncertainty appears to be largely the result of the unexpectedly large initial core expansion.

A two dimensional code may be necessary to predict vane worths in an accurate and consistent manner. WANL is currently using the CURE two dimensional diffusion theory code to predict the worth of vanes. This matter is discussed in Section 5.2.2.4.

5.2.1.2 RADIAL POWER DISTRIBUTION

The LASL predicted hot critical power distribution for assumed vane positions of 137° is shown in Figure 5.4.

5.2.1.3 AXIAL POWER DISTRIBUTION

The high temperature axial power distribution reported in LASL Progress Report N-2-7095 is reproduced in Figure 5.5. LASL indicates that this fission distribution was obtained by fission chamber measurements in the (cold) ZEPO assembly. LASL adjusted the amplitudes to agree with Figure 5.4.

5.2.1.4 MATERIAL WORTHS AND COEFFICIENTS

The following calculated temperature dependent reactivity contributions reproduced in Table 5.2 were previously reported in LASL Progress Report N-2-7166.

TABLE 5.2

COLD TO HOT REACTIVITY CHANGES

	<u>Reactivity Contribution</u>	
	4100°R	3600°R
Longitudinal Expansion	- 1.0\$	- 0.86\$
Radial Expansion	- 0.5	- 0.43
Thermal Base Shift	- 0.3	- 0.26
H ₂ in the Core	+ 0.3	+ 0.34
	<hr/>	<hr/>
	- 1.5\$	- 1.2 \$

The reactivity change resulting from the fuel Doppler coefficient has not been determined.

It has been estimated that the KIWI B-1A core would lose 1.0\$ upon expansion from 35" to 35.5" diameter.

5.2.2 WANL CALCULATIONS

5.2.2.1 WANL B-1 ANALYTICAL MODEL

The analytical model used for the Westinghouse calculations is defined in Table 5.3. The general dimensions and construction features are described in Section 3.0.

The KIWI B-1A geometry has been used for the majority of the calculations performed to date; however, the differences between the NERVA B-1 and the KIWI B-1A are not very great from a nuclear standpoint. These differences are discussed in Section 3.4.

It should be noted that the WANL analytical model and the LASL analytical model for the KIWI B-1A are not in exact agreement. This is a result of the fact that the two models were independently derived. At this stage of the work the differences are not of immediate concern.

The core was treated as consisting of seven radial regions. Different void fractions existed in each region because of variation in channel sizes. The seventh and outermost fuel region had a fuel loading of 240 mg/cc in the fueled graphite whereas the others had 396 mg/cc. A graphite sleeve 2.375 inches was adjacent to the core, followed by a beryllium reflector 4.5 inches thick.

In two dimensional calculations the control drums were explicitly represented while in one dimensional calculations, the control drums were homogenized with beryllium in the beryllium region using weighting factors derived from the two dimensional calculations.

5.2.2.2 RADIAL POWER DISTRIBUTION

Early calculations indicated that the assumption of region-wise constant cross sections in the thermal group resulted in an under estimate of the power peaking.

TABLE 5.3
COMPARISON OF KIWI B-1A ANALYTICAL MODEL

LASL						WANL						
Region	Densities, atoms/cm ³ x 10 ²⁴					Densities, atoms/cm ³ x 10 ²⁴						
	cm. Radius	°C Temp.	Oy	C	Nb	cm. Radius	°C Temp.	U235	U238	C	Nb	H
	0					0						
Core	38.997	2000	.000313	.07362	.000562	17.800	1838	.000270	.000019	.068972	.000339	.000057
	43.070	"	.000188	.07362	.000562	23.200	"	.000273	.000019	.069323	"	.000056
	44.450	"	-----	.09478	-----	26.700	"	.000277	.000019	.069630	"	.000055
						33.040	"	.000281	.000020	.069981	"	.000053
						35.260	"	.000277	.000019	.069630	"	.000055
						40.000	"	.000262	.000018	.068315	"	.000060
						44.960	"	.000160	.000011	.068490	"	.000059
Pyro C	45.418	1950		.03455		C ref 51.032	16			.088727		
C Sleeve	46.548	1072		.8776								
CCylinder	51.067	17		.07372								
SS Liner	51.118	17		.08355	Fe*							

*This stainless steel liner is in the graphite surrounding the core.

(Table 5.3 Continued)

COMPARISON OF KIWI B-1A ANALYTICAL MODEL

Region	LASL				WANL				
	cm. Radius	°C Temp.	Be	Al	cm. Radius	°C Temp.	Be	Al	H
Reflector	62.548	17	.1149		62.462	-73	.1002	.001214	.001992
Pressure Shell	64.113	17		.05432					

Basic Differences: LASL divided the lightly loaded core region into fueled and unfueled parts. WANL treated everything from the core edge to the beryllium reflector as one graphite region with uniform composition and temperature. The pressure vessel was not included in WANL calculations. The WANL model was refined to permit consideration of spatial variation of the thermal neutron spectrum using the extended CALAME method. The further core sub-division used in these calculations is not included in this table.

With respect to power peaking, recent calculations have been primarily concerned with taking the spatial variation of the thermal neutron spectrum into account. For illustrative purposes, a comparison between the power distribution calculated using region-wise constant cross sections and one calculated taking the spatial variation of the thermal neutron spectrum into account is shown in Figure 5.6.

To permit comparison with the LASL calculations, WANL hot power distribution calculations have been carried out for an assumed hot critical vane position of 137° to the extent that the spatial variation of the thermal neutron flux has been accounted for with an assumed vane position of 137° . The procedure used to date is an extension of the method developed by CALAME⁽¹⁾ which is outlined in Westinghouse Progress Report, WANL-TNR-003.

Use of a series of multigroup thermal codes for this problem is beginning. Subsequent to the preparation of spatially varying thermal cross sections for a vane position of 137° , LASL experiments indicated that the hot critical vane position was 165° . Two power plots were obtained using the spatially varying cross sections calculated for an assumed drum position of 137° . These represent the power distributions for two vane positions; i.e., 137° and 165° . The AIM-5 one dimensional diffusion theory code was used to obtain these results. For a more accurate calculation, a set of thermal constants is required for each vane position. The boral vanes were represented by homogenization over a portion of the reflector using weighting functions derived from similar CURE two dimensional calculations.

The curves shown in Figure 5.7 provide an indication of the accuracy with which the radial power distribution can be calculated at the present time. Shown in

Figure 5.7 are measured cold and hot power distributions as well as two LASL calculations

and one WANL calculation. However, as indicated previously, the measured hot power distribution may not be that corresponding to the full power temperature profile.

The measured hot and cold power distributions are not very different except near the core-carbon interface. The WANL prediction is reasonably accurate at this point.

The WANL distribution is repeated in Figure 5.8. The power distribution plotted directly from the AIM-5 output is included for both 137° and 165° drum positions although, as mentioned previously, the thermal cross sections were prepared for an assumed drum position of 137°. As can be seen from Figure 5.8, the power distribution is then obtained by drawing a smooth curve through points located at the center of a discontinuity as well as through points at the center of a sub-interval. The exceptions are discontinuities caused by changes in the composition of a region. Note that the smooth curve indicates greater peaking than the discontinuous curve. This behavior is consistent with the model used.

5.2.2.3 AXIAL POWER DISTRIBUTION

Axial power distributions were calculated both with and without a shield. The results of these calculations are shown in Figure 5.5. As can be seen, the power distribution in the unshielded core does not differ significantly from that in the shielded core except near the inlet where a maximum increase in power density of approximately 25% occurs as a result of shield backscatter and neutron moderation. The curves shown in Figure 5.5 do not account for the spatial variation of the thermal neutron energy distribution. This effect is much smaller in the axial direction than it is in the radial direction.

5.2.2.4 CONTROL VANE WORTHS

The hot control vane worths in the KIWI B-1A core have been calculated by WANL using the CURE two-dimensional diffusion theory code; however, there is some indication that these calculations may be predicting worths which are too large. The hot calculations predict that the total worth of the vanes is $0.09 \Delta k$ whereas the LASL cold measurements indicate that the vanes are worth $0.068 \Delta k$.

A plot of the WANL calculational results for the hot case is shown in Figure 5.9; also plotted are the LASL cold measurements. The spatial variation of the thermal neutron spectrum was taken into account in these calculations. It appears reasonable to expect that the vanes will be worth more (for given vane positions) when the core is hot than when the core is cold provided reflector conditions remain relatively unchanged; however, as mentioned previously, it appears that present WANL calculations are predicting worths which are too large. Work is in progress to obtain an improved representation for the control vanes in the CURE two-dimensional geometry.

6.0 LIFETIME REACTIVITY EFFECTS

6.1 FUEL DEPLETION

The mass of U^{235} in an operating rocket reactor may be depleted by two means which are discussed briefly here.

1. Burnup - Despite the high fission densities characteristic of propulsion reactors, very little fuel is consumed because the operating life of a rocket reactor is generally very short if shutdown periods are subtracted. The fraction of U^{235} consumed in the NERVA B-1 during 20 minute operation at full power is about 0.021 per cent.

2. Mechanical Loss - U^{235} may be lost to the system through erosion by the propellant stream or by diffusion through the fuel surface. Since the coolant channels are niobium clad for most of their length, in particular, over the hottest portion of the channel, this effect has been assumed negligible.

6.2 FISSION PRODUCT WORTH

The only fission product which is considered significant in fission product worth is Xe^{135} . Both Xe^{135} and its precursor I^{135} are formed in fission with yields of 0.3 per cent and 5.6 per cent respectively. During the relatively short operating times encountered in typical rocket missions, the buildup of I^{135} is effectively linear.

$$N_I(t) = \frac{\mu_I F}{\lambda_I} (1 - e^{-\lambda_I t}) \approx \mu_I F t \text{ for } \lambda_I t \ll 1$$

$$N_I(t) = I^{135} \text{ concentration as a function of time, atoms/cm}^3$$

$$F = \text{fission density, fissions/cm}^3 - \text{sec.}$$

$$\mu_I = \text{fission yield of } I^{135}$$

t = time, sec.

λ_I = I^{135} decay constant, sec.^{-1}

The production of Xe^{135} during reactor operation is given by

$$N_{Xe}(t) = \frac{\mu_{Xe} F}{M_*} \left\{ 1 + \frac{\lambda_I \mu_I}{L_* \mu_{Xe}} + \left[\frac{\lambda_I \mu_I}{\mu_{Xe} (M_* - L_*)} - 1 \right] e^{-M_* t} - \frac{\lambda_I \mu_I M_* e^{-L_* t}}{L_* \mu_{Xe} (M_* - L_*)} \right\},$$

where symbols $N(t)$, μ , F and λ are as defined above with appropriate subscripts for xenon and iodine and

$$M_* = \lambda_{Xe} + \int_E \sigma_a^{Xe}(E) \phi(E) dE, \text{ sec.}^{-1}$$

$$L_* = \lambda_I + \int_E \sigma_a^I(E) \phi(E) dE, \text{ sec.}^{-1}$$

$\sigma_a(E)$ = microscopic absorption cross section, cm^2

$\phi(E)$ = Neutron flux of energy E , neutrons/ cm^2 - sec.

Burnout of Xe^{135} and I^{135} is negligible. After shutdown the decay of I^{135} to Xe^{135} continues. The Xe^{135} concentration at any time $\tau = t - T_0$ after shutdown is given by the equation

$$N_{Xe}(\tau) = \frac{\lambda_I N_I(T_0)}{\lambda_I - \lambda_{Xe}} (e^{-\lambda_{Xe} \tau} - e^{-\lambda_I \tau}) + N_{Xe}(T_0) e^{-\lambda_{Xe} \tau},$$

where T_0 is the time of operation in seconds. The concentration reaches a maximum at about 11.25 hours after shutdown. Substituting this time τ_m in the above equation, the

peak Xe^{135} concentration is approximately given by

$$N_{\text{Xe}}(\tau_m) \approx .0240 F T_o + 0.428 N_{\text{Xe}}(T_o) .$$

The negative reactivity associated with xenon poisoning is approximately given by the ratio of xenon to uranium absorption, integrated over the reactor spectrum. However, such estimates appear to be low by as much as a factor of two. Improved estimates of reactivity losses due to peak xenon can be obtained by simply running a criticality problem with the proper xenon concentration. The per cent loss in effective multiplication factor for the NERVA B-1 reactor at peak xenon concentration after operating 20 minutes at full power is calculated to be 0.161.

An additional parameter of importance in control studies is the rate of burnout of Xe^{135} upon restart. This rate is approximately greatest at the time of peak Xe concentration, τ_m . At that time the rates of I^{135} and Xe^{135} decay are equal. The initial rate of reduction in xenon concentration (to which the reactivity loss is proportional) is then equal to burnout alone:

$$\frac{d N_{\text{Xe}}(\tau_m)}{dt} = - N_{\text{Xe}}(\tau_m) \int_E \sigma_a^{\text{Xe}}(E) \phi(E) dE ,$$

where $\phi(E)$ is the neutron flux and $\sigma_a^{\text{Xe}}(E)$ is the absorption cross section of Xe^{135} .

In thermal reactors epithermal absorption by xenon is nearly negligible, while in the NERVA reactors, epithermal absorption appears to be of the order of 10 per cent of the total xenon burnout. The rate of reduction of xenon burnout in the B-1 reactor at peak xenon after 20 minutes of full power reactor operation is calculated to be 1.1×10^{11} atoms/cm³ - sec.

7.0 RADIATION HEATING OF REACTOR COMPONENTS

7.1 GENERAL

Integrated heating rates have been computed for components of the B-1 reactor in order to make preliminary estimates of cooling requirements. These computations were performed by hand, and where possible they were compared with machine calculated values. This comparison indicates that these calculations are adequate for preliminary design purposes. Future computations for specific reactor designs will be accomplished using machine codes. The total power absorbed as a fraction of full power is given for the following components:

- 1) Core
- 2) Graphite Sleeve
- 3) Graphite Reflector
- 4) Beryllium Reflector
- 5) Aluminum Pressure Vessel
- 6) Aluminum Support Plate
- 7) Li H - SS Shield

These heating rates have been obtained for startup, steady-state, and programmed shutdown conditions. The nuclear heating in the lithium hydride-stainless steel shield is discussed in more detail in report WANL-TNR-002. Estimates of nuclear heating at the following specific points have also been calculated:

- 1) Hydrogen in the reflector
- 2) Hydrogen in the nozzle

3) Control drum actuator

4) Tungsten/Tungsten-Rhenium thermocouple in the nozzle

The heating due to the (n, α) reaction in B^{10} has been calculated for the bore of the control drums. Also the heating due to this same (n, α) reaction has been estimated for a pressure vessel bore lining that was considered for the proposed heated bleed cycle.

7.2 ASSUMPTIONS AND METHODS OF HEATING RATE CALCULATIONS

The reactor core is a cylinder 45 cm in radius and 128 cm in height. The material composition of the core weighted by volume fraction is:

- 1) Nb: 0.0565 gm/cm^3
- 2) C: 1.408 gm/cm^3
- 3) U: 0.1148 gm/cm^3
- 4) H: $0.362 \times 10^{-3} \text{ gm/cm}^3$

The reactor components are characterized by the following geometry and material densities:

<u>Component</u>	<u>Density</u>	<u>Inner Radius</u>	<u>Outer Radius</u>	<u>Height</u>
Graphite sleeve	1.818 gm/cm^3	45.0 cm	46.47 cm	128 cm
Void	-----	46.47	46.84	128
Graphite reflector	1.818	46.84	50.64	128
Void	-----	50.64	50.96	128
Beryllium reflector	1.49	50.96	62.39	128
Void	-----	62.39	62.55	128
Aluminum pressure vessel	2.7	62.55	63.98	128
Aluminum support plate	1.16	0	45.0	15.2

A uniform power distribution was assumed for a nuclear power of 1120 mw. This is an average power density of 1378 watt/cm^3 .

The spectrum of prompt gamma rays resulting from fission was approximated by three energy groups. It was assumed that each fission results in two gamma rays of 0.8 Mev, 1.7 gamma rays of 2.5 Mev, and 0.5 gamma ray of 6.0 Mev. This grouping includes an allowance for radiative capture in U^{235} . The delay gamma spectrum that is due to the decay of fission products is broken down into energy groups of 0.8 Mev, 2.5 Mev, and 4.0 Mev. The magnitude of the delay gamma sources is computed from data contained in XDC 60-1-157, "The Activity of the Fission Products of U^{235} ".

The gamma sources that would result from 20 minutes of operation at 1120 nuclear megawatts are:

TIME AFTER SHUTDOWN				
Energy	Operation	10 sec	10^3 sec	10^4 sec
0.8	$11.8 \times 10^{13} \text{ Mev/cm}^3 \text{ sec}$	3.90×10^{13}	0.458×10^{13}	5.20×10^{11}
2.5	28.8×10^{13}	8.66×10^{13}	0.961×10^{13}	6.64×10^{11}
4.0	5.03×10^{13}	3.29×10^{13}	0.152×10^{13}	0.216×10^{11}
6.0	12.8×10^{13}	0.691×10^{13}	-----	-----

The gamma energy flux at a point due to the core gamma sources being attenuated by geometry and shield materials was computed by standard formulae amenable to hand calculation⁽²⁾. A two-exponent fit to the energy buildup factor for a point isotropic source in water was assumed. This assumption tends to over-estimate the effect of energy buildup.

The power resulting from β decay of fission products at a time Υ days after startup due to operation at power P for a time T_o days is

$$P_{\beta} = 0.328 P \left[(\Upsilon - T_o)^{-0.2} - \Upsilon^{-0.2} \right].$$

It is assumed that all beta particles resulting from fission and the decay of fission products are absorbed in the reactor core.

The heating effect of neutron elastic scattering was computed by normalizing the P1MG few-group fluxes at a point to the PDQ few-group fluxes at the same point. This normalization factor was assumed to be energy independent and was then applied to the multigroup P1MG fluxes. The heating is then

$$H \left[\frac{\text{Mev}}{\text{cm}^3 \text{ sec}} \right] = k \sum_{i=1}^n \phi_i(u) \Delta E \Sigma_s(u) f(u),$$

where

k = P1MG - PDQ normalization constant

$\phi_i(u)$ = P1MG lethargy flux for i th group

ΔE = energy corresponding to Δu of i th group

$\Sigma_s(u)$ = scattering cross section

$f(u)$ = average loss of energy per collision .

In practice it is found that the contribution to the total heating of the energy groups beyond the twenty-fifty group is negligible.

The heating effect of the (n,γ) reaction was evaluated for the reactor components. In the most significant case this reaction was found to increase the total heating by less than 10%.

The 14-0 code will not compute fluxes in a radiation source region. Therefore, hand calculations were used in the core region to compute heating rates. Two values of heating rates are reported for the core region. The lower values are the results of using average hand computed values of gamma flux. The highest value is the maximum value that can be computed from hand calculations.

In the radial direction these quasi-hand calculations of nuclear heating compare favorably with the heating rates obtained by using the more refined 14-0 shielding code. This comparison is shown in Figure 7.1 for the midplane heating rates in the beryllium reflector and aluminum pressure vessel.

Heating due to the (n,α) reaction in the boral of the control drum vanes and a proposed boral lining of the pressure vessel was calculated using neutron fluxes obtained from the PDQ four-group diffusion code. It was assumed that each neutron absorption produces a 2.5 MEV alpha particle which is absorbed in the boron containing material.

7.3 DECAY POWER VERSUS TIME AFTER SHUTDOWN

The decay power versus time after shutdown is shown on Figure 7.2. Three possible modes of shutdown from a full power of 1120 nuclear megawatts were studied. In each case shutdown is due to a step input of -3.5% in reactivity. The shutdown modes studied are:

- 1) Full power operation for 20 minutes followed by shutdown
- 2) Full power operation for 15 minutes followed by half-power operation for 10 minutes

- 3) Full power operation for 15 minutes followed by quarter-power operation for 20 minutes.

In each of the above cases there is an equivalent 22,400mw-minutes of operation.

Figure 7.3 is a plot of fission product decay power versus time after shutdown from full-power operation for 20 minutes.

The curves of Figures 7.2 and 7.3 can be used to determine after-shutdown power as a per cent of operational power. Tables 7.2 and 7.3 can then be used to determine what per cent of this decay power is absorbed in a given component.

TABLE 7.1
PER CENT OF REACTOR POWER ABSORBED FOR
20 MINUTE OPERATION AT FULL POWER

Component	Time After Shutdown, Sec.			
	0	10	10^3	10^4
Core	91-93	91-93	78-93	78-94
Graphite Sleeve	0.28	0.66	1.34	1.26
Graphite Reflector	0.55	1.36	2.62	2.33
Be Reflector	0.83	1.88	3.46	2.98
Al Pressure Vessel	0.092	0.27	0.51	0.43
Ti Pressure Vessel	0.063	0.18	0.35	0.29
Al Support Plate	0.20	0.62	1.28	1.16
Li H - SS Shield	0.28	0.48	0.81	0.68

TABLE 7.2
PER CENT OF REACTOR POWER ABSORBED
AFTER PROGRAMMED SHUTDOWN

Component	15 Min. Full-Power 10 Min. Half-Power		15 Min. Full-Power 20 Min. Quarter-Power	
	Operation	10^3 sec	Operation	10^3 sec
Core	91-93	77-93	91-93	79-93
Graphite Sleeve	0.29	1.34	0.29	1.39
Graphite Reflector	0.56	2.89	0.59	2.78
Be Reflector	0.84	3.74	0.88	3.73
Al Pressure Vessel	0.094	0.61	0.10	0.59
Ti Pressure Vessel	0.064	0.41	0.068	0.40
Li H - SS Shield	0.23	0.92	0.24	0.90

7.4 HEATING RATES IN REACTOR COMPONENTS DURING PROGRAMMED STARTUP

The startup program investigated requires that one per cent power be attained within 30 seconds and that full power be attained within 60 seconds after startup. Table 7.3 lists the per cent of total power generated in the core which is absorbed in a given reactor component. For each component a maximum and minimum heating value has been computed. The maximum value is comprised of heating due to neutron elastic scattering, prompt gamma rays from fission, and beta and gamma decay of fission products. The contribution from fission product decay was excluded in computing the minimum value of heating which might occur on a first reactor startup.

TABLE 7.3

PER CENT OF REACTOR POWER ABSORBED AT STARTUP

Component	1% of Full Power at 30 sec		100% of Full Power at 60 sec	
	Min.	Max.	Min.	Max.
Core	90-92	90-93	90-92	90-93
Graphite Sleeve	0.20	0.23	0.20	0.25
Graphite Reflector	0.40	0.46	0.40	0.49
Be Reflector	0.60	0.71	0.60	0.72
Al Pressure Vessel	0.064	0.074	0.064	0.079
Ti Pressure Vessel	0.044	0.050	0.044	0.054
Al Support Plate	0.14	0.17	0.14	0.19
Li H - SS Shield	0.19	0.20	0.19	0.22

7.5 HEATING RATES IN REACTOR COMPONENTS DURING FULL-POWER
OPERATION AND AFTER STEP SHUTDOWN FROM FULL POWER

Nuclear heating rates in the various components resulting from full-power operation at 1120 nuclear megawatts for 20 minutes are reported in Table 7.2. Figures 7.4 through 7.7 are curves of the radial heating rates at full power and 10 , 10^3 and 10^4 seconds after shutdown. The heating rates for the operating case are computed for a time at which the reactor has been at full power for 20 minutes.

Approximately 85% of the fission energy release manifests itself in beta particles and the kinetic energy of fission fragments. This energy is absorbed in the core.

Of the 8 to 9% of fission energy release appearing as gamma rays and kinetic energy of neutrons, more than 7% is absorbed in the core to give a total in-core heating of approximately 93% of the total nuclear power produced during operation.

According to Table 7.1 approximately 95% of the total generated power during operation is absorbed in the core and reflector. In order to extract 1120 thermal megawatts from these reactor components, the total generated power must be approximately 1.07×1120 or 1200 mw.

As an example of the use of Table 7.1, consider the heat absorbed in the graphite reflector at 10^3 seconds after shutdown. From Figure 7.3 the total generated power at 10^3 seconds after shutdown is 0.0022 of 1200 mw or 2.6mw. Table 7.1 gives the per cent of the power generated at a given time after shutdown that is absorbed in a specified component. At 10^3 seconds the heat absorbed in the graphite reflector is $.0262 \times 2.6$ mw or approximately 70 kw.

7.6 HEATING RATES IN REACTOR COMPONENTS AFTER PROGRAMMED SHUTDOWN

At times greater than 100 seconds after shutdown the power being produced is almost totally due to beta and gamma decay of fission products. It is assumed that all of the beta particles are absorbed in the active core. Calculations show that approximately 80% of the gamma energy is absorbed in the core.

There is a decrease in per cent of power absorbed after 10^3 seconds. This decrease is almost pronounced in reactor components farthest from the core and is due to a spectral shift with time towards lower energy gamma sources.

Heating values for two possible shutdown programs are reported in Table 7.2.

The criterion for picking a shutdown program is that there shall be 22,400 mw minutes of

operation. The heating value reported for operating conditions is at a time just prior to shutdown.

Gamma heating rates at several specific points have been computed for the full-power operating condition. The heating of the hydrogen one foot below the core is 2 watt/gm. The hydrogen at core midplane in the beryllium reflector will absorb approximately 15 watt/gm from gamma rays. A tungsten/tungsten-rhenium thermocouple placed 4 inches below and 4 inches from the edge of the active core will be heated at the rate of 15 watt/gm.

If shielded by the Li H - SS Shield which reduces tank heating by 90%, the aluminum control drum actuator will absorb a total of 1.1 w/gm due to the combined heating effects of neutrons and gamma rays. This heating rate is equivalent to 1.3×10^8 RAD for 20 minutes operation at full power.

The heating from the (n, α) reaction in the boral vane of the control drums is approximately .53 mw per drum when the drums are at 90° and the reactor is at full power. A plot of vane heating as a function of control drum angular position is presented in Figure 7.8. The peak-to-average heating rate in a control vane is estimated to be approximately 4.5.

7.7 HEATING IN PROPOSED BORAL LINING OF PRESSURE VESSEL FOR HEATED BLEED CYCLE

A boral lining of the pressure vessel was proposed in order to obtain a heat source for the heated bleed cycle. Two types of boral were investigated as possible linings. The composition by atom density of these two types is:

	Atom Density (atom/cm ³)	
	Boral 187 B	Sigma Boral
B10	0.0234×10^{24}	0.0284×10^{24}
C	0.00585×10^{24}	-----
Al	0.0452×10^{24}	0.0421×10^{24}

The thickness of boral lining as a function of power produced from the (n, α) reaction in the boral is plotted in Figure 7.9. The weight of B¹⁰ necessary to obtain 1 mw of power from a boral lining of the pressure vessel is 1.6×10^3 gm.

Figure 3.1

KIWI B-1A Reactor Assembly

702J052

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SECTION A-A

NERVA REACTOR BASED ON KIWI B 1

GRAPHITE FELT LATERAL SUPPORT
TOP LOADED CORE

FIGURE 3.2

702J063

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Figure 3.3

702J116

Figure 3.4

367D654

NERVA Control Drum Assembly

Figure 3.5

975D880

Heated Bleed Shield Schematic

FIGURE 4.1
INTEGRATED RADIAL COLD FISSION DISTRIBUTION IN KIWI-B-1A

FIGURE 4.2
KIWI - B - 1A
INTEGRAL CONTROL VANE WORTH

FIGURE 4.3
KIWI B-1A DIFFERENTIAL CONTROL VANE WORTH

FIGURE 4.4
ROSSI α MEASUREMENTS ON KIWI-B-1A

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FIGURE 4.5
ROSSI- α MEASUREMENTS ON KIWI-B-1A

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Figure 4.6

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k_{eff} vs. Per Cent Hydrogen in Be Reflector
for B-1

FIGURE 4.7
KIWI B-1A RADIAL HOT FISSION DISTRIBUTION AT NTS INTEGRATED OVER CORE LENGTH

DSN MODEL OF
 KIWI-B REACTOR GEOMETRY
 FIGURE 5.1

FIGURE 5.2
 COMPARISON OF DSN RESULTS WITH COLD EXPERIMENTAL TRAVERSES

FIGURE 5.3
TEMPERATURE AND CONTROL VANE POSITION CORRECTIONS FOR
FISSION DISTRIBUTION IN KIWI-B-1A

KIWI B-1-A, HIGH TEMPERATURE (FISSIONS / FUEL
ELEMENT) / FISSIONS IN CENTRAL FUEL ELEMENT
INTEGRATED OVER COURSE 1 + COURSE 2

FIGURE 5.4

Figure 5.5

LASL and WANL Axial Power Distributions

FIGURE 5.7
HOT RADIAL POWER DISTRIBUTION
KIWI-B-1A

Figure 5.8

WANL Calculated Radial Hot Power
 Distribution

Figure 5.9

547566

Integral Control Vane Worth Curve
 (Hot, Calculated)

Figure 7.1

537432

Comparison of Nuclear Heating Rates at Midplane

TOTAL DECAY POWER VS TIME AFTER SHUTDOWN (10^{-1} - 10^3 SEC)

Figure 7.2

537433

Total Decay Power vs. Time After Shutdown
(10^{-1} to 10^3 Seconds)

Figure 7.3

537634

Figure 7.4

537434

20 Minutes Full Power, Radial Heating During
 Operation

Figure 7.5

537435

20 Minutes Full Power, Radial Heating 10
 Seconds After Shutdown

Figure 7.6

537436

20 Minutes Full Power, Radial Heating 10^3
 Seconds After Shutdown

Figure 7.7

537437

20 Minutes Full Power, Radial Heating Outside
 Core 10^4 Seconds After Shutdown

FIGURE 7.8
VANE HEATING VS VANE ANGLE

Figure 7.9

537438

Thickness of Boral vs. Power

9.0 REFERENCES

1. Calame, "A Few Group Theory of Water Gap Peaking," Nuclear Science and Engineering 8,400 (1960).
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