THERMAL-HYDRAULIC ANALYSIS OF THE ITU TRIGA M ARK-11 REACTOR

Tayfun BÜKE[1] and Hasbi YAVUZ[2]

[1] Muğla University, Faculty of Science and Letters, Physics Department
48000-Mugla-Turkey
[2] Istanbul Technical University, Institute for Nuclear Energy
80626-Maslak, Istanbul, Turkey

Experimental and analytical studies have been performed to find out the temperature distribution, as a function of reactor power, in the TRIGA Mark-II reactor at Istanbul Technical University. A two-dimensional computer code was written in FORTRAN-77 language numerically solves heat conduction equation using finite difference method at the steady state. The calculated results for fuel temperature and coolant temperature distribution in the reactor core for different reactor power were compared with the experimental data. Agreements between experiment and results from the computer program are fairly good.

1. INTRODUCTION

Research reactor users have been developed reactor calculations computer codes, particularly for personal computers since the computer codes for transient and steady-state thermal-hydraulic analysis are constructed to power reactors.

Some of the thermal-hydraulic computer codes developed for the research reactors are given below.

The coupled mass, momentum and energy equations for the annular core were solved simultaneously using an explicit forward marching numerical technique [1].

A two dimensional conduction code for a finned fuel elements analysis were developed and that computer code was applied to a 30 MW research reactor located at Korea Atomic Energy Research Institute [2],

A computer program TRISTAN was developed for calculating natural convection parameters in TRIGA cores [3] and an experimental and analytical program has been performed to study the temperature distribution, as a function of reactor power, in the TRIGA reactor at the Pennsylvania State University [4].

All of the above mentioned analytical models were carried out for different types research reactors. Therefore it is difficult to apply one of the above computer codes to our TRIGA reactor since the TRIGA reactor cores have complicated core geometry (mixed core, water gaps, irradiation channels etc.).

In this present paper, experimental and analytical studies have been performed to find out the temperature distribution, as a function of reactor power, in the TRIGA Mark-II reactor at Istanbul Technical University. A computer code was written in FORTRAN-77 language numerically solves heat conduction equation using finite difference method at the steady state in (r,0) dimensions. The calculated results for fuel temperature and coolant temperature distribution in the reactor core for different reactor power were compared with the experimental data to check the accuracy of the computer program.

ITU TRIGA Mark-II reactor core is located at the bottom of a 6.096 m deep, 0.6096 m thick concrete annulus tank of 2 m inside diameter. The tank is filled with approximately 20 m^3 of water, which cools the core, by natural convection. The reactor is licensed to operate at a maximum steady-state thermal power level of 250 kW and 1200 MW pulse power with 3 $ excess reactivity. It has a cylindrical configuration with an annular graphite reflector. The top view of the reactor core is presented in Fig. 1. In total there are 91 locations in the core, which can be filled either by fuel elements or other components like control rods, a neutron source, irradiation channels, etc. Elements are arranged in six concentric rings: A, B, C, D, E, and F having 1, 6, 12, 18, 24 and 30 locations respectively. Each location corresponds to a hole in the aluminum upper grid plate of the reactor core. The diameter of the holes in the upper grid plate of the reactor core is 3.8227 cm. Cooling water passage through the top plate is provided by the differential area between a triangular spacer block on top of the fuel element and the round hole in the grid. The top grid plate is 1.905 cm thick and its also contains 16 foil insertion holes 0.8052 cm in diameter. The distances between locations in a given ring are equal according to Safety Analysis Report of the reactor [5]. The reactor core has 67 standard stainless steel-clad fuel elements with 20 % enrichment and 8.5 wt % uranium, 2 instrumented fuel elements, 16 graphite elements, 3 control rods, 2 irradiation channels and a neutron source. Uranium contents of the fuel elements in the reactor core are given in Table 1. A graphite reflector enclosed in aluminum casing surrounds the core. Standard fuel elements are cylindrical rods. The active section of the fuel element is 38.1 cm long and 3.6322 cm in diameter. The hydrogen to zirconium atom ratio of the fuel material is about 1.6 to 1. To facilite hydriding, 0.635 cm diameter hole is drilled through the center of the active fuel section. A zirconium rod is inserted in this hole after hydriding is complete. Graphite slugs are approximately 9 cm longs act as top and bottom reflectors. The active fuel section, top and bottom graphite slugs are contained in a 0.0508 cm thick stainless steel clad. The stainless steel clad is welded to the top and bottom end fittings. The top-ending fitting incorporates a triangular spacer block that positions the top of the element in the grid and yet provides passage for cooling water flow through the grid. An instrumented fuel element is in identical to standard fuel elements, except that it is equipped with three chromel-alumel thermocouples, embedded in the fuel meat. The sensitive tips of the thermocouples are located 0.689 cm from the center of the fuel element in the radial direction. The sensing tips of the fuel element thermocouples are located at the vertical centerline of the

fuel element, one at the center of the fuel section and the other two 2.54 cm above, and one 2.54 cm below the center.

All experimental measurements have been performed for the reactor core configuration given in Fig.1.

2. EXPERIMENTAL MEASUREMENTS

Fuel element temperature was measured at different radial location in the reactor core is given in Fig 1. As shown in Fig.1 thermocouple instrumented fuel elements within fuel elements identification numbers 8259 and 8258 are located in rings B1 and F9 respectively. We were only used the fuel element originally located in B1 for fuel temperature measurements in the reactor core since thermocouples of the fuel element located in ring F9 is not properly works. Fuel element located in B1 was shuffled to several different locations in different rings. Its temperature was measured in each location at different reactor powers. The core cooling system was on during the measurements. The regulating control rod regulated the reactor power while other two control rods were fixed positions. The temperature signal was taken from the thermocouple at the vertical center of the fuel element during the measurements. The fuel element thermocouples were tested and calibrated by using the digital voltmeter before the measurement. The fuel temperature was measured at two different locations in rings B, C, D, E and F for different reactor powers.

The coolant temperature can be easily measured at the core inlet and at its also making accurate measurement at the core outlet for the coolant temperature since the water mixes thoroughly within the coolant channel. On the other hand; direct measurement of the coolant temperatures within the reactor core is difficult; a thermocouple placed in a channel will be disturbed by the flow and occasionally touches the surface of a fuel rod, resulting in erroneous data. Therefore, the accurate experimental data, which is required for a thorough evaluation, can not be easily obtained. Because of the above mentioned difficulties we were repeated our measurements for several times to get the accurate experimental data which may be used as benchmark test cases for TRIGA reactor calculations.

The coolant temperature were measured as a function of axial and radial positions at different reactor powers in the core by using the experimental setup as shown in Fig.2. We were used to top grid plate foil insertion holes 0.8052 cm in diameter which are numbered 1, 2, 3 and 15 through which the thermocouple is inserted for making measurements at different core radius. At different radial position in the reactor core, the chromel-alumel thermocouple mounted at the tip of a 1.5 m long, 0.65 cm diameter aluminum tube was axially inserted 72 cm in 3 cm increments. Measured values are stored in a personal computer by using the data acquisition system installed to the reactor console [6].

Chromel-alumel thermocouple was calibrated by using the digital voltmeter before the measurements.

All measurements were within the experimental error interval of ± 2 °C .This error due to calibrating uncertainty of the fuel element thermocouple and also chromel-alumel thermocouple that is used in the experimental setup to measure coolant temperature in the reactor core.

3. THEORETICAL MODELLING

Assumptions are made to develop a computational model for calculating the temperature distribution in the reactor core. Those are thermal conductivity is constant, axial heat conduction is neglected, water temperature, convective heat transfer coefficient, and thermal conductivity's are uniform around a fuel element and in addition those assumptions the measured coolant temperatures are used in the analytical model. With these assumptions the fuel temperature distribution can be calculated throughout the reactor core.

The fuel temperature distribution within a fuel element is determined from the heat conduction equation in cylindrical coordinates. Since the fuel thermocouples are near the core midplane, the expressions are initially restricted to this region [7]:

$$\frac{d^2T(r)}{dr^2} + \frac{1}{r}\frac{dT(r)}{dr} + \frac{1}{r^2}\frac{d^2T(r)}{d\theta^2} = -\frac{q'''(r)}{k_f} \tag{1}$$

where

$q'''(r)$: volumetric heat generation rate at power P and position r within the fuel element

k_f : thermal conductivity of the fuel meat

Boundary conditions across the cladding and the cladding-coolant interface completely determine the temperature distribution in the reactor core. The boundary conditions are given as follows:

$$h[T_c(r = r_c,0) - T_\infty] = -k_c \frac{dT_c(r = r_c,0)}{dr} \tag{2}$$

$$\int_0^{2\pi} k_{cf} \frac{dT_c(r = r_f, 0)}{dr} d\theta = q' \tag{3}$$

$$k_f \vec{\nabla} T_f(r = r_f,0) = k_c \nabla T_c(\vec{r} = r_f,0) \tag{4}$$

$$T_f(r = r_f,0) = T_c(r = r_f,0) \tag{5}$$

$$T_f(r = r_z,0) = T_z(r = r_z,0) \tag{6}$$

$T_z(r = 0, 0)$ must be finite (7)

where

h : convective heat transfer coefficient across cladding-coolant interface

$T_c(r = r_c, 0)$: surface temperature of the cladding at angle 0

$T_c(r, 0)$: cladding temperature

$T_z(r, 0)$: zirconium rod temperature

$T^\wedge$: coolant temperature at the center of the coolant channel

k_c : thermal conductivity of cladding

r_z : radius of zirconium rod

r_f : radius of fuel meat

r_c : outer radius of fuel element

q' : heat generated per unit length

The volumetric heat generation rate is a function of the nuclear properties of the core and fuel element composition. It is defined in the fuel region of the core midplane by

$$q'''(P, R, \bar{r}) = C_F [X_{f1}(P, R, r)\phi_1 + X_{f2}(P, R, r)\phi_2 + X_{f3}(P, R, r) + X_{f4}(P, R, r)] \quad (8)$$

where

$q'''(P, R, \bar{r})$: volumetric heat generation rate at power P and position $(R, \bar{r})$

C_F : conversion factor

X_{fi} : macroscopic fission cross section at i^{th} group

$\phi_i(P, R, \bar{r})$: neutron flux at i^{th} group at power P and position $(R, \bar{r})$

R : position of fuel element centerline relative to core centerline

$\bar{r}$: position in fuel relative to fuel element centerline

TRIGLAV computer code is based on four-group time independent diffusion equation in two-dimensional cylindrical (r,0) geometry [8]. Diffusion equation is solved using finite difference method with iteration of fission density. Material constants are assumed to be step functions of local variables r and 0 . Every fuel and non-fuel element position in the core is treated as a unit cell. Macroscopic cross sections and diffusion coefficients for all unit cells are calculated with transport code WIMS-D/4 that is integrated in the program package. WIMS-TRIGA library is used in WIMS-D/4 calculations where isotopes specific to TRIGA core fuel were added (Er-166, Er-167, Sm and Hydrogen in ZrH). Macroscopic cross sections and diffusion coefficients are calculated for every unit cell in dependence of fuel or non-fuel element geometry, material composition, actual fuel element burnup, temperature, water temperature and density, cladding temperature and of xenon concentration.

Four-group neutron flux values for the reactor core are calculated by using the computer code TRIGLAV. 102 intervals in angular direction and 7 (A,B,C,D,E,F rings and graphite reflector) intervals in radial direction are used in the flux calculation of the reactor core. There are totally 90 mesh points in radial direction that are distributed 10 points per ring and 30 points per graphite reflector. 92 homogeneous unit cells are used to get the macroscopic cross sections and diffusion coefficients. Water channel cross sections are used instead of control rod cross sections in the neutron flux calculations.

The neutron flux used to calculate volumetric heat generation rate is a numerical function; hence the heat conduction equation is written by using the finite difference method in matrix form. After that, the finite difference equations and the necessary boundary conditions are solved iteratively [9].

To completely specify the fuel temperature distribution within the fuel element, We need two other thermal parameters namely; the temperature of the reactor coolant, determined experimentally and the convective heat transfer coefficient. The convective heat transfer coefficient is determined from the relation given as follows:

$$h = \frac{q\left[- r_z \right]^{2\,1}}{r_c \, (r = r_c) - T} \tag{9}$$

where

$T_c \ (r = r_c)$ the surface temperature of the cladding in the midplane of the core is determined by normalizing the shape of T_f (r) to be measured fuel temperature T_f (r = r_{tc}) .

On the other hand an empirical value of heat transfer coefficient was used in the thermal hydraulic analysis of the reactor [10]. There is a lot of empirical relation for the heat transfer coefficients, which are, depend on the special experimental measurement. Each one causes to get a different result when they are used in the calculation.

4. RESULTS AND CONCLUSIONS

In this present paper, experimental and analytical studies have been performed to find out the temperature distribution, as a function of reactor power, in the TRIGA Mark-II reactor at Istanbul Technical University. A computer code was written in FORTRAN-77 language numerically solves heat conduction equation using finite difference method at the steady state in (r,0) dimensions.

Measured values of the fuel temperature at different reactor powers are shown in Table 2. These results are also consistent with the measurements of the fuel element reactivity worth in the reactor core [11].

The results of the coolant temperature measurements as a function of radial and axial position in the reactor core at 100 kW and 250 kW are given in Table.3 and Table.4 respectively. As seen these tables axial increase in coolant temperature in hole number 15 is less than according to the others. This is an expected result because of the hole number 15 is surrounded with 1 fuel element and 2 graphite dummy elements while the other coolant channels are surrounded with three fuel elements.

The calculated results of the fuel temperature in the reactor core for different reactor powers are also given in Table 5.

When we compared the calculated results with the experimental values. Agreement exists between temperatures calculated with the two-dimensional computer code for the position of the thermocouple within the fuel and that measured with the instrumented fuel element thermocouple on rotating the instrumented fuel element. Some deviation exists between the calculated and measured fuel temperature as a function of angle 0. This is probably comes from the coolant temperature around a fuel element is not constant as assumed in these analysis and experimental determination of the heat transfer coefficient.

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Table 1. Uranium content in each fuel element in the reactor core

Fuel element location	Fuel element identification number	Uranium (g)	U-235 (g)
B1	8259	193.56	39.01
B2	8241	196.02	39.01
B3	8210	195.79	38.96
B4	8211	196.05	39.01
B5	8212	195.79	38.96
B6	8226	194.17	38.64
C1	8224	193.76	38.56
C2	8198	193.59	38.53
C3	8246	188.60	37.53
C4	8200	193.33	38.47
C5	8225	193.14	38.44
C6	8215	193.12	38.43
C8	8190	193.06	38.42
C9	8216	192.69	38.34
C10	8223	192.56	38.33
C11	8191	192.42	38.28
C12	8199	192.31	38.27
D1	8213	192.09	38.23
D2	8214	192.09	38.23
D3	8217	192.09	38.23
D4	8227	191.83	38.18
D5	8188	191.47	38.10
D6	8195	191.37	38.09
D7	8196	191.38	38.08
D8	8232	191.07	38.03

Table 1. (Continue)

Fuel element Location	Fuel element identification number	Uranium (g)	U-235 (g)
D9	8231	191.00	38.01
D10	8233	190.97	38.01
D11	8228	190.92	37.99
D12	8242	190.64	37.93
D13	8222	190.49	37.91
D14	8194	190.45	37.90
D15	8239	190.47	37.90
D17	8197	190.34	37.89
D18	8808	190.38	37.88
E1	8203	190.38	37.88
E2	8209	190.24	37.87
E3	8235	190.30	37.86
E4	8208	190.25	37.86
E6	8189	190.08	37.83
E7	8202	190.12	37.83
E8	8220	189.98	37.81
E9	8218	189.95	37.80
E10	8205	189.96	37.80
E11	8238	189.95	37.80
E12	8204	189.87	37.79
E13	8206	189.88	37.78
E14	8207	189.82	37.78
E15	8245	189.88	37.78
E16	8230	189.85	37.78
E17	8221	189.65	37.74
E18	8809	188.31	37.47
E19	8201	189.38	37.70
E20	8807	189.32	37.68

Table 1. (Continue)

Fuel element Location	Fuel element identification number	Uranium (g)	U-235 (g)
E21	8234	189.21	37.65
E22	8738	189.03	37.62
E23	8229	188.97	37.61
E24	8737	188.74	37.56
F1	8237	186.03	37.02
F2	8812	187.47	37.30
F4	8240	187.09	37.23
F5	8811	187.08	37.23
F6	8244	188.68	37.55
F7	8219	188.74	37.55
F8	8243	188.60	37.53
F9	8258	191.44	38.10
F27	8193	190.20	37.85
F28	8192	187.76	37.36
F29	8236	188.06	37.42
F30	8810	188.30	37.47

Table 2. Fuel temperature distribution measurements at different reactor powers

Fuel element location	T (^{O}C)						
	25 kW	50 kW	75 kW	100 kW	150 kW	200 kW	250 kW
B1	75.07	101.50	123.20	146.50	182.50	216.60	243.50
B3	74.96	100.82	121.97	144.55	179.60	213.20	237.93
C1	72.87	95.06	115.39	132.40	167.10	196.20	219.20
C5	69.89	89.27	108.93	124.55	156.48	182.34	202.25
D1	67.15	86.10	103.23	120.12	148.27	174.02	204.86
D7	60.43	76.92	92.83	107.53	132.02	153.43	179.27
E1	61.80	76.82	91.20	104.43	130.60	150.40	176.72
E9	54.07	67.98	79.33	89.69	113.19	129.17	152.05
F1	52.50	63.60	74.80	84.66	104.25	121.00	141.40
F28	50.87	60.68	71.67	81.38	99.81	115.25	134.26

Table 3. Coolant temperature distribution measurements at 100 kW

Axial height (cm)	T (^{O}C)			
	Hole number 1	Hole number 2	Hole number 3	Hole number 15
3	25.20	25.20	25.20	25.20
6	25.20	25.20	25.20	25.20
9	25.20	25.20	25.20	25.20
12	25.20	25.20	25.20	25.20
15	25.20	25.20	25.20	25.20
18	25.43	25.40	25.37	25.24
21	26.29	26.13	26.02	25.37
24	27.35	27.04	26.82	25.54
27	28.59	28.10	27.76	25.74
30	29.99	29.29	28.80	25.96
33	31.46	30.56	29.92	26.19
36	33.00	31.87	31.07	26.44
39	34.54	33.19	32.23	26.68
42	36.02	34.45	33.35	26.92
45	37.41	35.64	34.39	27.14
48	38.66	36.71	35.33	27.33
51	39.73	37.62	36.14	27.50
54	40.58	38.36	36.79	27.64
57	40.83	38.57	36.97	27.68
60	40.83	38.57	36.97	27.68
63	40.83	38.57	36.97	27.68
66	40.83	38.57	36.97	27.68
69	40.83	38.57	36.97	27.68
72	40.83	38.57	36.97	27.68

Table 4. Coolant temperature distribution measurements at 250 kW

Axial height (cm)	T (^{O}C)			
	Hole number 1	Hole number 2	Hole number 3	Hole number 15
3	25.20	25.20	25.20	25.20
6	25.20	25.20	25.20	25.20
9	25.20	25.20	25.20	25.20
12	25.20	25.20	25.20	25.20
15	25.20	25.20	25.20	25.20
18	25.55	25.50	25.46	25.26
21	26.85	26.61	26.44	25.46
24	28.46	27.99	27.66	25.72
27	30.35	29.60	29.08	26.02
30	32.45	31.40	30.66	26.35
33	34.70	33.32	32.35	26.71
36	37.02	35.31	34.10	27.07
39	39.34	37.29	35.85	27.44
42	41.59	39.21	37.54	27.80
45	43.69	41.01	39.12	28.13
48	45.59	42.63	40.55	28.43
51	47.21	44.02	41.78	28.69
54	48.52	45.14	42.76	28.90
57	48.89	45.46	43.04	28.96
60	48.89	45.46	43.04	28.96
63	48.89	45.46	43.04	28.96
66	48.89	45.46	43.04	28.96
69	48.89	45.46	43.04	28.96
72	48.89	45.46	43.04	28.96

Table 5. Calculated fuel temperature distribution at different reactor powers

Fuel element Location	T (^{O}C)						
	25 kW	50 kW	75 kW	100 kW	150 kW	200 kW	250 kW
B1	79.33	106.82	129.37	153.95	188.71	224.26	253.25
B3	77.86	105.21	126.95	151.27	186.68	220.71	245.26
C1	76.21	100.86	120.92	138.28	174.62	203.05	225.87
C5	74.37	94.98	115.65	130.42	162.52	188.63	208.02
D1	71.66	90.65	109.16	126.52	153.94	179.98	210.03
D7	65.49	81.93	97.98	114.26	136.92	158.57	184.54
E1	66.10	81.43	96.42	109.15	136.57	155.42	181.91
E9	56.95	71.65	83.29	93.62	117.58	133.34	156.48
F1	55.37	67.15	78.95	88.19	108.64	124.84	145.35
F28	53.84	63.28	74.21	85.20	104.72	118.86	137.96

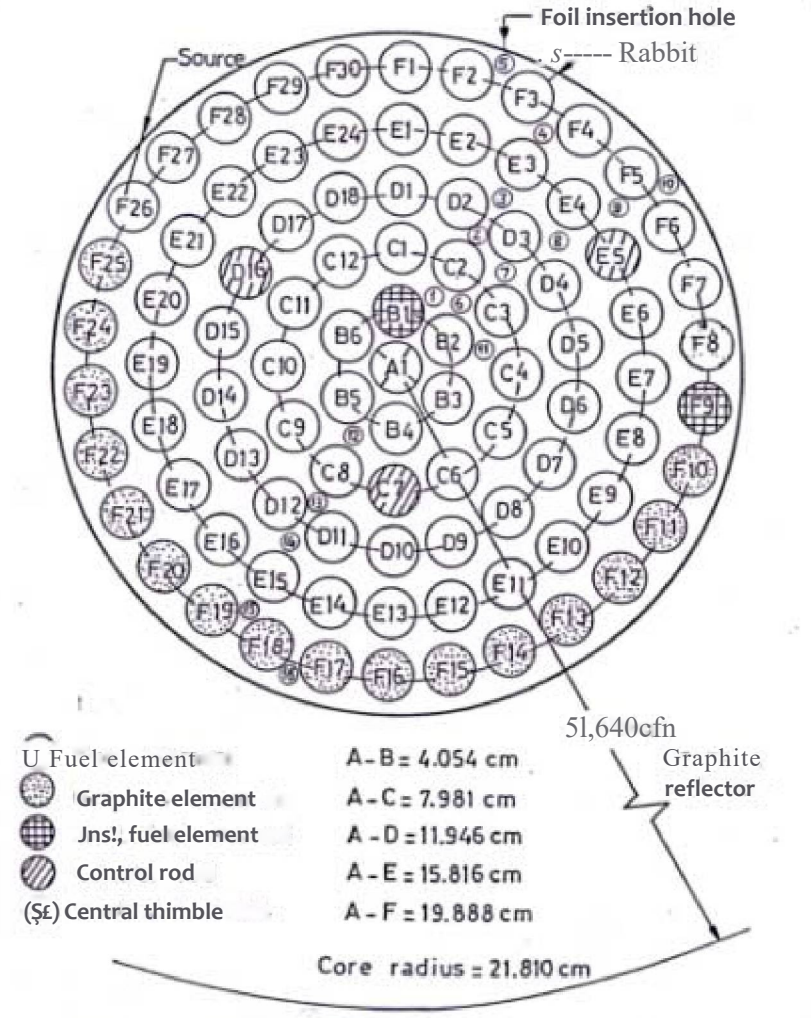
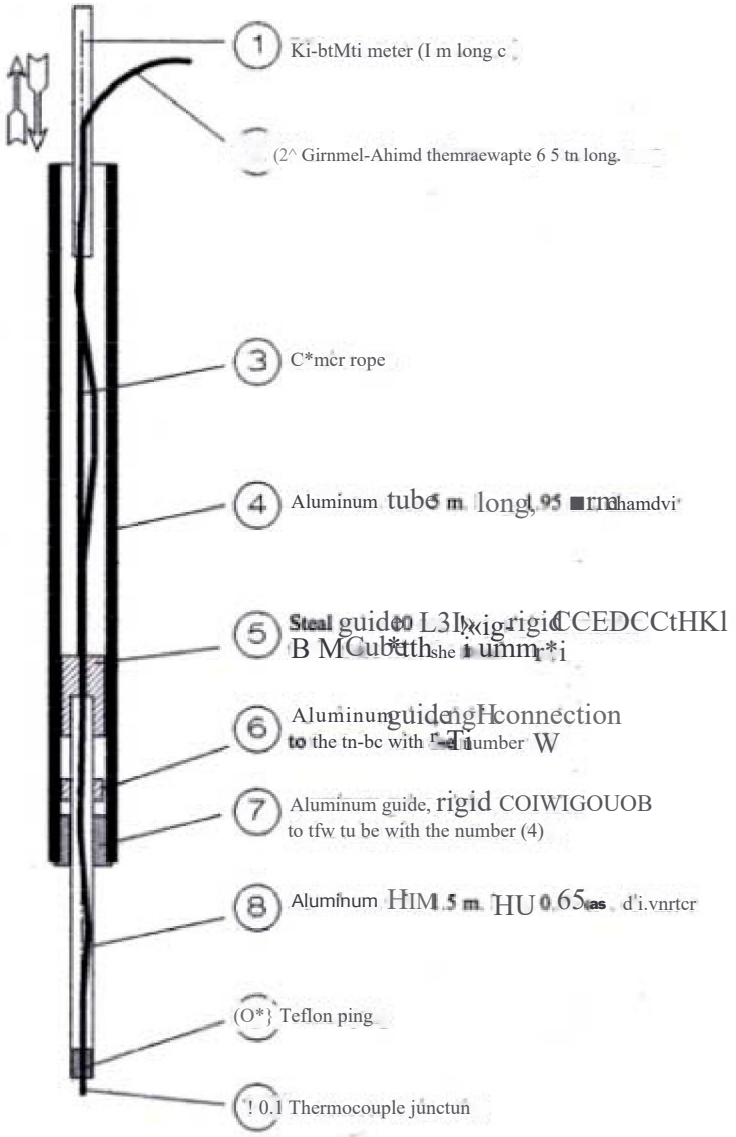


Fig. 1 nV-TRH core configuration



FIJT 2 Exptnmc/ual setup for the measurement of the a küttu temperature