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LOS ALAMOS SCIENTIFIC LABORATORY
OF THE UNIVERSITY OF CALIFORNIA ○ LOS ALAMOS NEW MEXICO

LOS ALAMOS MOLTEN PLUTONIUM REACTOR EXPERIMENT (LAMPRE) HAZARD REPORT

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LOS ALAMOS MOLTEN PLUTONIUM REACTOR EXPERIMENT (LAMPRE) HAZARD REPORT*

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ABSTRACT

The first experiment (LAMPRE I) in a program to develop molten plutonium fuels for fast reactors is described and the hazards associated with reactor operation are discussed and evaluated. The reactor description includes fuel element design, core configuration, sodium coolant system control, safety systems, fuel capsule charger, cover gas system and shielding. Information of the site comprises population in surrounding areas, meteorological data, geology, and details of the reactor building.

The hazards discussion considers the probable consequences of loss of coolant pumping, electrical power failure, and the malfunction of the several elements comprising the reactor system. A calculation on the effect of fuel element bowing appears in an appendix.

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1. INTRODUCTION, SUMMARY, AND CONCLUSIONS

1.1 Introduction

The Los Alamos Scientific Laboratory is investigating plutonium-fueled reactors for possible application to fast breeder systems. The fuels presently being considered are molten alloys of plutonium which offer the possibility of high burn-up and in-place reprocessing.

The first reactor in this program is called the Los Alamos Molten Plutonium Reactor Experiment Number One (LAMPRE I). The design power of the reactor is 1 Mw, which will result in an average specific power of 40 w/g of fuel. This reactor is intended to serve a vital function in providing a facility to test materials under operating conditions.

The reactor is presently being installed in an existing underground cell located at Site TA-35 of the Los Alamos Scientific Laboratory. Power operation is planned for early 1960.

1.2 Summary

The fuel, Pu - 2.5 w/o Fe, is contained in tantalum capsules with a 0.376 in. i.d. x 0.025 wall and 8 in. length. There will be 175 g of fuel in each capsule and the calculated number for criticality is 143. The internal reflector and core are cooled by series flow of sodium and are doubly contained by a vessel having an 8-3/4 in. i.d. At full power, the coolant flow rate is 133 gpm and the average Δt across the core is 113°C.

The sodium coolant system, of 316 ELC stainless steel, employs a.c. electromagnetic pumps and a finned tube sodium-to-air heat exchanger. Zirconium-filled hot traps are used to reduce sodium oxide content in the coolant. Coolant cover gas is helium passed through a NaK bubbler. Cover gas pressure is about 25 psi.

Reactivity is controlled by the use of a shim and four control rods which are external to the sodium vessel. Control element actuation is hydraulic. A scram drops the shim under acceleration by gravity only.

Fuel capsules are replaced in the core using a fixed charger which, in addition to gamma shielding, is designed for containment of alpha-active contaminants.

A gas disposal system has been installed to handle safely fission product gas resulting from a capsule rupture.

The possible hazards associated with the start-up and operation of LAMPRE I have been analyzed with consideration of the effect of various temperature coefficients of reactivity, fuel element bowing, and rupture of the coolant system or of fuel capsules. Attention has also been given to the effect of failure of various components, to the consequences of large positive reactivity insertions, and to the establishment of safe handling techniques for the fuel.

1.3 Conclusions

Analysis of hazards potentially present in the LAMPRE I system leads to the conclusion that the large negative prompt temperature coefficient resulting from the thermal expansion of the liquid fuel makes this reactor unusually safe. The maximum credible malfunctions of the system are estimated to lead to reactivity insertions of the order of several hundred dollars per second, an order of magnitude less than that required for formation of an explosive shock. Operational procedures and sequencing have been designed to minimize to the point of impossibility any power excursion of sufficient magnitude to vaporize the sodium coolant or to melt the fuel container. As with any system using high-temperature sodium, sodium flammability represents one of the principal dangers to the experiment. A sodium fire might terminate the experiment, but release of radioactive smoke to the atmosphere is prevented so that personnel will not be endangered. There are provisions for alpha particle containment in various phases of fuel handling in order to guard against any hazard resulting from rupture of a fuel element.

2. DESCRIPTION OF THE REACTOR

2.1 Type and Purpose

LAMPRE I is an experimental test reactor which will have a molten plutonium alloy fuel operating with a fast neutron spectrum. The reactor will be used to investigate (a) the feasibility of using molten alloys of plutonium as reactor fuels, (b) the satisfactory containment of such fuels, (c) fission gas disengagement from the molten fuel, and (d) the suitability of this reactor concept for future power breeder reactors using similar fuels. Figures 2.1 and 2.2 are plan and elevation layouts, respectively, of the reactor installation.

2.2 Core Configuration

The core of the reactor is made up of an array of tantalum capsules containing plutonium-iron alloy fuel. There are locations for 199 capsules, some of which will be occupied by unfueled reflector pins. Figure 2.3 is a cutaway view of the core region. Coolant sodium enters the reactor vessel and flows downward through an annulus to the bottom of the vessel. From the bottom the sodium flows up through the bottom reflector into a plenum, and then through the capsule locator plate into the core. After flowing past and receiving heat from the capsules, the sodium passes into the upper reflector region and out of the vessel. A tantalum catchpot, for containing fuel in the event of a capsule rupture, is located just below the turn-around plenum. Within the catchpot a

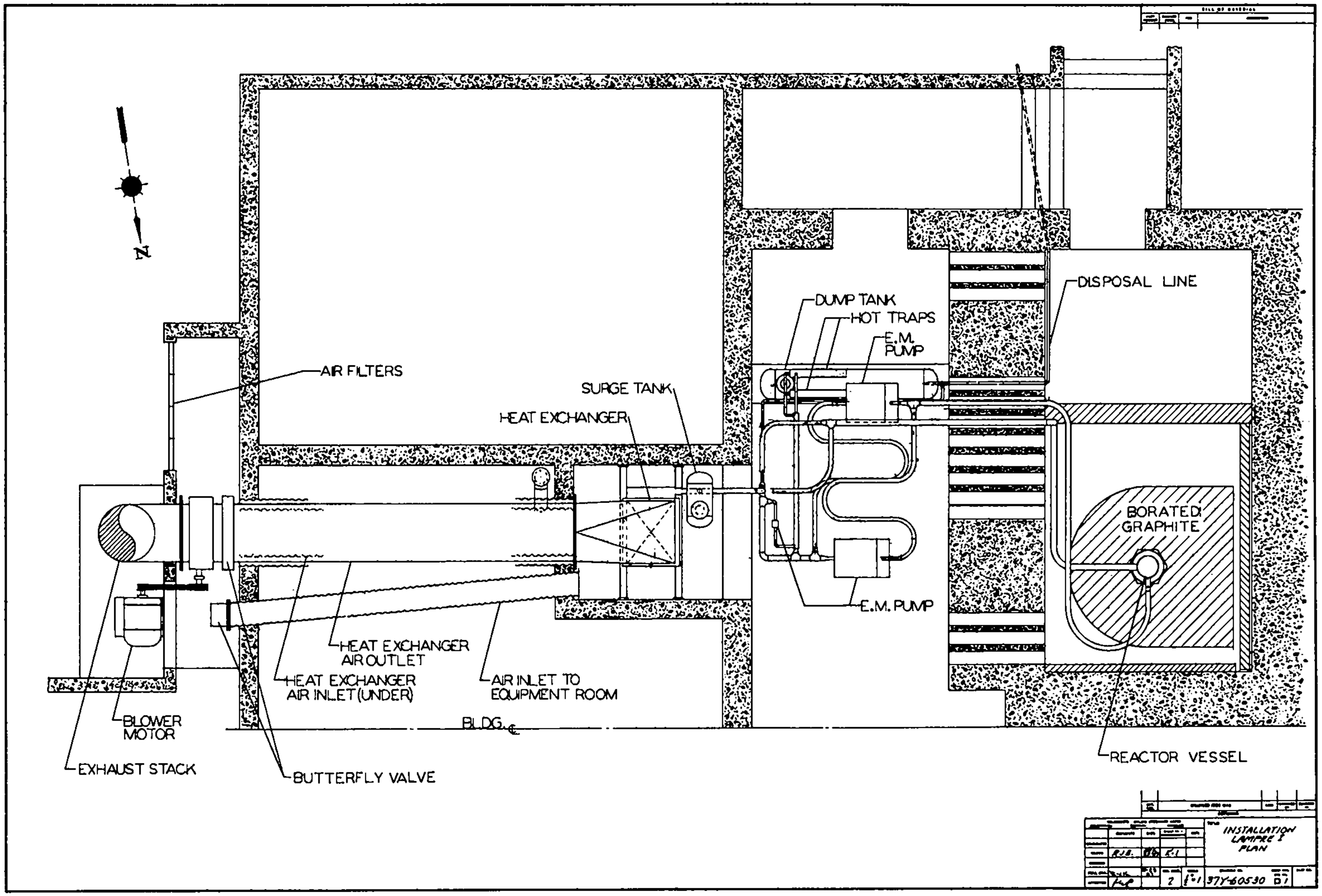

Fig. 2.1 Plan view of reactor installation.

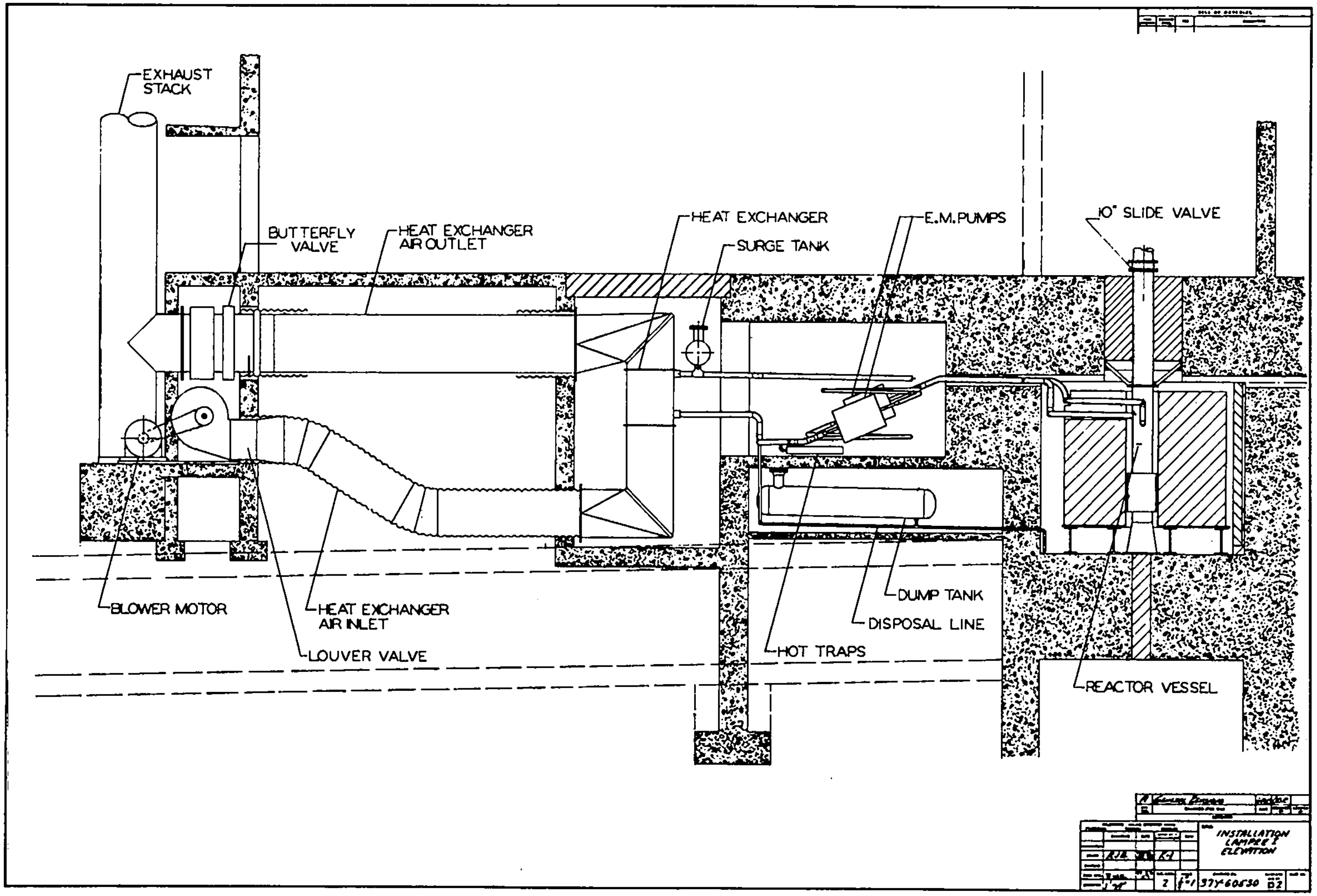

Fig. 2.2 Elevation view of reactor installation.

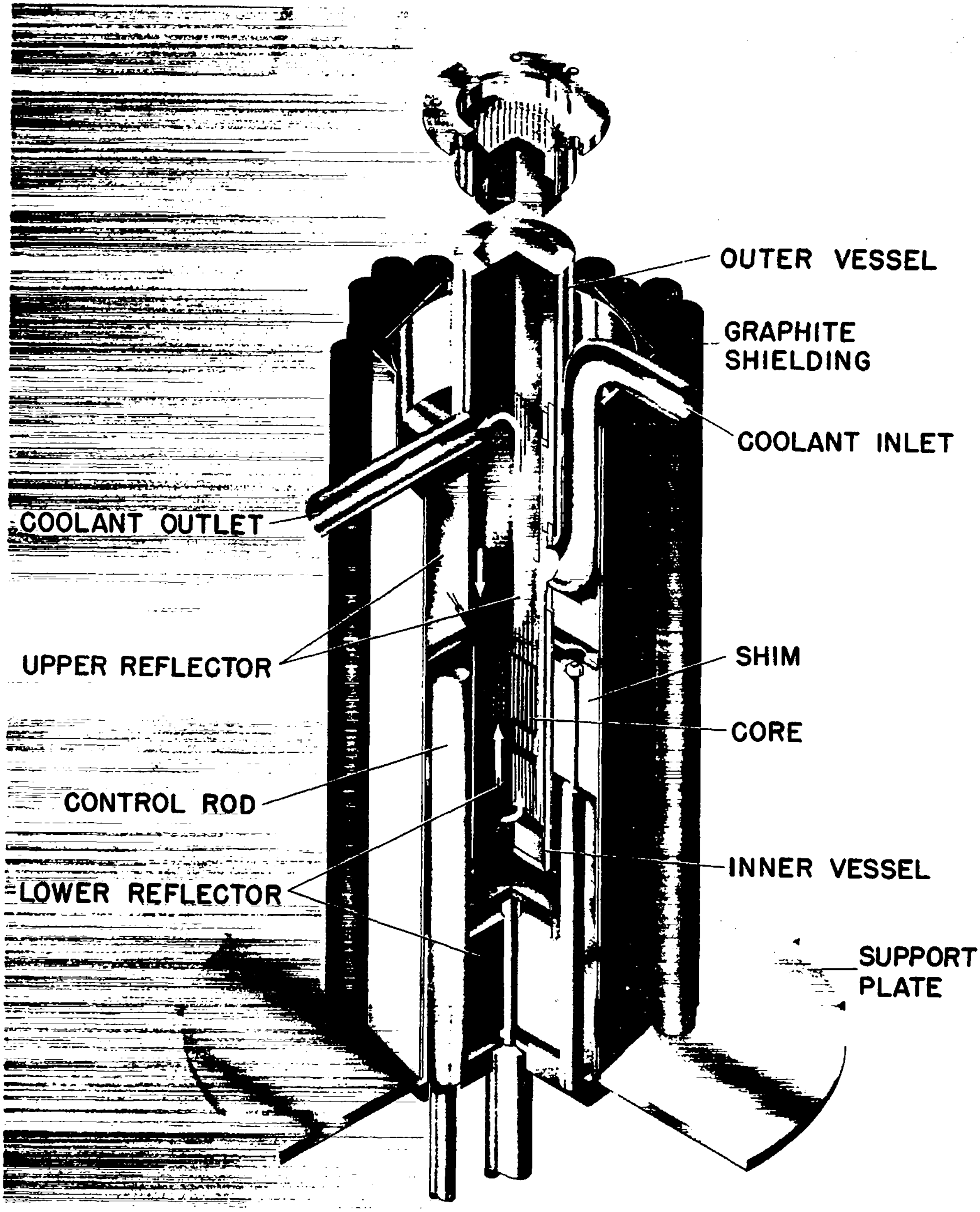

Fig. 2.3 Cutaway view of vessel region.

diluent plug of Armco iron is provided, with holes drilled so that there is about a 35% void. The plug serves to dilute fuel in the catchpot to a subcritical concentration, and also serves to alloy with and "dry up" the fuel by raising its melting point.

The bottom reflector is also made of Armco iron drilled for sodium passage, with about 17% of the cross sectional area for sodium flow. The capsule locator plate is 1-3/4 in. thick stainless steel, the top of which has been faced with a 1/4 in. thick sheet of tantalum. Sockets are drilled in the tantalum plate on a 0.497 in. triangular pitch to locate the conical capsule bottoms. The purpose of the tantalum face on the locator plate is to prevent leaking fuel from soldering the capsule tip to the stainless steel plate in the event plutonium enters the socket.

The lower part of the reactor vessel is double-walled, with no pipes entering the region. The core cannot accidentally be drained of coolant unless there is a leak in both walls of the double containment. A small line, by which the vessel may be drained if necessary, extends from the bottom of the catchpot through the sodium inlet pipe.

Capsule handles, described below, constitute the upper reflector and also part of the radiation shield. Above the vessel sodium outlet the capsule handles have a hexagonal cross section and extend to the top of the vessel. The top of the vessel is fitted with a 10 in. slide vacuum valve which will be closed when the reactor is operating.

An annular shim moving outside the sodium vessel furnishes coarse reactivity control. Vernier reactivity control is provided by four separately actuated control rods which are portions of the shim annulus. Above and below the shim are stationary steel shields. Surrounding these two shields and the shim is a flue which aids in cooling the boron-impregnated graphite immediately outside the flue. Figure 2.4 shows a horizontal cross section through the core. Figures 2.5, 2.6, 2.7, and 2.8 show the bottom reflector, catchpot and diluent plug, flow divider, and locator plate, respectively. Figure 2.9 is a photograph of the vessel.

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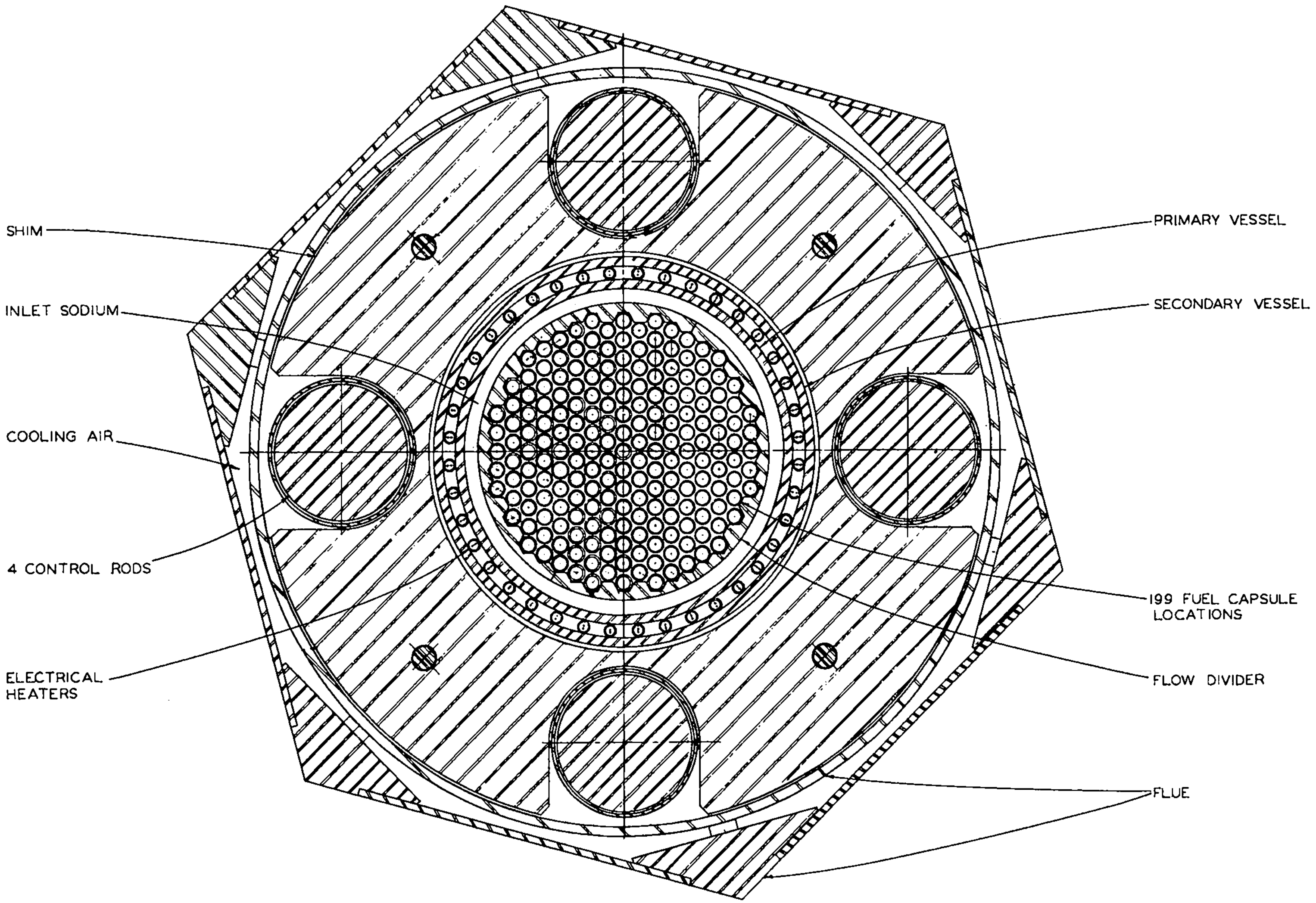

Fig. 2.4 Horizontal cross section through reactor core.

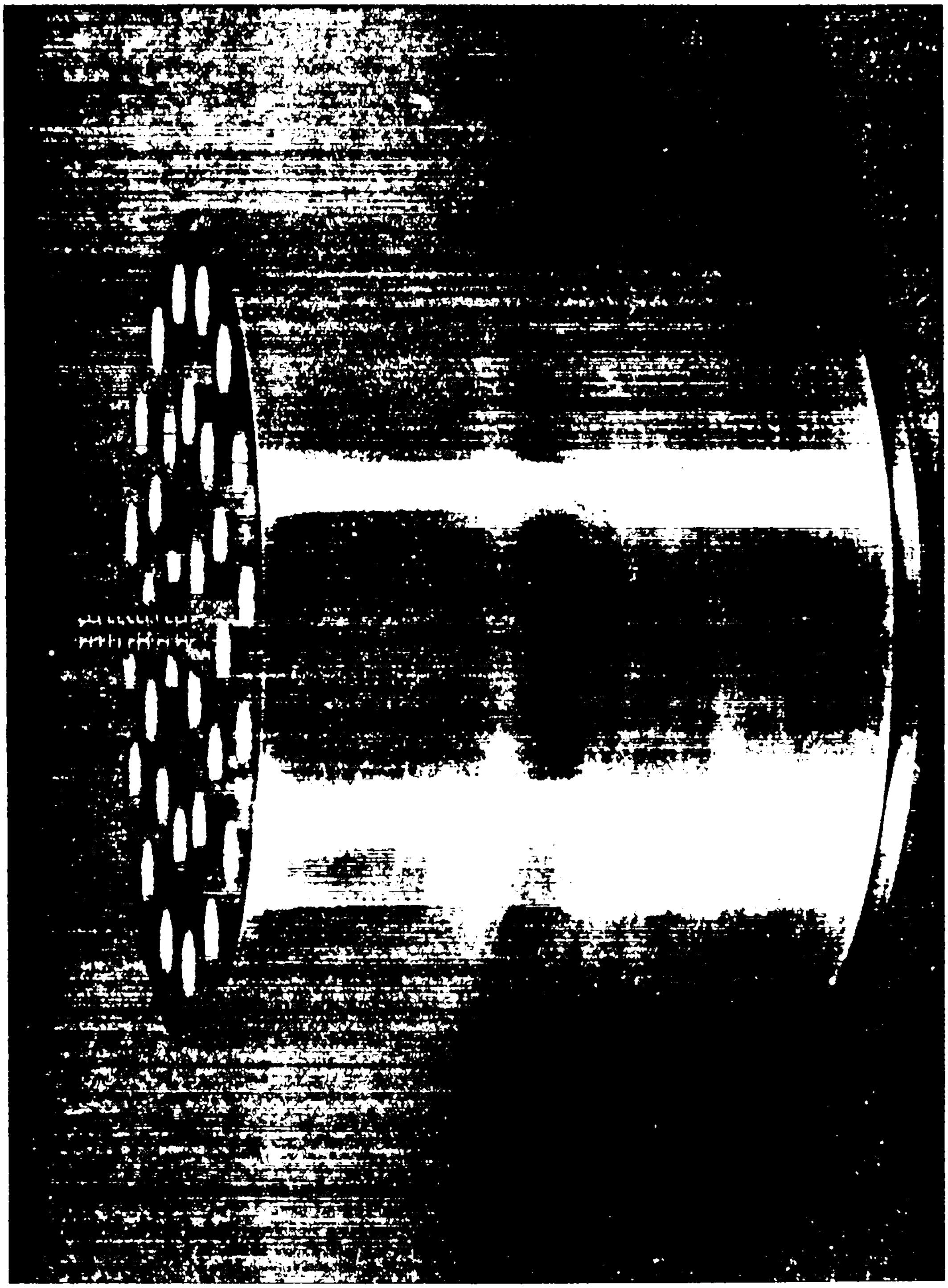

Fig. 2.5 Bottom reflector.

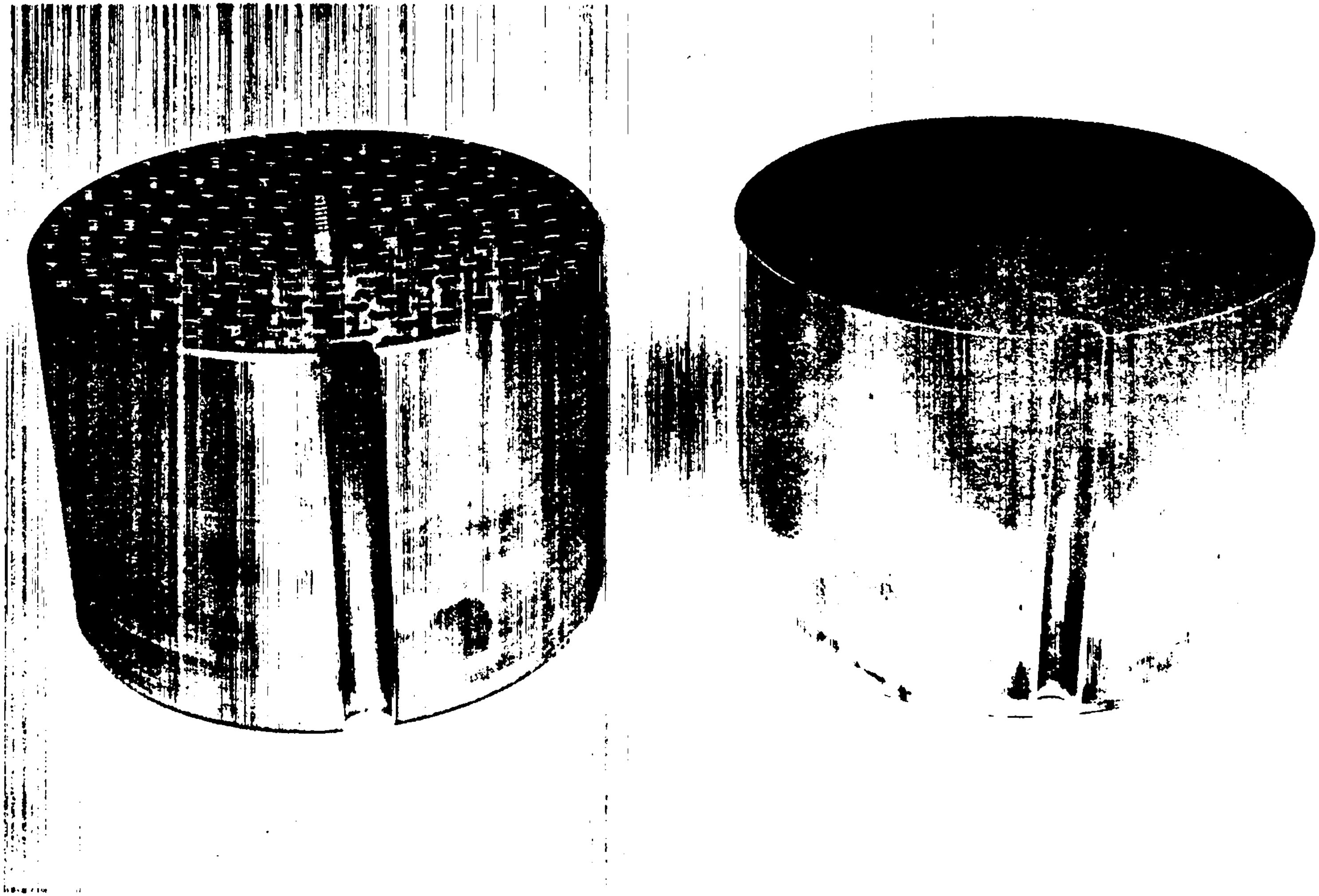

Fig. 2.6 Diluent plug and catchpot.

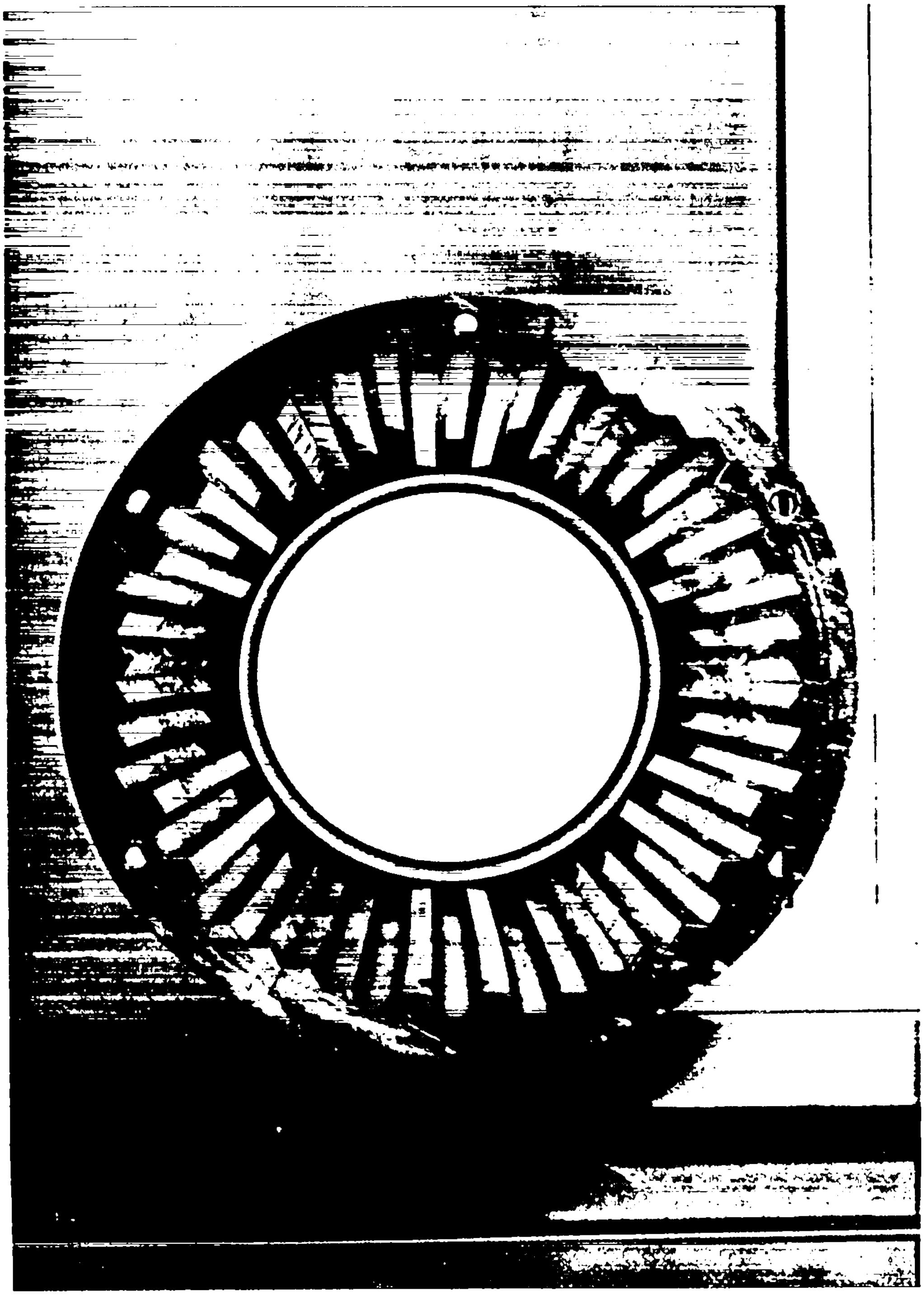

Fig. 2.7 Flow divider.

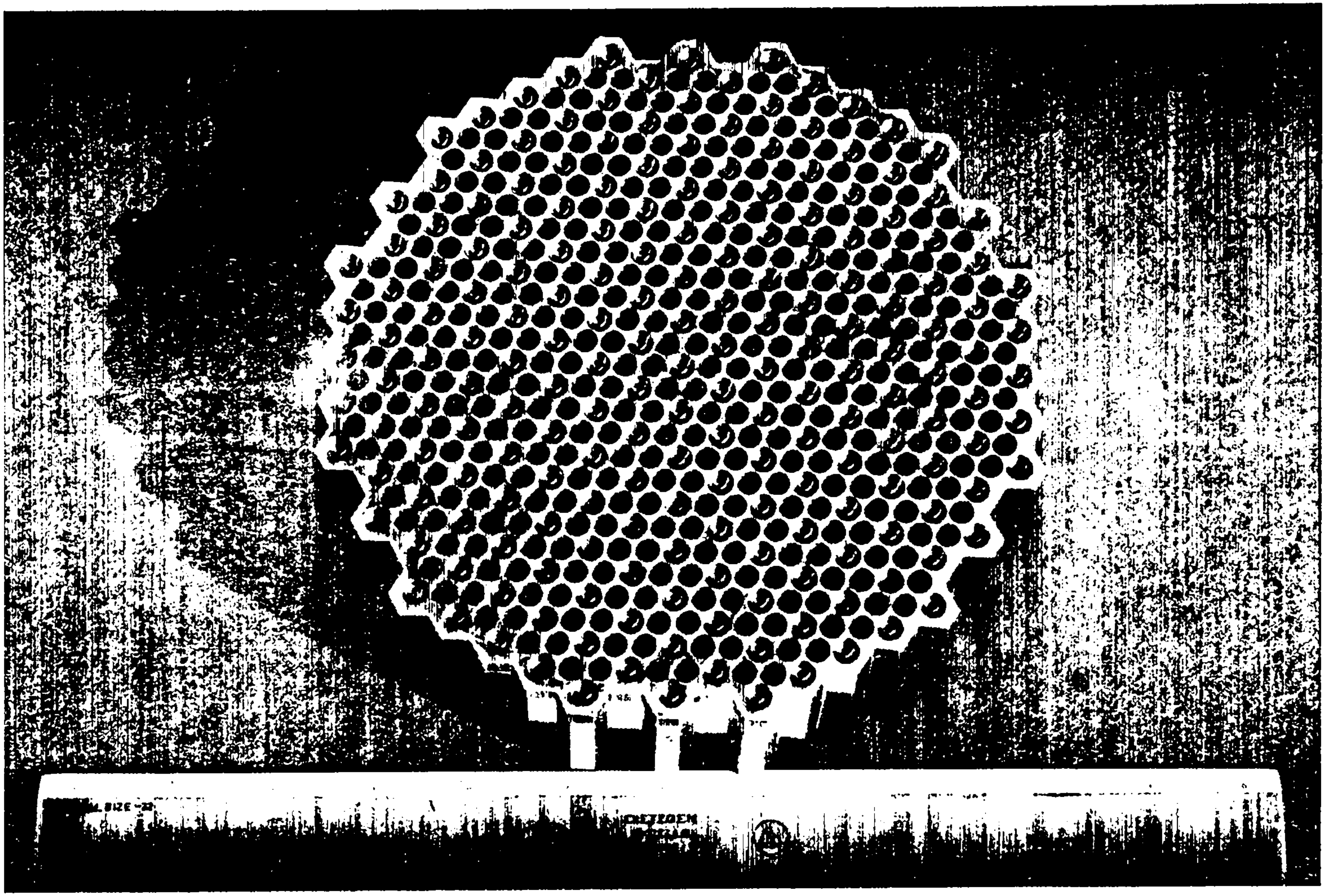

Fig. 2.8 Capsule locator plate.

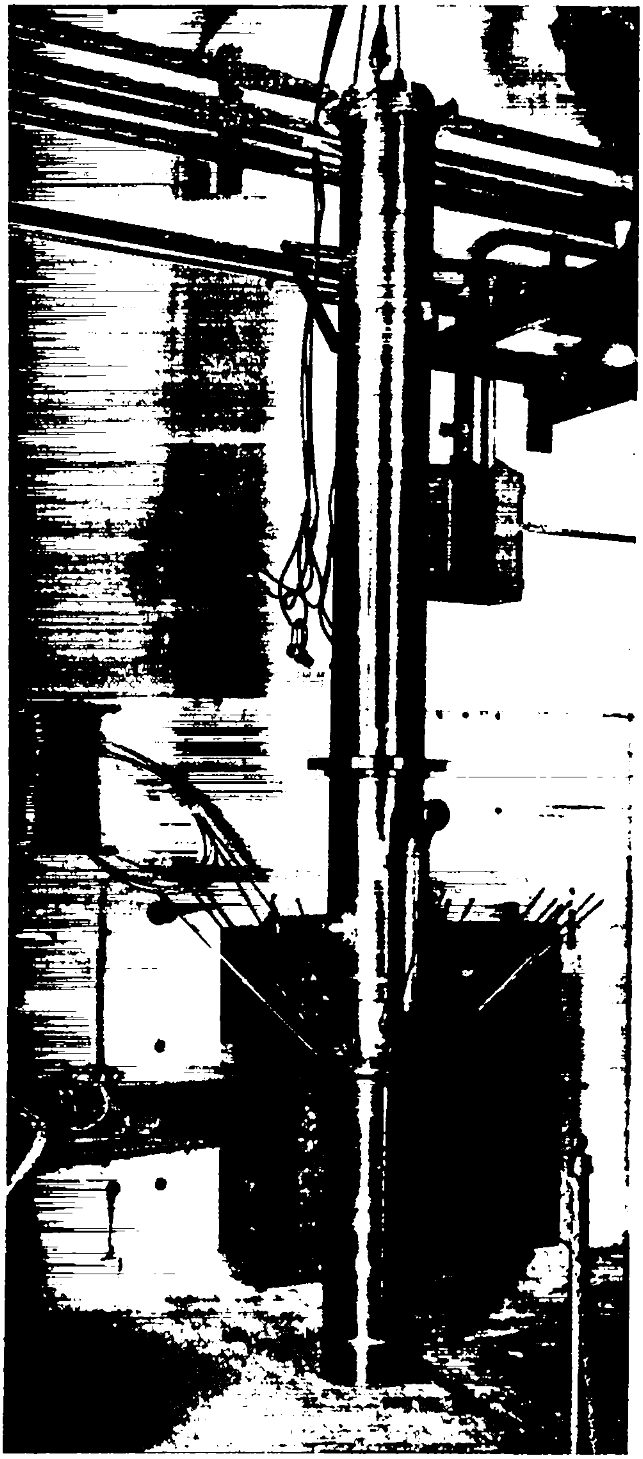

Fig. 2.9 Reactor vessel.

2.3 Fuel Element Design

Figures 2.10 and 2.11 show details of the fuel capsule; Figs. 2.12 and 2.13 show the complete element. In Fig. 2.13 a fuel element is suspended outside the vessel (to the left) but at the same elevation relative to the vessel that it would have if it were in the core. The fuel capsule is deep drawn from high purity tantalum and has an inside diameter of 0.376 in. and a 0.025 in. wall. The bottom six inches of the capsule will contain fuel, and fission product gases will accumulate in the remaining top two inches of the capsule. A tantalum closure plug is welded into the capsule top and has pinned to it a stainless steel adapter with a male thread which screws into the handle. Just above the point where the capsule is screwed to the handle is a locator section which positions the capsule tops. Sodium flows through this locator section in six splines and then into the top reflector region. The handle in the top reflector has a diameter of 0.420 in. In the outlet plenum region the handle has a diameter of 0.375 in. The remainder of the handle length has a hexagonal cross section of 0.483 in. across flats except for the handling and locking details at the upper end. Capsule locations in the core not used for fuel-filled capsules will be occupied by reflector pins. The reflector pin is a solid stainless steel cylinder having a conical tip and screwing into the handle in the same fashion as a fuel capsule. Pin diameter is 0.485 in. Three grooves in the pin allow passage through the wrench which unscrews the handle.

Specifications for the plutonium alloy fuel are listed in Table 2.1. Corrosion of tantalum by this fuel has been investigated using rocking bomb furnaces with a temperature differential of 150°C over several temperature ranges. With a high temperature of 700°C, corrosion is less than 1 mil/yr. Tests show that a trace amount of carbon in the fuel inhibits solution attack of the tantalum, but that a very low level of interstitial impurities in the metal is essential to prevent intergranular attack. The heat-affected zone of a fusion weld is more subject

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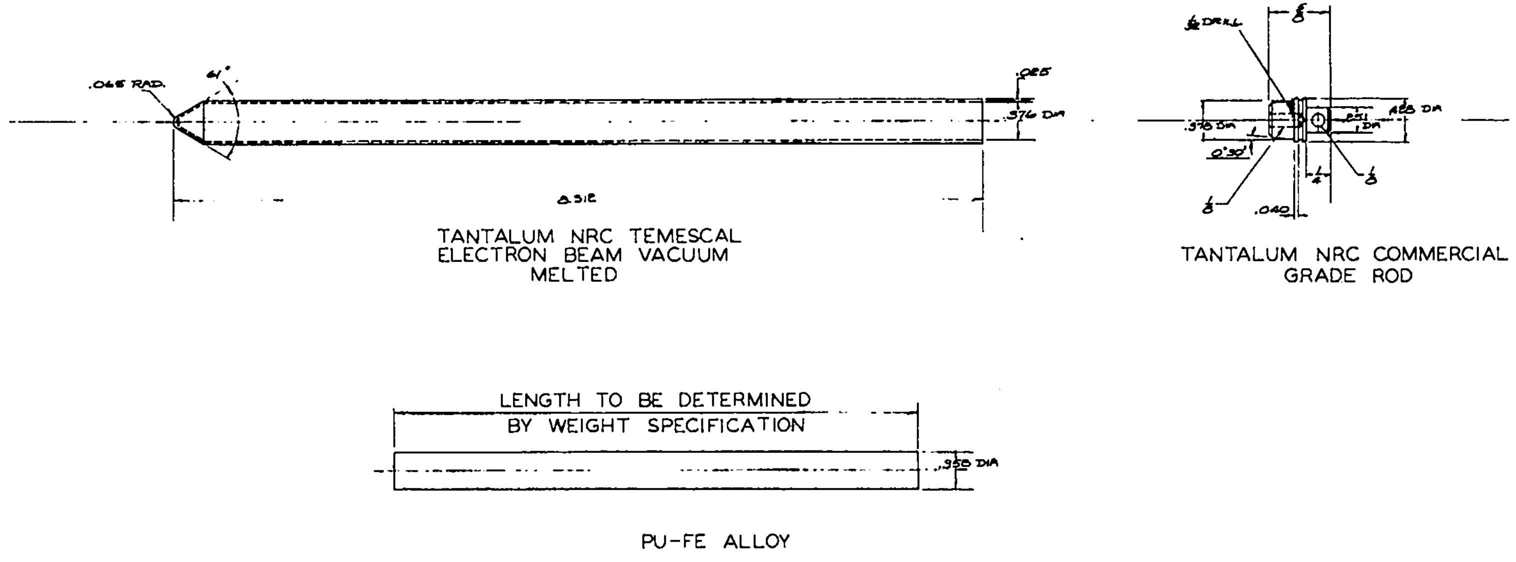

Fig. 2.10 Fuel capsule.

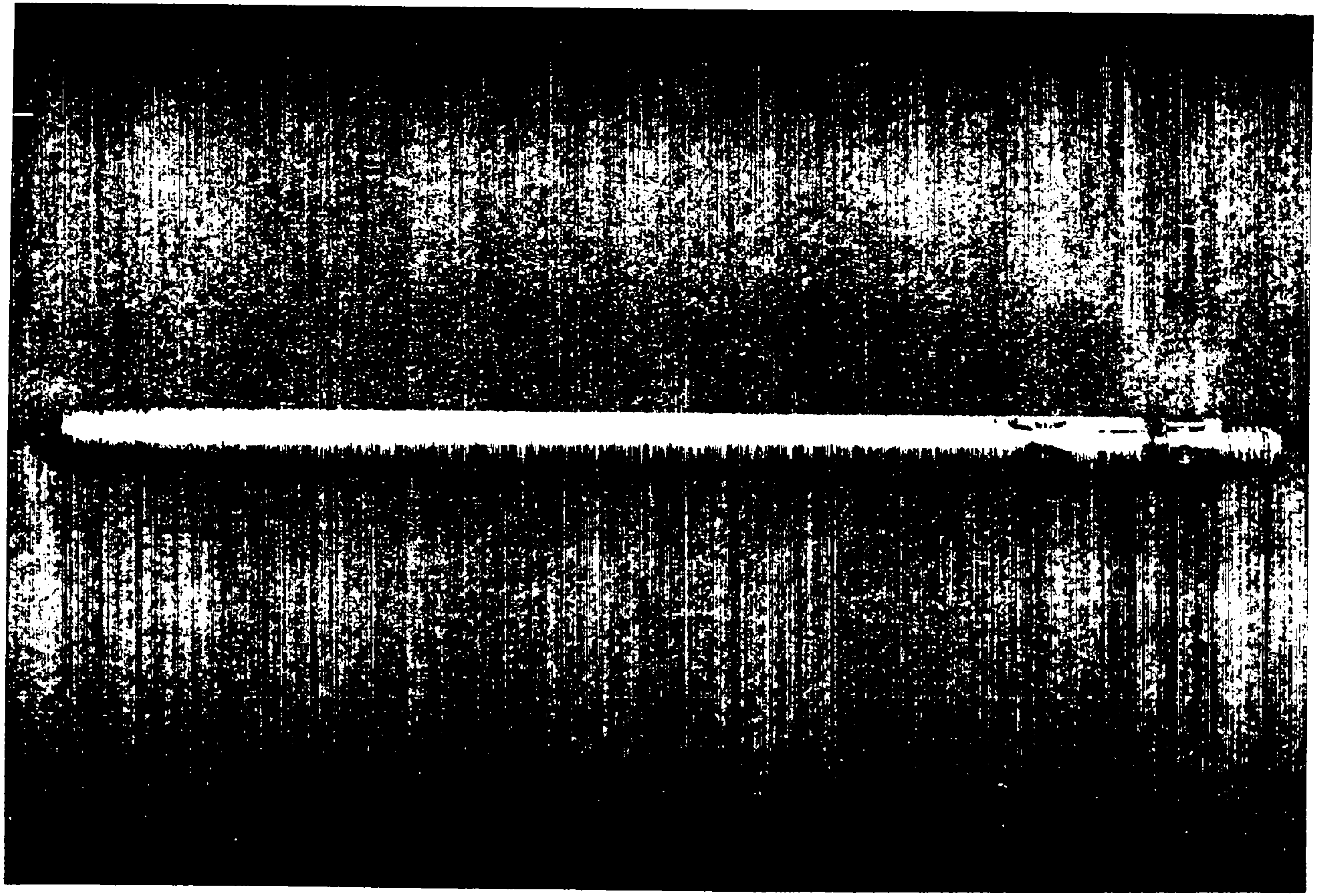

Fig. 2.11 Fuel capsule.

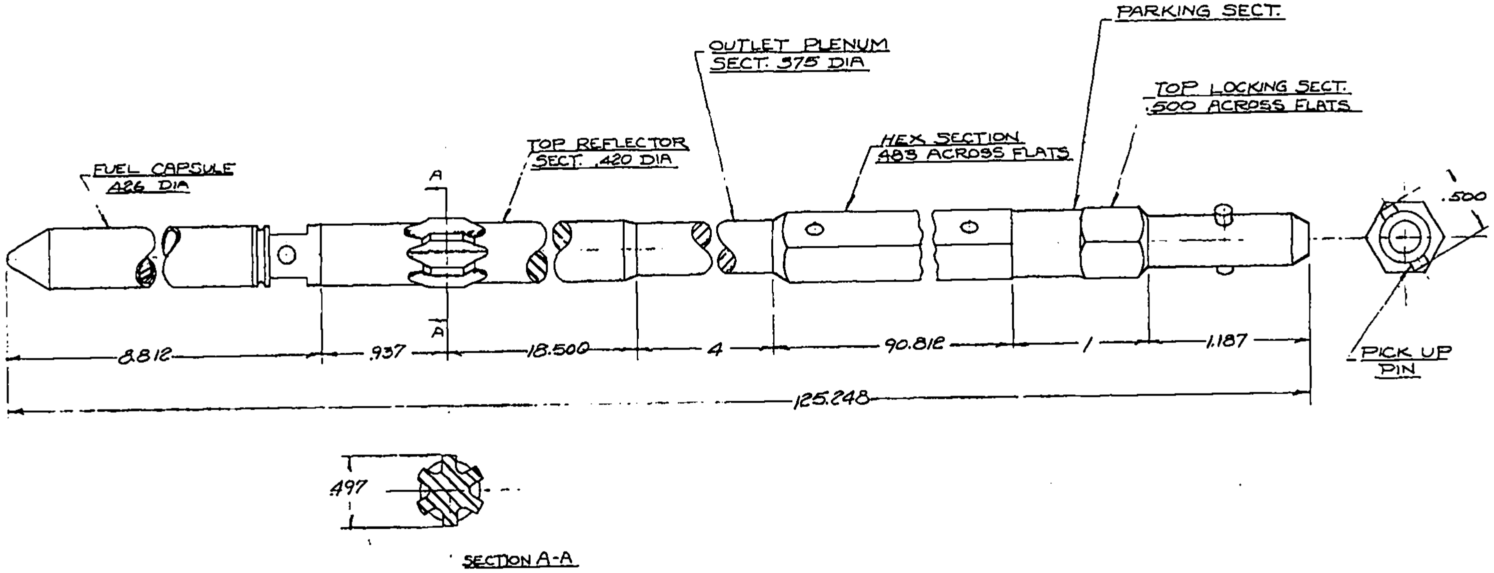

Fig. 2.12 Fuel element.

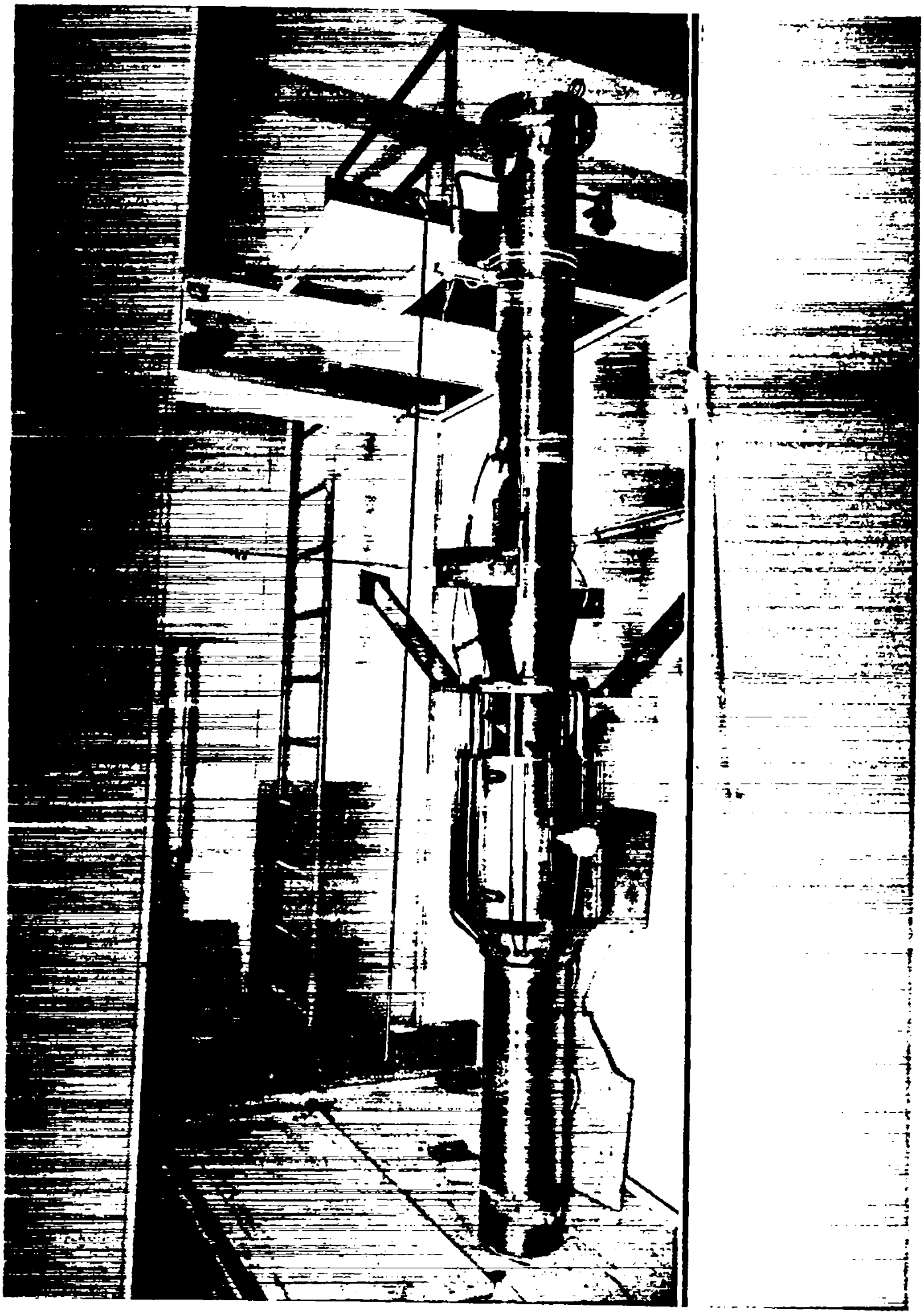

Fig. 2.13 Fuel element suspended beside vessel.

Table 2.1

FUEL SPECIFICATIONS

a. This specification defines the fuel composition, impurity limits, and analytical results for acceptance of LAMPRE I fuel slugs.

b. A melt analysis will be accomplished and submitted to K Division before final machining of the fuel slug. The following defines the impurity limits of certain elements.

Element	Concentration (ppm)
O	< 50
C	> 4 x the O_2 content
P	Reported

c. The iron content will be within 9.5 to 10.5 a/o based on 100% plutonium-iron material.

d. The source of the iron will be cast iron powder as was used in melts 3280 and 3466.

e. Fuel production techniques are to be similar to those of melts 3280 and 3466.

f. The Pu^{240} content will be reported on the basis of the production history of the plutonium used.

g. The weight of each fuel slug will be 175 ± 2 g.

h. As a result of the preceding fuel requirements a typical chemical analysis on an acceptable fuel lot should be as follows:

Element	Concentration (ppm)	Element	Concentration (ppm)
Li	< 0.2	La	< 10
Be	< 0.2	Si	400-800
Na	< 10	Pb	< 20
Mg	< 40	Cu	< 10-50
Ca	< 5	Mn	100-500
Al	< 35	Sn	< 20
Bi	< 20	Cr	30-70
Ag	< 10	Ni	100-900
Zn	< 100	C	> 150
Co	< 10-50	O	40
B	< 5	H	< 15
F	< 5	N	< 20

i. Deviations from the above specifications will be reviewed by K Division and will be subject to waiver.

to intergranular attack than other regions; for this reason the capsule is deep drawn and the fusion weld at the cap is made in the gas region which is out of contact with fuel.

The plutonium-iron phase diagram is shown by Fig. 2.14. Thermal expansion properties of the fuel are given in Fig. 2.15. The thermal conductivity of the molten fuel alloy is about the same as that of stainless steel. The measured value is 0.20 ± 0.01 w/cm °C at 500°C.

2.4 Core Physics

The calculational methods employed in estimating some of the basic nuclear parameters of the LAMPRE I core are based upon the S_4 transport scheme for spherical systems.[1] Empirical methods were used to convert spherical system results to the corresponding values for cylindrical systems. The values of some LAMPRE I parameters were experimentally determined from studies made with a critical assembly. The program carried out with the assembly designated LAMPRE Critical Experiment Number Two, or LCX II, is described briefly in Appendix A.

The following steps were involved in the calculations and conversions.

1. Spherical S_4 core calculations were performed for constant, specified thicknesses of reflector materials, using, in turn, the reflector thicknesses taken from LAMPRE design in three different directions: top, side, and bottom.

2. From the above calculations, the infinite media buckling and reflector savings were obtained for the spherical systems. The material buckling for the cylinder is computed from the average of the spherical system bucklings (reduced by 2% to account for inherent systematic errors). To obtain the reflector savings for the cylindrical system, the spherical reflector savings were multiplied by empirical factors 0.866 for the ends of the cylinder and 0.937 for the side.

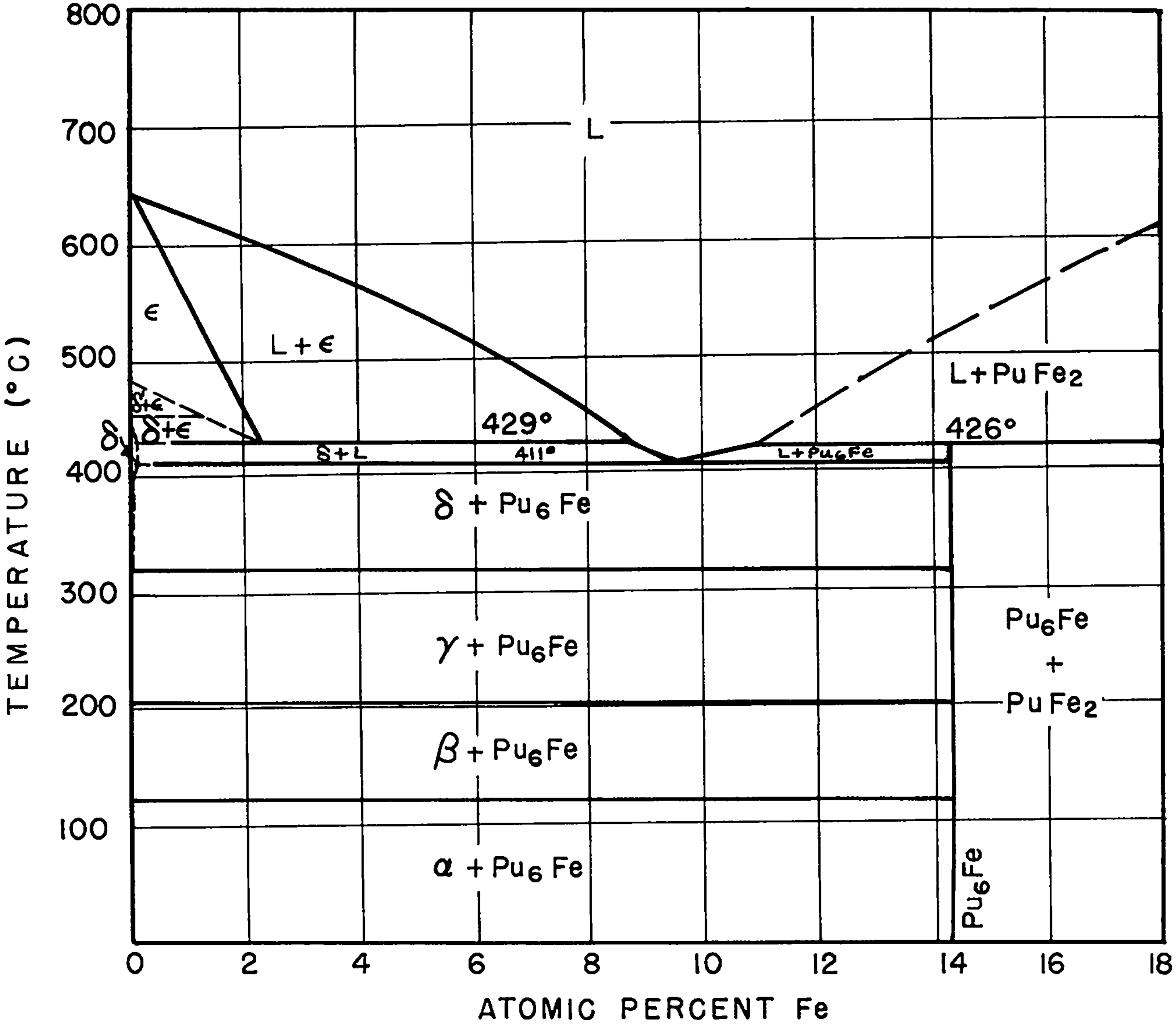

Fig. 2.14 Pu-Fe phase diagram.

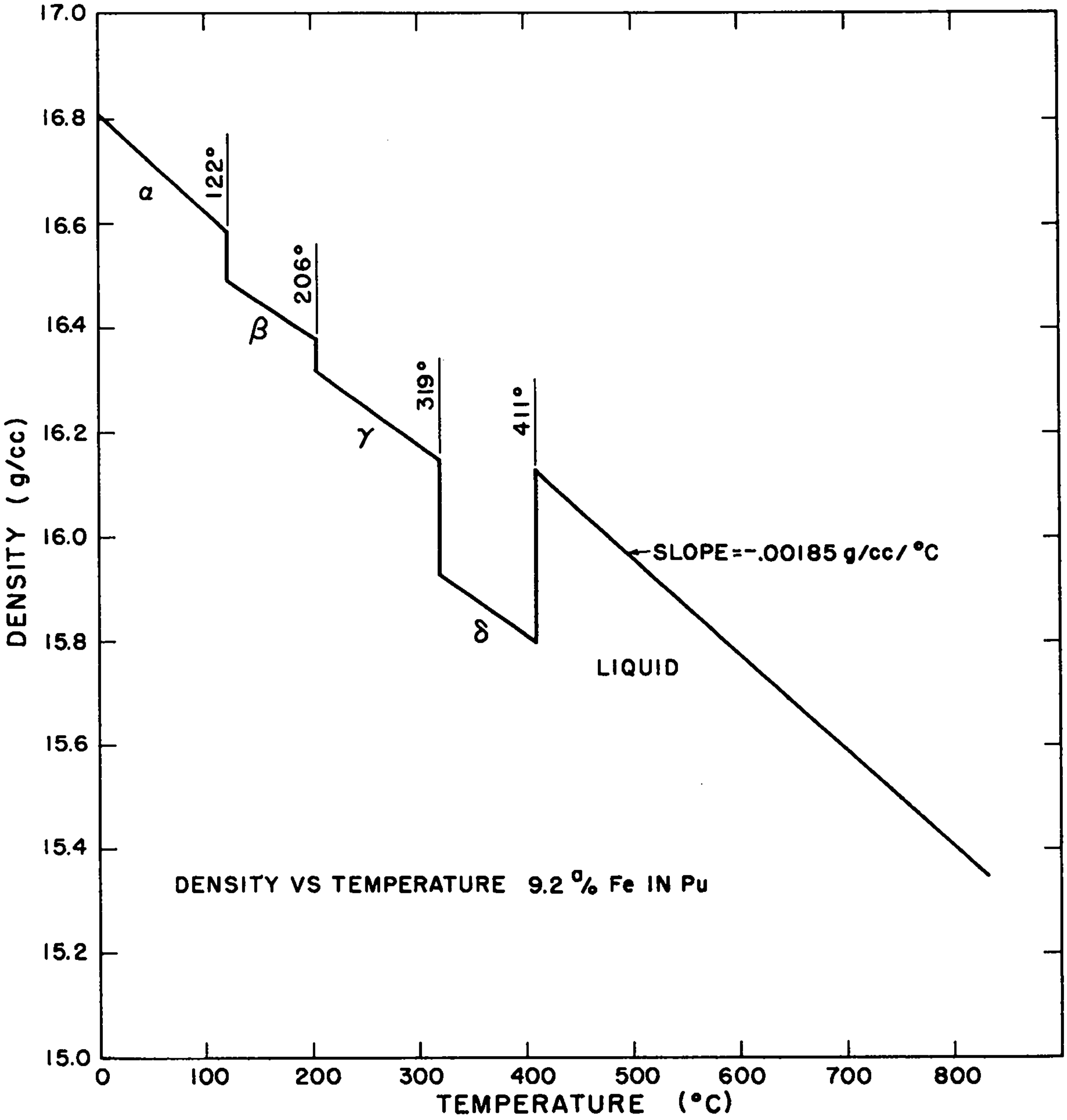

Fig. 2.15 Pu-Fe fuel density vs temperature.

3. The cylindrical mass was then computed from the formula

$$B^2 = \left(\frac{\pi}{H + \delta_T + \delta_B}\right)^2 + \left(\frac{2.405}{R + \delta_r}\right)^2$$

The core volume fractions used in the computations were:

Fuel alloy	0.5001 ($\rho = 16.051$ g/cm^3 at 550°C)
Tantalum	0.15403
Sodium	0.34587

Slightly different constants will apply to the as-built core.

The thicknesses and compositions of the side and end reflectors used in the computations are those listed in Table 2.2. For each of the three cases in the table, core composition remained the same. The results of these calculations indicate that for the hot (1 Mw) LAMPRE I configuration the important nuclear parameters have the following values:

Core buckling	$B^2 = 0.048549$ cm^{-2}
Side reflector savings	$\delta_r = 5.374$ cm
Top reflector savings	$\delta_T = 4.312$ cm
Bottom reflector savings	$\delta_B = 5.080$ cm
Cylindrical radius	$R = 7.758$ cm
Cylindrical height	$H = 16.249$ cm
Fuel alloy mass	$m = 24.99$ kg alloy
Core volume	$V_c = 3.06$ liters
K_∞ core	$(\nu\Sigma_f)/(\Sigma_a) = 2.62$
K_{eff}	Adjusted to 1.00 by shim
Central median fission energy	$\overline{E}_f \sim 1$ Mev
Prompt neutron lifetime	$\ell^* = \nu\Sigma_f\overline{V} = 8.9 \times 10^{-9}$ sec

The system temperature coefficients were determined by a series of mass calculations in which the material densities were varied in a suitable manner. The expansion coefficients used in obtaining the temperature coefficients were:

Table 2.2

MATERIALS AND THEIR THICKNESSES USED FOR S_4 CALCULATIONS

Material or Region	Thickness (cm)	Composition
	SIDE REFLECTOR	
Reflector pins	1.27	0.34587 Na, 0.65413 S.S. (430)
Flow divider	0.635	S.S. (304)
Na (inlet flow region)	0.9525	Na
Inner containment vessel	0.635	S.S. (304)
Heaters	0.9525	0.15 S.S. (304)
Outer containment vessel	0.635	S.S. (304)
Air gap	0.3175	Air
Shim	11.7475	0.98 S.S. (430)
Air gap	0.15875	Air
Circular flue wall	0.635	S.S. (430)
Air gap	1.27	Air
Hexagonal flue wall	0.635	Carbon steel
Shield	50.8	0.9108 B^{10} + C (0.4 w/o boron)
	TOP REFLECTOR	
Gas space in fuel capsules	5.08	0.15403 Ta, 0.34587 Na, 0.5001 air
Capsule fuel capsule plugs	1.27	Ta
Capsule handles	50.8	0.61518 S.S. (430), 0.38482 Na
	BOTTOM REFLECTOR	
Locator plate facing	0.508	Ta
Locator plate	4.572	0.23795 Na, 0.76205 S.S. (304)
Locator plate plenum	1.905	Na
Bottom reflector	12.70	0.1422 Na, 0.8578 Armco Fe
Turn-around plenum	1.905	Na
Catchpot	15.24	0.47 Na, 0.53 Armco Fe
Inner containment vessel	1.905	S.S. (304)
Air gap	6.35	Air
Bottom vessel shield	11.43	S.S. (304)

$$\left(\frac{\Delta\rho}{\rho}\right)_{\text{fuel}} = -89 \times 10^{-6}/^{\circ}\text{C}$$

$$\left(\frac{\Delta\rho}{\rho}\right)_{\text{Ta}} = -19.5 \times 10^{-6}/^{\circ}\text{C}$$

$$\left(\frac{\Delta\rho}{\rho}\right)_{\text{S.S. (304)}} = -36 \times 10^{-6}/^{\circ}\text{C}$$

$$\left(\frac{\Delta\rho}{\rho}\right)_{\text{Na}} = -300 \times 10^{-6}/^{\circ}\text{C}$$

The results of the computations are expressed in terms of changes in critical mass of fuel alloy per unit change in the temperature of the appropriate material. These derived constants are listed below.

Change in critical mass due to fuel temperature variation in an otherwise isothermal system:

$$\left(\frac{\partial m}{\partial \theta}\right)_{\text{fuel (prompt)}} = -2.72 \text{ g}/^{\circ}\text{C}$$

This may also be expressed as a function of fuel density:

$$\left(\frac{\partial m}{\partial \rho}\right)_{\text{fuel}} = -1.91 \times 10^{3} \text{ g}/(\text{g}/\text{cm}^{3})$$

Change in critical mass due to variation in the temperature (isothermal distribution) of all core materials:

$$\left(\frac{\partial m}{\partial \theta}\right)_{\text{all core materials}} = -4.27 \text{ g}/^{\circ}\text{C}$$

Critical mass change associated with a space independent, sodium-coolant temperature variation in an otherwise isothermal system:

$$\left(\frac{\partial m}{\partial \theta}\right)_{\text{Na}} = -0.80 \text{ g}/^{\circ}\text{C}$$

If the locating splines on the fuel pin assemblies (see Fig. 2.12) are assumed to be in contact, there is an additional temperature coefficient. This is produced by the outward displacement of fuel pins as the temperature increases. The displacement is different at the top and the bottom of the core, however, since the bottom ends of the fuel pins are spaced by the tantalum locator plate, which expands at a rate different from the stainless steel parts (splines) which position the fuel elements at the top end. An average expansion coefficient

$$\Delta \ell / \ell = 9.5 \times 10^{-6} / {}^{\circ}\mathrm{C}$$

was assumed to be applicable to the effective radial expansion of the core. The temperature coefficient of reactivity produced by the resulting variation in average fuel density was found to be, for an isothermal temperature distribution,

$$\left(\frac{\partial m}{\partial \theta} \right)_{\text{location geometry}} = -0.57 \ \mathrm{g}/{}^{\circ}\mathrm{C}$$

These estimated mass relations may be converted to temperature coefficients of reactivity by using the experimentally determined value (see Appendix A).

$$\frac{\Delta k}{\Delta m} = 1.18 \ ¢/\mathrm{g\ fuel}$$

In summary, the zero power, isothermal reactor system is characterized by these coefficients:

$$\left(\frac{\partial k}{\partial \theta} \right)_{\text{fuel (prompt)}} = -3.21 \ ¢/{}^{\circ}\mathrm{C}$$

$$\left(\frac{\partial k}{\partial \vartheta} \right)_{\text{Na}} = -0.94 \ ¢/{}^{\circ}\mathrm{C}$$

$$\left(\frac{\partial k}{\partial \vartheta} \right)_{\text{location geometry}} = -0.67 \ ¢/{}^{\circ}\mathrm{C}$$

$$\left(\frac{\partial k}{\partial \theta} \right)_{\text{reflector}} = -0.88 \ ¢/{}^{\circ}\mathrm{C}$$

The power coefficient of the system may be estimated from the above reactivity coefficients. It is necessary only to sum the effects of the listed coefficients, taking proper account of the fact that the temperatures of the fuel, the coolant, and the reflector structure increase by different amounts as operating power is raised. (For instance, it is calculated that for full coolant flow the average fuel temperature increases about three times as fast as the average sodium temperature when power demand is increased.) The total power coefficient is, on this basis, found to be

$$\left(\frac{\partial k}{\partial P}\right) \cong -0.69\ \text{¢/kw}$$

or

$$\left(\frac{\partial k}{\partial \vartheta}\right) \cong -12.3\ \text{¢}/{}^{o}\text{C rise in Na temperature}$$

Approximately three-fourths of the power coefficient results from the temperature rise in the fuel and is therefore a prompt effect. If there is a significant amount of convective circulation of fuel in the capsules, the average temperature of the fuel will be lower (relative to the coolant temperature at a given power), and the absolute value of the power coefficient would be smaller than the estimated value given here. Mechanical clearances in the core structure, not considered in the above power coefficient estimates, could produce a similar effect through their relation to core distortion (capsule bowing) as a function of power.

Since the LAMPRE I fuel is molten under operating conditions, a radial temperature gradient across the core does not produce strong bowing effects in the fuel capsules. Some bowing may occur, however, since the radial temperature variation will tend to introduce bowing forces in the tantalum fuel capsules themselves, and in the fuel capsule handles. The amount of bowing which can take place is a complicated function of assembly clearance on the fuel element assemblies and of the actual temperature distribution in the reactor. These variables, and their effects on the over-all power coefficient, are considered in Appendix B which may be summarized briefly as follows:

1. In the bowing calculation it was assumed that assembly clearances between fuel capsule handles are uniformly distributed, and that initially there is no contact between adjacent fuel capsules at the spline sections of the handles. If, as will almost certainly be the case, a few per cent of the fuel capsules are initially constrained by contact between adjacent spline sections, the effect of bowing is markedly reduced.

2. With the uniform clearance assumption, both capsule and handle bowing make positive contributions to the power coefficient up to about 10% of full power. At higher power levels only capsule bowing persists and it produces only a small reactivity effect.

3. On the basis of the pessimistic distribution of mechanical clearances assumed, the over-all power coefficient, including the bowing contribution, is calculated to be positive up to a power level of 100 kw, with a value of about 0.025 ¢/kw. Above the 100 kw power level, bowing is of little consequence, and the over-all power coefficient becomes approximately -0.7 ¢/kw.

4. The prompt component of the power coefficient is negative over the entire power range, with a value of about -0.55 ¢/kw.

There is one other recognized source of a small positive component of the over-all power coefficient. Relative motion between the reactor core and the control elements occurs as the average sodium temperature changes with variation in power. The motion is such as to increase reactivity as power, and hence the average sodium temperature, increases. A design feature which has to a large measure eliminated this effect is described in Section 2.7. It is estimated, however, that a positive component of the power coefficient of ≤ 0.015 ¢/kw exists because of this thermally induced, relative motion between the reactor core and the control elements.

Calculations indicate that with the reactor at operating temperature, but at low power, withdrawing the shim and control rods reduces reactivity by an amount equivalent to about 6 kg of fuel, or by roughly $70. Although the relative effectiveness of the control rods and the control shim is not known precisely for LAMPRE I, it appears that about $60 of this shutdown control is produced by the shim.

It has also been estimated that if the reactor is shut down from full power operating conditions by withdrawing the shim and the control rods, the reactor will be subcritical at room temperature by an amount equivalent to 2.8 kg of fuel, or by ~ $33.

The dry (no sodium) cold critical loading of the reactor is calculated to be 25.48 kg of fuel with the rods and shim inserted. The corresponding value with all control elements withdrawn is 31.68 kg.

Reactivity worth of the sodium coolant (including that in the inlet and outlet plenums) is listed below, expressed in terms of the equivalent change in critical mass:

Na worth at 20°C, shim and rods in $\cong$ 3.1 kg
Na worth at 20°C, shim and rods out $\cong$ 3.9 kg
Na worth at 550°C, shim and rods in $\cong$ 2.6 kg
Na worth at 550°C, shim and rods out $\cong$ 4.1 kg

2.5 Operating Conditions

2.5.1 General

LAMPRE I start-up and approach to operation at full power will be carried out in four major steps. First, a set of cold critical experiments will serve to determine critical mass and control element worth for comparison with calculated estimates based on LCX II observations. During this period no sodium coolant will be present in the reactor, and the entire system will be at room temperature. Second, after the conclusion of the cold critical measurements, all the fuel will be removed from the reactor, and the fuel capsules will be replaced by stainless steel dummies; the reactor system will then be filled with sodium and operated at an appropriate temperature, and for a suitable length of time, to clean up all system components. Third, the dummy elements will be replaced one by one with fueled elements, and a determination of the hot, wet critical mass and the control element worth will be made at a reactor system temperature in the range of 450 to 500°C, and at "zero" nuclear power. The fuel in each capsule will melt as it is inserted into the core. The value of the negative temperature coefficient will also be measured under these conditions. Finally, the reactor power will be raised stepwise, and at each power level reactor stability will be investigated by a series of reactor oscillator experiments.

Table 2.3 gives values of some core parameters.

2.5.2 Reactor Start-up and Approach to Full Power Operation

Cold Critical. The design of the mechanized fuel element handling system, or capsule charger, is such that it cannot load fuel into an empty reactor core. Therefore, an initial loading of dummy elements must be made manually before the capsule charger is attached to the reactor during the final stages of system assembly. At the start of the cold critical measurements, the reactor will therefore be completely filled with dummy fuel elements and reflector pins. Dummy capsules will be replaced manually, one at a time, by fuel capsules until a multiplication of 10 (with shim and rods down) is achieved. At this point the capsule charger will be installed and further replacement will be accomplished using the charger. Rules governing personnel access to reactor areas will be applied during the cold critical phase of operations. Protective instrumentation and interlocks described in Section 2.5.3 will be connected if they are related to nuclear safety considerations. Start-up detectors will consist of at least three B^{10}-lined proportional counters. A start-up neutron source of sufficient intensity is provided by Pu^{240} in the fuel itself.

The actual charger manipulations involved in replacing the dummy fuel elements with fuel-filled capsules are as follows: The vertical lift mechanism is lowered to engage a fuel element handle; the handle with its dummy element is lifted into the charger housing and the element then unscrewed from the handle. The dummy element is transferred out of the housing through a port, and a real fuel element inserted into the housing and attached to the handle by the reverse process. The element is then lowered into the reactor vessel and is positioned in the space formerly occupied by the dummy.

Table 2.3

VALUES OF CORE PARAMETERS

Capsule material	National Research Corporation high-purity Ta refined by Temescal electron beam melting
Capsule size	0.376 in. i.d. x 0.025 in. wall
Core capacity	199 capsules in hexagonal array
Number of capsules for criticality	143 (calculated)
Capsule spacing	0.497 in. triangular pitch
v/o fuel	51.5
v/o Na	33.5
v/o Ta	15.0
Fuel height	6 in.
Fission gas volume height	2 in.
Design power	1 Mw
Average fuel temperature	637°C
Maximum fuel temperature	870°C
Na inlet temperature	450°C
Na outlet temperature	563°C
Na flow rate	133 gpm
Central-to-edge power ratio	1.8
Axial power ratio	1.8

	Average	Maximum	Minimum
Specific power (w/g)	40	61	19
Heat flux (w/cm^2)	145	220	68
(Btu/hr/ft^2)	460,000	700,000	214,000
Na outlet temperature (°C)	563	597	531
ΔT in fuel (°C)	200	307	93
ΔT in Ta (°C)	17	24	8
Ta thermal stress (psi)	2800	4300	1300

All fuel additions are to be made with the control elements (shim and rods) in their least reactive, or down, position. When about two-thirds of the estimated 25.48 kg critical mass has been loaded, the shim and rods will be inserted, and the neutron level observed. This procedure will be repeated at appropriate intervals during the rest of the fuel loading operation, until sufficient data are available to estimate satisfactorily the actual critical mass with shim and rods fully inserted. The final approach to delayed critical will be made by control rod manipulation.

Critical mass at partial shim insertions may be determined in an exactly similar fashion after making the appropriate adjustments to the mechanical stop which limits available range of shim travel. An extrapolation of the data to the shim-down case will determine the worth of the shim; for the cold critical case, its value has been estimated to be equivalent to 6.2 kg of fuel, or about 39 elements.

Fuel inventory records kept during these, and subsequent, loading procedures will list the identifying number of each element inserted into the reactor, together with its position in the core. The total loaded fuel inventory will be physically checked against the number of dummy elements removed from the core during loading operations.

Sodium System Cleanup. When the cold critical experiments have been finished, the fuel will be removed and replaced by the dummy capsules, one by one. The reactor system by this time will be ready to charge with sodium. Test operation, with electrical heating to about 500°C, will then begin, and will continue until satisfactory performance of all coolant loop components has been demonstrated, and until the desired degree of sodium purity has been obtained through the use of cold and hot traps. The fuel capsule charger will be tested while the reactor system is at operating temperature, and the various sodium-system malfunction detectors checked out, insofar as possible, under actual operating conditions. Satisfactory mechanical functioning of the reactor control elements at elevated temperature will also be ensured.

Zero Power Hot Critical. When the coolant system cleanup and checkout program has been completed, the core will be loaded with fuel and the zero power, hot critical mass determined. During this operation, the reactor system will be held at a temperature of 450 to 500°C by the use of electrical heaters. Zero power, or isothermal, operation above 500°C is not advisable, since the sodium pumps (in the reactor inlet leg of the coolant loop) are not designed for temperatures in excess of 550°C. There will be no air flow through the sodium-to-air heat exchanger but, in other respects, the coolant system will be fully operative.

The insertion of fuel into the core will be accomplished in the same fashion as in the cold critical phase; i.e., a dummy capsule will be removed by the capsule charger, and a fuel capsule put in its place. It is expected that the fuel in a capsule will become fully molten during the time interval between its insertion and the loading of the succeeding fuel element.

A shim in, all rods in, critical configuration will first be determined. It will then be desirable to increase the fuel loading, one capsule at a time, to obtain a preliminary calibration of rod worth. A similar preliminary value of the temperature coefficient of reactivity will then be obtained by varying the electrical power input to the system and noting the configuration changes necessary to maintain criticality. Finally, critical mass as a function of shim insertion will be determined in the same fashion as during the cold critical tests, and intercalibration of the shim and the control rods will be carried out.

The final fuel loading will be adjusted to such a value that, at full power, a critical configuration should be reached with the shim and two rods fully inserted, and the third rod partially inserted.

Approach to Full Power. The following basic plan will be followed in achieving full power operation. With the shim fully inserted (corresponding to a slightly subcritical configuration at the 450°C, isothermal condition), the reactor will be made critical and coolant outlet temperature will be raised in small increments by inserting rods. A constant coolant inlet temperature will be maintained by decreasing the electrical heating power as nuclear power is increased.

The present estimate of over-all power coefficient for LAMPRE I is 0.7 ¢/kw. Therefore, $7 of reactivity control must be available in the rods to reach the nominal 1 Mw design power. The estimated value of the total worth of three control rods is very approximately $10. It appears, therefore, that sufficient reactivity control is available to reach design power without shim adjustment. However, the power coefficient depends strongly on the actual value of parameters such as the thermal conductivity of the fuel, which is not accurately known at present; also, the power coefficient of the reactor will be influenced by the amount of natural convection of the molten fuel within the capsules.

2.5.3 Protective Instrumentation

General Description of Safety Circuits. The safety instrumentation of the LAMPRE I system may be grouped into four classes: scram channels, rundown channels, annunciator channels, and interlocks.

Scram channel signals will produce reactor shutdown by causing the control shim to drop to its least reactive, or down, position. Rundown channel signals initiate automatic rod withdrawal with an attendant reduction in reactor power. Annunciator channel signals operate visible and audible alarms which indicate to operating personnel the existence of abnormal values of system parameters; corrective action must be initiated by the reactor operator. Interlocks provide sequencing of certain operations affecting reactor criticality; they also regulate coolant system start-up procedures and provide for ventilation system control in the event of fission product and sodium smoke release in certain areas.

Scram Channel Signals. The sources of scram signals are:

a. Reactor period less than preselected value

b. Neutron level above preselected value (two independent channels)

c. Reduction of coolant flow to ~1% of full flow

d. Excessive temperature indication from thermocouple located in heat exchanger air outlet duct

e. Slide valve at top of reactor vessel open

f. Manual scram switch

g. Electrical power failure in control element drive system

The period signal is obtained in a conventional manner from a log amplifier circuit containing appropriate differentiation. The amplifier input signal originates in a neutron-sensitive ion chamber. The differentiated amplifier output signal is fed into a fail-safe electrical switching circuit, i.e., a Schmidt trigger employing a normally open plate relay whose contacts are connected in series with the scram buss.

Neutron level signals are obtained from ion chambers feeding linear amplifiers which drive trigger circuits similar to those noted above. Coolant flow signals are obtained from a permanent magnet flowmeter whose output is recorded on a line drawing, strip chart recorder. A low-limit switch on the recorder triggers the scram.

A gross leak in the heat exchanger would result in a sodium fire which would raise the air outlet temperature well above normal operating values. A thermocouple will trip a scram channel when the temperature in this region is about 100°C above the maximum expected normal temperature.

It is considered undesirable to operate the reactor unless the slide valve which isolates the fuel charger from the reactor is closed. An electrical interlock will prevent control shim operation if this valve is open. The reactor operator may shut down the reactor at his discretion by means of the console scram switch. Electrical power failure in the shim-actuating hydraulic system will automatically produce a shim drop by de-energizing a normally open solenoid-operated valve.

Tests have indicated that 0.015 sec after the scram signal has reached the scram solenoid valve, the shim will have begun to drop. An additional delay of not more than 0.050 sec is introduced by the relay circuits associated with scram channels a and b above. Short delays in the transmission of scram signals from sources c, d, and e are of no consequence, since these scram demands do not originate as a consequence of the nuclear behavior of the reactor.

Rundown Channel Signals. Certain malfunctions, or variations in reactor system operating parameters, make it desirable to reduce the reactor power until the source of the trouble can be recognized, evaluated, and removed. The occurrences which have been selected to initiate automatically a reduction in reactor power are:

a. Reduction of sodium flow below a desired level
b. Reactor coolant outlet temperature higher than a preselected value
c. Stoppage of air flow through heat exchanger air ducts
d. Excessive radiation level in heat exchanger air outlet duct
e. Presence of smoke in heat exchanger air outlet duct, in the sodium equipment room, or in the reactor cell
f. Temperature of sodium at heat exchanger outlet plenum higher than a preselected value

Signals indicating that any of the above conditions exist will initiate automatic, sequenced withdrawal of control rods. The withdrawal will continue until all rods are out unless the operator has been able to reset the rundown channel circuits. He can do this only if the system has returned to a normal condition. Any additional corrective measures must be chosen and initiated by operating personnel.

Annunciator Channel Signals. The annunciator channels which warn operating personnel (via visible and audible signals) of low urgency, off-normal conditions in the reactor system are:

a. Leak in coolant piping system (signal comes from detection system which traces the piping); this method of leak detection is separate from the smoke detectors which can initiate a rundown

b. High gamma radiation levels at selected (seven) locations in reactor area

c. Sodium temperature at heat exchanger outlet below preselected value

d. Graphite shield temperature above preselected value

e. Leakage of sodium from inner reactor vessel

f. Door between control room and corridor to reactor cell open

g. Temperature of main loop heating transformer above preselected value

h. Poor vacuum in gas disposal system holding tank

i. Sodium level in surge tank too high

j. Sodium level in surge tank too low

k. Cover gas pressure too high

l. Cover gas pressure too low

m. Improper voltages on neutron level detectors

n. Improper voltages in remote-area gamma radiation monitoring system

o. Instrument air supply pressure too low

p. Pressure in shim drive hydraulic accumulator too low

q. Loss of cell blower power

Interlocks. Electrical interlocks in the LAMPRE I instrumentation and control systems automatically enforce certain restrictions on operational procedures. These interlocks may be grouped in the following fashion with functions as indicated.

1. Shim and control rod sequencing interlocks provide these five operational restrictions:

a. All rods must be down before shim can be raised

b. Shim must be up before any rod can be raised

c. Rods can be raised only one at a time and then only in a predetermined order

d. Only one rod (at a time) can be held in a position between up and down, and then only if the preceding rods in the sequence are up

e. Automatic rod rundown is initiated in case of a scram signal

2. A vessel slide valve interlock exists between the shim drive and the 10 in. slide valve which isolates the reactor vessel from the capsule charger. The shim cannot be up, i.e., reactor cannot be operating, unless the valve is closed. Therefore, the capsule charger cannot insert fuel into the core unless the reactor has been shut down to a scrammed configuration. This interlock is also listed in the scram channel tabulation (e), since opening the valve will produce a scram if the reactor is operating.

3. A cell ventilation system interlock assures that an abnormally high gamma radiation level in the reactor cell, assumed to be evidence of a reactor leak with subsequent escape of sodium and/or fission products, will stop the cell exhaust blower.

4. Heat exchanger blower control and sequencing interlocks are primarily to ensure that the heat exchanger does not cool the sodium to a temperature close to the fuel freezing point.

5. Smoke and gamma ray detector interlocks provide that the presence of smoke in the heat exchanger air outlet duct, in the sodium equipment room, or in the reactor cell, will cause the inlet and outlet vanes in the heat exchanger air ducts to close and the blower to be shut off. An abnormally high gamma ray level in the heat exchanger exhaust duct will also close the vanes and stop the blower. Note that these conditions also initiate a reactor rundown.

Neutron Detector Channels. The neutron-sensitive detectors which will be used during LAMPRE I start-up and operation are listed below and their ranges are shown in Fig. 2.16.

1. Two channels of neutron-sensitive (B^{10}-lined) pulse chambers will be utilized for normal operation. The operating range of one of the counters will overlap with one of the ionization chambers of 2 below. For start-up there will be two additional pulse-counting channels.

2. Two neutron-sensitive ionization chambers with dc amplifiers will be available; these will cover (with some overlapping) the power range from "high multiplication" to design power. Power indication from these chambers will be displayed on a linear curve drawing recorder.

3. A third neutron-sensitive ionization chamber will supply a power level signal to a logarithmic amplifier; the amplifier output will be recorded on a line drawing, strip chart recorder. The instrument range will extend from 10 to 10^{-3} times design power. The logarithmic amplifier signal will be used in a conventional circuit to provide the source of a period scram signal.

4. Two independent, neutron-sensitive ionization chambers will be used for level scrams. The level scram can be set at points ranging from full power to 10^{-2} of full power. Signals from these detectors will be displayed in the control room, but not recorded.

2.6 Coolant System

2.6.1 Components

The system which contains the sodium coolant is, in general, fabricated from Type 316 ELC stainless steel. Figure 2.17 is a schematic of the system and Figs. 2.18 and 2.19 show the coolant system layout. Components will be described individually.

Piping. Sodium piping is 2 and 3 in., Schedule 40, Type 316 ELC stainless steel. All weldments are made with an inert-arc root pass

(Text continued on page 53)