4.1 STARTUP OF TORREY PINES MARK III AND PUERTO RICO NUCLEAR CENTER REACTORS WITH TRIGA-FLIP FUEL, R. H. Chesworth (Gulf E&ES, San Diego, California, U.S.A.)

1. INTRODUCTION

In the continuing program to realize the fullest potential of the TRIGA reactor concept and to provide the research community with an extremely versatile tool, efforts at Gulf Energy & Environmental Systems (Gulf E&ES) have been directed over the past several years toward the development of a longer lifetime, higher-power-capability fuel which we have designated FLIP (Fuel Lifetime Improvement Program). The FLIP concept differs from standard TRIGA fuel in that the U^{235} enrichment in the fuel is increased from the nominal 20% to 70% or 93% (Table 1). In addition, natural erbium is added homogeneously to the fuel moderator matrix in order to enhance the prompt negative temperature coefficient and as a burnable poison to provide a relatively flat reactivity inventory over the reactivity lifetime of the fuel. With the 70% enrichment, reactivity lifetimes of 7-1/2 to 10 MW years are predicted; for 93% enrichment, 15 to 20 MW years are predicted.

TRIGA-FLIP fuel is available in essentially all of the geometries of the standard fuel. This paper will discuss the characteristics of TRIGA FLIP cores in two different geometries: the normal TRIGA single-rod geometry as typified by the installation in the Torrey Pines Mark III reactor; and the four-rod cluster geometry as typified by the conversion core installed in the Puerto Rico Nuclear Center reactor at Mayaguez. In both of these reactors the fuel is 8-1/2 wt % uranium, 70% enriched in U^{235}. The hydrogen-to-zirconium atom ratio is 1.5 to 1.65 and the cladding material is stainless steel. The basic neutronic characteristics of the fuel in both reactor installations will be briefly discussed.

TABLE 1

COMPARISON OF STANDARD AND FLIP*
TRIGA FUEL SPECIFICATIONS

	STANDARD	FLIP
Fuel length (in.)	15 in.	Same
Fuel composition	U-$ZrH_{1.6}$	Same
Wt % U^{235}	8.5	Same
Cladding (in.)	0.020 stainless steel	Same
U^{235} enrichment (%)	20	70 or 93
Approx single element U content (gm)	193	Same
Erbium content (wt %)	0	1.6 or 2.0

*Fuel Lifetime Improvement Program

2. STARTUP OF TORREY PINES MARK III REACTOR

2.1 GENERAL DESCRIPTION OF TORREY PINES MARK III REACTOR

The Torrey Pines TRIGA Mark III (Ref. 1) is located in a below-ground reactor tank in the TRIGA facility at Gulf E&ES laboratories, San Diego. The reactor is cooled by natural circulation of the pool water, which in turn is cooled and purified by external coolant circuits. The reactor experimental facilities included several special in-core irradiation positions, a fixed radial beam tube, a portable tangential beam tube, a portable thermal column, collimators for neutron radiography, and a variety of irradiation holders. The reactor began operation in January 1966, using standard TRIGA fuel. Prior to conversion to FLIP fuel, the reactor had a steady-state operating history of approximately 19,000 MW-hours. During most of the operating hours, the reactor was at power levels of 1.0 to 1.5 MW.

Because of the high use factor of the Mark III reactor, necessitating frequent fuel replacement, the decision was made to convert the core to the long-life TRIGA-FLIP fuel. This fuel would provide a much more economic fuel cycle and would permit extended operations without the requirement for shutdown and refueling. This is a desirable characteristic for irradiation of the in-core thermionic cells being tested in the TRIGA Mark III. The conversion to a FLIP core was preceded by an extensive and successful testing of 18 FLIP fuel elements in the TRIGA Mark F reactor (Ref. 2).

The principal design parameters of the converted reactor are presented in Table 2.

TABLE 2

PRINCIPAL DESIGN PARAMETERS

Reactor Type	TRIGA Mark III
FLIP Fuel Element Design	
Fuel-moderator material	U-$ErZrH_{1.6}$
Uranium content	8.5 wt %
U^{235} enrichment	70%
U content (avg) per element	193
Burnable poison	natural erbium
Erbium content	1.58 wt %
Shape	cylindrical
Length of fuel meat	15 in.
Diameter of fuel meat	1.434 in.
Cladding material	Type 304 SS
Cladding thickness	0.020 in.
Core Characteristics	
No. of fuel elements (2 MW)	100
Vol % water in core	33
No. of control rods	6
Total reactivity worth of control rods	8.7% $\delta k/k$
Neutron absorber	Boron in B_4C
Excess reactivity	6.1% $\delta k/k$
Core lifetime	8.7 MW-yr
Prompt negative temperature coefficient	
Averaged over fuel temperature range 20-310°C	-6.2×10^{-5} $\delta k/k°C$
Averaged over fuel temperature range 20-700°C	-10.3×10^{-5} $\delta k/k°C$

2.2 STARTUP OF THE TORREY PINES MARK III REACTOR WITH FLIP FUEL

2.2.1 Criticality

The Torrey Pines Mark III reactor was taken critical with FLIP fuel elements on June 8, 1971. A standard approach to criticality was made using the inverse multiplication technique $\left(\frac{1}{m}\right)$. The initial critical loading was 53 fuel elements. The Safety Analysis Report included an estimate of 64 elements as the critical core loading (Ref. 1). The calculations in the SAR assumed a uranium content of 8.5 wt % (70% enriched), an erbium content of 1.58 wt %, and a nominal level of hafnium in the zirconium used in the zirconium-uranium alloy. The core averaged value for uranium was actually 8.42 wt %; the core averaged value of erbium was 1.48%, and the hafnium in the zirconium used for this batch of FLIP fuel elements was only about 1/2 the value assumed as the impurity level for calculating the zirconium absorption cross section.

A recalculation of the critical core loading, with the actual content of uranium and erbium, gives a critical mass corresponding to 57 fuel elements (Ref. 3). Adjusting for the effect of the low hafnium level and the variation in accuracy of the erbium cross section (±10%) indicates a quite reasonable correlation between calculation and measurement for the critical loading of the Mark III core with FLIP fuel.

2.2.2 Physics Measurements

Following the achievement of criticality and preliminary control rod calibration, the core was loaded to a total of 67 fuel elements and a number of tests were performed on this core, which is shown in Fig. 1.

2.2.2.1 Neutron Lifetime Measurements. Measurements of the prompt neutron lifetime were made at four reactor power levels (100, 500, 2000, and 5000 W) using an in-core reactivity oscillator. The prompt neutron lifetime determined by this technique was established as 20×10^{-6} seconds. This compares to predicted values ranging from 16×10^{-6} to 20×10^{-6}. However, the value

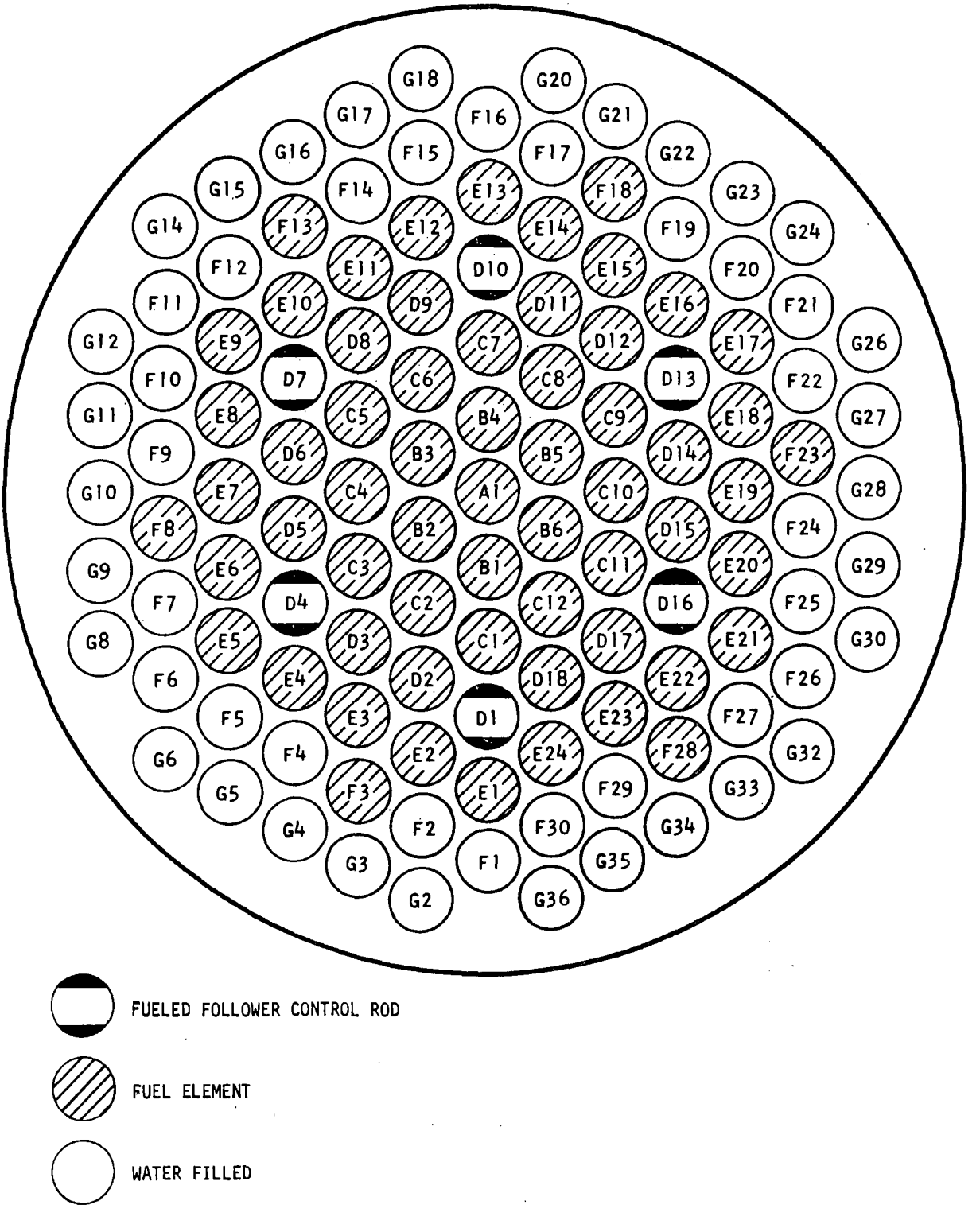

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Fig. 1. A core map of the FLIP TRIGA for a configuration using 67 fuel elements, including six fueled follower control rods

of 16 x 10^{-6} seconds was calculated for the same core parameters that led to a prediction of 64 fuel elements for the critical core loading. A recalculation of the prompt neutron lifetime with the parameters adjusted to reflect actual uranium and erbium contents gives a value of 17.7 x 10^{-6} seconds. Further adjusting for the 10% accuracy in the published cross sections of erbium, together with the lower-than-usual content of hafnium poison in the zirconium, brings the predicted prompt neutron lifetime calculated value into closer agreement with the measured value.

Furthermore, it is useful to note that the same experimental technique has been used to measure the prompt neutron lifetimes for the same Mark III and the Mark F reactors, both loaded with standard TRIGA fuel. The measured and calculated values for these cases agree quite well, as shown in Table 3.

2.2.2.2 Flux Measurements. A limited number of flux measurements were obtained during the startup tests for specific in-core locations to check predictions for specific experiments. These data are presented in Fig. 2.

A Reuter-Stokes self-powered rhodium detector was used along a radius to map the flux distribution for the 67-element core. The detector was roughly calibrated by inserting it in the lazy susan of the Mark I reactor, which has a known thermal flux. Although this calibration is used to establish the data presented in Fig. 2, it is recognized that considerable error may exist because of the harder spectrum in the FLIP core and because of the effect of the rhodium epithermal resonance. The shape of the curve around the fuel-followed D-ring control rod is thought to reflect the presence of more water than core average around the reduced diameter fueled followers.

2.2.2.3 Power Versus Reactivity Loss. The TRIGA reactors possess a large prompt negative temperature coefficient. For the steady-state operation of a TRIGA reactor, the above fact is manifested in a substantial reactivity loss associated with power generation. The larger the power level (and

TABLE 3

COMPARISON OF MEASURED AND CALCULATED PROMPT NEUTRON LIFETIMES FOR THREE DIFFERENT TRIGA CORES

Reactor	TRIGA Fuel	Core Configuration	ℓ (μsec)	
			Measured	Calculated
Mark F	High hydride, stainless-steel clad, H_2O reflected	Circular array	39.8 ±1.2	39
Mark III	High hydride, stainless-steel clad, H_2O reflected	Hex array	40.9 ±1.9	39
Mark III	FLIP, H_2O reflected	Hex array	20.0 ±0.2	17.7

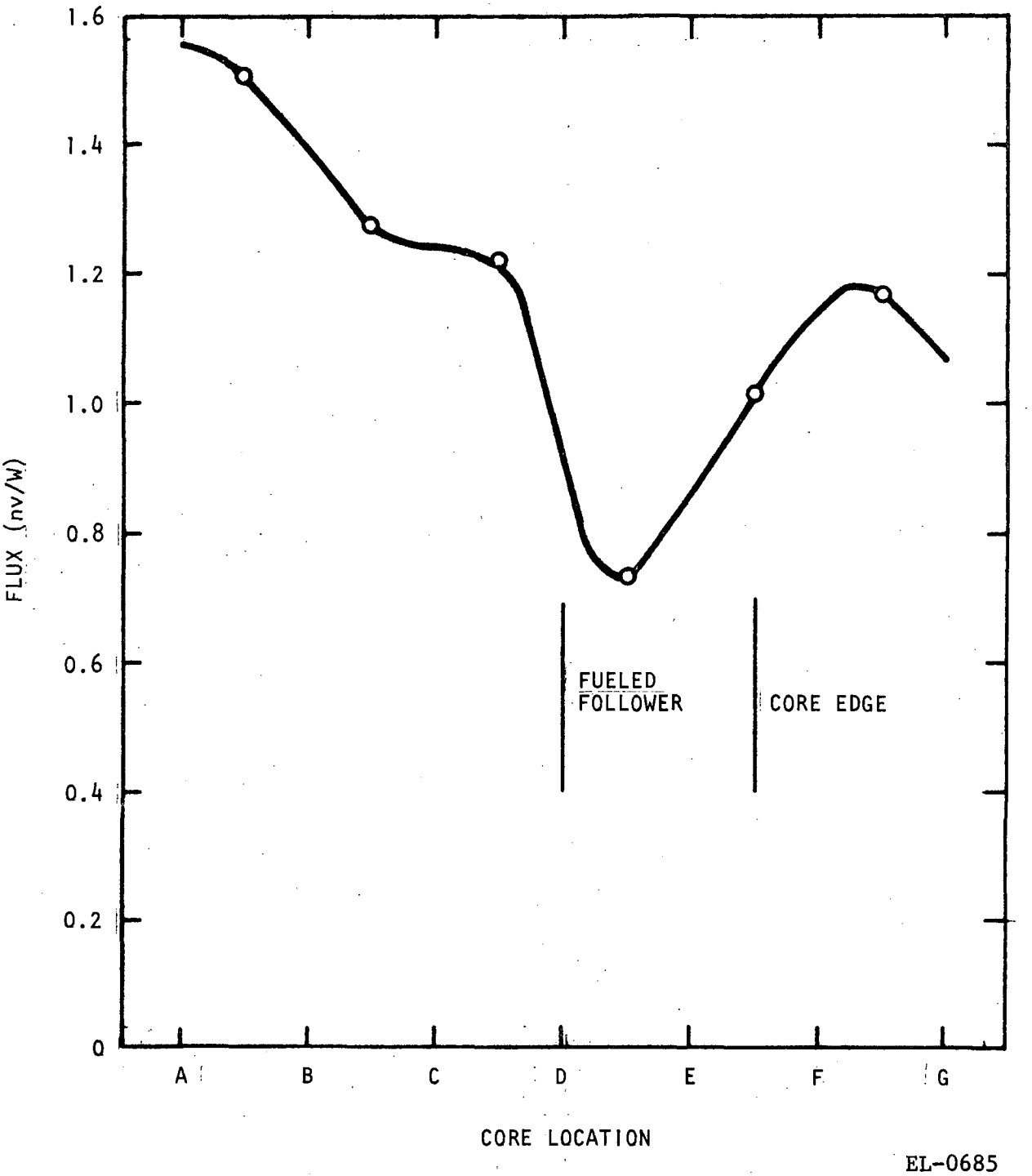

Fig. 2. Flux distribution obtained with a rhodium self-powered in-core detector in a 67-element FLIP core

hence fuel temperature) the larger the reactivity loss. This feature of the FLIP TRIGA reactor was investigated during the startup tests. Figure 3 shows the observed reactivity loss as a function of power level for both the 67- and the 100-FLIP fuel-element configurations. The highest power level of 1200 kW for the 67-fuel-element core (1200 kW/67 el = 18 kW/el) corresponded to a power level in the 100-fuel-element core of 1790 kW. It is therefore not surprising that the reactivity loss should be about the same for the 67- and 100-fuel-element cores at power levels of 1200 kW and 1800 kW, respectively.

2.2.3 Temperature Profiles

With the small core (67 fuel elements) and the larger core (100 fuel elements plus 11 in-core dummy elements), temperature profiles along a radius of the core were measured. Figure 4 shows the variations in temperatures measured by a typical instrumented element as it was located in different core positions (A, B-2, C-4, D-5, E-7, F-9). The temperatures were determined using a potentiometer and were measured with respect to the ambient temperature. Values corrected to 0°C are plotted in Fig. 4. Figure 5 shows the small skewing of the temperature profile due to the presence of a large graphite column surrounding one half of the core from positions G-2 through G-18.

2.3 GENERAL CONCLUSIONS

In summary, the TRIGA-FLIP core appears to function very much like the standard TRIGA fuel and to have similar thermal and neutronic characteristics for steady-state operation. In the 100-element loading, the power level was increased to 2 MW with natural convection cooling. The reactor operation was observed to be very stable. After over one year of operation (478 MW-days to July 1972) of the TRIGA-FLIP core in the TRIGA Mark III reactor, the observable reactivity change has been within the predicted 7-1/2 to 10 MW-year reactivity lifetime. In fact, data to date indicates a lifetime much closer to the higher value.

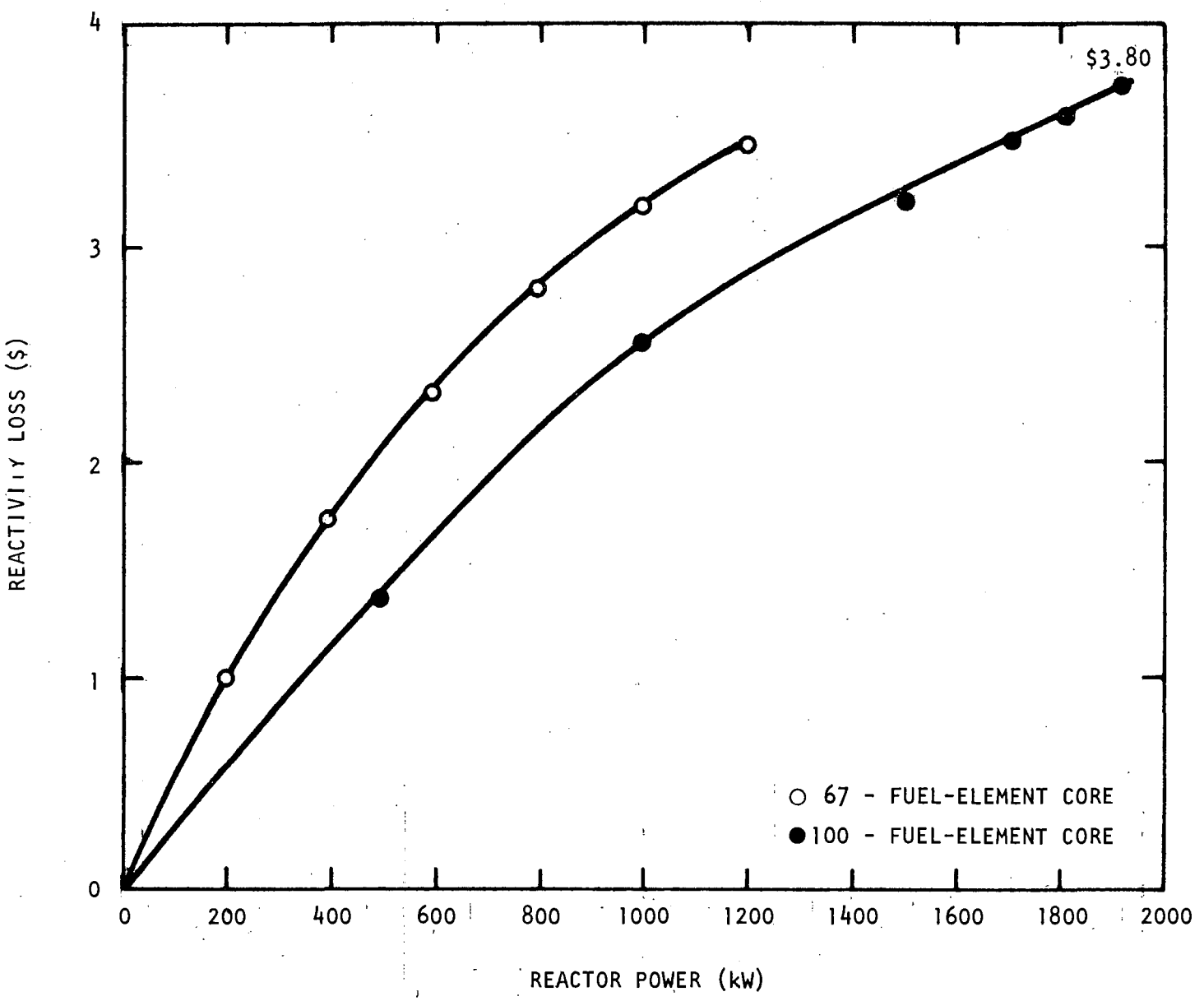

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Fig. 3. Reactivity loss as a function of reactor power level

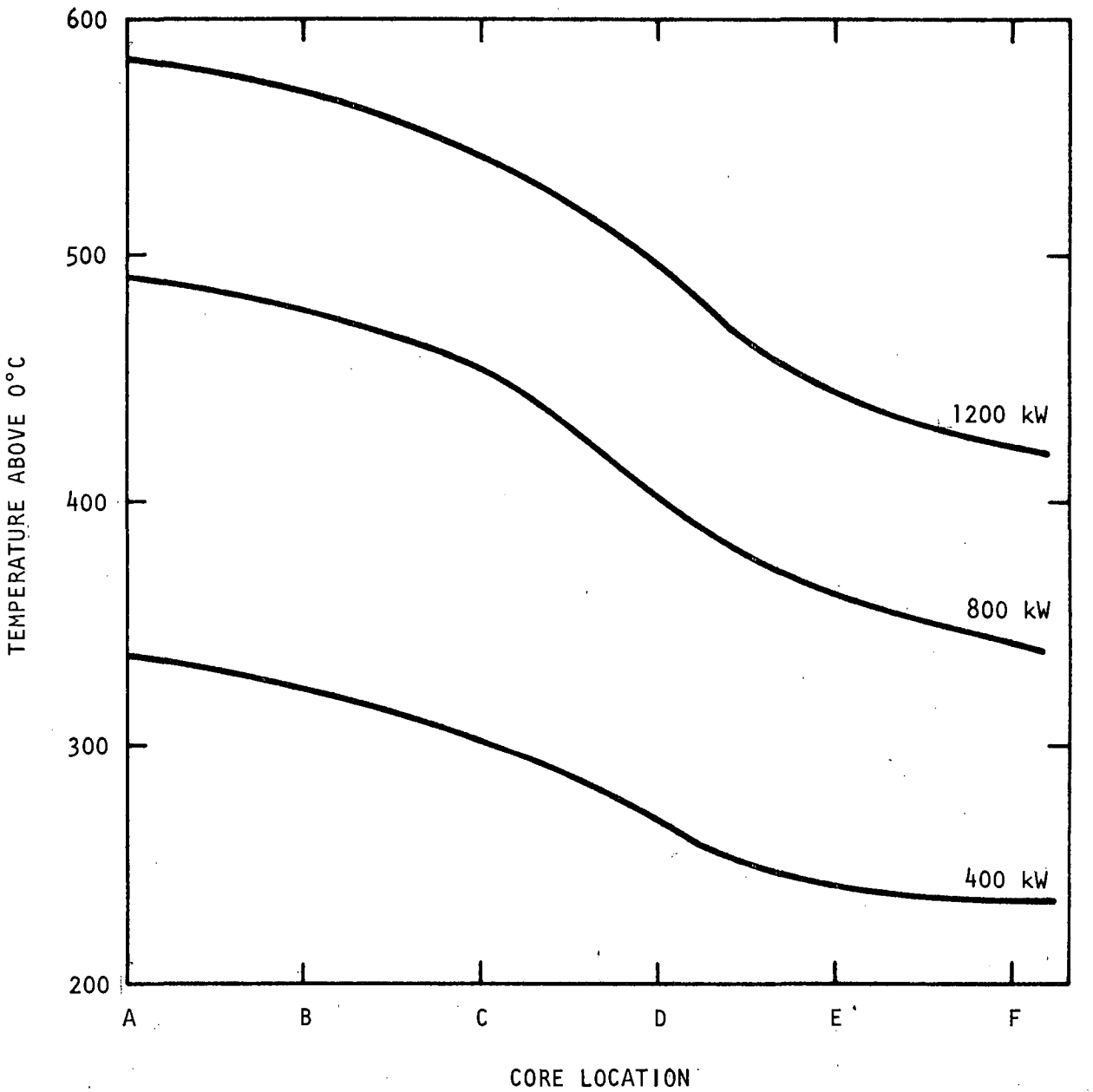

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Fig. 4. Temperature profile for a 67-element FLIP-fueled TRIGA reactor (temperature above 0°C for the center line thermocouple)

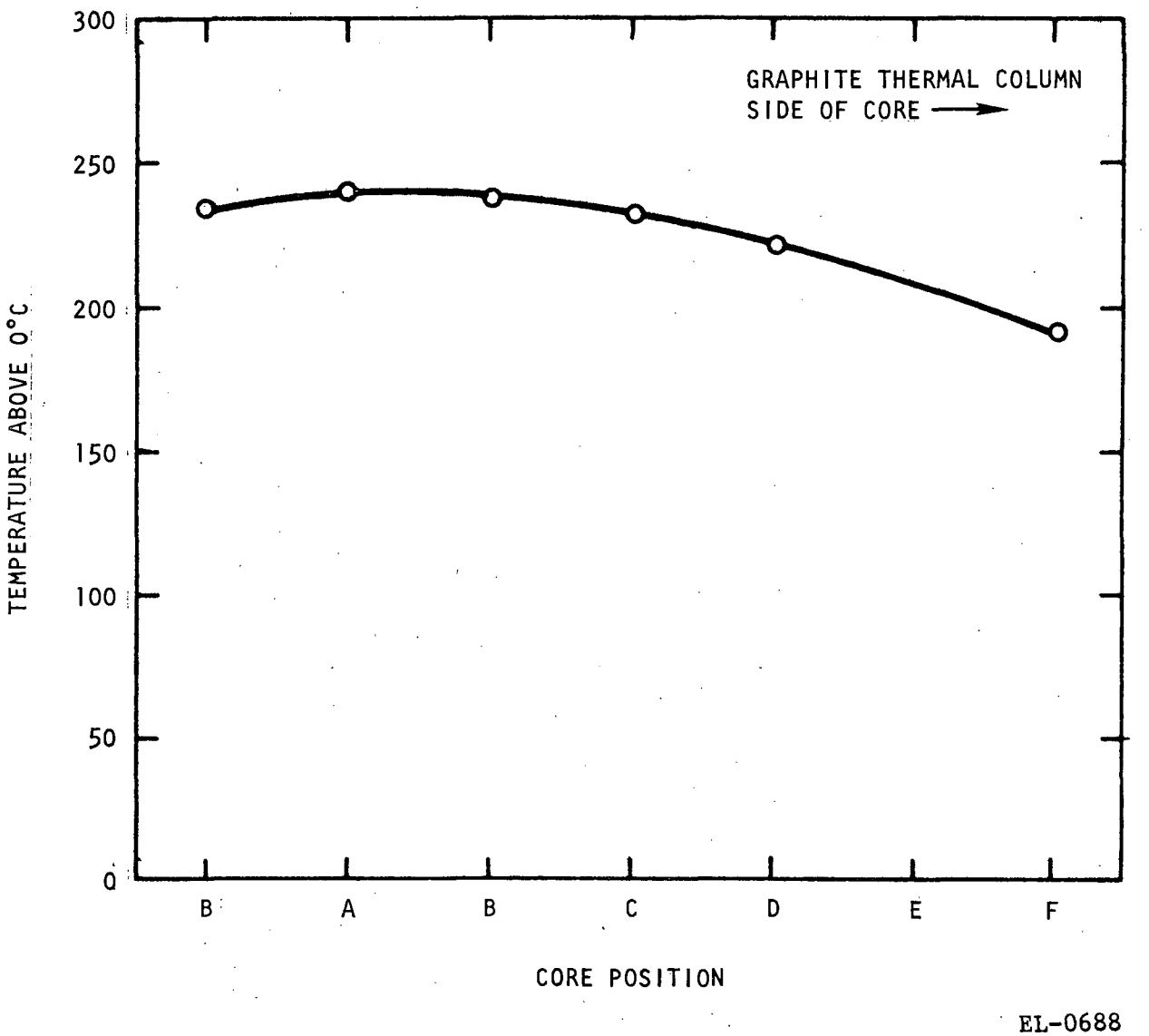

Fig. 5. Typical temperature profile for a 100-element FLIP-fueled TRIGA reactor (10 in-core dummies) with graphite thermal column temperature indicated by center line thermocouple of element

3. STARTUP OF PUERTO RICO NUCLEAR CENTER REACTOR WITH TRIGA-FLIP FUEL

3.1 GENERAL REACTOR DESCRIPTION

The Puerto Rico Nuclear Center (PRNC) reactor is located at Mayaguez, Puerto Rico and is operated for the U. S. Atomic Energy Commission by the University of Puerto Rico. The reactor (see Fig. 6), an open pool-type originally designed for 1 MW operation with plate-type fuel, was initially placed in operation in September 1960. The reactor has been used extensively for research and training and has also been a valuable tool in the area of nuclear applications, primarily agricultural, medical, and industrial.

After nearly ten years of operation at a relatively high duty cycle, the need was foreseen to replace the original plate-type core, to upgrade the instrumentation, and to provide for higher flux levels in the core (Ref. 4). Accordingly, the U. S. Atomic Energy Commission entered into a contract with Gulf E&ES to upgrade the reactor by installing TRIGA-FLIP type fuel which would provide for steady-state operation at 2000 kW with natural convection cooling and would provide for pulsing up to 2000 MW. The long-lived FLIP fuel was selected to provide the desirable 7-1/2 to 10 MW-years lifetime without the requirement for fuel replacement. The contract also provided for installation of a modern TRIGA control console, TRIGA-type control rods and drives and a primary-to-secondary tank water system heat exchanger. The control console provided was the first research reactor console designed in compliance with the specifications of the USAEC RDT standards for reactor plant protection systems. Gulf E&ES was responsible for a turn-key conversion, including preparation of the Safety Analysis Report, Technical Specifications, installation, and checkout.

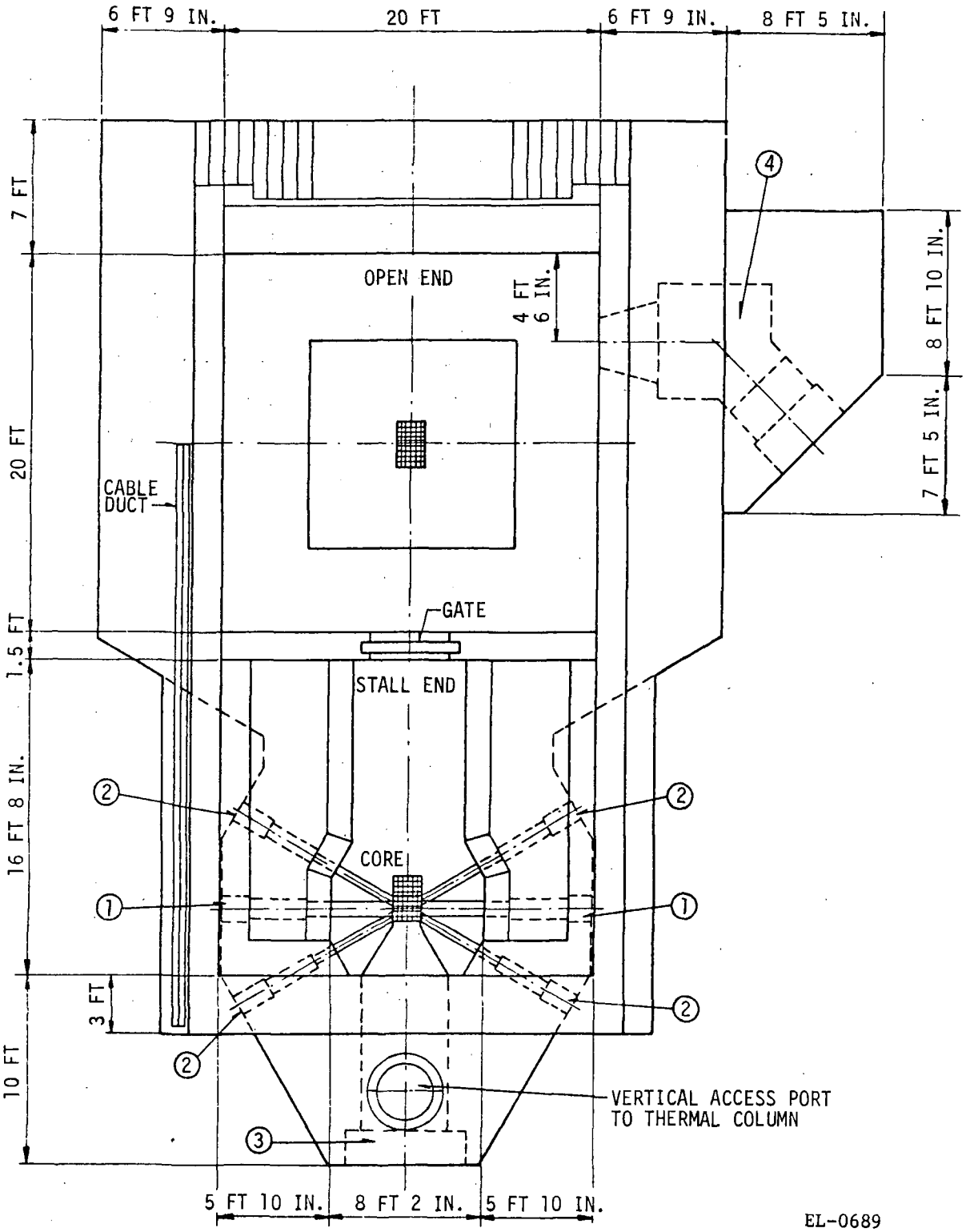

Fig. 6. Plan view of PRNC reactor

3.2 STARTUP OF THE PRNC NUCLEAR REACTOR WITH FLIP FUEL

3.2.1 Criticality

The PRNC reactor (Ref. 5) was taken critical with TRIGA FLIP fuel on January 19, 1972. The critical configuration, shown in Fig. 7, contained 62 fuel rods. The FLIP fuel for the PRNC reactor was arranged in four-rod fuel clusters. Throughout this report, each rod in the cluster will be referred to as a fuel element. As can be seen from Fig. 7, the abscissa of the core is designated by numbers (1 through 6) and the ordinate of the core by letters (A through F). Thus, an alpha-numeric designation identifies the position of a four-rod cluster. Within a cluster, the clockwise numbers 1 through 4 designate the location of a specific fuel rod within a cluster.

The criticality attained with 62 elements was reached with slightly fewer fuel elements than was originally calculated. Again this is due to the fact that these FLIP fuel elements were manufactured in the same batch as those manufactured for the Torrey Pines TRIGA Mark III reactor and contained on the average slightly less than predicted quantities of erbium. Also the hafnium was below normal in the zirconium used in the zirconium-uranium alloy.

3.2.2 Adjustment of Operating Core Loading and Control Rod Calibration

A core size of 94 or 95 elements was believed desirable for heat transfer considerations; therefore, the reactor was loaded in a compact array to 94 fuel elements. The experimentally determined worths of the control rods in this core configuration are summarized below:

Safety rod	$2.25
Transient rod	2.38
Shim 1	1.64
Shim 2	2.21
Regulating	2.22

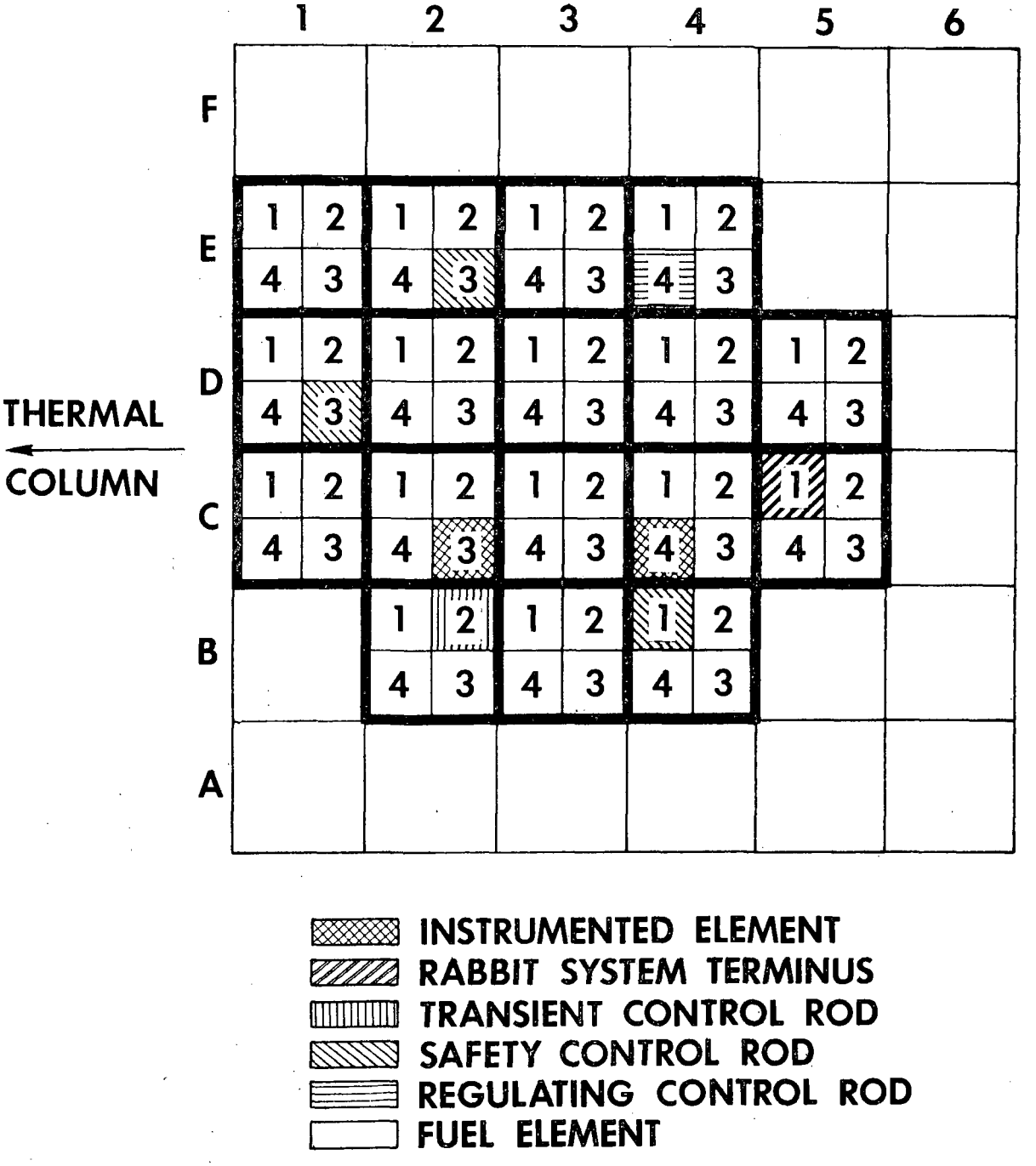

Fig. 7. Sixty-two-element critical core configuration

Based upon reactivity losses at power measured at the Torrey Pines TRIGA Mark III FLIP reactor and calculated Xenon reactivity losses, a core excess reactivity of $5.00 to $5.30 appeared desirable. The excess obtained at PRNC with the initial 94 element loading was higher than desirable, so steps were taken to reduce the core excess while maintaining about the same number of fuel elements in the core. This was accomplished as shown in Fig. 8 by replacing the fuel rods in positions D3-4 and C3-2 with dummy rods. The fuel rods which had been removed were relocated to positions A6-1 and F6-4.

3.2.3 Power Calibration

A power calibration was performed by measuring the rate of temperature rise of the known volume of well mixed water in the reactor tank. This method has proved successful for previous TRIGA installations and gives an accurate calibration provided that:

1. The volume of water involved in cooling the reactor is known and constant.

2. Heat gains or losses from the tank water due to conduction through tank walls, evaporation, auxiliary systems, such as water treatment loops, or others, is either negligible or taken into proper account in the calibration.

3. Thorough mixing of the heated volume is maintained.

4. Adequate temperature measuring equipment is used.

The PRNC reactor facility provided a somewhat greater challenge than many installations with regard to assuring that all reactor heat was properly considered in the measured rate of temperature rise of the tank water. Figure 9 shows elevation views of the reactor facility. For all power calibrations, the small-volume tank section was used, since it is the

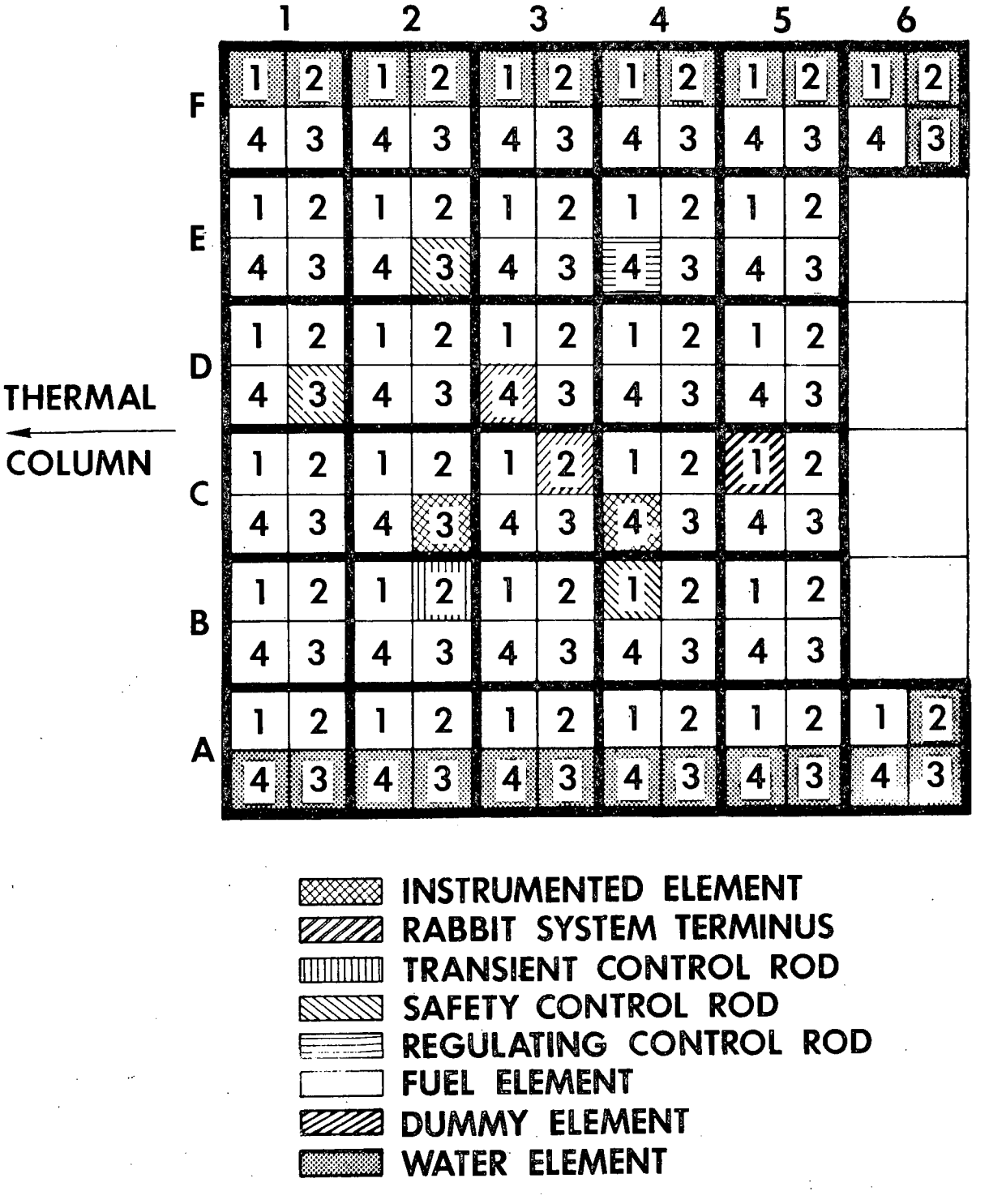

Fig. 8. Ninety-four-element core with two dummy elements

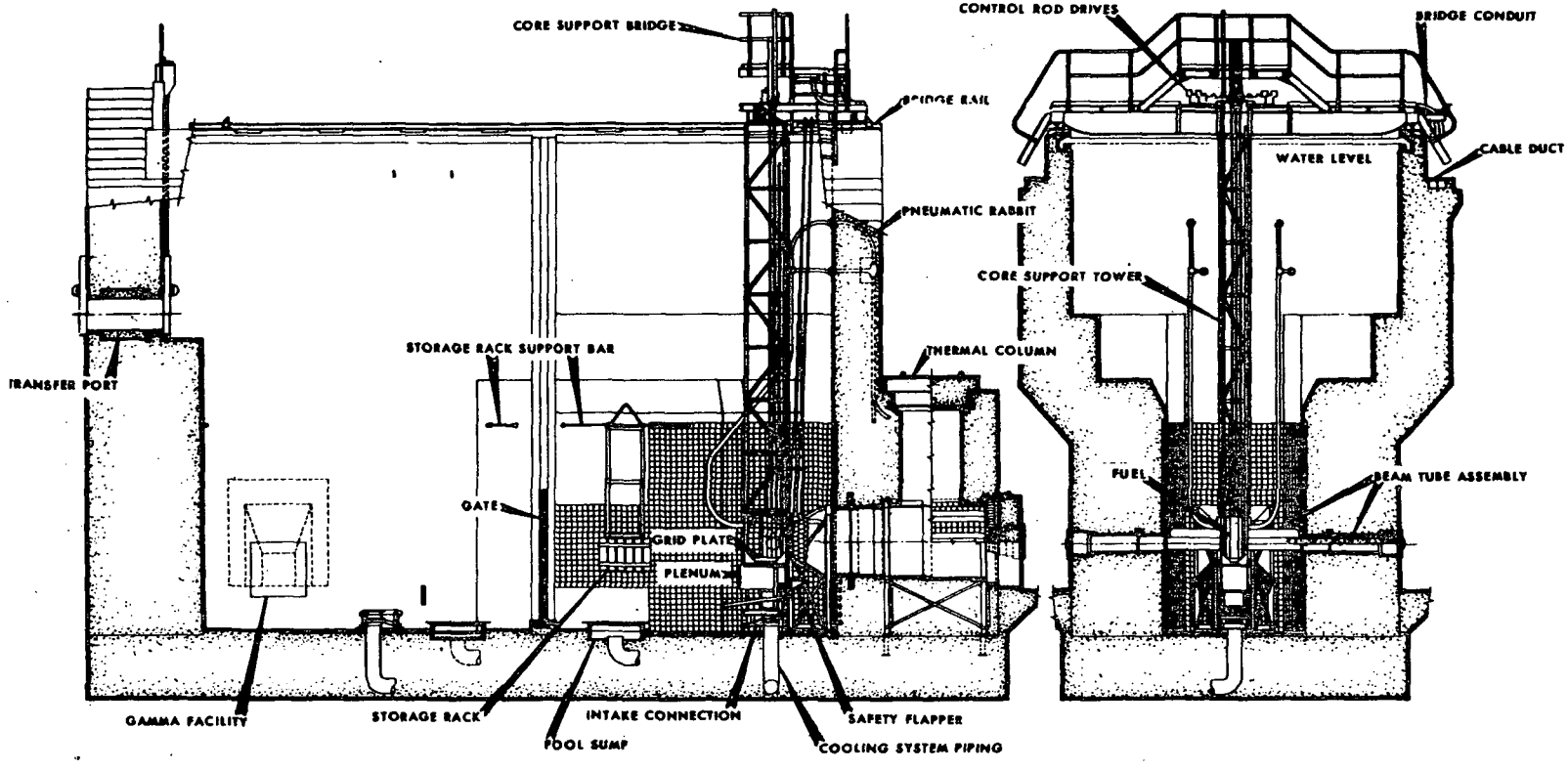

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Fig. 9. PRNC reactor, front and side elevations

normal operating location for the reactor and its smaller volume allowed more rapid water heating rates. For all power calibrations the gate was in place to separate the two tank sections and to restrict the water heating to the small-volume section.

The tank volume (small side) was determined from the facility construction drawings to be 51,000 gal. Tank dimensions, depths, and location of insets were verified on the existing tank. The calculated heat capacity of the water in this tank section is 0.22392 MW-hr/°C or 4.4618°C/MW-hr. This value was used for all calibrations performed.

After the gate was installed and a preliminary calibration underway, some transfer of water was found to have occurred between the two tank sections. This seems to have been due entirely to the relatively loose-fitting gate which is not provided with inflatable seals. A tight seal is obtained only if a large differential force is applied to the gate forces such as those that occur when one side of the tank is emptied. In the power calibration measurements the leakage around the gate edges was unknown and unavoidable. In order to eliminate errors in power calibration, several tank-water-heating runs were made under varying conditions to determine a calibration procedure which eliminated uncertainties resulting from water transport across the gate.

For all calibration runs, the tank water mixer (a propellor and shaft attached to an electric motor) was used and water temperature measurements made by use of a Digitec Model 501 digital thermometer. Two temperature measuring probes were used, the accuracy of which had been verified by water-bath tests at various temperatures and compared with two mercury thermometers. Agreement between the thermometers and the Digitec probes was found to be very good.

From the different tank water heating runs made, tank wall contributions were shown to be negligible if the water temperature is approximately the same as that of the walls and surrounding concrete. The main heat losses are the result of water and/or heat transfer through the gate. The final

calibration was made with the small tank section at approximately 18°C and the large section at approximately 20°C at the start of the reactor run. Water heatup data was taken for 75 minutes before starting the reactor run and showed a temperature-rise rate of 0.3°C/hr. The reactor then was operated at ~600 kW (indicated) for a time sufficient to bring the small tank section up to ~22°C. After shutdown of the reactor, the cool-down rate was found to be 0.3°C/hr. The reactor power was measured to be 658 kW at the midpoint in this heating curve.

3.2.4 Preliminary Steady-State Operation

After calibration of the reactor power level, the reactor was prepared for initial operation up to full authorized power. Instrument linearity tests were performed to verify tracking of all detector channels. Results of the test are shown in Table 4 for powers up to 1400 kW. The response of the safety channels was tested during the initial run to higher power since these channels are single range units indicating 100 percent at full power (or 10 percent of scale for each 200 kW of power).

Table 4 shows the results of the initial high power operation, and Figs. 10 and 11 are plots of reactivity loss and measured fuel temperature versus reactor power.

During the tests at increasing power, small fluctuations in power level were observed at 1.4 MW. As power level was varied, the magnitude of these fluctuations increased sharply. At 1.6 MW, the console recorder indicated a fluctuation of approximately 4% on the 3 MW range (i.e., ~120 kW peak values). Tests conducted to date have shown that the fluctuations are not caused directly by unbalanced control rods, flow blockages, or improper power calibration. It should be noted that similar power fluctuations were observed when the Torrey Pines TRIGA Mark III reactor was operated at power levels above 1.4 MW with a small (67 fuel-element) core. The Korean and Torrey Pines TRIGA Mark III reactors were built in the conventional TRIGA

TABLE 4

CONTROL SYSTEM LINEARITY DATA

Linear Recorder Values	Safety Channels (%)		Log Channel (%)	Fuel Temp (TC) (°C) Location		Banked Rod Positions	Reactivity Loss Relative to 100 W ($)
	No. 1	No. 2		C4,4	C2,3		
200 (20%-1 MW)	~10	~10	10	155	145	458	0.60
400 (40%-1 MW)	20	20	20	218	208	475 (Reg 496)	
600 (60%-1 MW)	29	31	30	262	262	496	1.32
800 (80%-1 MW)	38	40	40	300	308	512	1.63
1000 (33%-3 MW)	46	50	50	340	342	525	1.84
1200 (40%-3 MW)	56	60	60	380	380	541	2.13
1400 (46%-3 MW)	65	70	70	415	421	557	2.39

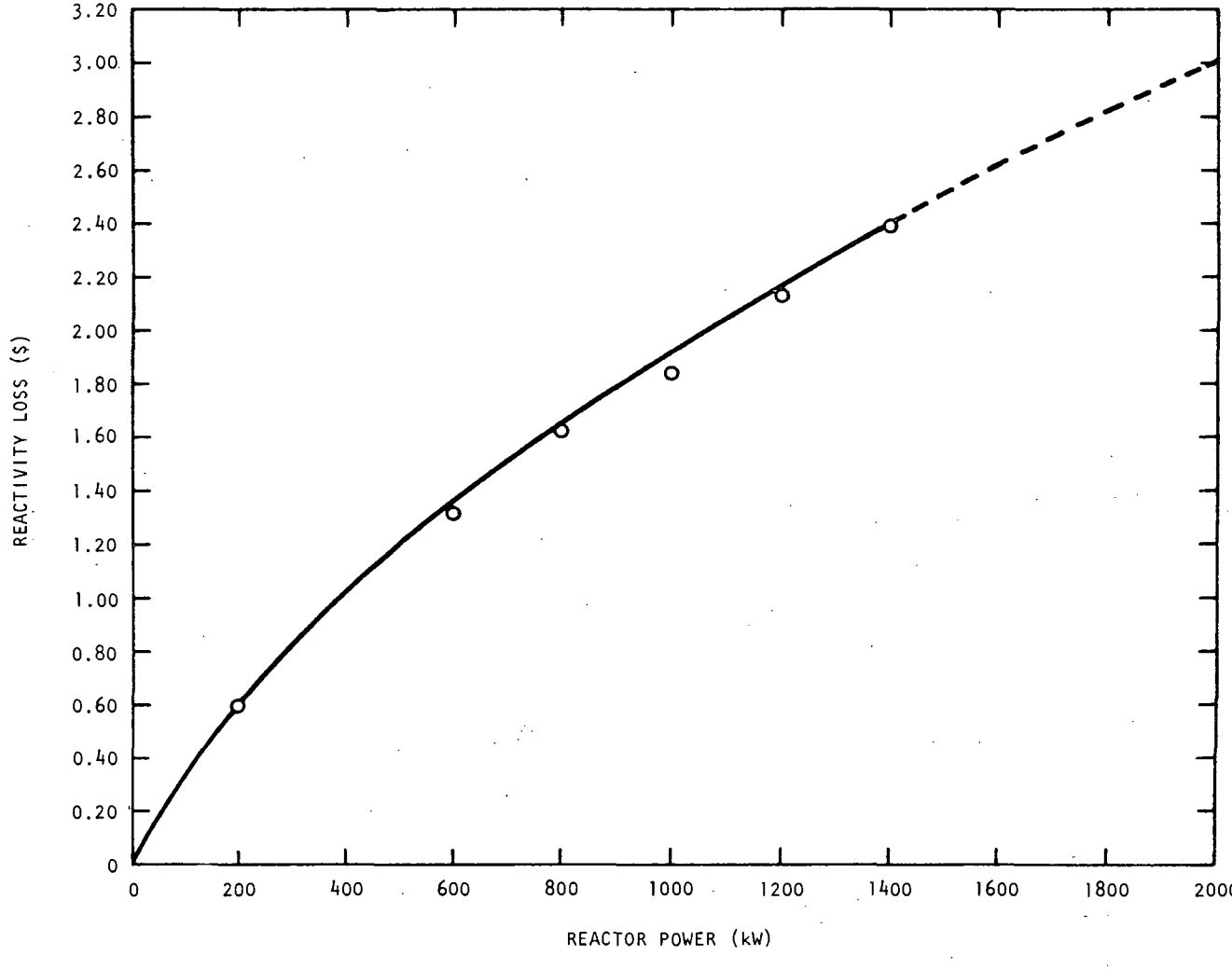

Fig. 10. Reactivity loss vs. reactor power (94-element core at 2 MW in mid-pool position)

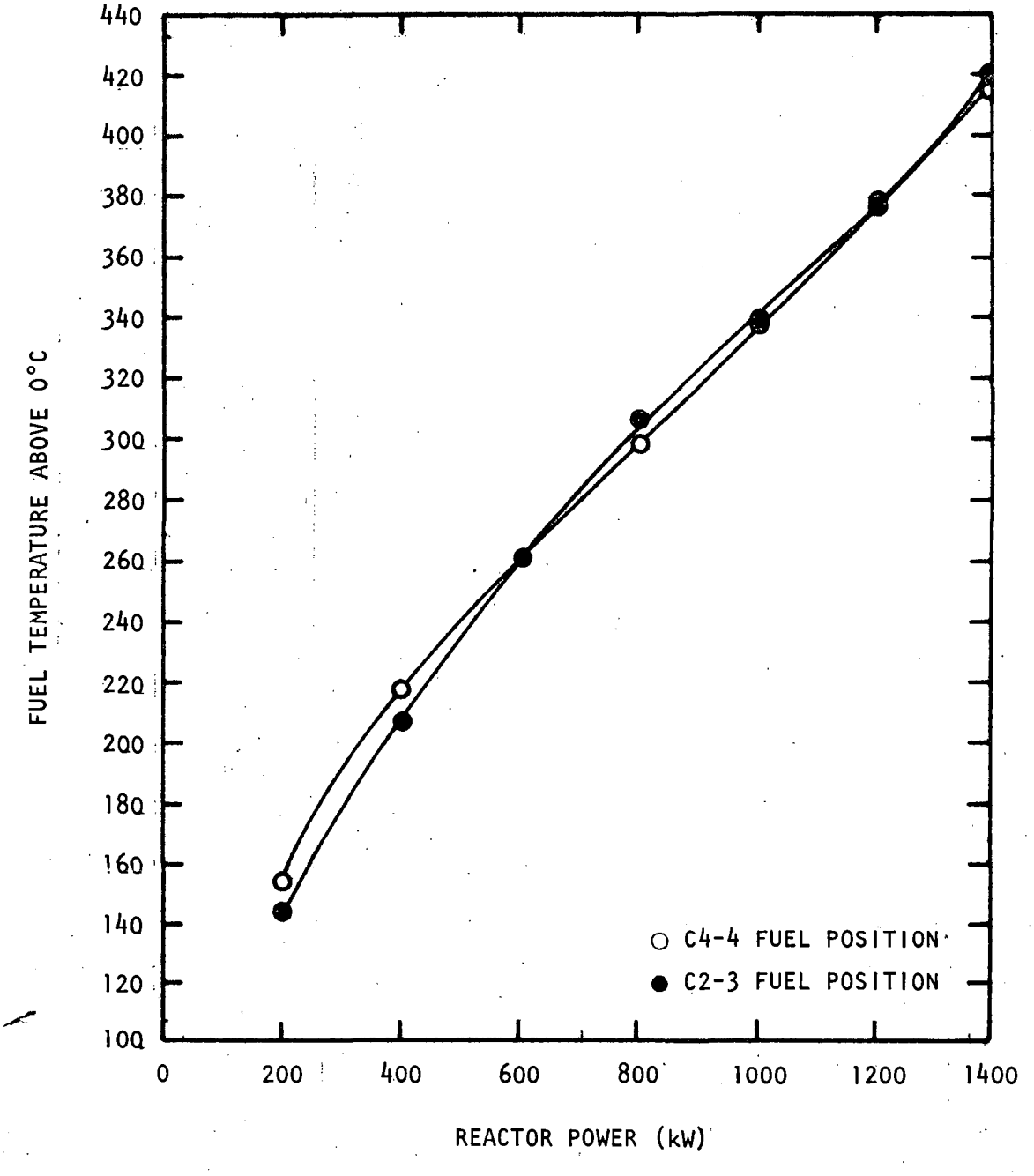

Fig. 11. Fuel temperature vs. reactor power

configuration and are presently operating at 2 MW steady state without power fluctuations, whereas the PRNC-TRIGA is the first four rod cluster-type conversion to operate over 1 MW.

It is postulated that the power fluctuations might be caused by the relatively small gap between adjacent fuel rods in the PRNC core. The lattices in both the Korean and Torrey Pines Mark III reactors provide fuel spacing in an equidistant triangular pattern. Although the water fraction in the PRNC TRIGA is the same as these two Mark III reactors, the separation distance in one direction between rows of fuel rods in the four-rod cluster with the square array is substantially smaller (0.119 inches versus 0.239 inches with a triangular lattice). Studies are underway to evaluate the improvements that could be effected by changing the fuel lattice configuration to increase the inter-element gap.

3.2.5 Pulse Tests

The first tests of the reactor in the pulsing mode were made in the midpool position, giving a totally water reflected core. The goal of these tests was to determine the major pulsing parameters for pulses up to 2000 MW peak power. The parameters of interest for varying reactivity insertions are peak power, maximum measured fuel temperature, energy release, and pulse width.

Measurements were made with both the console instrumentation and auxiliary systems consisting of an auxiliary ion chamber channel and a Visicorder fast recorder. Several pulses at each level were performed and the measuring channels and recording apparatus changed for each to cover all desired measurements for each pulse size.

Table 5 shows the data obtained during pulsing and Figs. 12 and 13 are plots of these results.

TABLE 5

PULSING RESULTS FOR PRNC FLIP REACTOR
(NINETY-FOUR ELEMENTS, TWO STAINLESS-STEEL DUMMIES)

Δk_{TOTAL} ($)	$\hat{P}$ MW	E[a] MW-Sec	T_{MAX} Near TR	T_{MAX} Near Shim 2
1.50	400	12.75	290	316
1.75	790	15.60	340	367
2.00	1330	-	384	420
2.20	1770	-	420	452
2.28	1950	21.1	434	-

[a] Includes pulse tail energy to 1 sec after transient rod insertion.

Measured values of the various pulsing parameters of the PRNC reactor differed somewhat from pre-startup predictions. The magnitude and direction of the discrepancies probably represent the results of effects due to actual fuel material content (notably erbium) versus assumed content and rod calibration techniques. In general, the PRNC core achieved pulsing temperatures (measured) for the 2000 MW peak power pulse which were in close agreement with prestartup expectations. The magnitude of reactivity insertions required for the maximum pulse was somewhat larger than anticipated, and the energy release was measured to be smaller than anticipated. Specifically, $\delta k/k$ of \$2.30 gave 1980 MW $\hat{P}$ with a measured temperature rise of ~465°C and an energy release of 20.6 MW-sec. Calculations suggested ~\$1.95 $\delta k/k$ would give 2000 MW $\hat{P}$ with an energy release of ~25 MW/sec.

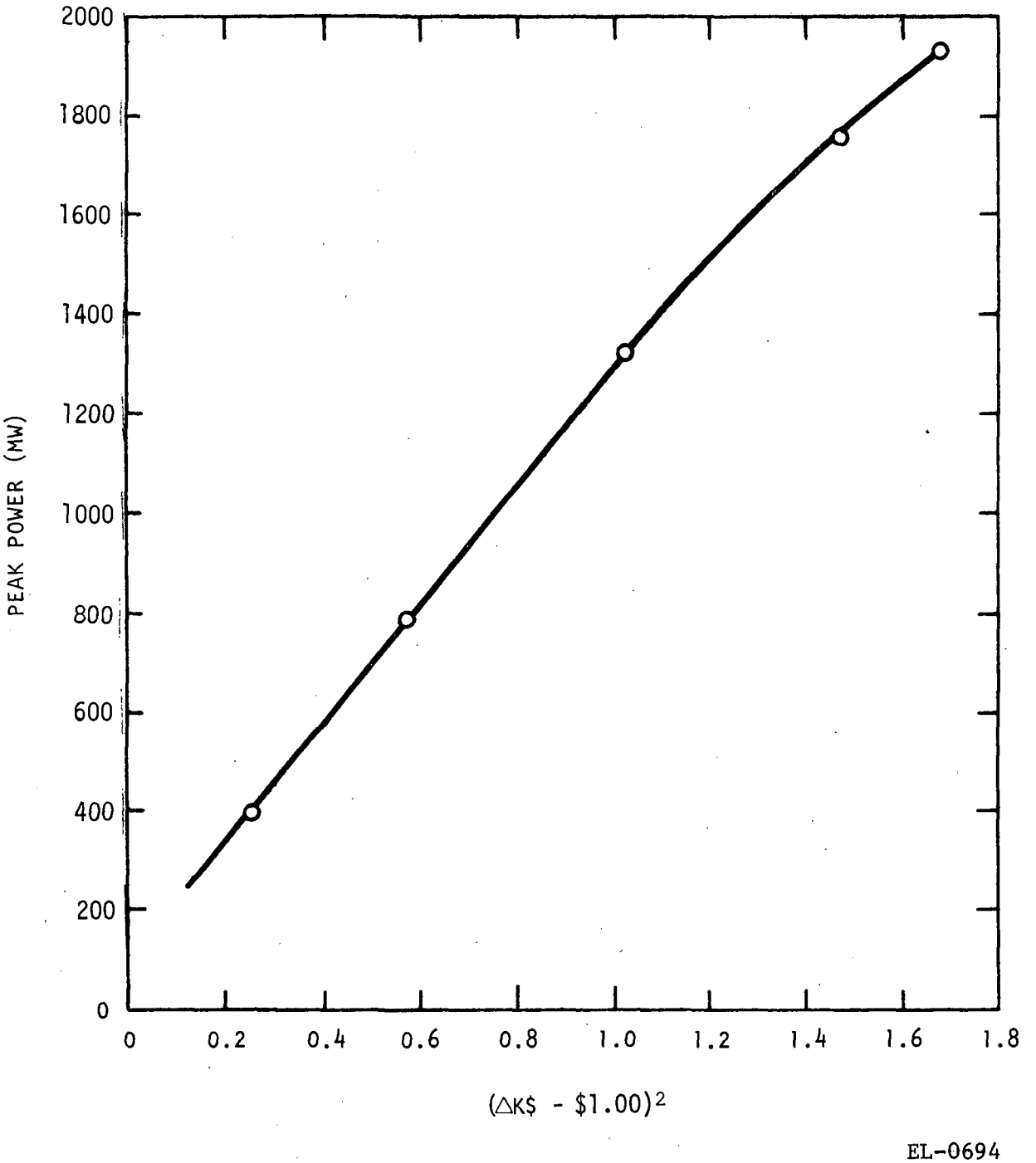

Fig. 12. Midpool pulsing - peak power vs. reactivity insertion

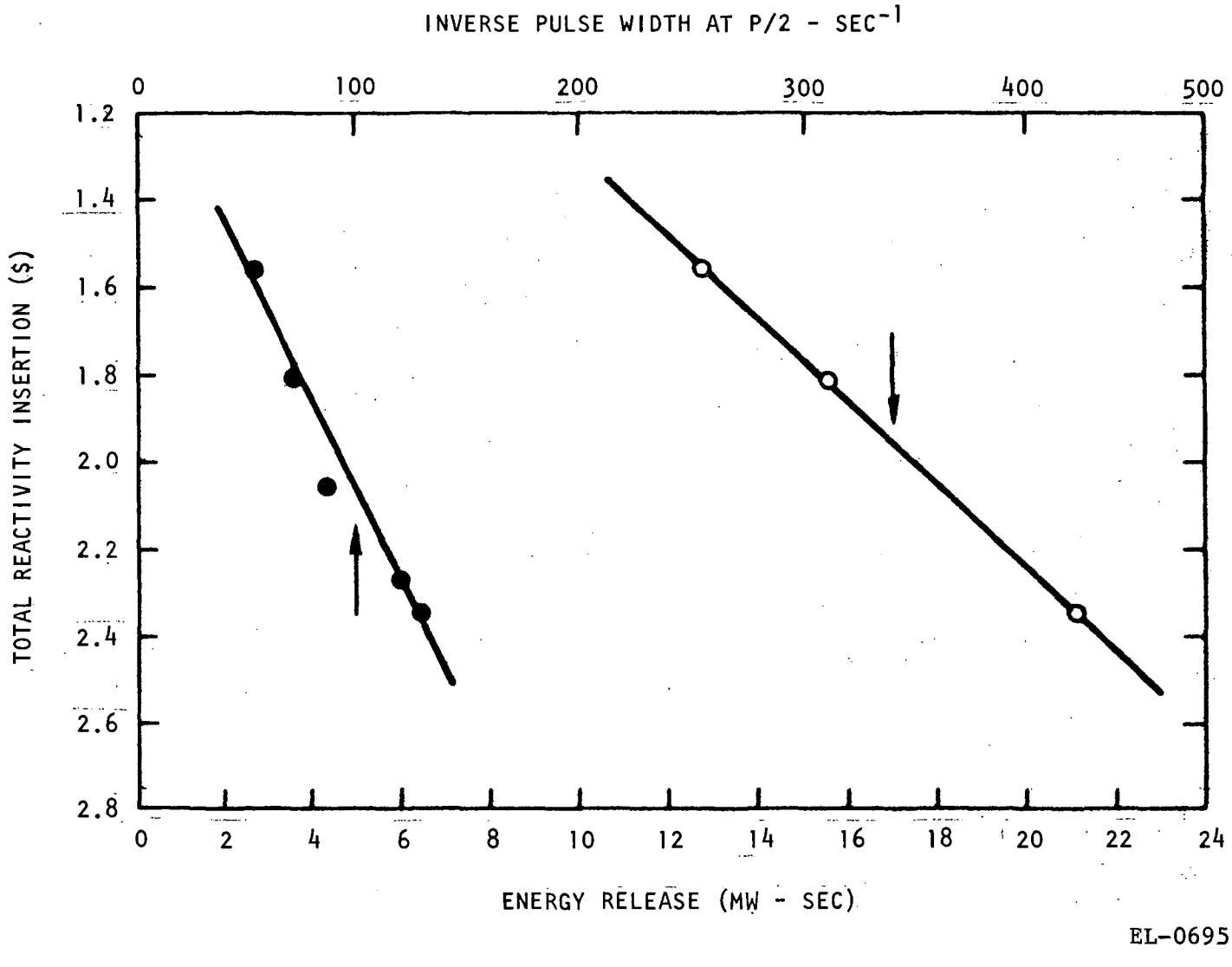

Fig. 13. Inverse pulse width and energy release vs. total reactivity insertion

3.3 SUMMARY AND CONCLUSIONS

Following installation of the TRIGA-FLIP core loading, the PRNC TRIGA was operated in a series of calibration tests. The reactor performed close to prediction in all operating modes, including pulsing to 2000 MW, except that power fluctuations were observed in the steady-state mode of operation around 1.5 MW. Studies are continuing to determine the optimum methods for modifying the core arrangement to improve the thermal-hydraulic characteristics so that stable operation at 2.0 MW can be achieved.

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4.2 THE TRIGA REACTOR FRANKFURT CONSTRUCTION AND EXPERIMENTAL FACILITIES, D. Rossberg, A. Wensel, G. Wolf (University of Frankfurt, Germany)

The first reactor of the Institut für Kernphysik, the FRF 1, was a Homogeneous Water Boiler Reactor L 54 of Atomics International with a maximum power level of 50 kW. After 10 years of operation it had to be shut down permanently because of a failure in the hydrogen-oxygen recombination system. In view of the limited experimental utility of the FRF 1 (maximum thermal neutron flux $7 \cdot 10^{11} n/cm^2 s$) it was decided to replace it by a modified TRIGA reactor.

This new reactor FRF 2 was designed by Gutehoffnungshütte Sterkrade AG in cooperation with the reactor group of the Institut für Kernphysik. The maximum power level of the reactor will be 1 MW; installation of facilities for pulsed operation is possible at a later date. Performance and design data of the FRF 2 are given in Tab.1.

The reactor is expected to get in operation in 1973.

Since the FRF 2 will be installed inside the biological shield and reflector of the FRF 1, the diameter of the FRF 2 core tank has to correspond to the dimensions of the FRF 1 core structure. Hence, the FRF 2 differs from standard TRIGA reactors in several significant characteristics.

Design of Reactor Components

The reactor tank with the core is inserted into a vertical hole of 57 cm inner diameter in the biological shield. The tank is surrounded by a graphite reflector with a minimum thickness of 35 cm.

The biological shield consists of two parts, the inner layer and the outer main shield. In order to prevent excessive