A TEMPERATURE STUDY OF

THE V.P.I. TRAINING AND RESEARCH REACTOR (UTR-10)*

by

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Thesis submitted to the Graduate Faculty of the

Virginia Polytechnic Institute

in candidacy for the degree of

MASTER OF SCIENCE

in

Nuclear Science and Engineering

May 1964

Blacksburg, Virginia

*Supported in part by the United States Atomic Energy Commission

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I. INTRODUCTION

The University Training and Research Reactor (UTR-10), owned and operated by Virginia Polyetechnic Institute, was designed and constructed by Advanced Technology Laboratories, a division of American Standard. The reactor is a modified Argonaut design with highly enriched uranium fuel, graphite reflector, and light water moderator-coolant. The present licensed power of the reactor is 10 Kw.

The reactor fuel is contained in two groups of six fuel elements which are arranged in two parallel rows as shown in Fig. 1. Each element is composed of twelve plates of enriched U^{235} alloyed with aluminum (UAl_4) and clad with 0.02 inches of aluminum. Fig. 2 is a photograph of a typical element.

Cooling of the fuel plates is accomplished by the transfer of heat from the plates to the moderator-coolant which flows between the plates. The coolant flows in a closed system, depicted in schematic form in Fig. 3, composed of a dump tank, a circulating pump, a heat exchanger, the reactor core, and the associated piping. The system is closed to prevent contamination of the reactor area in case of a ruptured fuel plate.

Plans are now in progress to increase the licensed power of the V.P.I. reactor to 100 Kw. This increase in power by a factor of ten will increase the amount of heat evolved by the same factor. Since the present cooling system is designed for a power of 10 Kw,

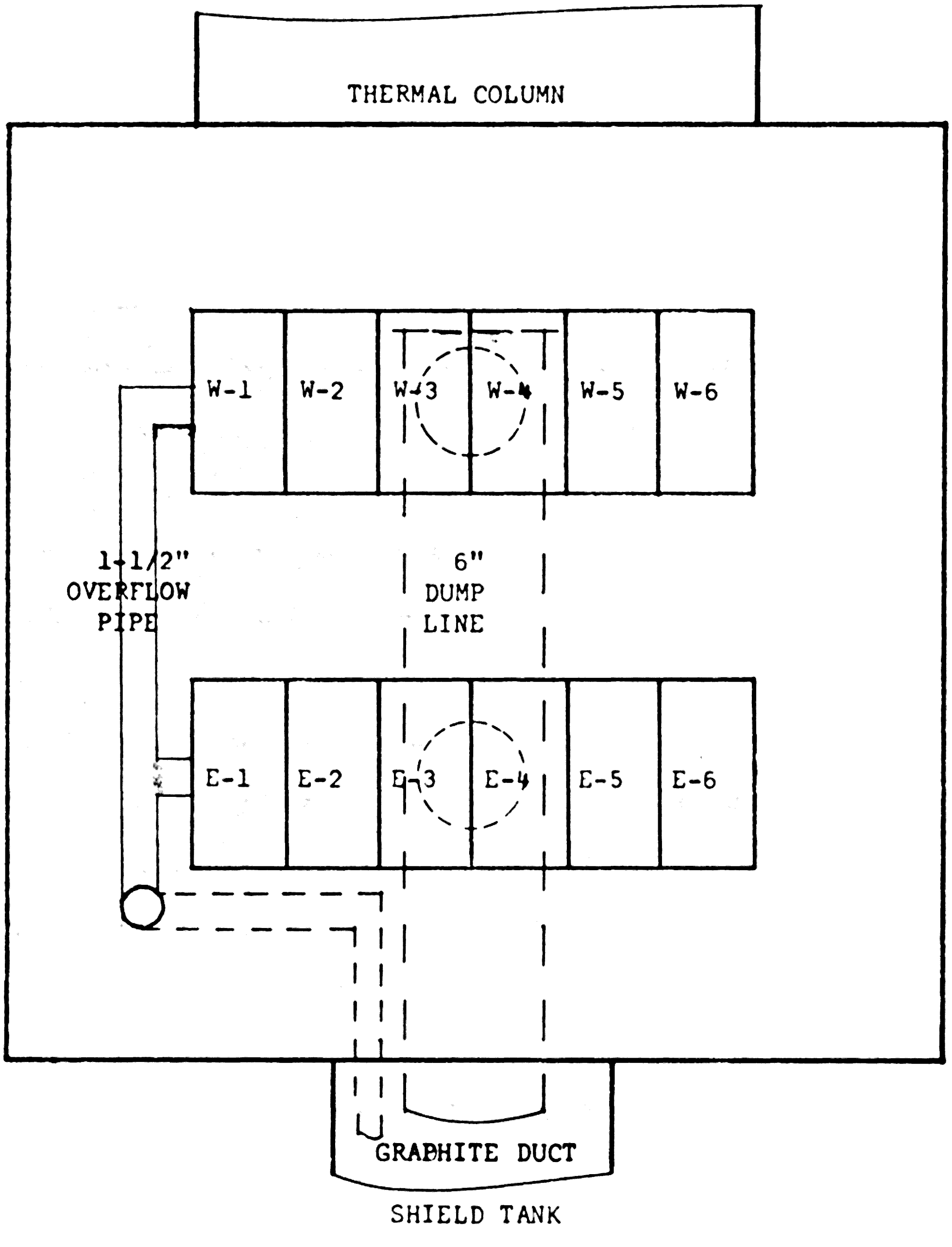

FIGURE 1 CORE PLAN

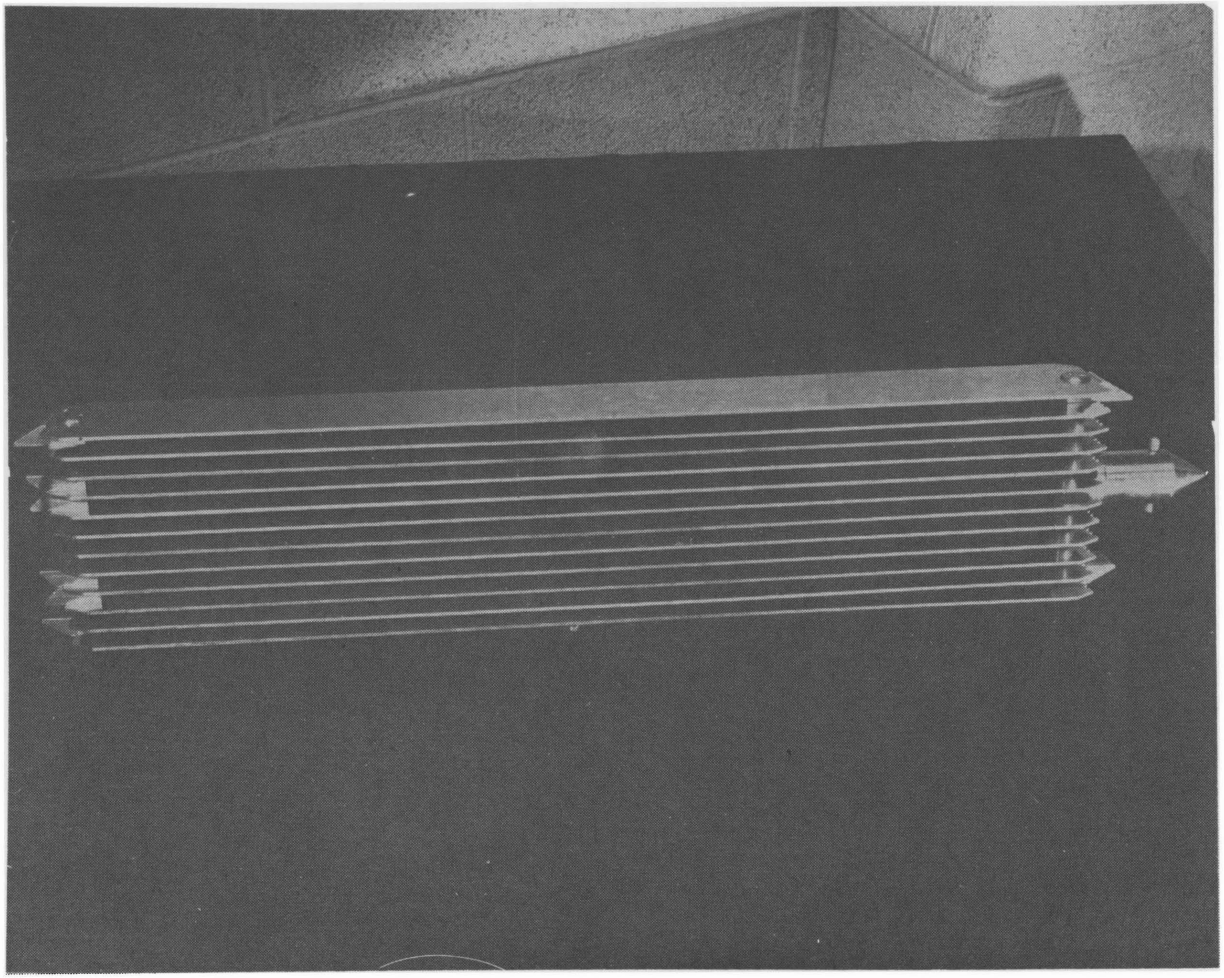

FIGURE 2   UTR-10 Fuel Element

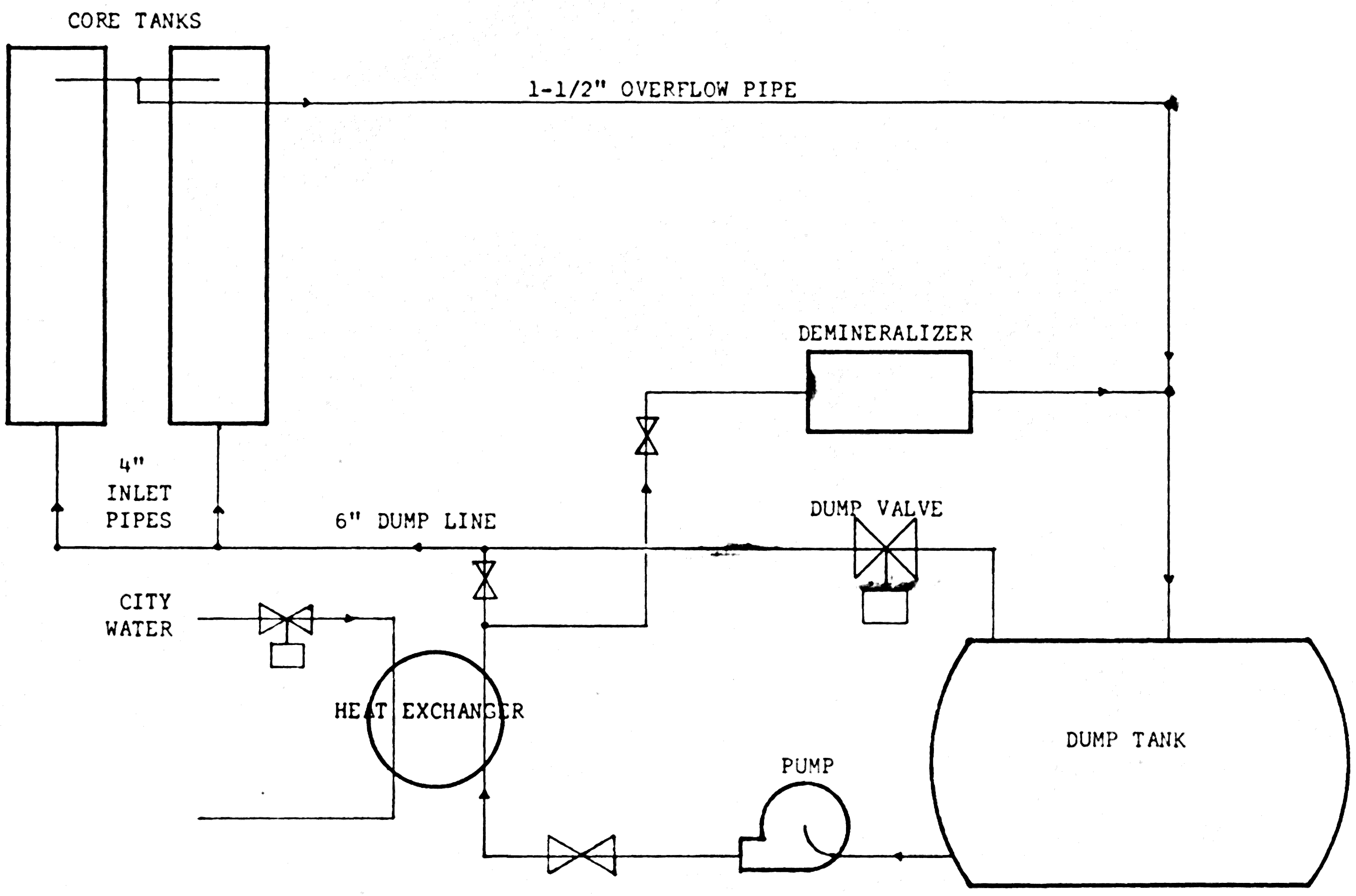

FIGURE 3 PROCESS SYSTEM

a knowledge of the coolant temperatures within the reactor core is a necessity in order to determine the needed changes in the cooling system for the higher power level.

The main cause of concern in the operation of the reactor at the higher power, as far as temperature is concerned, is the maximum coolant temperature to which the aluminum cladding of the fuel elements will be subjected. The maximum temperature of both the fuel cladding and the fuel matrix may be computed if the temperature of the surrounding moderator-coolant and the fission rate in the fuel matrix are known.

This thesis covers the temperature measurements of the entire reactor core and the predictions which are made on the basis of these data.

II. PROCEDURE AND RESULTS

Temperature measurements were made with the V.P.I. reactor core using copper-constantan thermocouples as temperature sensors. This type of thermocouple was chosen for its high thermoelectric output, 22 μvolts per °F, and a supply of No. 24 thermocouple wire with thermoplastic insulation was readily available. The thermocouple assemblies made of this wire were found to be durable and provided a satisfactorily large signal over a large range of reactor powers.

There was no indication that the output of the thermocouples was affected by the neutron and gamma radiation to which they were exposed or by the induced radioactivity in the wire and insulation. The absence of any radiation effects on the thermoelectric emf agrees with the earlier results reported by Palladino[5]. There were, however, marked color effects on the plastic insulation of the thermocouples making the insulation appear bleached where the flux had been the highest.

The thermocouples were positioned in the fuel elements by means of the assembly shown in Fig. 4. The eleven aluminum rods, held at the top by the aluminum plate, are inserted between the twelve fuel plates of each element. The rods are located in the center of the coolant channel by spacers on the bottom of the rods and are positioned with respect to the width of the channels by members which slip over the lower connecting shaft of the fuel element. The top

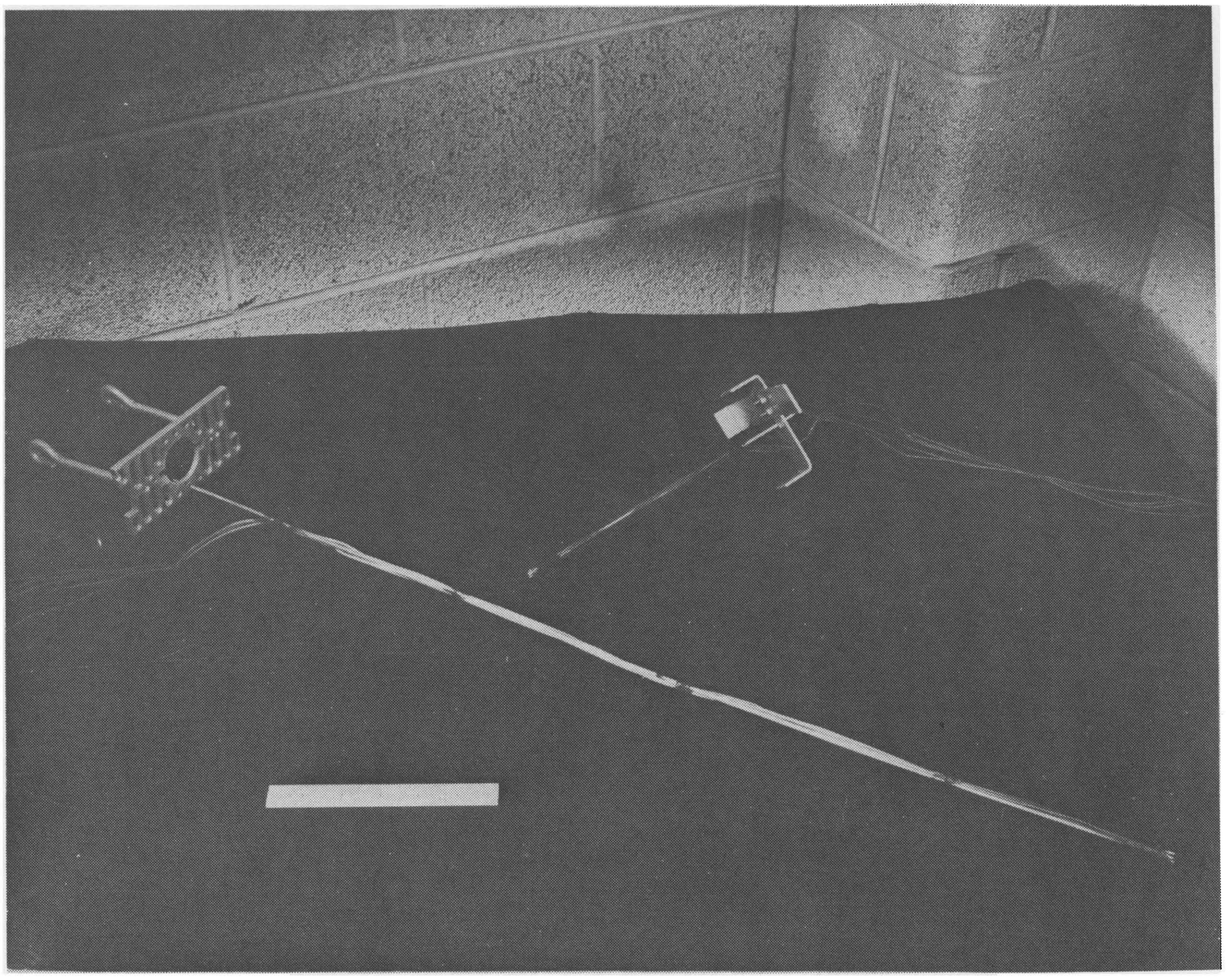

FIGURE 4 Fuel Element Thermocouple Assembly

plate was cut so that it fit snugly on the top of the fuel element and numerous holes were drilled in the plate so as to reduce its impedance to the coolant flow in the element. The thermocouples were fastened to the rods in such a manner that the hot junctions were located adjacent to the bottom, middle, and top of the fuel matrix in the fuel plates.

The power of the reactor, as indicated by the level of the neutron flux, can be accurately controlled; however, the coolant flow rate to the core is adjusted by a coarse system of valves and there is no provision for regulating or measuring the individual coolant flows to the two core tanks. The inlet and outlet temperature for each core tank were monitored during the temperature measurements for each element by the addition of the thermocouples shown with their holders in Fig. 5. The inlet hot junctions were actually placed within the 4" inlet pipe for each core. The outlet junctions were located adjacent to the outlet pipes and directly over elements W-1 and E-1 as shown on page 6. The inlet and outlet thermocouples were moved only when their presence actually interferred with the loading of the thermocouple assembly into a fuel element and were then returned to their original positions as closely as possible. After all of the fuel element profiles had been determined, the average inlet-outlet temperature difference for the entire series of measurements was used to normalize the

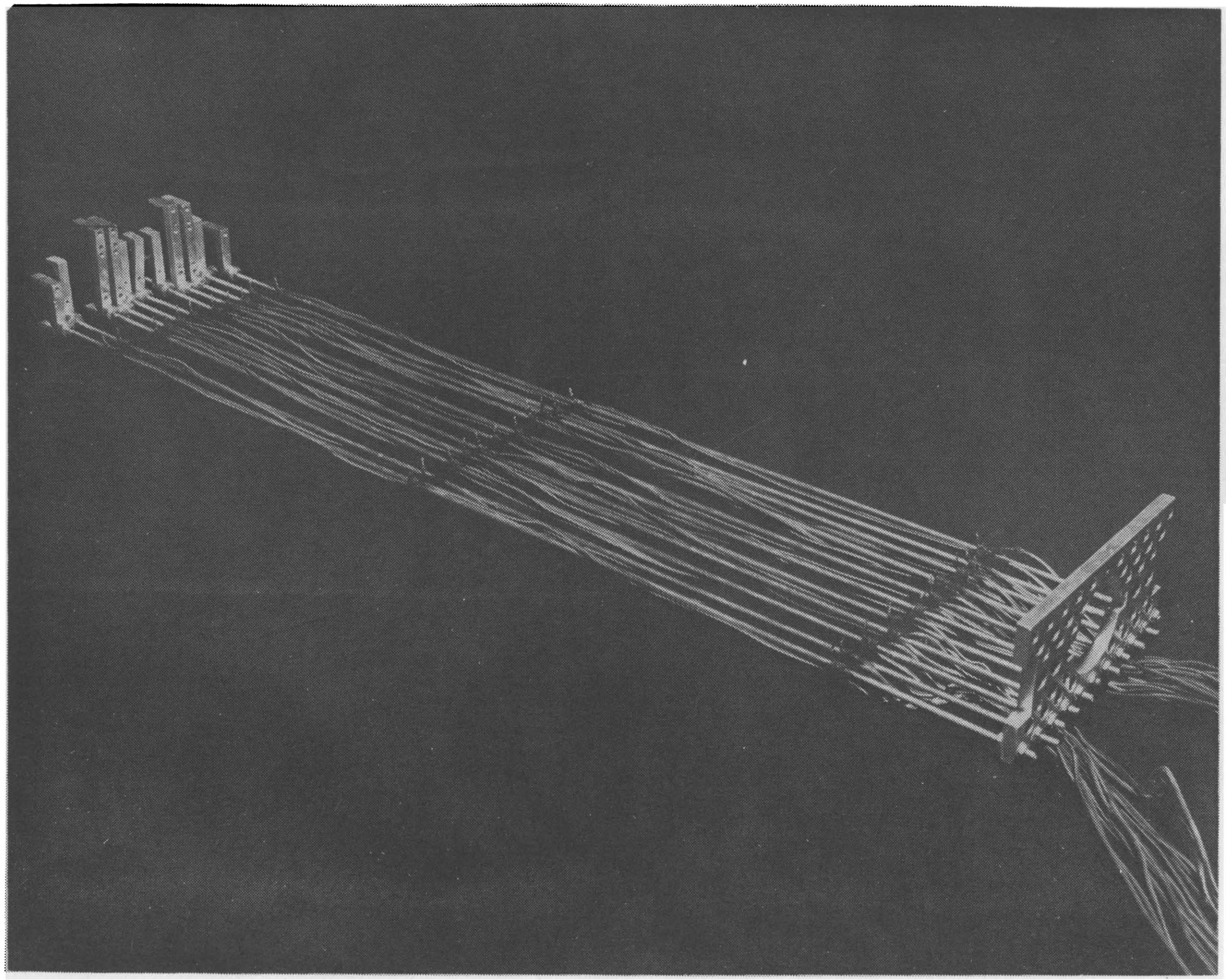

FIGURE 5 Inlet and Outlet Thermocouple Assemblies

average temperature profiles for the individual elements into a composite profile of the entire reactor core.

The thermoelectric emfs were measured with the circuit shown in Fig. 6. The Type K-3 Potentiometer was capable of reading directly to within 0.25 μvolts. The linearity of the thermocouple outputs was checked by placing the hot and cold junctions of the thermocouples in isolated insulated containers (Dewars) and varying the temperature of the hot junctions. No deviations from a linear output were observed for any of the thermocouples for the temperature range of the incore measurements.

When the thermocouples were in use within the reactor and after thermal equilibrium had been attained, slight variations in the outputs of the sensors located at the middle and top of the coolant channels were noticed. It is believed that turbulence, resulting from the construction of the fuel elements and possibly the thermocouple assembly, is responsible for these variations. The variations in the center readings, ± 1°F in as little as 10 seconds, were so large in comparison to the total bottom to top temperature difference that the center readings were not used in determining the profiles of the fuel elements. The top variations were much smaller than the middle ones. This is believed to be due to the flow patterns in the coolant channels becoming more stable as the water procedes up the channel.[3]

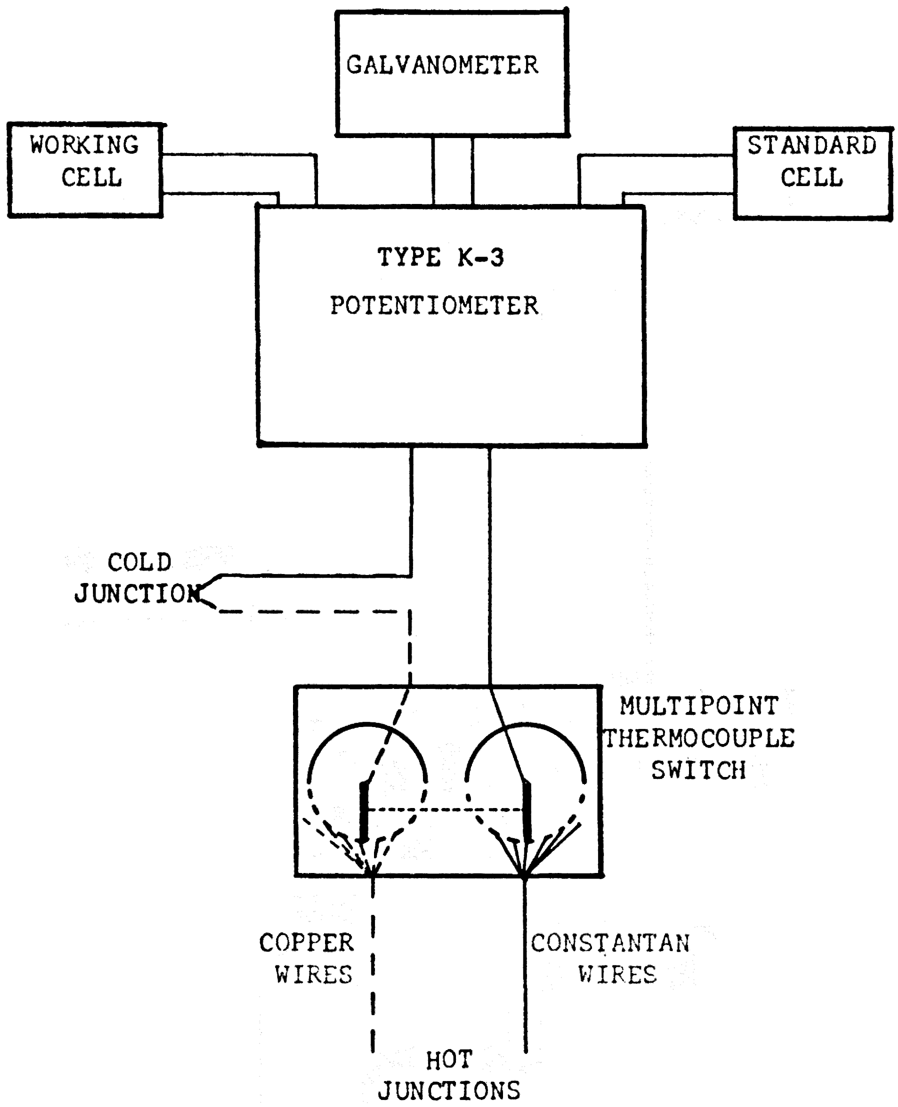

FIGURE 6 ELECTRICAL CIRCUIT

The temperature differences, which are shown in Table 1, were observed while the reactor was running at full power, 10 Kw, and with the flow rate to the core being maintained as close at 10 gpm as possible. The data shown have been normalized as discussed on page 14.

The temperature differences between the inlet and outlet of the west core tank was found to be much higher than the corresponding difference for the east core tank under normal operating conditions. This is also true for the individual elements as can be seen in Table 1. Since the thermal flux in the east core is approximately 85% of that in the west core,[6] and since the energy released and hence the observed temperature differences are directly related to the flux, the east core temperature rise should have been approximately 85% of the west core temperature rise. The actual ratio of the observed temperature rise in the east core to that in the west core at both the normal flow rate of 10 gpm and a higher flow of 15 gpm was 0.6:1. At approximately 2.5 gpm, however, the ratio was found to be 0.9:1. This latter ratio is well within the measurement error of the flux ratio. These data are summarized in Table 2.

It is hypothesized that for flow rates in which the outlet pipes are filled, the water from the east core will enter a low pressure region caused by the water from the west core flowing past the outlet pipe of the east core. This reduced pressure would cause

TABLE 1

TEMPERATURE PROFILES

Channel #	E-6	E-5	E-4	E-3	E-2	E-1	Channel #	W-6	W-5	W-4	W-3	W-2	W-1
			ELEMENT #						ELEMENT #				
E-5	6.1°F	5.3°F	5.4°F	4.5°F	5.0°F	4.1°F	E-5	9.1°F	9.0°F	8.5°F	9.0°F	8.5°F	9.1°F
E-4	---	5.5	5.5	4.9	5.6	5.1	E-4	9.1	8.9	9.2	9.3	8.2	8.9
E-3	6.1	5.4	5.0	4.0	5.5	5.2	E-3	9.1	8.9	9.1	9.3	8.6	8.8
E-2	5.7	5.4	5.6	4.5	6.3	4.9	E-2	9.3	9.1	9.3	9.2	8.6	8.9
E-1	5.8	5.4	5.0	5.0	6.2	5.7	E-1	8.3	8.7	8.5	9.4	8.5	9.1
C	5.9	5.9	5.6	5.3	7.4	---	C	9.0	9.1	9.1	9.3	9.0	9.1
W-1	5.8	5.5	5.6	4.9	5.7	5.0	W-1	8.8	---	---	9.1	8.5	9.0
W-2	6.1	5.3	4.6	5.0	6.0	5.2	W-2	9.0	9.3	9.3	9.2	8.0	8.6
W-3	6.3	5.9	4.9	4.6	6.0	6.0	W-3	9.2	8.8	8.4	9.0	8.9	9.1
W-4	6.3	5.0	5.2	4.7	5.7	5.3	W-4	9.3	8.6	8.6	9.4	9.1	9.1
W-5	6.0	5.7	5.0	4.7	5.0	4.8	W-5	9.2	8.8	---	9.1	8.6	9.0
Average	6.0	5.5	5.3	4.7	5.9	5.1	Average	9.0	9.1	9.1	9.2	8.6	9.0
Core Average			5.4				Core Average			9.0			

TABLE 2

FLOW-TEMPERATURE RISE VALUES

Flow Rate	Temperature Rise East Core	Temperature Rise West Core	East-West Ratio
15 gpm	4.1°F	6.8°F	0.6:1
10	6.4	10.7	0.6:1
2.5	28.5	31.2	0.9:1

a higher flow from the east core and the heat produced in this core would thus be removed faster. This could well be responsible for the extreme ratio of temperature rises for the two cores. Observation of the water level in the core may be carried out using an eye level gauge in the process pit. By means of this gauge it was determined that the outlet pipes were indeed full for the two higher flow rates in the Table 2. If the pressure drop is the cause of the anomalous temperature difference in the core, the difficulty can be remedied by taking the overflow pipe from a point midway between the core tanks.

The temperature profile for each of the cores would be expected to follow the flux distribution (an approximate cosine curve) if the same volume of coolant passed through each of the elements. It is apparent from the average temperature rises for the elements as shown in Table 1 that this is not the case. The profiles except for a slightly low value for element E-3 are very flat. This would indicate that the outer fuel elements are receiving less coolant than the elements which are located directly over the inlet pipes. There was a slight tendency for the outermost coolant channels in the two cores to be cooler than the inner channels.

After the temperature profile of the fuel elements and the reactor core had been determined and examined, a series of measurements were made to determine the temperature rises in the two core tanks as a function of the power level. The temperature at each of the inlets and outlets for the two tanks were measured using the same apparatus shown in Fig. 5.

Thermal equilibrium of the reactor, for any power level, was found to be reached at about three hours of operation at the level in question if the reactor had been previously heated by pumping hot water through the system. This equilibrium was observed by displaying the output of the thermocouples on chart recorders. The absolute values of the thermoelectric emfs were determined using the circuit shown in Fig. 6. The inlet and outlet temperatures, at each power level, were recorded over a thirty minute period following the three hours of operation. The long observation period was used in order to minimize the effects of heat exchanger fluctuations.

If the power temperature measurements had been made on different days there would have been appreciable errors in the measurements due to the differences in flow rates, ambient room temperature, and fission product decay. The measurements were therefore made in one continuous run. The effect of fission product decay was the only factor which would not affect all the measurements to the same degree. It was found that the fission product decay from the fission product buildup of previous power levels tended to make the final readings for a long run slightly high. This would change the slope of any curve which related temperature rise to reactor power. If the continuous run was made starting at the lower powers and finishing with the highest powers the decay effect was found to be negligible. After several preliminary runs to examine the fission product decay effect a final run was made. The results of this run are shown for each of the core tanks in Fig. 7. The temperature rises can be seen to vary linearly with power with the exception of the 1 Kw measurement. This is very likely because the reactor had not been warmed up before the 1 Kw measurement so that fluctuations were present which caused significant error in the measurements of the small temperature rise at this low power.

The quantities that were computed for the present reactor operating conditions were the maximum fuel plate surface temperature and the maximum fuel matrix temperature. These two temperatures were obtained by using the following equations:

$$t_o = t_m + \frac{Qa^2}{2K_f} + \frac{Qa(b-a)}{K_c} + \frac{Qa}{h}$$

$$t_2 = t_m + \frac{Qa}{h}$$

where the terms are defined as:

t_o - fuel matrix temperature

t_2 - fuel plate surface temperature

t_m - mixed mean coolant temperature

Q - volumetric heat source strength

a - half thickness of fuel matrix

b - half thickness of fuel plate

K_f - thermal conductivity of fuel matrix

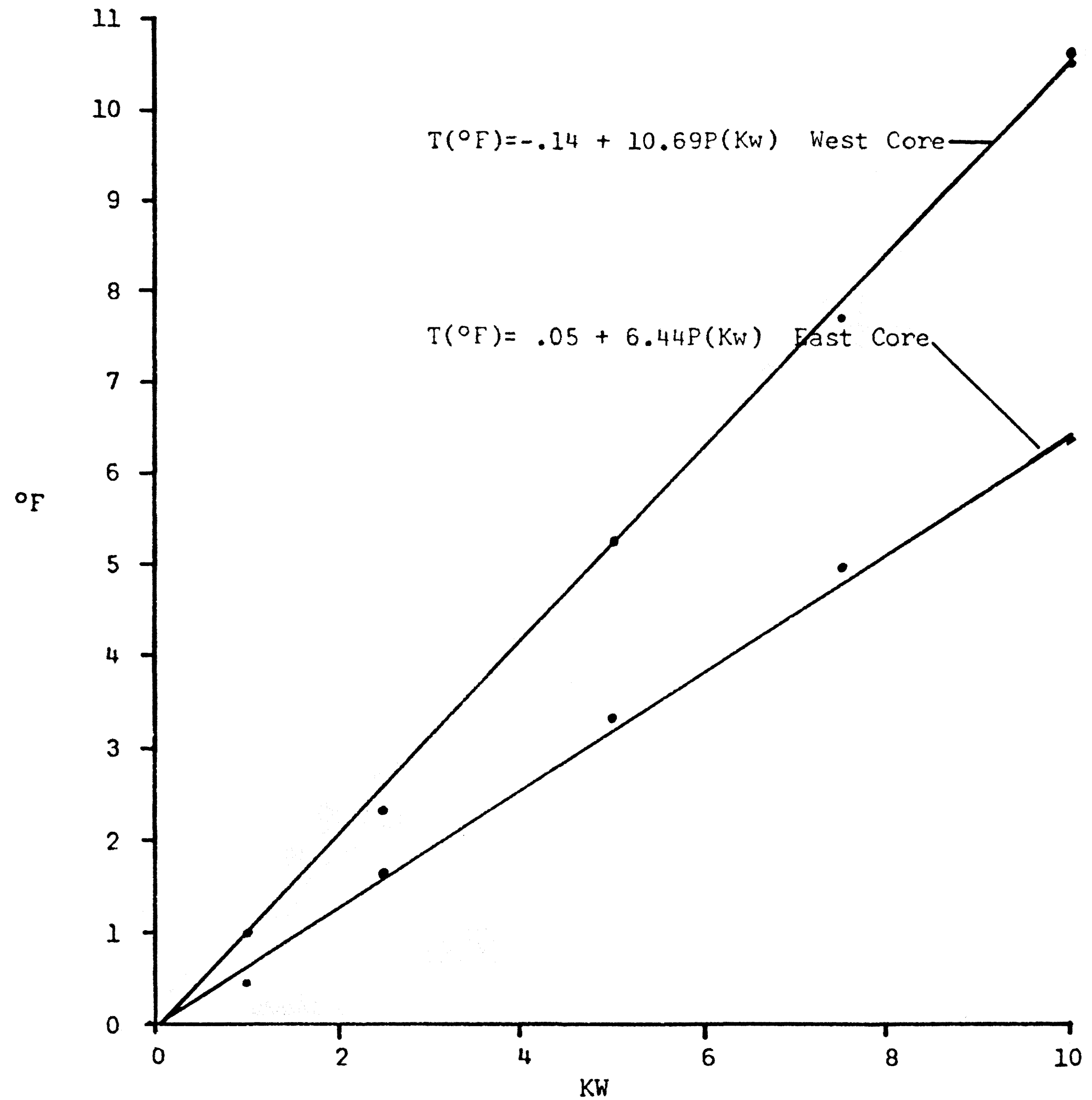

FIGURE 7 Power-Temperature Curve

K_c - thermal conductivity of cladding

h - heat transfer coefficient (surface to coolant).

The maximum observed values for the coolant temperature and the thermal flux were used in order that the largest possible values for the desired quantities would be obtained; this would be the most serious situation.

The following values were used to compute the fuel plate surface temperature and the fuel matrix temperature.

$t_m = 111°F$

$Q = 2.74 \times 10^5$ Btu/ft^2 hr (this corresponds to a thermal flux of 1.43×10^{11} n/cm^2 sec)

$a = 1.66 \times 10^{-3}$ ft

$b = 3.32 \times 10^{-3}$ ft

$K_f = 121$ Btu/ft hr °F

$K_c = 121$ Btu/ft hr °F

$h = 27$ Btu/ft^2 hr °F

The computed values for the temperatures were:

$t_2 = 127.85°F$

$t_o = 127.86°F.$

The calculated values were carried to enough places so that the difference between the surface temperature and the matrix temperature could be shown even though the use of five significant digits is not justified. It should again be noted that the above values are the extremum values and the actual temperatures throughout the reactor are usually much lower. The value of Q is certainly too high for even an extreme case, since the flux depression of the fuel element has not been taken into account.

The value of the heat transfer coefficient was computed by two different procedures. The first method was for forced convection in the laminar flow region and the second method was for free convection with no flow except that caused by the temperature rise of the water. In both methods the coefficient is computed from the Nusselt number by means of the equation $h = N_n K_w/D_e$ where

N_n - Nusselt number

K_w - thermal conductivity of water

D_e - effective diameter of the coolant channel.

For forced convection with laminar flow the Nusselt number is defined as $N_n = 2 N_g^{1/3}$ where N_g is the Graetz number which is given by $N_g = D_e N_r N_p/4L$. N_r is the Reynolds number, N_p the Prandtl number, and L is the length of the fuel plate. The value for h under forced convection was found to be 27.6 Btu/ft^2 hr °F. For free convection, $N_n = .52(N_g N_p)^{1/4}$. N_g is now defined as $D_e^3 p^2 B g dT/u^2$ with p the density of water, B the volume coefficient of expansion, g the acceleration of gravity, dT the temperature difference between the coolant and the wall, and u the viscosity of water. The computed value for h in free convection was 27 Btu/ft^2 hr °F. This latter value for h was used in the computations for the fuel plate temperature even though we have forced convection in the reactor since it gave a larger value for the temperature.

III. CONCLUSIONS

The temperature measurements that have been performed on the V.P.I. reactor indicate that no major changes are needed in the coolant system for the proposed increase in power from 10 Kw to 100 Kw, but a few minor changes are necessary.

For the present operating conditions with a flow rate of 10 gpm and the unequal coolant flows in the two core tanks, the predicted temperature rise for the west core at 100 Kw is 107°F and for the east core, 64°F. For the same conditions the temperature differences between the fuel plate surface and the coolant could be as much as 168°F in element W-3. All of the above predictions are based on the assumption that there is no boiling in the reactor system.

In order not to have the large difference in the temperature rise between the east and west cores, it is necessary that the two cores receive equal amounts of coolant. It should be possible to accomplish this by moving the overflow pipe to a point midway between the core tanks so that neither core outlet will cause a pressure drop at the other outlet. To assure that each core outlet will be at atmospheric pressure, the outlet pipes should be made large enough so that they will never be completely filled with water as they are at present when 10 gpm or more are pumped into the core.

Once the flow rates for the core tanks have been equalized, the maximum temperature rise for either core will be about 75°F at 10 gpm and a power of 100 Kw. This temperature rise can be further reduced by increasing the coolant flow to the core. The increased flow rate can be obtained by using larger control valves in the pumping system. The increased flow will necessitate replacing the 0-15 gpm pressure transducer with a transducer having the range 0-30 gpm. With the above changes, the flow rate to core can be increased to the capacity of the overflow pipes, under a gravity flow system, which is about 20 gpm. If a higher flow rate than this is desired, a suction pump can be placed on the overflow pipe and the flow rate may then be increased to the maximum capacity of the present circulating pump.

The estimated inlet-outlet temperature differences and the maximum fuel plate temperatures relative to the inlet temperature are shown in Table 3. These data are for an even distribution of the coolant between the two tanks with the assumption that no boiling occurs. The latter assumption is impossible as can be seen from column 3 of Table 3.

As just pointed out, there are no practical inlet temperature values which would keep the maximum fuel plate temperature below the boiling point of water. This is not as unfavorable a situation as one might think at first. Once the fuel plate temperature exceeds the boiling temperature of the coolant by a small amount, heat is

TABLE 3

PREDICTED TEMPERATURE RISES AND FUEL PLATE

TEMPERATURES FOR A REACTOR POWER OF 100 KW

Flow Rate	Inlet-Outlet Temperature Rise	Fuel Plate Temperature Relative to Inlet
10 gpm	75°F	+243°F
15	50	+218
20	38	+206

transferred from the fuel plate not only by convection but also by nucleate boiling. The heat transfer coefficient jumps tremendously at this point due to the additional heat of vaporization carried away by the coolant.

In all of the predictions made, the extreme values for all quantities have been used. The actual values for the temperatures of the fuel elements will be less than the extreme case for all of the elements except possibly W-3. It is very doubtful if even this element will attain the predicted temperatures due to the use of undepressed neutron flux and minimum values for the heat transfer coefficient. If the flux is depressed in the fuel matrix by only a factor of 2, the points on element W-3 where boiling might occur are very few or nonexistant.

Because of the high coolant temperatures which will be experienced at 100 Kw, there will be a very high humidity in the area over the core. There should be some type of ventilation to remove this humidity from the core area. There are plans for pumping air through the core to reduce the concentration of radioactive Argon. This pumping will probably solve the humidity problem.

IV. ACKNOWLEDGEMENTS

The author wishes to express his appreciated to Dr. A. Keith Furr for his guidance, assistance, and patience throughout this work. He would also like to thank Mr. Lewis J. Newmire, Dr. Andrew Robeson, Mr. John Tucker, Mr. Ephraim Stam, and Dr. James A. Jacobs for their help, advice, and suggestions on this thesis.

The author also wishes to express his appreciation for financial support for part of the research from the following source, Atomic Energy Commission Contract No. AT-(40-1)-2686.

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ABSTRACT

A major problem encountered in the designing and construction of any nuclear reactor system is the removal of heat from the core of the reactor. In the V.P.I. UTR-10, the heat is removed from the fuel plates by the light water moderator-coolant which is pumped between the plates.

The maximum temperature which the fuel will attain is determined by the coolant temperature, the coolant flow rate, and the reactor operating power. The latter two of these are given by the reactor instrumentation. No provisions were made which allowed the temperature of the coolant in the region of the fuel elements to be measured. In order to ascertain this coolant temperature, an array of copper-constantan thermocouples was inserted between the individual plates of the fuel elements. The temperature of the coolant near the bottom and top of the 144 fuel plates was found using this thermocouple arrangement.

These results were used to predict temperature of the coolant and fuel at a power level of 100 Kw. It is believed that with appropriate modifications of the present system no major difficulties should be encountered in increasing the reactor's licensed power from 10 Kw to 100 Kw.