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Experimental and analytical investigation of ITU TRIGA Mark-II reactor core

An experimental and analytical study 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 (ITU). The lumped parameter model with four governing equations was used in the analytical model. Based on the mathematical model, a computer code has been developed for calculating fuel and coolant temperatures in the reactor core. The calculated results for fuel and coolant temperature in the reactor core for different reactor power levels have been compared with the experimental data. Agreements between experiment and results from the computer code are fairly good.

Experimentelle und analytische Untersuchung des Reaktorkerns ITU TRIGA Mark-II

Um die Temperaturverteilung als eine Funktion der Reaktorkraft zu ermitteln, wurde in dem Reaktor TRIGA Mark-II an der Technische Universität Istanbul (ITU) eine experimentelle und analytische Studie durchgeführt. In dem analytischen Modell wurde das Klumpenparametermodell mit vier leitenden Gleichungen benutzt. Auf dem mathematischen Modell basierend wurde Computercode entwickelt, um die durchschnittliche Temperatur des Brennstoffs und des Kühlmittels im Reaktorkern zu berechnen. Die Ergebnisse der Berechnungen von durchschnittlicher Brennstoff- und Kühlmitteltemperatur im Reaktorkern unter verschiedenen Niveaus der Reaktorkraft wurden mit den experimentell ermittelten Daten verglichen. Die Übereinstimmungen zwischen den Daten des Experiments und den Ergebnissen aus dem Computercode sind bemerkenswerterweise gut.

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 for research reactor calculations have been developed given as follows:

A computer code THMOD2 has been developed for analyses of the steady state thermal hydraulic analysis of research reactors in which coaxial and/or plates type fuel elements are adopted [1].

A computer code has been developed for a finned fuel element analysis in two-dimensional situation and that computer code applied to a 30 MW research reactor located at Korea Atomic Energy Research Institute [2].

A computer code has been written namely TRISTAN [3] for calculation of natural convection parameters in TRIGA cores.

Experimental and numerical studies have been performed to calculate the temperature distribution, as a function of reactor power, in the TRIGA reactor at the Pennsylvania State University [4].

A computer code TRAP22 has been developed for reactivity transient analysis during natural core cooling operation of Egypt Test and Research Reactor Number 2 (ETRR-2) [5].

For the sensitivity analysis for those codes, comparisons with experimental data are required in order to verify the accuracy of the codes. For this reason, accurate and well-described benchmark experiments are performed in several research reactors.

For example, three-dimensional coupled kinetics thermal-hydraulic benchmark TRIGA experiments have been performed in one experimental study [6]. And another experimental study has been performed for TRIGA Mark-II reactor in Ljubljana to use the results of these measurements as a benchmark for validation of different computer codes and reactor calculation [7].

All of the above mentioned analytical models have been carried out for different types of research reactors and experimental studies for different types core configurations. Therefore it is difficult to apply one of the above computer codes to the ITU TRIGA Mark-II reactor since the TRIGA reactor cores have complicated core geometry (mixed core, water gaps, irradiation channels etc.).

In this study, experimental and analytical studies have been performed to find out the temperature distribution, as a function of reactor power, in the ITU TRIGA Mark-II reactor. A computer code has been developed for calculating fuel and coolant temperatures in the reactor core by using the lumped parameter model. The code simulates the reactor core by several channels. The calculated results for fuel and coolant temperature distribution in the reactor core for different reactor power levels have been compared with the experimental data to check the accuracy of the computer code.

2 Description of the ITU TRIGA Mark-II reactor core

ITU TRIGA Mark-II reactor is licensed to operate at a maximum steady-state thermal power level of 250 kW and 1200 MW pulse power with 2100 pcm 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 3.823 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 0.805 cm in diameter [8]. 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 3.632 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 0.051 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 0.762 cm from the center of the fuel element in the radial direction. As seen in Fig.2, 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.

3 Experimental measurements

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 measurements were repeated during the experiments for several times to get the accurate experimental data which may be used as benchmark test cases for TRIGA reactor calculations.

The coolant temperatures were measured as a function of axial and radial positions at different reactor power levels in the core configuration given in Fig.1 by using the experimental setup as shown in Fig.3.

1, 2, 3 and 15 numbered top grid plate foil insertion holes were used through which the thermocouple was 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 were stored in a personal computer by using the data acquisition system installed to the reactor console [9]. Chromel-alumel thermocouple was calibrated by using the digital voltmeter before the measurements.

Fuel element temperature was measured at different radial locations for the core configuration 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. The fuel element originally located in B1 was used for fuel temperature measurements in the reactor core.

Fuel element located in B1 was shuffled to several different locations in different rings of the reactor core. 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 power levels.

Fuel and coolant temperature measurements were within the experimental error interval of ± 2 ^{o}C. This error comes from calibrating uncertainty of the fuel element thermocouple for fuel temperature measurements and the error comes from the fluctuation uncertainty of the mixing behavior of inward and upward components of the coolant flow in the coolant channel for coolant temperature measurements.

4 Analytical model

A computer code has been developed for analysis of the steady state thermal hydraulic analysis of the reactor core by using the lumped parameter model. It simulates the reactor core by several channels. The analytical model is developed on the following assumptions. The axial power distribution of the reactor core is cosine and averaged over the channel height, there is no mass, momentum or energy transfer between the coolant channels, heat conduction in the axial direction is neglected, thermal-hydraulic properties of the coolant have a uniform distribution in any coolant channels, material properties of fuel and clad regions are constant. With these assumptions, according to the doubly lumped parameter model, the heat balance equations for the fuel element in the coolant channel (i, j) can be expressed as [10,11]:

$$\mu_1\left[\overline{T}_f(i,j) - \overline{T}_c(i,j)\right] = P_{av}(i,j) \tag{1}$$

$$\mu_1\left[\overline{T}_f(i,j) - \overline{T}_c(i,j)\right] = \mu_2\left[\overline{T}_c(i,j) - \overline{T}_m(i,j)\right] \tag{2}$$

Where (i, j) are radial and angular directions in the reactor core, $\overline{T}_f(i,j)$ and $\overline{T}_c(i,j)$ is average fuel meat and clad temperatures of the fuel element in the coolant channel (i, j) respectively, $P_{av}(i,j)$ average power of the fuel element in the coolant channel (i, j).

The power of all fuel elements in the midplane of the reactor core was calculated from the two-dimensional, TRIGLAV computer code [12]. It is based on four-group time independent diffusion equation in two-dimensional cylindrical (r,θ) geometry. 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 θ. 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 (^{166}Er, ^{167}Er, 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.

μ_1 and μ_2 are thermal resistance coefficients are obtained by solving the steady state, one dimensional heat conduction equation in the fuel element for cylindrical geometry. Referring to standard type TRIGA fuel element consisting of zirconium rod, fuel meat, cladding, and after several calculations:

$$\mu_1 = \frac{\pi H_f}{\dfrac{1}{4k_f} - \dfrac{r_f^2 + r_z^2}{8k_f(r_f^2 - r_z^2)} - \dfrac{r_z^4}{2k_f(r_f^2 - r_z^2)^2}\ln\dfrac{r_z}{r_f} + \dfrac{r_c^2}{2k_c(r_c^2 - r_f^2)}\ln\dfrac{r_c}{r_f} - \dfrac{1}{4k_c}} \tag{3}$$

$$\mu_2 = \frac{\pi H_f}{\dfrac{1}{2r_c h_m} + \dfrac{1}{4k_c} - \dfrac{r_f^2}{2k_c(r_c^2 - r_f^2)}\ln\dfrac{r_c}{r_f}} \tag{4}$$

Where h_m is the free convection heat transfer coefficient and the remaining notations in Eq.(3) and Eq.(4) are given in Table 2.

Heat balance equation for coolant in the coolant channel (i, j) is expressed as:

$$\mu_2\left[\overline{T}_c(i,j) - \overline{T}_m(i,j)\right] = 2\,\overline{u}_m(i,j)\overline{\rho}_m(i,j)c_{p,m}A_m\left[\overline{T}_m(i,j) - T_{in}\right] \tag{5}$$

Where $\overline{T}_m(i,j)$ and $\overline{u}_m(i,j)$ are average coolant temperature and velocity in the coolant channel (i, j) respectively.

Momentum equation along the coolant channel (i, j) is calculated according to the following equation:

$$\left(K_{in} + K_{out} + \frac{fH_c}{D_h}\right)_{i,j}\overline{u}_m^2(i,j) = 2gH_c\frac{\rho_{in} - \overline{\rho}_m(i,j)}{\overline{\rho}_m(i,j)} \tag{6}$$

Where g is the gravity acceleration, f is the friction loss factor which is calculated in the following functional forms [11]:

a) Laminar flow

$$f = \frac{64}{\mathrm{Re}}, \qquad \mathrm{Re} \langle 2000 \tag{7}$$

b) Turbulent flow

$$f = 0.184(\mathrm{Re})^{-0.2}, \qquad \mathrm{Re} \rangle 2000 \tag{8}$$

Where Re Reynold number that is defined as:

$$\text{Re} = \frac{\overline{\rho}_m \overline{u}_m D_h}{\mu_m} \tag{9}$$

Hydraulic diameter D_h in Eq. (9) is calculated for the rod centered sub channels. The free convection heat transfer coefficient h_m on a vertical cylinder is calculated in terms of the dimensionless groups from the literature in the following functional forms [5,13,14]:

a) Laminar flow

$$h_m = \frac{k_m}{H} 0.59(Gr\,\text{Pr})^{1/4} \quad , \quad 10^4 \le Gr\,\text{Pr} \le 10^9 \tag{10}$$

b) Turbulent flow

$$h_m = \frac{k_m}{H} 0.10(Gr\,\text{Pr})^{1/3} \quad , \quad 10^9 \le Gr\,\text{Pr} \le 10^{13} \tag{11}$$

Where H is the length and Gr and Pr Grashoff and Prandtl numbers those are defined as:

$$Gr = \frac{g\beta_m \overline{\rho}_m^2 \left(\overline{T}_c - \overline{T}_m\right) H^3}{\mu_m^2} \tag{12}$$

$$\text{Pr} = \frac{\mu_m c_{p,m}}{k_m} \tag{13}$$

In addition, empirical relations represent the temperature dependence for the physical properties of coolant.

The solution of the above –mentioned discretization equations are solved by the Gauss-elimination method for two-dimensional (r,θ) situation.

After the development of a FORTRAN program to solve the set of algebraic equations, it was run to obtain average fuel and coolant temperature distribution in the reactor core. The main input data are listed in Table 2 [8].

The computer code also contains the following analytical solutions for fuel meat and coolant temperatures in order to compare the calculated results with the experimental values:

$$\left[T_f(r_{tc})\right]_{i,j} = \overline{T}_f(i,j) - \frac{P_{av}(i,j)}{\pi H_f\left(r_f^2 - r_z^2\right)}\left[\frac{\left(r_{tc}^2 - r_z^2\right)}{4k_f} - \frac{\left(r_f^2 + r_z^2\right)}{8k_f} + \frac{r_z^2 r_f^2}{2k_f\left(r_f^2 - r_z^2\right)}\ln\left(\frac{r_f}{r_{tc}}\right) - \frac{r_z^4}{2k_f\left(r_f^2 - r_z^2\right)}\ln\left(\frac{r_z}{r_{tc}}\right)\right] \quad (14)$$

$$\left[T_m(z)\right]_{i,j} = T_{in} + \left(\overline{T}_m(i,j) - T_{in}\right)\left\{1 + Sin\frac{\pi z}{H}\right\} \quad (15)$$

Where $\left[T_f(r_{tc})\right]_{i,j}$ is temperature of thermocouple in the fuel meat at position r_{tc} for fuel element in the channel (i,j) and $\left[T_m(z)\right]_{i,j}$ is temperature of coolant at the position z for channel (i,j).

The above expressions permit to approximate thermocouple temperature of fuel element and axial temperature distributions of coolant with the help of the average temperatures $\overline{T}_f(i,j)$, $\overline{T}_m(i,j)$, i.e. the lumped parameters.

5 Comparison of experimental measurements with calculations

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 shown in Table 3.

As shown in Table 3, axial increase in coolant temperature in hole-15 is less than according to the others. This is an expected result because of the hole-15 is surrounded with one fuel element and two graphite dummy elements while the other coolant channels are surrounded with three fuel elements. And the coolant temperature difference between the channel inlet and exit increase with increasing heat input to the coolant channel from the fuel elements.

Calculated results for coolant temperature in the midplane of the reactor core at 100 kW and 250 kW are shown in Table 4. The calculated coolant temperatures are approximately 8 % greater than the measured coolant temperatures at the axial position z=36 cm which is midplane of the reactor core.

The experimental and calculated fuel temperature in dependence of reactor power is given in Table 5. The calculated fuel temperatures are approximately 5% greater than that measured fuel temperatures according to these values.

In general, both calculated and experimental values for fuel and coolant temperatures show good agreement. Some deviation exists between the calculated and experimental values. This is probably due to the fact that the assumptions are used to develop the analytical model.

6 Conclusions

In the present work, detailed measurements have been carried out for the fuel and coolant temperatures, as a function of reactor power, in the ITU TRIGA Mark-II reactor core. After that a computer code has been developed for calculating fuel and coolant temperatures in the reactor core by using the lumped parameter model.

The calculated and experimentally measured values of the fuel element and coolant temperatures show good agreement in general. These results for fuel temperatures are also consistent with the measurements of the fuel element reactivity worth in the reactor core [15].

Special effort was made to perform the temperature measurement experiments since the temperature measurements are difficult in the TRIGA reactor core coolant channels. For this reason, it is proposed to use the results of these measurements as a benchmark for validation of different computer codes and reactor calculations.

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Figure caption

Fig. 1. Core configuration of ITU TRIGA Mark-II reactor

Fig. 2. Instrumented fuel element

Fig. 3. Experimental setup for the measurement of the coolant temperature

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

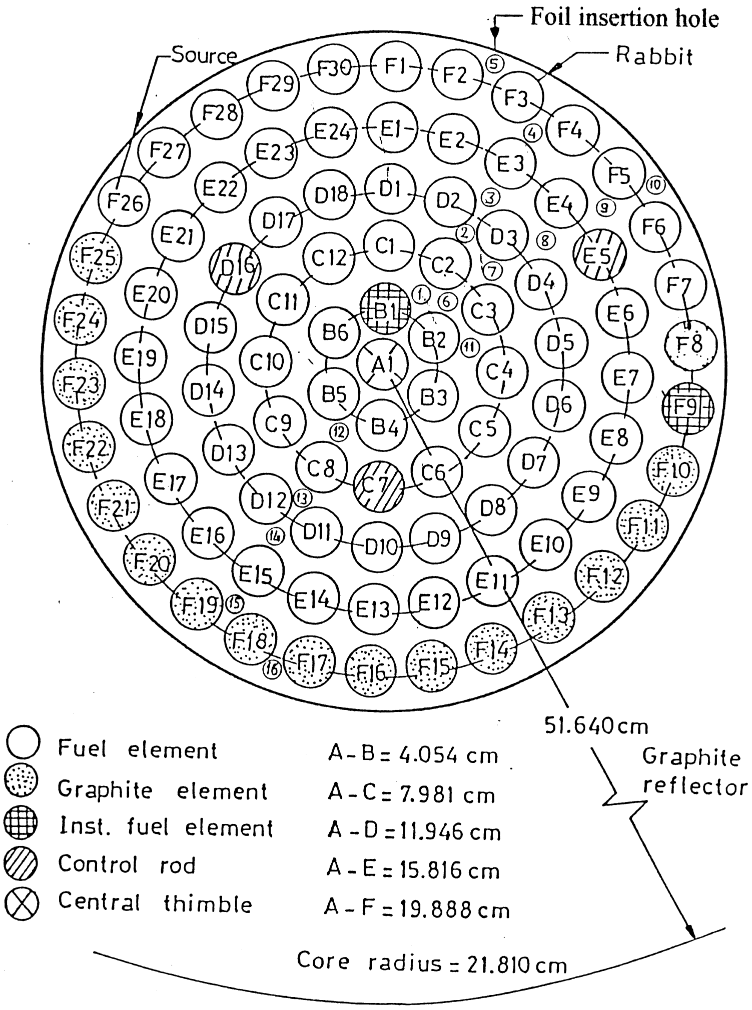
Table 2. Main input data

Table 3. Coolant temperature distribution measurements at 100 kW and 250 kW

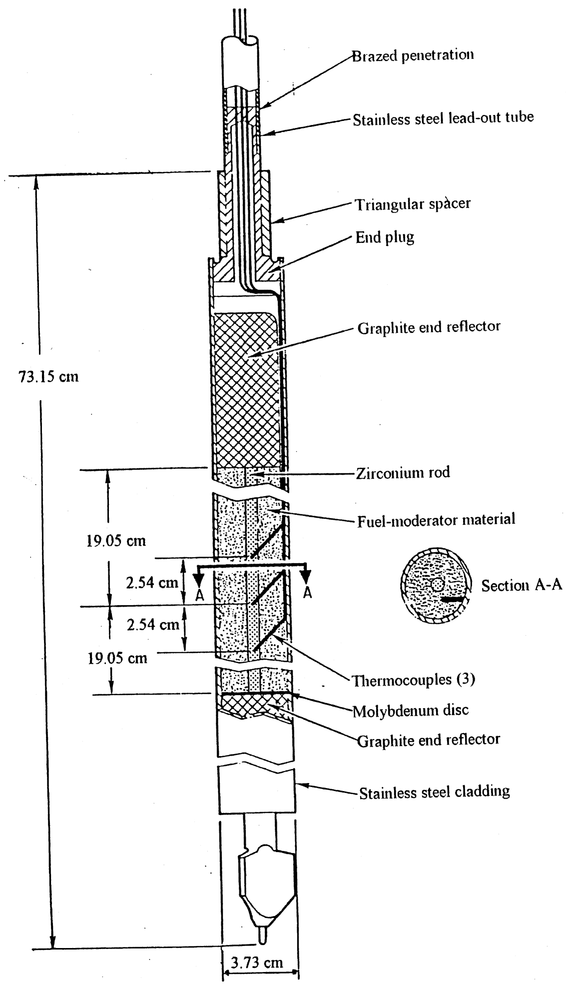
Table 4. Calculated coolant temperatures at 100 kW and 250 kW

Table 5. Experimental and calculated fuel temperature in dependence of reactor power

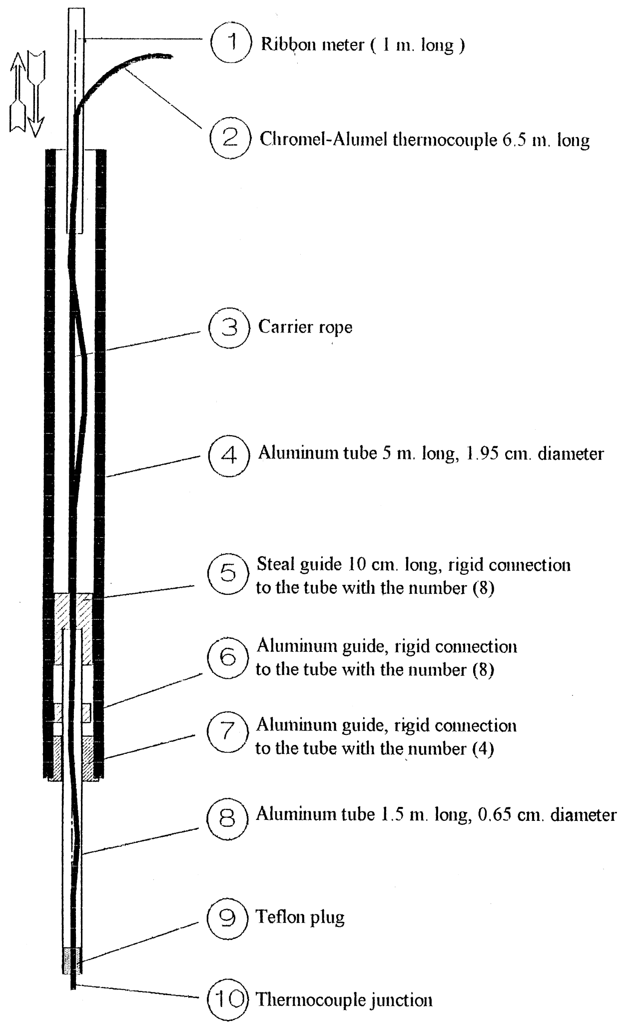
"Fig. 1"


Foil insertion hole
Rabbit
Source
51.640cm
Graphite reflector
Fuel element
Graphite element
Inst. fuel element
Control rod
Central thimble
A-B = 4.054 cm
A-C = 7.981 cm
A-D = 11.946 cm
A-E = 15.816 cm
A-F = 19.888 cm
Core radius = 21.810 cm

"Fig. 2"


Brazed penetration
Stainless steel lead-out tube
Triangular spàcer
End plug
Graphite end reflector
73.15 cm
Zirconium rod
Fuel-moderator material
19.05 cm
2.54 cm
A
A
Section A-A
2.54 cm
19.05 cm
Thermocouples (3)
Molybdenum disc
Graphite end reflector
Stainless steel cladding
3.73 cm

"Fig. 3"


1 Ribbon meter (1 m. long)
2 Chromel-Alumel thermocouple 6.5 m. long
3 Carrier rope
4 Aluminum tube 5 m. long, 1.95 cm. diameter
5 Steal guide 10 cm. long, rigid connection to the tube with the number (8)
6 Aluminum guide, rigid connection to the tube with the number (8)
7 Aluminum guide, rigid connection to the tube with the number (4)
8 Aluminum tube 1.5 m. long, 0.65 cm. diameter
9 Teflon plug
10 Thermocouple junction

" Table 1"

Fuel element Location	Fuel element identification number	Uranium [g]	^{235}U [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
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

" Table 1 (Continue)"

Fuel element Location	Fuel element identification number	Uranium [g]	^{235}U [g]
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
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"

Explanation	Symbol	Value
Radius of zirconium region [cm]	r_z	0.3175
Thermocouple location from the center of the fuel element [cm]	r_{tc}	0.762
Radius of fuel meat [cm]	r_f	1.8161
Radius of fuel element [cm]	r_c	1.8669
Fuel length [cm]	H_f	38.10
Coolant channel height [cm]	H_c	72.06
Coolant channel area [cm^2]	A_m	5.388
Thermal conductivity of fuel meat region [W/cm.^{o}C]	k_f	0.1758
Thermal conductivity of clad region [W/cm.^{o}C]	k_c	0.1665
Coolant inlet temperature [^{o}C]	T_{in}	25
Coolant inlet density [g/cm^3]	ρ_{in}	0.997
Total pressure loss factor at channel inlet and exit	$K_{in} + K_{out}$	2.6
Coolant density [g/cm^3]	ρ_m	$\rho_m(T)$
Coolant specific heat [J/g. ^{o}C]	$c_{p,m}$	$c_{p,m}(T)$
Coolant dynamic viscosity [g/cm. s]	μ_m	$\mu_m(T)$
Coolant thermal conductivity [W/cm.^{o}C]	k_m	$k_m(T)$
Volumetric expansion coefficient [1/^{o}C]	β_m	$\beta_m(T)$

“ Table 3”

Axial height [cm]	Hole-1		Hole-2		Hole-3		Hole-15	
	100 kW	250 kW	100 kW	250 kW	100 kW	250 kW	100 kW	250 kW
3	25	25	25	25	25	25	25	25
6	25	25	25	25	25	25	25	25
9	25	25	25	25	25	25	25	25
12	25	25	25	25	25	25	25	25
15	25	25	25	25	25	25	25	25
18	25	26	25	26	25	26	25	25
21	26	27	26	27	26	26	25	25
24	27	29	27	28	27	28	25	25
27	29	31	28	30	28	29	25	26
30	30	32	29	31	29	31	26	26
33	32	35	31	33	30	32	26	27
36	33	38	32	36	31	35	26	27
39	35	39	34	37	33	36	27	28
42	36	42	35	39	33	38	27	28
45	37	44	36	41	34	39	27	28
48	39	46	37	43	35	41	27	28
51	40	47	38	44	36	42	28	29
54	41	49	39	46	37	43	28	29
57	41	49	39	46	37	43	28	29
60	41	49	39	46	37	43	28	29
63	41	49	39	46	37	43	28	29
66	41	49	39	46	37	43	28	29
69	41	49	39	46	37	43	28	29
72	41	49	39	46	37	43	28	29

“ Table 4”

Location	P=100 kW	P=250 kW
Hole-1	35	41
Hole-2	34	39
Hole-3	33	38
Hole-15	28	29

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“ Table 5”

	P=100 kW		P=150 kW		P=200 kW		P=250 kW	
Location	E	C	E	C	E	C	E	C
B1	147	155	183	191	217	226	244	255
B3	145	152	180	189	213	221	238	247
C1	132	140	167	176	196	204	219	227
C5	125	131	157	164	182	189	210	217
D1	120	127	148	155	174	180	205	211
D7	108	114	132	138	153	160	179	186
E1	104	110	131	138	150	157	177	182
E9	90	95	113	119	129	135	152	158
F1	85	89	104	110	121	126	141	145
F28	81	86	100	106	115	120	134	138

E = Experiment and C = Calculated