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

LAMPRE I FINAL DESIGN STATUS REPORT

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OF THE UNIVERSITY OF CALIFORNIA LOS ALAMOS NEW MEXICO

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LAMPRE I FINAL DESIGN STATUS REPORT*

by

K Division Personnel

*This report complements LA-2327.

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ABSTRACT

A description is given of LAMPRE I as it was built including preliminary operating experience of the components. This report is intended to supplement LA-2327, the hazard analysis report prepared in 1959 in advance of construction.

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1. INTRODUCTION

LAMPRE I is the first reactor built by the Los Alamos Scientific Laboratory in its continuing program of developing plutonium fuels for fast breeder reactor applications. The initial conceptual form of LAMPRE I called for a design wherein the plutonium fuel was to be contained in a single connected region and cooled by sodium flowing through tubes welded to the top and bottom plates of a cylindrical container. When detailed design of the core calandria was started, it became apparent that knowledge of container and fuel materials was such that a design could not be prepared in a time compatible with that proposed for LAMPRE I.

Accordingly the objectives, and hence the design of the first reactor, were changed radically to serve as a radiation test reactor to obtain the necessary material experience that will permit design and fabrication of future reactors.

The power of the reactor was determined by the decision to locate the equipment in the available space of the existing building. At 1 Mw, the specific power is relatively low (~ 40 w/g of Pu) but is high enough to provide initial experience on the effects of fission product accumulations in the fuel. The flexible design of separate capsules for fuel containment allows the testing of variations in fuel and in capsule material.

In the description that follows an attempt has been made to incorporate all the changes and modifications that have been made up to time of the start of power operation. As with any experimental machine, it is expected that additional changes will be made as the desirability or necessity is indicated by operating experience. Rather than to attempt continuing amendments to this description of the reactor, subsequent modifications in the design and operation of the reactor will be recorded in reactor manuals.

2. CORE REGION

2.1 General

The core of the reactor is an array of tantalum capsules containing Pu-Fe alloy fuel. Of the 199 core locations, approximately 140 central ones are occupied by these fuel capsules, and the remaining locations are occupied by stainless steel reflector pins. Vertical and horizontal cross sections of the core region are shown in Fig. 1 and 2.

Coolant sodium enters the vessel and flows down through the 3/8 in. annulus formed between the vessel and a flow divider. At the bottom of the flow divider is a turn-around plenum where the coolant stream reverses to flow up through the bottom reflector, into another plenum, and then through a locator plate assembly into the core, where, after flowing past the fuel capsules and reflector pins, it flows through the top reflector region and into the outlet plenum. From this point it flows out of the vessel.

A catchpot for containment of fuel in the event of a capsule rupture is located immediately below the turn-around plenum.

2.2 Component Description

2.2.1 Locator Plate Assembly. The locator plate assembly, shown in Fig. 3, supports and positions the fuel elements and is supported in the vessel by a shoulder around the inside diameter of the flow divider described in Section 3.2.

Fig. 1. Vertical Cross Section of Core Region

Fig 2. Horizontal Cross Section of Core Region

Fig 3. Locator Plate Assembly

The assembly consists of two parts: 1) the 0.250 in. thick locator plate made of NRC commercial grade tantalum, and 2) the 2 in. thick core support plate made of Type 304 stainless steel. Tantalum was specified for the locator plate to prevent soldering of fuel elements to the assembly in the event of fuel leakage. The two parts were machined simultaneously to obtain the best possible alignment. The assembly is approximately 7-1/2 in. across the flats, is hexagonal in shape, and has an irregular periphery that conforms to the contour of the internal surface of the flow divider. The two parts are considered to be an integral unit although the only attachment used is two tubes through sodium flow holes to prevent relative angular displacement. Vertical displacement is prevented by the weight of the fuel elements upon the assembly.

At the bottom of the core, orientation and restraint of the fuel elements is done by locating the elements in 199 conical bores, or core locations, in the locator plate. These bores are 0.256 in. across corners with a nominal angle of 59 degrees, 52 min; the bottom of the bore is relieved with a 0.140, 118 degree drill. To allow for coolant flow through the assembly 348 holes, each with a 0.234 in. diam, were drilled through both parts. At the top of the core, orientation and restraint of the fuel elements is provided by the spline sections of the handles directly above the core (see Section 2.3.2, Capsule Handles).

2.2.2 Reflectors. Core neutron reflection is accomplished, to a greater or lesser degree, in three regions; top, bottom, and side. Top reflection is small; no material is added for reflection. The following components, described in Section 2.3, act as a top reflector, the fuel capsule wall tops (there is a 2 in. void in the capsule above the fuel), the capsule closure plug, and the lower end of the capsule handle.

Bottom reflection is furnished by the locator plate assembly and an Armco iron bottom reflector shown in Fig. 4. It is a cylinder of 6-7/8 in. diam by 6 in. high and is supported by a shoulder turned on the inside diameter of the flow divider. The thirty-two 1/2 in. diam holes drilled through this component for coolant passage represent 17% of its total cross sectional area. A threaded handling stud in the top face is used for assembly or removal.

Fig. 4. Bottom Reflector

Side reflection is accomplished by reflector pins in core locations not occupied by fuel capsules, the flow divider, the vessel walls, and the movable shim described in Section 9.1. The mass of the internal, fixed side reflector is minimized to have a maximum amount of reactivity control in the shim.

2.2.3 Catchpot and Diluent Plug. The purpose of the catchpot and diluent plug, shown in Fig. 5, is to contain in a noncritical geometry any fuel that leaks from the core. The catchpot is 8.64 in. i.d. by 6 in. high and is made from 0.040 in. tantalum sheet. The diluent plug is made from Armco iron and is 8.62 in. o.d. by 5-15/16 in. high. Holes drilled in the plug provide the dispersion for a non-critical geometry. The holes are drilled from the top to within 1/8 in. of the bottom so that molten fuel will not flow to the bottom of the catchpot and cause the plug to float.

In addition to dispersing fuel, the diluent plug will dissolve in molten fuel to form an alloy with a higher melting point. As solution continues, the alloy gradually becomes a solid.

A detail in the wall of the catchpot accommodates a 3/8 in. tube that may be used for siphoning the coolant from the vessel (see Section 3.1).

2.3 Fuel Elements

2.3.1 Capsule Assembly. The components of the capsule assembly are the thimble, fuel slug, closure plug, and adaptor. Figure 6 shows the four components separately and Fig. 7 shows an assembled capsule.

A standard thimble for Core I was fabricated from an alloy containing tantalum and 0.1 w/o tungsten. Its overall length is 8.312 in., its nominal outside diameter is 0.429 in., and its cone is a nominal 61 degree. The inside diameter is tapered because of fabrication considerations and is a nominal 0.377 in. at the top and 0.362 in. at the cone end. The hemispheric internal configuration at the cone end projects slightly into the cone.

Fig 5. Diluent Plug (Left) and Catchpot (Right)

Fig. 6. Fuel Capsule Components

Fig. 7. Assembled Fuel Capsule

The fuel slug, machined from cast rod, has a nominal 0.358 in. diam, and its length, determined by the fuel weight specification 175 ± 2 g Pu-Fe alloy), averages 6.33 in.

The closure plug is machined and ground to close tolerances from NRC commercial grade tantalum rod.

The adaptor is fabricated from Type 304 stainless steel rod and is attached to the closure plug with a 1/8 in. , Type 304 stainless steel dowel pin. A 5/16-18 thread on the adaptor screws into the capsule handle. Five-sixteenth inch wrench flats below the thread are used in loading and unloading operations described in Section 6.5.

The fuel slug is machined immediately before assembly into the thimble, the operation done in the ambient atmosphere. The slug, thimble, and matched closure plug are assembled in a collet, which is installed into a rotating spindle mounted in a plexiglass welding can. A total of three helium flushes are used with a pump-down to 100 μ pressure between flushes. The welding atmosphere is helium with an argon stream over the weld, and because gas impurities are detrimental to the weld, only selected lots of helium and argon are used in this operation.

After completion of the welding operation, the assembly is monitored for external alpha contamination and the adaptor is installed.

Before acceptance, the assembly is radiographically inspected, a helium leak check is made, and the adaptor thread and assembly are visually inspected.

2.3.2 Capsule Handles. Capsule handles are fabricated from Armco 17-4 PH stainless steel except for the top 2 in. section and dowel pins that are Type 304 stainless steel. Fuel capsules are inserted into the core and withdrawn from it by these handles, and in addition, they maintain the desired core configuration in a radial direction. The configuration of a fuel element showing the fuel capsule and handle is shown in Fig. 8.

The lower rod section of the handle is 23.44 in. long with a round cross section, and it has a 5/16-18 tapped hole to accommodate the fuel capsule.

Fig. 8. Fuel Element

The lower end of this section has a nominal 0.118 in. diam for 19.44 in., and the remaining 4 in. of this lower rod section has a 0.375 in. diam. This 4 in. part forms the coolant outlet plenum.

Lateral restraint of the core is provided by a splined section with a nominal 0.497 in. diam located 13/16 in. above the lower end of the handle. There are six splines in this section, each 0.109 in. wide, 0.250 in. long, and a root diameter of 0.297 in. The resulting diamond-shaped void between handles is the coolant channel through this section.

The lower rod section of the handle is attached to a hexagonal section with a light press fit on a 0.250 in. diam and is secured with a 1/8 in. dowel pin.

The hexagonal section of the handle, furnished by Armco to the finished dimension, is 90.81 in. long and is a nominal 0.486 in. across the flats.

The top 2.19 in. section of the handle serves two functions. First, its hexagonal part provides a tight fitting assembly that prevents racheting during handling, and secondly, it provides a means of withdrawing the fuel element from the vessel. The top 1-3/16 in. of this section has a 0.373 in. diam and a 1/2 in. long, 1/8 in. dowel pin at its midpoint. This part, with its associated pin, mates with the grabber of the fuel handling equipment described in Section 6.5. The locking feature that prevents racheting during handling is provided by the next 1/2 in. hexagonal part, 0.497 in. across the flats. The lower 1/2 in. of this top 2.19 in. section has a 0.478 in. diam and a round cross section.

After assembly and degreasing the capsule handles were degassed and the 17-4 PH components hardened at 950 F.

Dummy fuel capsules were made to accomplish the initial stacking of the elements in the vessel and to provide a satisfactory configuration for initial approaches to critical. These dummy capsules were fabricated from 17-4 PH stainless steel and they have the same outside dimensions as the fuel capsule assembly.

To achieve the most desirable core configuration, reflector pins that are made of the same material as the dummy fuel capsules are provided. These

reflector pins have the same configuration as the dummy fuel capsules; however, there are these exceptions: the diameter of the pin is 0.497 in. and the fuel transfer equipment requires that the pin has three grooves 1/16 in. deep, spaced 120 degrees apart.

2.3.3 Thermocouple Handles. Fifteen capsule handles were modified to provide thermocouple installations in the core region. Two handles have reflector pins permanently installed with the thermocouple extending to the lower end of the reflector pin to observe core inlet temperatures. One thermocouple terminates at the outlet plenum. The balance terminate at the core outlet immediately above the handle splines.

A groove 0.078 in. x 0.078 in. was milled in 15 assembled handles. This groove terminates immediately above the spline section and is aligned with one spline groove. The top end of the handle assembly was drilled to accept a Teflon insulator and a rhodium-plated miniature connector that was secured with two set screws. A stainless steel sheathed, chromel-alumel thermocouple with a 0.060 in. diam was staked into the milled groove.

2.4 Core I Capsules

2.4.1 Thimble Fabrication. The Ta fuel thimbles used in the first core loading were fabricated from two sources of starting stock.

Eleven thimbles originated with NRC tantalum powder alloyed at the Los Alamos Scientific Laboratory with a nominal composition of 2000 ppm yttrium and 1000 ppm tungsten. The ingot made from the alloy powder was electron beam melted; the final material showed tungsten and yttrium in average concentrations of 950 ppm tungsten and <40 ppm yttrium.

The balance of Core I thimbles originated with Temescal tantalum powder alloyed at LASL with Sylvania tungsten powder. The material showed 915 ppm tungsten after the ingot was electron beam melted.

Ingot material was rolled, swaged, and machined. This resulted in a 5/8 in. diam rod which was then cut into 1.3 in. lengths, the starting stock for

a thimble. A thimble 2.5 in. long with a 0.069 in. wall was formed by six impact stages; six ironing stages produced a full length thimble. Subsequent operations included straightening, machining the cone, and trimming to length.

After the trimming operation was completed, radiographic inspection of thimbles was made to determine thimble wall thickness (a minimum standard of 0.020 in. of metal without defects was maintained), variations, and the internal configuration at the cone end (the wall thickness standard was maintained). Surface defects such as scores, galls, and cracks were reported and used as guides in subsequent visual inspections.

Although 300 peak kilovolts inspection (0.003 in. minimum resolution) was used early in the Core I program, it was discontinued later because of interpretation difficulties. One thousand peak kilovolts inspection (0.005 in. minimum resolution) proved adequate for determination of defects of interest, because thimbles were not rejected for defects smaller than 0.005 in.

Ultrasonic inspection, having as its objective the determination of folds and cracks not observable by radiographic inspection, was discontinued because of the surface roughness and grain structure of the material. The background obtained was roughly the same size as the defect of interest. Ultrasonic inspection has not been abandoned, but must wait for more refined fabrication techniques.

The final operation was an anneal in a vacuum furnace with the temperature of the anneal 1450 °C for NRC thimbles and 1300 °C for Temescal thimbles.

Details of LAMPRE I thimble fabrication are in Los Alamos Scientific Laboratory Report LA-2434.

2.4.2 Tests. Tests performed in a sodium environment approximating that of the reactor indicate fuel capsule corrosion behavior is influenced by these variables: operating temperature, thimble heat treatment, and fuel composition.

The more positive test conclusions are based on alpha monitoring of capsules (usually every 700 hr) to determine penetration by fuel. Once full penetration is indicated, the extent and location of leakage are verified by

autoradiography. This sensitive detection method is required since these capsules show no visible leakage.

Results obtained by autoradiography are supplemented by radiography, metallographic examination, and chemical analysis. Tests details are given in Appendix A.

2.5 Core I Fuel

Core I fuel is composed of 90 a/o Pu - 10 a/o Fe. This alloy is characterized by its high plutonium density and relatively low melting point. A phase diagram for this fuel is shown in Fig. 9. Minor variations in fuel composition appear to influence the corrosive behavior of this alloy. Known fuel variations in Core I may be summarized:

- (1) Most of the fuel contains cast iron substituted for pure iron as a primary alloying ingredient. The balance of the fuel contains pure iron. (The cast iron fuel is characterized by the following impurities: manganese, 100-500 ppm; silicon, 400-800 ppm; and carbon, >150 ppm.)
- (2) Most of the fuel contains 300 ppm carbon added as PuC. The balance of the fuel contains 1000 ppm carbon added as graphite.

Operation with fuel containing cast iron results in impurity segregation near the liquid level. The tantalum thimble is intergranularly attacked in this region if the fuel is operated above 575 °C, or if the thimble is annealed below 1450 °C. Fuel with added carbon shows a segregated region below the fuel level. This region collects fission gases during reactor operation, resulting in fuel column separation and significant reactivity losses.

Capsule tests (Appendix A) indicate that the fuel made with pure iron is less corrosive than fuel with added impurities. A few capsules in Core I contain pure fuel to obtain preliminary information for future core loadings. Radiographic examination of this fuel after operation shows no column separation and limited fuel level segregation.

Fig 9. Pu-Fe Phase Diagram

2.6 Core Physics

Calculations made during the design period are described and estimated values of several parameters are noted. Measured values for some of the reactor system characteristics are listed and discussed.

2.6.1 Design Calculations. The calculational methods used in estimating some of the basic nuclear parameters of the 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 B.

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 the reactor design in three different regions: 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 $2\epsilon_c$ 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.
- (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 ($\rho = 16.051 \text{ g/cm}^3$ at 550°C)
Tantalum	0.15403
Sodium	0.34587

Slightly different constants apply to the as-built core.

The thicknesses and compositions of the side and end reflectors used in the calculations are listed in Table 1. 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) reactor configuration the important nuclear parameters have the following values:

Core buckling	$B^2 = 0.048549 \text{ cm}^{-2}$
Side reflector savings	$\delta_r = 5.374 \text{ cm}$
Top reflector savings	$\delta_T = 4.312 \text{ cm}$
Bottom reflector savings	$\delta_B = 5.080 \text{ cm}$
Cylindrical radius	$R = 7.758 \text{ cm}$
Cylindrical height	$H = 16.249 \text{ cm}$
Fuel alloy mass	$m = 24.99 \text{ kg alloy}$
Core volume	$V_c = 3.06 \text{ liters}$
K_∞ core	$(\nu\Sigma_f)/(\Sigma_a) = 2.62$
K_{eff}	Adjusted to 1.00 by shim
Central median fission energy	$\bar{E}_f \sim 1 \text{ Mev}$
Prompt neutron lifetime	$\ell^* = \frac{1}{\nu\Sigma_f \bar{V}} = 8 \times 10^{-9}$

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:

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

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

TABLE 1

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)
Sodium (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 closure plugs	1. 27	Ta
Capsule handles	50. 8	0. 61518 S. S. (430), 0. 38482 Na
BOTTOM REFLECTOR		
Locator plate	0. 508	Ta
Core support plate	4. 572	0. 23795 Na, 0. 76205 S. S. (304)
Support 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)

$$\begin{aligned}\left(\frac{\Delta \rho}{\rho}\right)_{\text{S.S. (304)}} &= -36 \times 10^{-6}/^{\circ}\text{C} \\ \left(\frac{\Delta \rho}{\rho}\right)_{\text{Na}} &= -300 \times 10^{-6}/^{\circ}\text{C}\end{aligned}$$

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:

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

These estimated mass relations may be converted to temperature coefficients of reactivity by using the constant (experimentally determined in the critical assembly mockup LCX II)

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

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

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

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

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

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

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

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

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

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.

Because the 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, because 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 overall power coefficient, are considered in Appendix C 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 percent 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 $\sim 3\%$ 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 clearance assumed, the overall power coefficient, including the bowing contribution, is calculated to be very small up to a power level of 30 kw, with a value of $\sim -0.04\%$ /kw. Above the 30 kw power

level, bowing is of little consequence and the overall power coefficient becomes approximately -0.44¢/kw .

- (4) The prompt component of the power coefficient is negative over the entire power range, with a value of about -0.41¢/kw .

There is one other recognized source of a small positive component of the overall power coefficient. Relative motion between the 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 9.2. It is estimated, however, that a positive component of the power coefficient of $\leq 0.015\text{¢/kw}$ exists because of this thermally induced, relative motion between the core and the control elements.

Calculations indicate that at operating temperature, but at low power, withdrawing the shim and control rods reduces reactivity by an amount equivalent to ~ 6 kg of fuel, or by roughly \$70. Although the relative effectiveness of the control rods and the control shim is not known precisely, it appears that $\sim \$60$ 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 $\sim \$33$.

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 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

2.6.2 Measurements. The basic measurements made during startup and early operation included those which established:

- (1) The critical mass with and without sodium in the core, and at various temperatures both below and above the melting point of the fuel.
- (2) Reactivity worth of the control elements.
- (3) β /gram worth of the fuel.
- (4) The isothermal temperature coefficient of reactivity.
- (5) The power coefficient of the system (up to 400 kw).
- (6) The transfer function of the reactor system.

2.6.3 Critical Mass. These determinations consisted of measurements performed with three basic core configurations:

- (1) A no-sodium, approximately 80 °C core, solid fuel. *
- (2) A with-sodium, approximately 160 °C core, solid fuel and molten sodium.
- (3) A with-sodium, approximately 480 °C core, molten fuel and sodium.

In all cases, except the first, the reactor system was equipped with three of the four available control rods. During the 80 °C measurements an oscillator rod was also in place and contributed reactivity equivalent to approximately 139 g Pu. Table 2 lists critical masses for the conditions specified. Numbers in parentheses are the calculated estimates corresponding to the measured quantities.

The 80 ° and 160 °C data permit an evaluation of the reactivity worth of the sodium. Corrections were applied to the 160 °C, with-sodium data, prior to comparing them with the 80 °C, no-sodium data. Table 3 illustrates the way in which these corrections were made.

* This core is sometimes referred to as LCX III (LAMPRE Critical Experiment No. III). Earlier critical experiments (Appendix B) were performed with a mockup of the reactor; in LCX III the LAMPRE I system was used.

TABLE 2
CRITICAL MEASUREMENTS

Configuration	Shim up Rods up	Mc (kg Pu) Shim up Rods down	Shim down Rods down
80 °C, solid fuel, no sodium	25.35 (25.48)	26.60	31.38 (31.68)
160 °C, solid fuel molten sodium	22.98	23.96	27.80
480 °C, molten fuel and sodium	23.37 (~ 25)		

2.6.4 Reactivity Worth of Control Elements. The worth of the shim was determined by a series of measurements of the extrapolated critical mass as a function of shim position. Data were obtained only at 160 °C, i. e. , below the melting point of the fuel. Results of these measurements are shown in Fig. 10.

The control rod calibrations were obtained from close extrapolated critical mass measurements performed during the final approaches to the initial configurations at 80 °, 160 °, and 480 °C. Reactivity data obtained from period measurements taken near the center of travel of rods No. 2 and 3 were used to correct the mass calibrations to reactivity calibrations. The results for the three basic configurations are compared below:

Case	Rod No.	ζ /gram	Total mass worth of rod	Total reactivity worth of rod
80 °C, LCX III	3	0.62	370 grams	\$ 2.29
160 °C	3	0.68	328 grams	2.23
480 °C	3	0.75	330 grams	2.47
480 °C	2	0.57	393 grams	2.24

It is suspected that the ζ /gram calibrations for the two rods at 480 °C are in error since capsules in equivalent core positions were used to obtain the extrapolated mass curves for the two rods; thus, the ζ /gram calibration should

TABLE 3

CORRECTIONS APPLIED TO EVALUATE SODIUM REACTIVITY WORTH

Item	Value	Source of data
(1) Fuel temperature coefficient at 80° C	-1.1 β /° C at 80° C	LCX III, measured
(2) Density change per ° C (alpha phase Pu)	-0.00183 g/cm ³ -° C	Fig. 11
(3) Fuel density reactivity coefficient	+6.01 β /(g/cm ³)	(1) and (2)
(4) Density change per ° C (beta phase Pu)	-0.00134 g/(cm ³ -° C)	Fig. 11
(5) Fuel temperature coefficient at 160° C	-0.81 β /° C	(3) and (4)
(6) Total temperature coefficient at 160° C	-1.32 β /° C	Measured at 160° C
(7) Sodium temperature coefficient	-0.51 β /° C	(5) and (6)
(8) Fuel worth	+0.68 β /g	Measured at 160° C, with sodium
(9) Worth of oscillator rod	-139 g of Pu	Measured at 160° C, with sodium
(10) Fuel density at 80° C	16.65 g/cm ³	Fig. 11
(11) Fuel density at 160° C	16.44 g/cm ³	Fig. 11
(12) Worth of fuel density change 160° C-80° C	-97 g	(3), (8), (10), and (11)
(13) Worth of Na density change 160° C-80° C	-60 g	(7) and (8)
(14) Total correction 160° C-80° C	-296 g	(9), (12), and (13)

The sodium worth is then evaluated as equivalent to:

2.94 kg Pu at 80° C, shim up

3.88 kg Pu at 80° C, shim down

Calculated values at 20° C (see Section 2.6.1) were, for the corresponding cases, 3.1 kg and 3.9 kg Pu.

Fig. 10. Extrapolated Critical Mass vs Shim Position at 160°C-All Rods and Oscillator Out

Fig. 11. Pu-Fe Fuel Density vs Temperature

have been equal for the two rods. The average of the two ζ /gram measurements agrees closely with the measurement at 160°C, and the total reactivity rod worths for all four cases are in reasonable agreement. Accordingly, the results for the two rods at 480°C have been averaged to obtain the reactivity versus rod position curve shown in Fig. 12. A curve of cents per inch of rod insertion is also shown in the figure. It is seen that at 15 in. of insertion of the rods the maximum reactivity point is encountered. This point, which was at 16 in. of insertion at 160°C, was evidently shifted downward due to the relative expansion of the core and rod supporting materials during the temperature ascension. Stops are now in place on the rod drives to limit the insertion to 12.5 in.

The experimentally determined worth of the shim and four rods is seen to be equivalent to ~ 5 kg Pu; the corresponding value obtained by design calculations was ~ 6 kg Pu.

2.6.5 Reactivity Worth of Fuel. An absolute reactivity calibration was made by observing the relationship of rod displacements to positive (and negative) periods. Because the rods had previously been calibrated in terms of peripheral Pu core mass, a $\sim \zeta$ /gram value for fuel was thus determined. Results of the measurements are summarized in Table 4.

TABLE 4
FUEL REACTIVITY WORTH

Core	ζ /gram Pu on core periphery
80°C (no sodium)	0.62
160°C (with sodium)	0.68
480°C (with sodium)	0.66

For purposes of design it had been assumed that the ζ /gram value of 1.18 experimentally determined on LCX II would apply approximately to the actual core. The fact that it does not, being in error by about a factor of 2, has not been investigated in any detail. The difference between the assumed and the

Fig. 12. Hot Critical Rod Worths

actual ζ /gram values of fuel in the reactor accounts in large measure for the difference in predicted shim worth ($\sim \$60$) and that actually observed ($\sim \$26$). (As already noted, the predicted shim worth of the control elements in terms of kg Pu agreed satisfactorily with the measured value.)

Similar differences appear, for the same reason, between the predicted and observed values of the temperature and power coefficients.

2.6.6 Isothermal Temperature Coefficient. This was measured by varying system temperature through adjustment of the electrical heat input and then balancing changes in reactivity by appropriate motion of the previously calibrated control rods. Reactor power during these experiments was a few watts; the observations were carried out at 80 °C (no sodium), at 160 °C (with sodium), and also at temperatures above the melting point of the fuel.

Variation in fuel density, in sodium density, and in structure density and dimensions contributed to the observed isothermal temperature coefficient. Estimated magnitudes of these effects are noted in Table 5.

TABLE 5
ISOTHERMAL TEMPERATURE COEFFICIENTS

System temperature	Temperature coefficient ($-\zeta/^\circ\text{C}$)			
	Total	Pu	Na	Structure
80 °C	1.1 (meas)	0.8		0.4
160 °C	1.3 (meas)	0.5	0.5	0.4
480 °C	2.9 (meas)	2.0	0.5	0.4
480 °C	3.2 (calc)*	1.8	0.5	0.9

2.6.7 Power Coefficient. This coefficient was evaluated by comparing the net change in reactor power (following a power demand) with net change in reactivity

* Original design calculations, corrected to the ζ /gram Pu value measured in LAMPRE I.

from the variation in the core inlet temperature resulting from the power demand. Power demands were produced by reducing the electrical heat input to the reactor system, by increasing the louvre opening in the sodium-to-air heat dump radiator, or by starting the air circulating blower in the heat dump. Initial powers prior to demand initiation have ranged from 0.1 kw (10 kw demand) to 50 kw (100 kw demand). Reactor response to power demand is indicated for two typical cases in Figs 13 and 14. Values of power coefficients derived from the power demand experiments are given in Table 6.

TABLE 6
POWER COEFFICIENTS AT FULL COOLANT FLOW (133 gpm)

Power demand (kw)	Core inlet temperature change ($^{\circ}\text{C}$)	Power coefficient (c/kw)
24	-3.9	-0.47
48	-7.7	-0.47
57	-9.9	-0.51
74	-12.8	-0.51
100	-17.3	-0.50

The power coefficient at half-normal coolant flow (67 gpm) was not extensively investigated, but the data obtained yielded a value of approximately -0.57c/kw .

2.6.8 Transfer Function. This has been measured at powers ranging from ~ 1 w to 400 kw and for input (oscillator rod) frequencies from 10^{-1} to 10^2 radians/sec. Results of the measurements are shown in Fig. 15. The gain curves indicate the increase in stability as the power level is raised. This increase in stability is also verified by the greater damping observed in the response to power demands as the initial power level is increased.

A good fit to all the oscillator experiment data is obtained by a feedback function of the type

Fig. 13. 10-kw Power Demand from 100 w

Fig. 14. Core Inlet and Outlet Temperatures for 100-kw Power Demand Starting from 45 kw

Fig. 15. Transfer Functions for Various Power Levels

$$H(w) = C/(1 + j w \tau)$$

where $H(w)$ is the frequency dependent transfer function, C is the power coefficient, τ is the reactor thermal time constant, and w is the input frequency in radians/sec. Effects due to the sodium coolant flow may be seen in a comparison in Table 7 of the results for full flow (133 gpm) and half flow (67 gpm).

TABLE 7
TRANSFER FUNCTION ANALYSIS

<u>Constant</u>	<u>Full flow</u>	<u>Half flow</u>
C	-0.47¢ /kw	-0.63¢ /kw
τ	1.3 sec	1.8 sec

The short time constant indicates that the principal contributions to the transfer function must come from core parameters.

Power coefficients derived from the transfer function analysis are in reasonable agreement with those determined by power demand measurements.

3. VESSEL AND ASSOCIATED COMPONENTS

3.1 Vessel

The reactor vessel is a Type 316 stainless steel welded assembly with an 8.750 in. i.d. in the core region and an overall height of 149 in. The lower portion of the vessel is double walled with no penetrations; therefore, coolant sodium cannot be accidentally drained from the core unless a leak occurs in both primary and secondary vessels. The coolant may be siphoned from the vessel through a 3/8 in. stainless steel tube that enters the vessel through the sodium inlet pipe and extends to the bottom of the vessel. Closure is accomplished at the top of the vessel by the mating of a standard 10 in. , 150 lb. slip-on flange with a 10 in. slide valve.

3.2 Flow Divider

Directly surrounding the core is a Type 430 stainless steel shell called the flow divider. The contour of the internal surface of this shell is identical with the contour of 199 tightly packed 1/2 in. hexagons. The primary function of the flow divider is to direct the coolant flow; it also provides direct support for the bottom reflector, direct support for the locator plate assembly, and indirect support for the fuel elements. A top view of the component is shown in Fig. 16.

A flange at the top of the flow divider rests on a shoulder around the inside diameter of the transition piece. This shoulder is immediately above the

Fig. 16. Top View of Flow Divider

coolant inlet. Internal shoulders, at and near the bottom of the flow divider, support the bottom reflector and locator plate assembly.

The flow divider was fabricated from six slabs that, prior to machining, were outgassed in a vacuum furnace. The profile of the inside and edge surfaces are the same for each of the six pieces and are cut on a milling machine. The outside surfaces were rough cut on the milling machine to reduce the amount of material to be removed later by turning.

After the milling operation the six pieces were assembled around a hexagonal mandrel, were clamped together by a ring fixture, and were joined with an intermittent weld ($\sim 50\%$). The welding schedule minimized warping; cracks at the unwelded portions of the joint were less than 0.002 in. After welding, the mandrel was pushed out with a hydraulic press and round sections were turned on a lathe. Inspection of the finished part showed dimensional tolerances had not been exceeded. The inside profile was as specified - within ± 0.008 in.

3.3 Transition Piece

The transition piece, shown in Fig. 17, is a stepped, Type 304 stainless steel cylindrical sleeve that sits on a shoulder around the inside diameter of the vessel wall. This piece supports the flow divider and vessel liner and also establishes an upper boundary for inlet coolant flow. Exit coolant passes through the hole in its circumference.

3.4 Vessel Liner

The vessel liner is a Type 304 stainless steel sleeve that functions as a spacer between the inside diameter of the vessel and the bundle of capsule handles. The liner was fabricated from six identical pieces and has a cylindrical outside diameter and an inner contour matching that of the flow divider. The component is shown in Fig. 18.

Fig. 17. Transition Piece

Fig. 18. Vessel Liner

3.5 Vessel Heating

In the annulus formed between the primary and secondary vessels are nine 230 v, 3000 w, resistance heaters that assist in keeping the core molten when the reactor is subcritical. Figure 19 shows the bare vessel with the core heaters. They are normally operated at reduced power, and they emerge from the top of the annulus through Swagelock sealed fittings.

Electrical heaters surrounding the upper 6 ft of the vessel maintain the temperature in this region above the melting point of the coolant.

Heat conduction through the capsule handles may raise the temperature of the valve flange above a desirable level. If this occurs, the region can be cooled through a coil of stainless steel tubing surrounding the vessel immediately below the upper flange. It has not been necessary to use this feature.

3.6 Leak Detection

Two leak detectors are located at different heights in the annulus formed between the primary and secondary vessels. Each detector is a pair of stainless steel-sheathed, ceramic-insulated wires. The wire ends are separated by a small air gap. If a leak occurs in the primary vessel, an electrical circuit completed between the two wires will initiate an annunciator signal and light display on the reactor console. The detectors emerge from the top of the annulus through Swagelock sealed fittings.

3.7 Thermocouple Feedthrough Assembly

The thermocouple feedthrough assembly is shown in Fig. 20. The thermocouple feedthrough bulkhead, which is installed on a flanged opening immediately below the top flange of the vessel, is made from stainless steel bar and

Fig. 19. Bare Vessel with Core Heaters

Fig. 20. Thermocouple Feedthrough Assembly

has a 4-1/2 in. diam flange with a neoprene O-ring. Each of the four feed-through connectors (Conax high-pressure fittings) has 8 terminals; 16 temperature points are available in all; on the internal side 8 support points relieve handling loads on the thermocouple wires. The stainless steel-sheathed, thermocouple wires and bulkhead connection were potted in Silastic to prevent sodium vapor damage and reduce the possibility of damage from residual sodium carried into the top vessel region upon withdrawal of the fuel elements.

4. COOLANT SYSTEM

4.1 General

A diagram of the system is shown in Fig. 21, a plan view in Fig. 22, and an elevation in Fig. 23. Type 316 and, in general, Type 316 ELC stainless steel was used in fabricating the system, including the sodium piping (2 and 3 in Schedule 40), tanks, hot and cold trap systems, and drain lines.

All welds were made with an inert-arc root pass followed by coated rod filler passes. Root passes and the completed weld were checked by dye penetrant, and welds in piping larger than 1 in. were tested radiographically. All welds were leak tested with a helium mass-spectrometer leak detector.

4.2 Component Description

4.2.1 Piping. The piping loop is supported from spring hangers with anchors at the reactor vessel and at selected points in the loop. See Figs 22 and 23. The loop is designed to maintain an internal stress level no greater than 14,000 psi at 650 °C. For the most part, the loop is insulated with a two-layer construction; the inner layer is Superex for high temperatures, and the outer layer is 85% magnesia for more effective insulation once the temperature has been decreased to a reasonable level. The overall thickness is between 3.5 and 4 in.

A leak detection system consisting of a pair of high temperature, loose insulation thermocouple wires (located below the piping) traces all piping and divides it into ten zones. This was done with the intention that any sodium leak

Fig 21. Coolant System Flow Diagram

Fig. 22. Plan View of Coolant System

Fig. 23. Elevation of Coolant System

will cause a short between the wire pair and indicate which section of the system contains the leak.

Two methods are used for heating the coolant system. First, a calrod trace system is used for preheating of the entire system. Secondly, the temperature can be raised to the melting point of the fuel when the system has been filled with sodium and is flowing by using a heating transformer, a full-flow loop in the system constructed to be the secondary of the transformer.

4.2.2 Pumps. Coolant pumping is done by two Callary AC electromagnetic conduction pumps in parallel, each rated at 100 gpm at 20 psi head. Power to the pumps is controlled by motor-driven adjustable autotransformers. Check valves of the flapper type in the outlet of each pump prevent bypass flow.

4.2.3 Heat Exchanger. The heat exchanger, built by Griscom-Russell, is made of finned sections that transfer heat from sodium to air, which, in turn, is exhausted up a stack outside the building. Airflow through the unit is controlled by sodium temperature at the outlet from the exchanger. The temperature is sensed by a thermocouple connected with a self-balancing potentiometer. This potentiometer controls a pneumatic operator, which positions a louver-type damper at the blower inlet.

4.2.4 Hot Trap. Three getter hot traps connected in parallel with a heater control for each trap are installed in a 1 in. bypass loop on the main system. It is estimated that the Na_2O content in the coolant can be reduced to 1 or 2 ppm by use of these traps. The metal used for gettering is zirconium foil heated to 700°C . The hot trap loop has its own electromagnetic pump and flowmeter. Capacity of the traps is estimated to be tenfold greater than necessary. The hot trap bypass flow is $\sim 5\%$ of full-power coolant flow.

4.3 Dump Tank and Cold Trap

The main coolant system is filled from or drained to a tank located at the lowest part of the system. Sodium is pushed from the tank to the main

system by gas pressure and dumping is usually done by gravity, although gas pressure can be used to speed up the operation. Isolation of the dump tank from the main system is done by a motor-operated, bellows-sealed valve in the drain line.

A cold trap and plugging indicator are located in a side loop on the dump tank. Cold trapping of the main system can only be done by returning it to the dump tank. During reactor operation the cold trap, as well as the dump tank, are valved off from the main system. The plugging indicator allows the Na_2O content to be measured down to ~ 15 ppm.

4.4 Instrumentation

The coolant system instrumentation is conventional in most respects. Figures 24 and 25 indicate instrumentation locations. In addition to the core thermocouples, there are thermocouples on the 2 and 3 in. piping. Remote indicating transmitters measure pressures at the surge tank, pump inlet and outlet, and dump tank. Within the combination surge and expansion tank installed upstream from the heat exchanger inlet and the dump tank are variable reluctance sodium level indicators. Flow measurement is made by three electromagnetic flowmeters, one in each of the 2 in. pump loops, and one in the 3 in. piping between reactor outlet and surge tank. The 3 in. flowmeter was made and calibrated by Atomics International.

4.5 Sodium Disposal

A method of disposal of contaminated sodium from the dump tank and reactor vessel, if this ever becomes necessary, is built into the coolant system. The sodium disposal system consists of 300 ft of insulated 1 in. pipe enclosed in 3-1/2 in. conduit, which extends from the sodium room and reactor cell south from Building TSL-2 to the canyon rim (Fig. 26). The disposal line ends in two

Fig. 24. Plan View of Coolant System Showing Instrumentation

Fig 25. Elevation of Coolant System Showing Instrumentation

Fig. 26. Ten Site Layout

120 ft wells. In the well shack at the end of the line is a 6 kv-a heating transformer that serves the wells and a heavy-duty vacuum pump for evacuating the line and wells. The disposal line is heated by a 25 kv-a transformer that utilizes the pipe resistance as a secondary load. This transformer is housed in a manhole near the center of the disposal line, as are all switch gear, remote operating controls, monitoring equipment, and communication facilities.

4.6 Operating Experience

At the end of the initial cleanup and shakedown period the end hot trap was cut free from the system and the remaining manifold stub ends were sealed. The examination of the zirconium showed that oxygen had been absorbed, but the limit on the hot trap's capacity for oxygen absorption had not been reached.

The coolant system has been operating since October, 1960, at temperatures from 160 ° to 500 °C and with flows from 15 to 200 gpm. Temperatures have been maintained by electrical power only, nuclear power only, and several combinations of both. Electrical heating can maintain the temperature above 500 °C with all units on and above 450 °C with all means available minus the heating transformer. Limitation on flow has been governed by two parameters: 1) total flow of more than 165 gpm causes a dynamic head rise in the vessel resulting in sodium rising to the level of the thermocouple connections, and 2) a power greater than 140 amp to either pump causes excessive mechanical vibration in the pump structure.

5. COVER GAS AND GAS DISPOSAL SYSTEM

5.1 General

The system controlling the condition, i.e., pressure, temperature, purity, etc., of the cover gas and disposing of gaseous fission products that might evolve from the reactor is shown in Fig. 27. The main subsystems are:

- (1) The low pressure, high-purity helium manifold in Room E.
- (2) The main vacuum and helium distribution manifold in Room E.
- (3) The pressure equalization line with associated heaters and motor-operated valves in the fuel transfer area.
- (4) The contaminated helium and vacuum manifold in the reactor cell.
- (5) The underground holdup tank (T-7) for storage of contaminated helium.

Helium is introduced into the system through the low pressure, high-purity helium manifold, which consists of four bottles of high-purity helium valved into the system via four pressure regulators. Helium flow is regulated by a flow controller and is monitored by a flowmeter.

The main distribution manifold serves as a helium routing facility and a central high-vacuum pumping station. Vacuum is provided by a Kinney KC-15 vacuum pump run in series with a diffusion pump and associated equipment.

The pressure equalization line allows the pressure in gas phases of the vessel and surge tank to be the same at all times. A fission gas monitor that consists of a small, static volume of the cover gas is appended to the line. A scintillation counter close to this volume detects any significant level of

Fig 27 Cover Gas and Gas Disposal System Diagram

radioactivity in the gas phase. A signal from this counter is continuously displayed in the control room.

All gases that have been in the reactor are considered as contaminated, and for this reason, are handled behind the shielding of the cell wall in the contaminated helium and vacuum manifold.

The underground holdup tank (nominal 1000 gal) is located near the filter building (Fig. 26). In case of necessity, fission gases removed from the reactor may be dumped to this tank until their activity decreases to a level allowing them to be vented to the atmosphere.

Piping and valves are 18-8 stainless steel and are of welded construction, with the exception of the low pressure helium manifold and the vacuum side of the 2 in. manifold in Room E, which are copper with fitted and silver-soldered joints. Valves in the system have both metal-to-metal seats and Teflon seats and seals; both ball and needle valves are included in the system. Welds in the system were inert gas arc-welded followed by dye penetrant leak checks of the first pass. A final mass-spectrometer leak check was made of the entire system.

Certain valves located in the fuel transfer area are motorized for remote operation, and all motorized valves are Teflon seated with Teflon gasketed bellows seals. Valves in the cell are operated from the control room with reach rods. The pressure equalization line and the dump tank gas line are heated by Calrod heaters to keep sodium vapor from condensing, freezing, and forming a plug in them.

5.2 Helium Purification and Analysis

Because the helium used in the system is obtained in a state of relatively high-purity, through special handling at the source, the only treatment is for removal of water, which is done by a molecular sieve tower containing Linde 5A molecular sieve crystals. A typical analysis of the helium supplied for the system is given in Table 8.

TABLE 8
TYPICAL HELIUM ANALYSIS

<u>Impurities</u>	<u>ppm by volume</u>
H ₂	0.5
O ₂	1.0
N ₂	7.0
H ₂ O	2.5 before molecular sieve
H ₂ O	< 1.0 after molecular sieve

A Beckman gas chromatograph is set up to sample the cover gas in the surge tank and in the capsule charger (see Section 6). Normal analysis shows concentrations in amounts on the order of 5 ppm by volume of hydrogen, nitrogen, oxygen, and methane in the cover gas.

5.3 Radioactive Gas Disposal

The gas handling system has been designed to dispose of 35.6 ft³ of helium containing 0.173 g-moles of fission gas resulting from the failure of 140 fuel capsules after the reactor has been at a power of 1 Mw for 6 months. This includes these gas volumes--the reactor cover gas phase, the capsule charger, and the dump tank gas phase.

5.4 Operating Experience

The first cover gas sample, which was taken approximately two weeks after the initial sodium fill, showed hydrogen present in quantities on the order of 1%. The system was flushed numerous times and the impurity concentrations were monitored as they built up; hydrogen was the most prevalent impurity.

Zirconium metal in various forms was placed into the core as a getter, and flushes of the gas phase were continued with the temperatures of the sodium system varying from 150 °C to 480 °C. During the loading of fuel elements into the vessel, the temperature of the sodium system was set at 160 °C and remained there for a considerable time. At about the time the temperature was dropped to 160 °C, the hydrogen levels dropped, but methane appeared in high concentrations (~ 500 ppm) in the cover gas. Purges were continued in the attempt to remove methane until the core was melted. Impurities in the gas phase dropped off as the temperature was increased, and although they sometimes built up to ~ 100 ppm (at which time the cover gas was purged), the concentrations continued at low values for long periods of time.

6. CAPSULE CHARGER

6.1 General

The capsule charger (Fig. 28) is located in the fuel transfer area shown in Fig. 29 and is used for all loading and unloading of the core. By suitable mechanical operations it is possible to (a) select and remove a single fuel element from any one of the 199 core locations, (b) remove the fuel capsule (or dummy capsule, reflector pin, test specimen, etc.) from the selected capsule handle, and (c) transfer the fuel capsule out of the capsule charger through the transfer valve port into a shielded alpha container. By reverse operations it is possible to transfer a fuel capsule into the capsule charger for insertion into the core. (The vacant core location is always refilled with a new fuel capsule, dummy capsule, etc., before a new transfer operation is started.)

All operations required in the selection, removal, and transfer of a fuel capsule are done remotely in a helium atmosphere, behind adequate shielding, and with visual reference possible at all times by means of several periscopes. Each operation is selected by the operator and is mechanically or electrically activated. A small control console in the fuel transfer area contains position lights and indicators used during transfer operations; no automatic sequencing or interlocks are used with the exception of a few limit switches.

6.2 Housing

The main capsule charger housing contains the vertical lift and grabber units. A rectangular side chamber, approximately 2 ft above the housing base,

Fig. 28. Capsule Charger Schematic

Fig. 29. Building Layout

contains the horizontal transfer and vertical transfer units. The 10 in. slide valve described in Section 3.1 connects the base of the housing to the top of the vessel. (This valve is closed during reactor operation and most of the fuel transfer operations.)

The housing has (a) eight penetrations in the housing wall for viewing windows, (b) six rotary shaft seals for the vertical lift, (c) one 2 in. diam connection for evacuation, (d) one 1/4 in. diam line for gas filling and purging, and (e) one 3/4 in. transfer valve port. (An identical valve port is located in the top of each capsule loading and transfer alpha container, which is described in Section 7.2.) Electrical leads are brought through the housing wall by means of three, hermetically sealed AN connectors.

Design pressure of the housing components is 40 psig, with a normal operating pressure equal to the cover gas of ~ 10 to 12 psig high-purity helium. The housing can be evacuated below 50μ , for purging, in ~ 20 min.

6.3 Viewing

Six lucite windows and three periscopes allow viewing of the housing interior. The two lower rectangular windows, 90 degrees apart, are equipped with periscopes mounted through an offset penetration of the lead shielding that is described in the next section. The periscopes provide an erect image by use of two front silvered mirrors and a 6X monocular eyepiece. The next higher window is in line with the horizontal transfer unit and has a simple periscope with an approximate 6 in. offset in the lead wall. The two upper rectangular windows are used to set the grabber unit coordinates during insertion of a fuel element. Three round windows at the top of the housing are normally used only for inspection of the grabber unit and the electrical cable unit.

6.4 Shielding

Eight inches of lead is stacked around the base of the capsule charger to a height of 5 ft, with a reduced thickness continuing to about the 8 ft level. No shielding is used above this point with the exception of the 18 in. of concrete in the roof of the fuel transfer area.

When the capsule charger shield pot (see Section 7.2) is in place for unloading operations, its 5 in. tuballoy walls form a continuation of the lead shielding that surrounds the horizontal transfer section of the capsule charger.

Clearance is provided between the capsule charger and the lead shielding to allow vertical movement due to temperature changes of the vessel.

6.5 Fuel Capsule Transferring

The sequence of removing a capsule begins by equalizing cover gas and housing pressures; after this is done the 10 in. slide valve is opened. The grabber of the vertical lift unit is capable of motion in a cylindrical coordinate system and is positioned by changing its r , θ , and z location. It is moved to engage its bayonet socket with the mating end of the desired capsule handle. Alignment for engagement is done visually by means of the two periscopes that extend from the two lower rectangular windows in the housing.

After engagement of the bayonet socket, the vertical lift unit raises the fuel element to a station where the capsule is unscrewed from the handle. The splined locator section on the handle is rotated by a motor on the horizontal transfer unit, while wrench flats on the capsule top are restrained by a stationary wrench. When the capsule is free of the handle, the horizontal transfer unit moves it to a position above the capsule charger shield pot that is sealed to the transfer valve port. At this point the vertical transfer unit engages the capsule, the horizontal transfer unit disengages, and the capsule is lowered

into the alpha container in the shield pot. Transfer valve ports (housing and alpha container) are closed, the space between the transfer valves is evacuated, and the shield pot is disconnected and lowered from the port. For loading the sequence is reversed; however, during insertion of a capsule into the core the speed is restricted to ~ 5 fpm in the last 10 to 12 in. of travel.

Power for vertical motion of the vertical lift unit is supplied by a hydraulic motor. Hydraulic pressure to the motor is adjustable and, during normal operation, will be only high enough for the unit to exert a lifting force of ~ 75 lb on a handle. The components are designed however to withstand forces required to exert a 1000 lb pull in case a sticking handle makes it necessary.

Position of the vertical lift is indicated by a counter to 0.10 in. and may be controlled within 0.01 in. over the full travel.

7. IRRADIATED FUEL EXAMINATION AND STORAGE

7.1 Capsule Storage Facility

Fuel capsules in alpha tight containers are stored in this facility to allow the capsules or sections of capsules to decay to tolerable limits for disposition. The storage facility is constructed as an addition to the main building (TSL-2); its location in respect to the reactor vessel is shown in Fig. 29 and an interior view of the facility is shown in Fig. 30.

The facility consists of 114 mild steel sleeves of 4.760 in. i.d., imbedded 4-1/2 ft in magnetite concrete of 240 lb/ft³ density. These sleeves, shown in Fig. 31, are welded upon a 6 in. x 3 in. mild steel rectangular duct through which water flows for cooling of the capsule. Eleven other sleeves (8, 10, and 12 in. i.d. x 12 in. deep) are not cooled and are used for storing very low level sources; these sleeves are located in one corner of the facility.

The 4.760 in. i.d. sleeves are positioned 12 in. on centers in six rows; the six rows are 17 in. on centers. This array is nuclear safe with two capsules located in each sleeve. The maximum radiation at floor level for this 17 in. x 12 in. array is calculated to be less than 1-1/2 mr/hr. This is based on a capsule source strength of 8×10^3 Mev/sec (1.80-Mev photons) following 1 Mw operation for six months and 1 hr after shutdown. The curie source strength of hard gammas 1 hr after shutdown is therefore ~ 1200 curies; the corresponding figure 16 hr after shutdown is 600 curies. Shielding above the capsule in the sleeve consists of a 4.718 in. diam tuballoy plug that is described in Section 7.2.

Fig. 30. Interior View of Capsule Storage Facility

Fig 31 Capsule Storage Facility Sleeves Mounted on Cooling Ducts

The water cooling system is closed with water circulated by pump and cooled by a two-speed circulating water cooling unit. A temperature controller operates the cooling unit at low speed when the water temperature has reached 70 F and at high speed when the water temperature is 80 F. Flow in the system is 20 gpm. A check valve keeps the water being circulated from re-entering the building supply system.

An exhaust blower rated at 611 cfm maintains a negative pressure in the facility. The air is exhausted from the floor and flows through a pre-filter and absolute filter to the exhaust fan on the roof of Building TSL-2.

The floor is painted with two coats of Plasite protective coating for ease of cleanup in the event of a radiation spill.

A 3-ton crane with hand-racked bridge and trolley and electric hoist is used for handling a radiological shield pot and lowering the sources into the storage sleeves.

7.2 Irradiated Fuel Handling

Fuel capsules leaving the vessel go directly to storage. Figure 32 shows the capsule loading and transfer alpha container and the capsule sectioning and transfer alpha container which are used for all transfers and storage. Figure 33 shows the tuballoy shield cask (the "capsule charger shield pot") which is used for alpha container transfers.

7.2.1 Vessel to Storage Facility. When a fuel capsule leaves the vessel for storage, the capsule is lowered from the capsule charger through the transfer valve port into the capsule loading and transfer alpha container that is in the up position in the capsule charger shield pot. The fuel capsule in its alpha container is lowered by the elevator to the down position and the sliding top entry door of the shield pot is closed. A special cart is used to transport the shield pot to the storage facility.

In the storage sleeve is a tuballoy shield plug, 7 in. long with a 1.718 in. diam, that has a stepped central rod to prevent gamma streaming through

Fig. 33. Capsule Charger Shield Pot

the clearance between the rod and the shield plug body. This central rod is pinned on the upper end and threaded on the lower end. The lower end screws into a matching female thread in either the capsule loading and transfer or capsule sectioning and transfer alpha containers.

The shield plug is removed from the sleeve, and a locating ring that positions the cart is placed into a step at the top of the sleeve. After the shield pot in its cart is positioned over the sleeve, the shield plug is lowered to within 1/32 in. of the top door of the pot to provide biological shielding when the door is opened. The shield plug is lowered after the door is opened until it rests upon the alpha container. The shield plug rod is then screwed into the alpha container and the shield plug and container are raised slightly to clear the bottom door of the shield pot. The bottom door of the pot is opened and the plug-container assembly is lowered to its storage position within the sleeve.

7.2.2 Storage to Sectioning and Metallographic Caves. When a fuel capsule goes to the sectioning and metallographic caves (see Sections 7.5 and 7.6), the capsule charger shield pot is used for all transfers. The capsule in its alpha container is taken from the storage sleeve and placed in the shield pot in a manner reverse to that described in the previous section. The shield pot is taken from the storage facility to the sectioning cave and is drawn into the cave in a manner reverse to the capsule charger unloading operation. The capsule is removed from its alpha container and the required sections are cut from the capsule.

The section is placed in the capsule sectioning and transfer alpha container, which is then placed in the shield pot and taken to the metallographic cave. The alpha container is transferred to the sectioning cave in the shield pot after completion of the metallography operations, and the remaining sections of the capsule are picked up.

The procedure for transferring the alpha container from the shield pot to the storage sleeve is the same as that used in the direct storage operation.

7.3 Radiography

The autoradiographic techniques used to locate sections of interest in test fuel capsules cannot be applied to irradiated capsules. Thus, it is necessary to use the information gained from radiographs and gamma scannings (see Section 7.4) of the irradiated capsule to locate sections for metallographic study and chemical analysis.

All radiographic and other nondestructive inspection services are performed by group GMX-1 of LASL. Fuel capsule radiography is done using a betatron with a 150 R exposure.

An irradiated capsule receives special handling because of health reasons and possible contamination due to release of fission products resulting from capsule damage. An irradiated capsule in an alpha container is transported to the betatron site in a radiographic shield pot (Fig. 34) designed especially for this transfer operation. At the betatron site the shield pot is appropriately located with respect to the x-ray beam. A remotely operated hoist is fitted to the pot and secured to a shield plug, which, in turn, is attached to the alpha container holding the capsule. To reduce film fogging to a minimum, the shield plug-alpha container-capsule assembly is withdrawn from the shield pot only long enough to cover the betatron exposure time.

Although the maximum source strength of Core I fuel capsules inspected in this manner was 140 curies, this inspection is expected to yield good results with capsules to 500 curies.

7.4 Scanning

Irradiated fuel capsules are examined with the scanning device shown schematically in Fig. 35. A capsule is positioned vertically with respect to a variable-width collimating slit by an elevator, which is remotely controlled by

Fig. 34. Radiographic Shield Pot

Fig. 35. Block Diagram of Scanning Apparatus

means of a servo loop from which a signal is fed into the X axis of a plotter. The activity is observed with a scintillation counter positioned behind the slit; the amplified signal from the counter is fed into a count rate meter (CRM). Output from this meter is fed to the Y axis of the plotter; thus, a plot can be made of the activity of a selected fission product vs position along the axis of the fuel capsule. Figure 36 is an example of such a plot.

Pulses from the amplifier can also be fed into a 400-channel analyzer, which produces a pulse-height distribution curve (count rate vs pulse height) for any fixed position along the axis of the fuel capsule. Well-defined peaks appearing on these curves correspond to discrete gamma-ray energies that are characteristic of one or a number of fission products. Analysis of pulse-height distribution curves gives evidence of the spatial distribution of selected fission products in the fuel capsule. From information such as this, inferences may be made regarding aspects of the behavior of the fuel while it is in the core.

7.5 Capsule Sectioning Cave

A view of the partially assembled capsule sectioning cave is shown in Fig. 37. The cave is located in Laboratory A (Fig. 29) and is used for (a) sectioning of capsules into discs cut transverse to the axis and of approximately 1/4 in. length, (b) recovery of the radioactive gas from the capsule and charging it through a charcoal trap, (c) cleaning of sodium from the capsule, (d) sectioning low level radioactive capsules into lengths to permit several capsules to be stored in the sectioning and transfer alpha container, and (e) visual inspection of the capsule.

The air in the cave is exhausted through an absolute filter in the cave and then into the Laboratory A exhaust system that has a washer and dry filter system in the Filter Building (Fig. 26).

The cave consists of (a) an entry port similar to that on the capsule charger, (b) a capsule lift rod, (c) a brass heat sink, (d) an off-gas capsule chamber with puncture mechanism, (e) a cut-off saw, (f) an alpha container

Fig. 36. Plot of Gross Activity vs Axial Position of Fuel Capsule

Fig. 37. Partially Assembled Capsule Sectioning Cave

in-cell transfer mechanism, (g) a transfer trolley to position and remove the alpha container in and from the shield pot and transfer mechanism, and (h) a pair of Castle manipulators.

Waste material from the sawing operation is placed in the alpha container.

7.6 Metallographic Cave

The cave is located in Laboratory C (Fig. 29) and is used for (a) separation of the tantalum container material from the fuel, (b) straightening and bend testing the tantalum specimen, (c) polishing the plutonium specimen, (d) acid etching the plutonium specimen, (e) metallographic inspecting the plutonium, (f) photographing the plutonium specimen, (g) hardness testing the plutonium specimen, and (h) polishing and metallography of tantalum and plutonium in duplicate discs. The cave is shown in Fig. 38.

The air in the cave is exhausted through absolute filters into the laboratory exhaust system.

The cave is divided into two sections; a top opening one where the capsule sectioning and transfer alpha container with the disc is removed from the shield pot by a manipulator and placed on a horizontal transfer mechanism, and a closed alpha cell where the metallographic and bending test operations are performed. A transfer mechanism moves the disc into the manipulator operated alpha cell.

Contaminated waste material is dropped through a cell floor opening into a sealable plastic bag.

Fig. 38. Metallographic Cave

8. SHIELDING

8.1 General

Shielding requirements were determined by calculation of the radiation dose rates at selected points in the vicinity of the reactor for specified operating and/or shutdown times. Calculations were also made to determine the dose rates due to Na^{24} activity in areas adjacent to the coolant loop. Radiation surveys have been made at reactor powers of 50 and 100 kw to determine the adequacy of the shielding in areas where personnel access is required during reactor operation, i. e. , the control room and block storage (Figs. 39 and 40).

8.2 Radial Shielding Toward Control Room

In the radial direction the shielding between the vessel and the control room consists of 3-1/2 ft of borated graphite plus 8 in. of lead plus 5-1/2 ft of normal concrete. The borated graphite shield is made of graphite cylinders 3-5/8 in. diam by 72 in. long containing an average of 0.4% by weight natural boron. Forced air cooling removes the 25 kw of heat which, it is estimated, will be deposited in the shield at 1 Mw operation. The graphite in the high flux region will be maintained at 150 °C to 200 °C to minimize radiation damage.

Calculations indicate that at 1 Mw operation the thermal neutron flux at the outer radius of the borated graphite shield will be $\sim 10^8$ neutrons/cm²-sec; the calculated flux in the energy range between thermal and 0.5 Mev is 2.7×10^9 neutrons/cm²-sec. The neutron flux above 0.5 Mev is 1.5×10^8 neutrons/cm²-sec.

Fig 39. Horizontal Cross Section Showing Reactor Shielding

Fig 40. Vertical Cross Section Showing Core Shielding

Results of measurements at a reactor power of 100 kw indicated that an additional decade of neutron and gamma ray attenuation at a power level of 1 Mw was required to achieve allowable dose rates in the control room. The shielding in some of the penetrations through the concrete wall was also found to be inadequate. The shielding in these penetrations has been modified to provide additional attenuation; following this, the dose rates in the control room were measured and supplementary lead shielding installed to reduce the radiation levels to below tolerance values.

A removable neutron shield consisting of normal concrete blocks is placed in the entrance to the reactor cell. At 1 Mw the gamma ray and fast neutron dose rates will be about 200 mr/hr and 500 neutrons/cm²-sec respectively. During operation, access to the area south of the reactor cell and sodium equipment room (Fig. 39) will be allowed only with the approval of health physics personnel.

8.3 Shielding Below Reactor

The main shield directly below the core consists of 16 in. of iron (bottom reflector) plus a 7 ft long floor shield plug. The bottom 2 ft of this plug is filled with lead shot; the remaining volume is filled with magnetite aggregate having a density of 230 lb/ft³. The shield plug is provided with holes through which the shim and control rods are moved.

During reactor operation the subbasement is part of the exclusion area and can be entered only with the knowledge and consent of both the reactor supervisor and health physics personnel. The radiation levels measured at the platform in the subbasement (Fig. 40) were:

Reactor power	=	100 kw
Gammas	=	15 mr/hr
Fast neutrons	=	750 neut/cm ² -sec
Thermal "	=	3500 neut/cm ² -sec

The gamma ray dose rate directly below the reactor, for 3-month operation at 1 Mw and immediately after shutdown, is calculated to be 2 mr/hr. Dose rates in those areas not shadowed by the 6-in. lead shield on the subbase-ment ceiling (Fig. 40) are estimated to be as high as several hundred mr/hr. Since the control mechanisms are located directly below the reactor, no difficulty in maintenance of these components after shutdown is anticipated. The control system hydraulic pumps are located at a point where inspection and maintenance are possible during full power operation.

8.4 Shielding Above Reactor

The fuel capsule handles provide ~ 9 ft of iron shielding directly above the core. The shielding between the reactor cell and the rest of the fuel transfer area is 5-1/2 ft of normal concrete or the ceiling shield plug. The latter consists of a cylinder 47 in. o.d., 17-1/4 in. i.d., and 50 in. long, with ferrophosphorus concrete in the annular region. The inner 3-1/4 in. thick annulus, with an o.d. of 17-1/4 in., is filled with stainless steel shot.

From results of a radiation survey at 50 kw it was estimated that an additional neutron attenuation of 10^3 to 10^4 would be required to reduce the levels outside the fuel transfer area to below tolerance values. This additional attenuation was provided by (a) a laminated iron-masonite shield (alternate 1-in. layers) on top of the shield plug and on the floor of the fuel transfer area and (b) a concrete shield around, and on top, of the fuel transfer area. A 12 in. thick iron-masonite shield, equivalent to about two decade lengths for fast neutrons, on the floor of the fuel transfer area was installed. The thickness of concrete required was determined after a radiation survey taken with the iron-masonite shield in place.

8.5 Coolant Loop Shielding

The shielding for the loop is shown in Figs 39, 40, and 41. The total volume of sodium in the reactor system is $\sim 7\text{-}1/2\text{ ft}^3$ (3 ft^3 in the sodium equipment room piping, 2 ft^3 in the piping from the vessel to the sodium equipment room, 1 ft^3 in the heat exchanger, and $1\text{-}1/2\text{ ft}^3$ in the vessel). The Na^{24} equilibrium activity induced in the sodium is calculated to be 6 mc/cm^3 . In estimating the radiation levels in the vicinity of the sodium system, the coolant loop was divided into four parts, each part considered as an independent source. The four sources considered were the sodium equipment room piping, the surge tank, the heat exchanger, and the dump tank.

Radiation levels were calculated for the block storage and boiler room areas and for the sodium equipment room. Typical values for the gamma ray dose rates above the sodium equipment room ($2\text{-}1/2\text{ ft}$ normal concrete ceiling) are 7 mr/hr with 6 in. of steel and 40 mr/hr with 2 in. of lead. Dose rates 1 ft from the 8 in. lead walls at the north and south ends of the sodium equipment room are 50 mr/hr .

The 40 mr/hr dose rate estimated for the boiler room is considered acceptable, because routine boiler servicing requires that personnel enter the room for only a few minutes during each 8-hour shift.

From data obtained following 50 kw operation for 60 hr , it is estimated that the maximum dose rate in block storage, due to Na^{24} activation at 1 Mw , will be $3\text{-}4\text{ mr/hr}$. These data also imply a maximum dose rate of 10 mr/hr in the boiler room. A check on these figures will be made at a power level of 200 kw . It should be noted that the usage of the block storage area is such that radiation levels several times tolerance are acceptable.

Fig. 41. Plan View of Shielding on Floor in Area Above Reactor

9. CONTROL SYSTEM

9.1 General

Reactivity adjustment of the reactor is done by means of control elements consisting of five removable reflector sections. Although the five control elements serve as shutdown devices, the fast shutdown or scram is accomplished by dropping the four control rods.

The shim, shown in Fig. 42, is the largest of the control elements and has the form of an annular cylinder concentric with the reactor vessel. It is made of Type 430 stainless steel 20 in. o.d., 10-3/4 in. i.d., and 16 in. long. It may be raised to a position surrounding the core or lowered to a position where its effectiveness as a reflector for neutrons escaping from the core is negligible. The differential worth of the shim over the range of its 16 in. travel is $\sim \$23$.

Adjustments to reactor criticality finer than those affected by shim motion are made by moving four control rods; these consist of high-purity nickel cylinders 3.8 in. o.d. x 16 in. long, which move vertically in appropriate channels provided in the shim. The maximum range of travel of these control rods is 16 in.

The entire control element assembly, together with the bottom vessel shield, is supported below the reactor by the floor shield plug. Figure 43 is a cross section through the complete control assembly. Drive mechanisms for the control elements are located in the subbasement below the reactor cell and are attached to the bottom of the shield plug. All parts of the drive mechanisms are accessible in the subbasement. Figure 44 shows the shim assembly with the shim at mid-position.

Fig. 12. Control Shim

Fig. 43. Section Through Control Assembly

Fig. 44. Shim Assembly

The bottom vessel shield, which is rigidly attached to the floor shield plug, locates the bottom of the reactor vessel laterally. Alignment of the shim with the vessel is accomplished by the shim actuating members, which pass through the bottom vessel shield and the floor shield plug. These rods connect the shim to its actuating unit.

Vertical sections of 4 in. i.d. stainless steel tube, rigidly attached to the bottom vessel shield, pass through the four channels in the shim. The control rods move inside these tubes and are guided by them.

Drive units for the shim and the control rods are double-ended hydraulic cylinders actuated by suitable hydraulic systems. The use of double-ended cylinders (within piston rods extending out of both ends of the cylinders) permits gauging the position of the control elements by means of indicator mechanisms attached directly to the lower end of the piston rods.

Two separate, low pressure hydraulic systems are used; each has its own pump. One system supplies oil to the shim drive cylinder and the other supplies oil to the four control rod actuating cylinders. Both pump systems are fitted with 250 psi rupture discs in the pump output lines. These prevent dangerous overpressuring of the drive units in the event of a pressure regulation failure. The hydraulic system for the shim is indicated schematically in Fig. 45 and 46 shows the corresponding circuit for a control rod.

9.2 Control Element Drive Systems

The shim drive system is operated from the console in the control room. Console switches control relays, located in a restricted access area, for energizing the necessary valves and the interlock circuits. Relay power is supplied through a locked switch on the console; the key may be removed from the lock only if the switch is off.

All valves operable from the console are of the solenoid-controlled, pilot-actuated type. Both normally open and closed valves are required in order to make the hydraulic system fail-safe: a power failure can result only

Fig. 45. Schematic of Shim Hydraulic System

Fig. 46. Schematic of Control Rod Hydraulic System

in a shim or rod motion that reduces reactor criticality. In no case will valve leakage result in addition of reactivity.

By means of switches on the reactor console the operator can control the shim through a four-way valve that admits oil (from the continuously operating supply pump) to the bottom or the top of the shim drive cylinder to produce upward or downward motion of the shim. Shim position may be changed in increments that can be read to 0.010 in.

The shim may be moved at either of two speeds. The operator selects the desired speed by appropriate manipulation of the console shim control switch. Speed control is done by regulating oil flow to the cylinders through two parallel lines. Each of these lines is fitted with an adjustable throttling valve and a solenoid-controlled, flow shut-off valve. If the slow rate of shim motion is desired, the slow speed channel is opened; the flow of oil in it occurs at a rate fixed by the setting of the throttling valve. When fast rate of motion is required, both oil channels are opened; the total flow rate is then dependent upon the setting of both throttling valves. The throttling valves are preset to give flow rates compatible with the desired maximum rate of reactivity addition. These rates (slow and fast) cannot be readily altered by the operator, because the throttling valves are located in a locked area remote from the control room.

Shim position is indicated on a console-mounted digital register reading to 0.010 in. The register is driven through synchros by a gear train actuated by the shim cylinder piston rod. Electrical power for the synchros system is independent of main control console power and is left on continuously whether the reactor is in operation or not. The occurrence of power failure (no voltage) in the synchros system is indicated on the control room annunciator panel.

In its up position the shim is in contact with a mechanical stop. When contact is made with this stop a limit switch is actuated controlling the valves necessary to maintain pressure continuously to the shim drive cylinder holding the shim up against this stop. Should the shim be lowered as little as 0.02 in. (either by the operator or through system malfunction) the limit switch is deactivated. The shim cannot be reinserted except in accordance with the normal sequence startup procedure.

The shim stop was designed so that variations in reactor temperature would not significantly affect the relative position of the shim to the core. This was done by limiting the upward shim motion with a stop abutting on the bottom of the vessel (Fig. 43). As the vessel temperature increases, the core and the bottom of the vessel move downward because of vessel thermal expansion. By means of the shim stop mechanism the shim is pushed down against the hydraulic system pressure by very nearly the same amount as the change in the core position. With the compensation provided by this shim stop design, the change in reactivity due to shim movement that can be produced by temperature variations in the vessel should not exceed $0.1\beta/^\circ\text{C}$. The precise value of a possible positive contribution to the overall temperature coefficient of reactivity has not been ascertained.

The position of the shim stop is adjustable in increments of 0.025 in. over an 8 in. range. This range of settings for the up position has been used to accommodate variations in core loading and also during intercalibration of the shim with the control rods. Adjustment of the stop is performed in the restricted access subbasement under the reactor cell. Design of the assembly is such that changing the shim stop setting automatically repositions the up limit switch (Fig. 43).

The reactor is scrammed by rapidly lowering the control rods. This is done by opening scram valves that are located in lines connecting the top and bottom ends of the rod drive cylinders. When the valves are opened the differential pressure across the piston is reduced to essentially zero, and the weight of the rod and its supporting members (approximately 400 lb) forces the piston down. Rod position as a function of time after the scram signal appears at the solenoid valve is shown in Fig. 47.

Because the scram valves are the normally open type, electrical power failure in the control circuits automatically scrams the rods. An electrical power failure (or a scram signal) results in a valve configuration that allows oil to escape from the bottom of the cylinders and return to the oil storage reservoir. This mechanism contributes slightly to the rate of rod drop that follows a power failure or a scram signal, but oil flow through the scram valves is by far the more important factor in determining rod drop time.

Fig. 47. Rod Position vs Time After Scram Signal

A gas-ballasted accumulator is fitted to the rod drive hydraulic system; the oil in this accumulator is held at pump pressure so long as the pump is operating. Should the pump fail, this accumulator continues to store oil at pressure equivalent to that put out by the pump until it is used. Should a scram valve fail to open in case of a power failure or the receipt of a scram signal, or should a rod fail to drop freely during a scram, this accumulator pressure would be applied to the top of the rod drive piston. The resulting force tending to lower a rod will then be about twice that due to the weight of the rod assembly. If the scram valve sticks in the closed position, this extra force on the top of the rod piston will be sufficient to lower the rod at a rate about equal to that normally obtainable by turning a console rod control switch to fast-down.

Several control system notations are:

- (1) There is no provision for scrambling the shim. Power failure does, nevertheless, result in the shim being withdrawn as oil is forced from the drive cylinders by the weight of the shim assembly.
- (2) The stops that determine the up positions of the rods are not adjustable.
- (3) There is no provision for temperature compensated stops of the type used for the shim, because control rod reactivity worth is much less than that of the shim.
- (4) A rod fully inserted to the up position is held there by continuously supplied hydraulic system pressure, just as in the case of the shim. However, at partial insertion a rod is held up by means of a closed lock-up valve in the line leading from the bottom end of the control rod hydraulic cylinder. The positions of other control valves in the system are such that should the lock-up valve leak, the control rod will be withdrawn, rather than inserted.
- (5) Rates of rod motion have been adjusted so that reactivity insertion rates will be about $2\text{¢}/\text{sec}$.

- (6) Control rod positions are displayed on digital registers reading to 0.010 in.

9.3 Sequencing and Interlocking

In the interest of safety, electrical interlocks have been incorporated into the control system to enforce certain restrictions on operational procedures. Each of the control elements is furnished with up and down limit switches that provide a visual indicating at the console of the location of the particular element in question at the time. There are three indicator lights for each control element. A green light indicates that the element is fully down, an amber light indicates that the element is in an intermediate position, and a red light indicates that the element is in a full up position. The control system interlocks may be grouped as follows:

- (1) All control rods must be in the down position before the shim can be raised from its lower limit.
- (2) The shim must be in its fully up position before the first control rod can be brought off its lower limit.
- (3) The control rods can be raised only one at a time and then only in the following sequence: 1, 2, 3, 4.
- (4) 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.

10. SAFETY INSTRUMENTATION

10.1 Scram System

Scram channel signals will produce reactor shutdown by causing the four control rods to drop quickly to the down or least reactive position. The rods drop a distance of 12.5 in. in approximately 1-1/2 sec. A scram signal also causes an automatic withdrawal of the shim at a rate of ~ 10 in./min. The origin of the scram signal is displayed on an array of lights at the operator's console. The conditions that will initiate a scram are:

- (1) Period
- (2) Neutron level 1
- (3) Neutron level 2
- (4) Manual
- (5) Low flow sodium
- (6) High temperature air - heat exchanger exhaust
- (7) Ten-inch slide valve open
- (8) High neutron level
- (9) Low temperature sodium - vessel inlet
- (10) High temperature sodium - vessel outlet
- (11) High temperature sodium - heat exchanger outlet
- (12) Smoke - sodium areas
- (13) High radiation - sodium areas

The period scram signal is obtained from the period amplifier section of a Keithley log-n period amplifier. This instrument consists of a logarithmic micro-micro-ammeter for indicating reactor power level, followed by a

differentiator for measuring the period. The manufacturer's specifications state that a 10 v output is developed by the period circuit; the output voltage is zero for an infinite period and plus 10 v for a 3 sec positive period. This voltage is fed to a sensitive relay mounted on the log-n chassis. The relay pull-in voltage is set in the vicinity of 7 v; consequently, the reactor is scrammed if the period is less than about 4 sec. This period scram channel may be bypassed to prevent spurious scrams when the reactor is operating at a constant power level.

The neutron level No. 1 and 2 scram signals are obtained from B¹⁰ lined, uncompensated ionization chambers feeding linear amplifiers. These instruments (2) are Keithly micro-micro-ammeters provided with 20 linear current ranges from 10×10^{-4} to 3×10^{-13} amp full scale. Range selection is accomplished by means of panel switches on the front of the instruments. Each of these electrometers has a contact meter relay with an adjustable trip-set that can be made to trip at neutron levels corresponding to ionization chamber currents from 10×10^{-4} to 3×10^{-13} amp.

The manual scram signal is obtained by pressing one of three scram buttons. One is located on the operator's console and the other two are located on the racks, one to the left and one to the right of the console.

The low flow sodium scram signal is obtained if the sodium flow is below a preselected minimum level. The signal for this channel is obtained from a mercury switch trip upon the 3 in. flowmeter recorder. The recorder reads from 0 to 10 Mv full scale, with one Mv equal to about 25 gpm. The trip signal for this channel is normally set at about 4 Mv or 10 gpm.

The high temperature air-heat exchanger exhaust scram signal is obtained if the temperature of the air in the heat exchanger exhaust is above a preselected value. The signal for this channel is obtained from a contact meter relay driven by a thermocouple located at the heat exchanger air exhaust duct.

The 10-inch slide valve open scram signal is obtained if the valve that isolates the vessel from the capsule charger is opened when the reactor is operating.

The high neutron level scram signal is intended for use during reactor operation at approximately 25% of design power and above. The high neutron level circuits consist of three Minneapolis-Honeywell safety trip units-Model 1910; each of these units receives a current from an ionization chamber identical to those used in neutron level No. 1 and 2 scram channels. A two out of three channel trip scheme is employed: two of three measuring and trip channels must generate trip signals, not necessarily simultaneously, in order to initiate a scram.

The low sodium temperature-vessel inlet scram channel signal is initiated provided the sodium system temperature is reduced below a preset point. This signal is obtained from an adjustable limit switch on the vessel sodium inlet temperature recorder.

The high temperature sodium-vessel outlet scram is obtained from an adjustable limit switch on the vessel outlet sodium temperature recorder. The temperature at which a scram signal is initiated is determined by the maximum allowable fuel temperature.

A scram is initiated if the heat exchanger outlet sodium temperature becomes too high. The signal is derived from a contact meter relay driven by a thermocouple located in the heat exchanger outlet sodium pipe.

Four smoke detectors, located in the sodium areas, provide a scram signal. Radiation detectors, placed in the sodium areas to detect radioactive sodium smoke, feed contact meter relays that, in turn, are connected to the scram circuits.

If a scram occurs during normal operation, an annunciator will alarm and a red light marked scram will be illuminated on the console. A pair of indicator lights, the scram channel lights, is associated with each channel; one of these lights is marked scram, the other OK to reset. Thus, in case of a scram the operator can look at the bank of scram channel lights and determine the reason for the scram. When the condition that caused the scram has been rectified, the OK to reset light will come on, and the reactor can be reset and again brought to a critical condition.

10.2 Annunciator System

In addition to the scram channels there is an annunciator system, which warns operating personnel (by visible and audible signals) of low urgency, abnormal conditions in the reactor system. These conditions are:

- (1) Power circuit de-energize (Any breaker in the reactor load center located outside TSL-2 will activate this channel.)
- (2) High gamma radiation levels at 7 locations in the reactor areas
- (3) Low pressure instrument air
- (4) Graphite shield temperature above a preselected value
- (5) Sodium leakage from primary vessel or coolant system piping
- (6) Open door between electrical equipment room and corridor to reactor cell
- (7) Temperature of main loop heating transformer above a preselected value
- (8) Poor vacuum in gas disposal holdup tank
- (9) Sodium temperature at heat exchanger outlet below a preselected value
- (10) Improper sodium level in surge tank
- (11) Improper cover gas pressure
- (12) Improper voltages on neutron level detectors
- (13) Radiation monitor failure
- (14) Contact meter relay chassis power too low
- (15) Smoke detector power supply voltage too low
- (16) Low pressure in shim hydraulic system
- (17) One of the Minneapolis-Honeywell 1910 units tripped
- (18) High sodium flow
- (19) Heat exchanger intake vane fully open
- (20) High temperature in the electromagnetic pump windings
- (21) Loss of cell blower power

- (22) Water level in wash pit too high
- (23) Low control-element synchro voltage
- (24) Receipt of scram signal
- (25) The period scram channel bypassed
- (26) Smoke in the sodium areas
- (27) High radiation in the cover gas system

10.3 Area Radiation Monitors

A system of radiation monitoring is installed for the protection of personnel. This system consists of ten remotely located radiation detector heads that feed information pertaining to the radiation level to a centrally located unit in the control room. All ten area monitor signals are recorded on the multi-point Brown recorder 9-3. Each of these channels has its own contact meter relay that can be preset at a level consistent with the area involved. When anyone of seven of these channels is tripped, an annunciator signal will occur. The other three channels will initiate an annunciator signal and in addition, will start a reactor scram. Each of the ten monitor channels is supplied with an audible alarm at the detector location; consequently, personnel working in these areas will be warned if an abnormally high radiation occurs in their area.

11. CONTROL ROOM

11.1 Distribution Box Layout

Because most of the control and instrumentation signals required for operation of the reactor system originate in many areas of the installation, a scheme employing two terminal junction boxes at each of these locations with matching boxes located in the control room is utilized for cable runs. One terminal junction box is used for low level DC instrumentation signals and the other is used for AC control signals. All cable runs (AC and DC) are physically separated to further reduce the possibility of AC electrical interference problems arising. This scheme has resulted in a total of 28 junction boxes being used in the system. The wiring used in the system consists of 203 thermocouple pairs, 1050 conductors, and 100 co-ax cable and others.

11.2 Rack Layout

The control room has 9 standard electrical equipment racks and 3 console type racks as shown in Fig. 48. The 4 racks to the left of the console contain the chassis that are utilized in the control of the coolant system.

The console contains the displays and controls that are most pertinent to the operation of the system. The left hand console contains the main electromagnetic pump and heat exchanger blower motor controls. The center console contains the controls and indicators for the shim and control rods, and the right hand console contains the visual displays for the scram channels. Located just

Fig. 48. Control Racks

above these three consoles are the recorders for the vessel outlet temperature, linear power level, and main loop coolant flow. Visual displays for the annunciator system are located just above these recorders.

The first racks to the right of the console contain the nuclear instrumentation, which includes power supplies, amplifiers, scaling circuits, and recorders. The remaining racks contain the area radiation monitoring circuits.

The opposite side of the control room contains one rack of electrical equipment that is used for the cover gas system. The equipment includes controls for motor-operated valves, pressure indicators, and heating controls. Adjacent to this rack is a bank of manually-operated controls for valves located in the reactor cell.

Adjacent to the control room is the electrical equipment room, which houses the variable transformers for heater control. Other electrical equipment in the room includes circuit breakers for the sodium system heater circuits, electromagnetic pumps, and control racks.

12. ELECTRICAL POWER DISTRIBUTION SYSTEM

Ten Site is served by two main independent 13.2 kv lines, designated S-3 and S-9 as shown in Fig. 49. S-9 feeds the transformer in Building TSL-2, and S-3 feeds the substation outside the new Test Pit Building, TSL-29 (Fig. 26).

The reactor is normally operated on S-9 and the transformer and breakers in TSL-2. The electrical load center for the reactor is located in the cabinets on the south exterior wall of TSL-2 above the cell access hallway. This load center is fed by a 480 v breaker, P-7, located in the north end of the basement of TSL-2.

If the S-9 line goes out of service, there is automatic transfer of the S-3 line and its transformer: the entire reactor load is transferred to the secondary of a different transformer. This transfer between lines takes place in about 30 sec. As soon as the S-9 line is restored, the system is automatically switched from S-3 to S-9.

A chassis located in the instrument rack in the control room contains lamps that indicate the existence of an electrical potential at various points in the circuits. Thus, if a breaker in the load center should open for any reason, the appropriate lamp indicates the lack of voltage at that point. The opening of a breaker will also result in an annunciator signal providing a visual and audible alarm to control room personnel.

A 75 kv generator is available to furnish power to a portion of the reactor electrical loads in special circumstances, such as a planned power outage of both the S-3 and S-9 lines. This generator will not automatically pick up loads if both S-3 and S-9 should fail.

Fig. 49. LAMPRE I Power Distribution

Appendix A

CAPSULE TESTS

Test results of particular interest to the LAMPRE I program are:

- (1) Thimbles fabricated from ingot K574 (Temescal beam-melted tantalum ingot containing 0.1 w/o tungsten) in Core I were annealed at 1300 °C. Most of the fuel is made with cast iron plus additions of stabilizer and PuC placed in the capsules with the fuel loading. Test information shows that this particular capsule-fuel combination operates without leakage from 2100 hr at 575 °C, but is definitely unsuitable for operation at 625 °C. Other inferior capsule-fuel combinations have also performed for 2100 hr at 575 °C after an initial 2500 hr at 525 °C.
- (2) Leakage location, as well as time required for leakage, at 625 °C is dependent on thimble annealing temperature. 1050 °C annealed thimbles show leakage below the fuel level at 700 hr. Those annealed at 1200 °C and 1350 °C perform longer and show leakage first at the fuel level and then in the gas phase. If 1050 °C annealed thimbles are run at 525 °C and 575 °C for several hundred hours and then run at 625 °C, leakage is also confined to the fuel level and gas phase. The use of stabilized fuel without additives or a 1450 °C annealing temperature has eliminated this effect thus far.
- (3) Much of the fuel in Core I contains calcium and magnesium as impurities. 1300 °C annealed thimbles containing fuel with

additions of these elements have been tested. Leakage at 625°C occurs sooner when these impurities are present.

- (4) K574 thimbles annealed at 1450°C perform far better than those annealed at lower temperatures. These thimbles have been operated for 2700 hr at 625°C plus 700 hr at 675°C without leakage.
- (5) Thimbles made from beam-melted unalloyed tantalum show promise for additional core replacement stock. Those annealed at 1200°, 1400°, 1600°, 1800°, 2000°, and 2200° C have been operated for 2100 hr at 625°C plus 700 hr at 675° C without leakage.
- (6) At 625° C, prestabilized fuel without additives proves considerably less corrosive than fuel with added PuC. This fuel is also satisfactory at 675°C thus far. A few capsules containing prestabilized fuel have been operated in LAMPRE I. This fuel does not show separation as observed in fuel columns containing PuC.

The following general information concerning capsule and fuel behavior is noted:

- (1) The use of stabilized fuel has thus far eliminated melt-freeze damage. The result is important since gross leakage is the end product of the damage.
- (2) At 625° and 675° C, even the most severe intergranular corrosion noted to date requires alpha counting or autoradiography for detection. This leakage has not been observed visually. (Certain special tests run at 750° C in vacuum, helium, and sodium have produced visible leakage, but fuel losses appeared slight and capsules remained intact.)
- (3) Leakage has occurred mostly at the fuel level and in the gas phase. Leakage has also occurred at bubbles trapped in capsule tips or contacting walls. The influence of impurities either entering the fuel or there initially cannot be overlooked. Certain fission products may similarly influence capsule corrosion behavior.

- (4) The search for a fuel additive to reduce temperature gradient mass transfer without increasing intergranular attack is continuing. Fuel alloyed with silicon has not penetrated 1450°C annealed thimbles as yet, but a definite segregated phase has been detected radiographically.

Appendix B

LAMPRE I CRITICAL EXPERIMENTS

In support of the LAMPRE I program, two critical experiments were performed at the N Division (Pajarito Site) remotely controlled critical assembly facility. LCX I (LAMPRE Critical Experiment No. 1)² mocked up an early LAMPRE I design. Since, in the development of the final LAMPRE I design, extensive modification was made of the core and other components, the need arose for additional critical experiments. These experiments (LCX II) were performed using the machine depicted in Fig. B. 1.

The LCX II core consists of approximately 24 sandwiches, each of which contains four 5.94 in. o.d. disks of the following materials:

Material	Thickness (in.)
Pu (clad with 0.005 in. Ni)	0.122
Al (perforated to make number of Al atoms per cm ³ equal to atomic density of Na at 560 °C)	0.075
Ta	0.022
S.S.	0.014

As seen in Fig. B. 1 the core is reflected on all sides by iron, which was more readily available in the required shapes than the stainless steel selected for use in LAMPRE I. The side iron reflector consists of a 3-in. thick fixed reflector situated adjacent to the core, and a 3-in. thick, 12-in. long, movable reflector surrounding the fixed reflector.

Fig. B.1. Schematic of LCX II

This movable reflector serves as the reactivity control shim. Critical masses measured over the 12-in. length of shim travel are shown in Fig. B.2. It is seen that the shim worth is 1050 g, or $\sim \$12.4$ on the basis of the measured mass-dollar relation:

$$\frac{\Delta m}{m} / \$1.00 = 0.41\% / \$1.00; \frac{\Delta k}{\Delta m} = 1.18\text{¢} / \text{g fuel alloy}.$$

As a consequence of this measurement of shim worth, both the thickness and length of the LAMPRE I shim were increased (over the LCX II dimensions) in order to obtain the desired amount of shutdown Δk . Similarly, the worth of the LCX II vernier (control) rod, an independently actuated small section of the shim, had a measured worth of 30¢. The control rods in LAMPRE I are larger than those in LCX II and are made of nickel to further increase their effectiveness.

The shield in LCX II consists of the same type graphite cylinders as are used in LAMPRE I; however, only 18 in. of the nominal LAMPRE I 3-1/2 ft shield thickness is used in LCX II, since it appeared this amount would serve to evaluate the shield contribution to reactivity. This contribution was measured to be \$8.

Other experiments performed on LCX II include:

- (1) An examination of the effect of plenums situated at the ends of the core.
- (2) A measurement of the critical mass of the core containing only the nickel-clad plutonium disks, and no diluent disks.
- (3) A comparison of the effectiveness of stainless steel, nickel, and copper as control rod materials. (A relative effectiveness of 1:1.62:1.68 was observed for equal volumes of stainless steel, nickel, and copper.)
- (4) An investigation of the effect of increasing the density of the core plutonium. (A 1% increase in volume-averaged plutonium density decreased the critical mass by $\sim 1\%$.)
- (5) The examination of several oscillator rod designs.
- (6) The testing of the transfer function analyzer equipment which is designed for use on LAMPRE I.

Fig. B. 2. Critical Mass vs Shim Insertion, LCX II

Appendix C

FUEL ELEMENT BOWING

C.1 Introduction

Data from LCX II lead one to expect a power density in the LAMPRE I core which is well approximated by the expression

$$P(r, z) = P_o \cos \left(\frac{r}{3.06} \right) \cos \left(\frac{z - 0.5 H}{3.34} \right) \quad (C.1)$$

where r , z , and H are in inches. Here P_o is the power density at the center of the core, H is the core height, and the origin of the cylindrical coordinate system is taken at the center of the bottom of the core. If we let $\theta(r, z)$ represent the temperature rise in the coolant above that entering the bottom of the core,

$$\theta(r, z) = \theta(O, z) \cos \left(\frac{r}{3.06} \right). \quad (C.2)$$

This leads to a radial temperature gradient

$$\theta'(r, z) = \frac{-\theta(O, z)}{3.06} \sin \left(\frac{r}{3.06} \right). \quad (C.3)$$

Taking $\theta(O, H) = 147^\circ\text{C}$ ($\theta_m = 113^\circ\text{C}$ = mean rise through core) as representative of nominal LAMPRE I operating conditions, one obtains the values given in Table C.1.

TABLE C.1
NOMINAL CORE EXIT CONDITIONS
($\theta_m = 113^\circ\text{C}$)

r (in.)	0	1	2	3
$\theta(r, H) (^\circ\text{C})$	147	139	117	92
$\theta'(r, H) (^\circ\text{C}/\text{in.})$	0	-15.4	-29.4	-39.8

The radial temperature gradients of Table C.1 will be superimposed upon the local temperature variations in the molten fuel, tantalum capsules, and coolant stream. Specifically, any particular tantalum capsule will be hottest nearest the core center line and will, therefore, bow inward with a curvature C and radius of curvature R_c given by

$$C = \frac{1}{R_c} = \alpha \theta' \quad (\text{C.4})$$

where α is the thermal expansion coefficient of the metal.³

From Table C.1 it is noted that θ' is roughly proportional to radius. Hence, capsule curvature and deflections toward the core center line are also about proportional to radius, leading to approximately uniform density variation over the cross section (at a given value of z) as bowing occurs.

The tantalum capsules are taken to be simply supported at their lower ends, where their conical tips fit into mating holes in the locator plate. Their tops are rigidly attached to long 17-4 PH stainless steel handles which are themselves subjected to radial temperature gradients over part of their lengths. The round part of the handle is in contact with a flowing sodium coolant, in which the radial gradient at the core exit may be assumed to exist. The hexagonal part of the handle is bathed by stagnant sodium, so it may be assumed to be essentially free of radial temperature gradient and bowing. When assembled at room temperature, the handles are expected to fit snugly against each other at

the spline section just above the core. The spline metal (17-4 PH stainless steel) has a slightly higher temperature coefficient of expansion ($11.9 \times 10^{-6}/^{\circ}\text{C}$) than that ($11.0 \times 10^{-6}/^{\circ}\text{C}$) of the Type 430 stainless steel flow divider surrounding the fuel element array, so that at operating temperature the splines become slightly tighter (by ~ 3 mils differential across the entire core). If we assume the cumulative clearance across the core diameter (at the splines) at room temperature to be 0.010 in., the expected clearance at operating temperature will be 0.007 in. across the core, or about 0.0005 in. per gap.

The nominal clearance per gap at the lower ends of the hexagonal parts of the handles is 0.017 in., some 30 times that at the spline section. Because the long handles will not be perfectly straight, actual gaps existing at either spline or hexagonal sections will be subject to large variations from nominal values. The lower ends of the hexagonal section have about 10 times the nominal separation of the splines, and they are only about 3 times as far from the pivot points (tantalum capsule tips). Hence, the splines are more likely to provide the constraint of warped fuel element assemblies than are the lower hexagonal sections.

Maximum bowing reactivity rate (β/kw) is produced if each and every fuel element is free of constraint except at its lower end (conical tantalum capsule tip) and upper end (top locking section). In the calculations reported below, the splines are assumed to be initially free of constraint and in their nominal positions. This assumption leads to maximum rate, but largest total bowing reactivity would result if one assumed all splines to be initially pressing outward against the core container. In the latter case the total reactivity due to bowing would be somewhat larger than that calculated below, but the rate (β/kw) would be about the same.

With the splines free of constraints at the start, radial temperature gradients far less than those of nominal operating conditions of Table C. 1 are sufficient to displace the spline sections to the extent that all splines will be in contact. For temperature gradients above those required for spline constraints to act, bowing effects are much smaller. Under these conditions, actual bowing becomes a complicated function of section moduli and elastic constants in the

entire fuel element assembly. If one considers the inward bowing that would occur in a simple fuel capsule simply supported at both its lower tip and at the spline section, the inward bowing will be overestimated, since the bowing of the handle will certainly tend to transmit through the spline section an effect at least partially opposing the inward bowing so calculated.

C.2 Results

Estimates of bowing have been made using LAMPRE I constants under the two assumptions outlined in the preceding paragraph. In addition, the following assumptions have been made: (a) Vertical variation in power density has been neglected so that coolant temperature is assumed to rise linearly with height from bottom (inlet) to top (outlet) of the core. This means that in Eq. C.2

$$\theta(O, z) = \frac{z}{H} \theta(O, H), \quad 0 \leq z \leq H. \quad (C.5)$$

(b) Radial temperature gradient is assumed to follow Eq. C.3 within the core. The fuel element handles up to the top of the round section are assumed to be subjected to the same radial gradient as that existing at core outlet plane. (c) The long hexagonal sections of the fuel handles are considered to be subject to no radial gradient and to remain straight extensions of the round handle sections. (d) Results are expressed in terms of average coolant temperature θ_a , the coolant temperature rise (above entrance temperature) averaged over the entire core--that is, half the temperature rise θ_m of the mixed outlet coolant stream. For example, the average coolant temperature θ_a for the nominal operating conditions of Table C.1 would be $\theta_a = \theta_m/2 = 113^\circ\text{C}/2 = 56.5^\circ\text{C}$. (e) Dimensions (measured from lower end of tantalum capsule) and materials data used in the calculations are as given in Table C.2 (f) Reactivity worths are based on the experimental LAMPRE I factor of \$13 per 1 g/cc fuel density change. This corresponds to +210¢ for +1% density change or +420¢ for -1% $\Delta R/R$ at constant core height.)

TABLE C. 2
DIMENSIONS AND MATERIALS OF FUEL ELEMENTS
AND FLOW DIVIDER

Range (in.)	Part	Material	Expansion coefficient α
0 - 6	core capsule	Ta	$6.6 \times 10^{-6}/^{\circ}\text{C}$
6 - 9.75	core to spline	S.S. + Ta	$9.1 \times 10^{-6}/^{\circ}\text{C}$
9.75 - 32	round handle	17-4 PH (S.S.)	$11.9 \times 10^{-6}/^{\circ}\text{C}$
32 - 123	hexagonal handle	17-4 PH (S.S.)	$11.9 \times 10^{-6}/^{\circ}\text{C}$
	flow divider	430 (S.S.)	$11.0 \times 10^{-6}/^{\circ}\text{C}$

A summary of the results is given in Table C. 3. The positive power coefficients listed have been broken down into (a) a slightly delayed component due to tantalum capsules bowing in the core, and (b) a delayed component due to bowing of the handles. The first component, effective at all values of θ_a , has the value $+0.026\text{¢/kw}$. The second component is present only up to $\theta_a = 1.8^{\circ}\text{C}$, and has the value $+0.408\text{¢/kw}$.

TABLE C. 3
SUMMARY OF BOWING-EFFECTS ESTIMATES

Range of interval θ_a ($^{\circ}\text{C}$)	P (kw)	Mils displacements at R = 3 in.			Reactivity effects		Full flow power effects ($\Delta k/\Delta P$ in ¢/kw)	
		Spline ΔR	Effective core ΔR and $\Delta R/\Delta\theta_a$		Δk (¢)	$\Delta k/\Delta\theta_a$ (¢/ $^{\circ}\text{C}$) ^a	Slightly delayed	Delayed
0 - 1.8	0 - 30	-3.0	-0.92	-0.51	+13	+7.2	0.026	0.408
1.8 - 56	30 - 1000	0	-1.80	-0.032	+25	+0.46	0.026	0

It is estimated that of the -0.47¢/kw measured, the prompt power coefficient associated with fuel expansion amounts to about -0.414¢/kw (based on estimates that the average fuel temperature rise, θ_f , is equal to $3.1\theta_a$ for

nominal operating conditions). The capsule bowing coefficient is necessarily slightly delayed as compared to the prompt fuel coefficient, as the capsule bowing is driven by fuel temperature rise effects; the magnitude of this delay is about 0.1 sec. The handle bowing power coefficient is still further delayed by a time estimated to be of the order of 2 or 3 sec--because of sodium transit time effects coupled with thermal capacity of the handles.

There are other sources of negative power coefficient having delay times as short or shorter than the delayed handle bowing coefficient. Among these are density changes of the coolant in and above the core and of the fuel element handles (acting as reflecting material). These effects account for the remainder (-0.056¢/kw) of the measured coefficient.

C. 3 Discussion

A summary of the reactor power coefficients discussed above is presented in Table C. 4. It is seen that the net prompt coefficient has a large negative value at all power levels, whereas if one includes delayed effects (several seconds time constant) the overall power coefficient is nearly zero at low power. It must be remembered, however, that we have assumed the worst possible situation--namely, that all fuel elements are completely free of constraints over their entire length of ~ 10 ft.

It is likely that there will be sufficient deviation from straightness of the fuel assemblies to cause a fair number of them to be in contact at the spline section, where nominal clearances are only about 0.5 mil. Also, about half of the 0.408¢/kw contributed to the power coefficient by the handles is due to the long straight hexagonal sections acting as extensions of the round sections where the bowing occurs. This contribution will vanish to the extent that these hexagonal handles are constrained by warpage.

It should be pointed out, in conclusion, that the above analysis applies only to full sodium flow conditions. Consider, for example, the conditions at $\theta_a = 1.8^\circ\text{C}$ at reduced flow. Because the temperature rise within the fuel

TABLE C. 4

SUMMARY OF REACTIVITY EFFECTS

Range of θ_a ($^{\circ}\text{C}$)	0 - 1.8	1.8 - 56
Power at full flow (kw)	0 - 30	30 - 1000
Power reactivity coefficients (ζ /kw)		
(1) Negative		
Fuel expansion (prompt)	-0.414	-0.414
Other negative effects (delayed)	-0.056	-0.056
Total negative	-0.470	-0.470
(2) Positive		
Capsule bowing (slightly delayed)	+0.026	+0.026
Handle bowing (delayed)	+0.408	0
Total positive	+0.434	+0.026
(3) Totals		
Net total power coefficient	-0.036	-0.444
Net prompt power coefficient	-0.414	-0.414

$(\theta_f - \theta_a)$ is proportional to power level (now reduced in proportion to the flow), the fuel is now cooler than for $\theta_a = 1.8^{\circ}\text{C}$ at full flow. Whereas for full flow it was estimated that $\theta_f = 3.1 \theta_a$, we get instead $\theta_f = \theta_a$ in the limit of very slow flow. The prompt negative power coefficient due to fuel expansion is, therefore, reduced a factor 3.1 relative to all other power coefficients, the latter are, of course, inversely proportional to the flow. The bowing reactivity at $\theta = 1.8^{\circ}\text{C}$ remains at +13.0 ζ , while the compensating negative reactivity is reduced from -14.1 ζ (full flow value) to -5.7 ζ (slow-flow value). It should also be recalled that the 2 or 3 sec time constant associated with handle bowing changes in inverse proportion to the flow.

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