IEA-R1 NUCLEAR REACTOR: FACILITY SPECIFICATION AND EXPERIMENTAL RESULTS

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Abstract

The report provides technical details on the IEA-R1 nuclear reactor core and immediate structure for analysis purposes, and describes experimental benchmark problems as conducted at the IEA-R1 nuclear reactor. The goal of the report is to provide sufficient geometric and material data to build a computational neutronic model of the facility, and to provide sufficient experimental details to enable simulation of the experiments.

1. GENERAL DESCRIPTION

The Nuclear and Energy Research Institute (Instituto de Pesquisas Energéticas e Nucleares, IPEN) IEA-R1 is a 5 MW pool type research reactor classified among the Material Testing Reactor (MTR) type. The core consists of 20 fuel assemblies, each with 18 fuel plates assembled on two lateral support plates forming 17 independent closed flow channels. Figure 1 shows a simplified scheme of the primary and secondary loop.

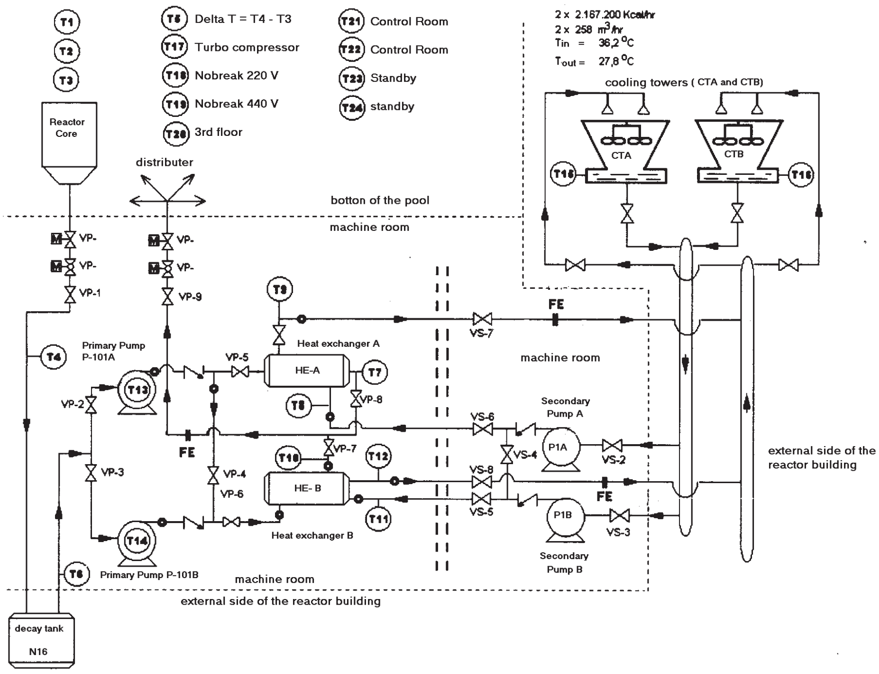


FIG. 1. Schematic process diagram of the IEA-R1 reactor.

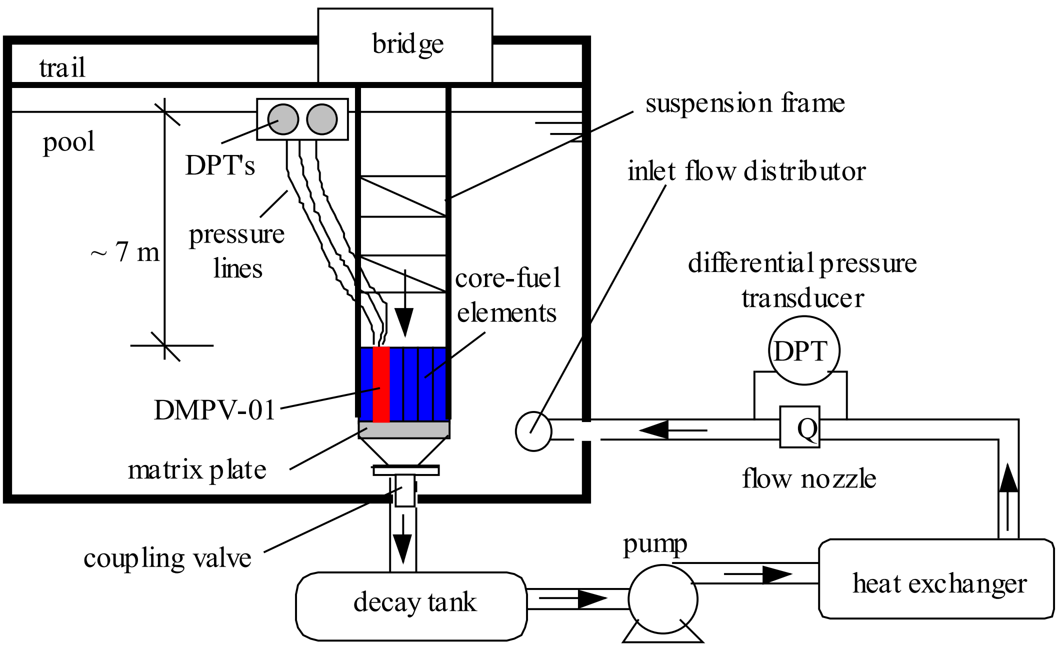
Continuous operation of the reactor, at high power levels, is permitted by removing the generated nuclear heat with the following auxiliary systems:

(a) During the reactor operation, a forced circulation coolant system pumps pool water down through the fuel assemblies to remove the fission heat from the reactor.
(b) A water to water heat exchanger transfers the generated heat to a secondary water coolant system.
(c) The secondary system carries heated water to the cooling tower, which dissipates the heat to the atmosphere.

The primary water is returned to the reactor pool. Water from the cooling tower is re-circulated through the secondary system.

For 5 MW operation, loop A is involved (pump P-101A or P-101B, heat exchanger HE-B, pump P1A or P1B, and cooling tower CTB). The other heat exchanger and cooling tower are used only for low power operation (<3 MW). Temperatures T1, T2 and T3 (T1 and T2 are redundant) are the water pool temperature at the inlet of the core, and T4 is the outlet temperature.

The primary cooling system of the IEA-R1 consists of a pool, piping, decay tank, pumps, heat exchangers, flow meter system, distributor, valves and structures (see Fig. 1). Figure 2 is a simplified depiction of the primary system. Figure 3 shows simplified illustration of the primary loop. The primary pump circulates water through the core to remove the heat generated during the reactor operation. The water then flows through the decay tank to decrease the N16 activity before entering the heat exchanger, which transfers the heat to the secondary cooling system.



DPT — differential pressure transducer; DMPV-01 — pressure and flow measure device; Q — volumetric flow nozzle.

FIG. 2. Simplified drawing of primary system showing the coupling valve.

A manually actuated pneumatic system lifts a device, called a header, coupling the outlet nozzle to the core matrix plate. The pump is then turned on, and the primary operating flow rate is adjusted. The pneumatic system is turned off, and the device is kept coupled by the hydrodynamic force resulting from the pressure difference. The reactor power operation is adjusted. If the primary flow rate decreases below the setpoint value (90%), the reactor is shut down and the coupling device falls due to gravity, and then the residual heat is removed through natural circulation in the reactor pool.

Figure 3 shows the location of the reactor core inside the reactor pool. The total volume of water in the pool is approximately 272 m^3. Figure 4 shows a detail of the coupling valve and matrix plate, and Table 1 lists the lengths and diameters of the pipes. Figures 5–8 show the isometric pipe of the IEA-R1.

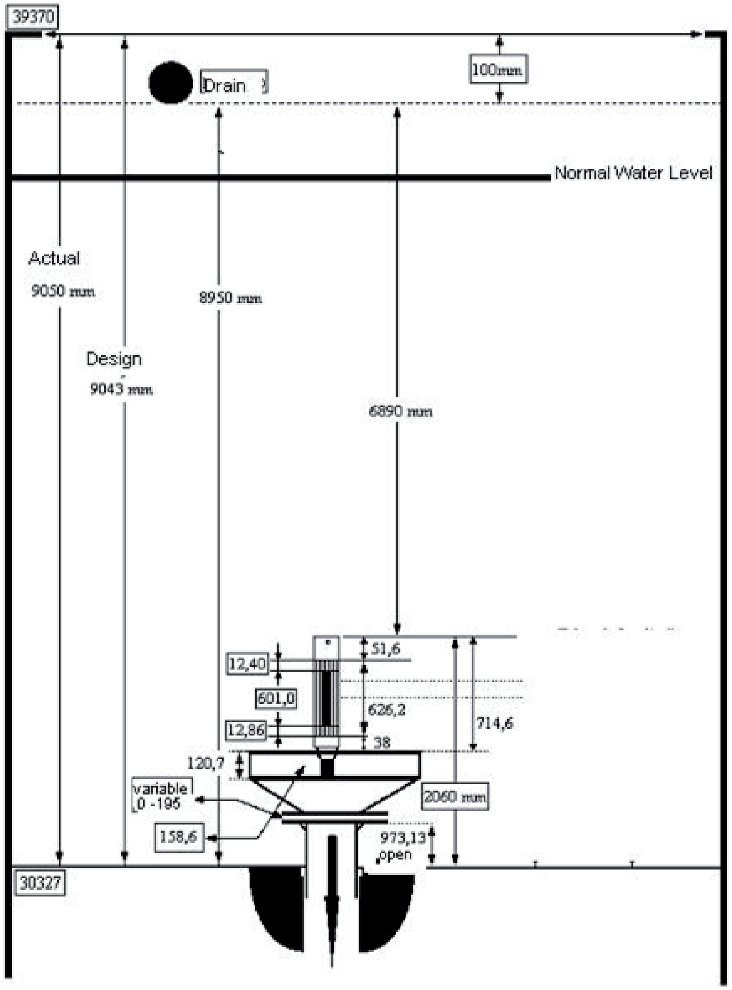


FIG. 3. Transverse view of reactor pool.

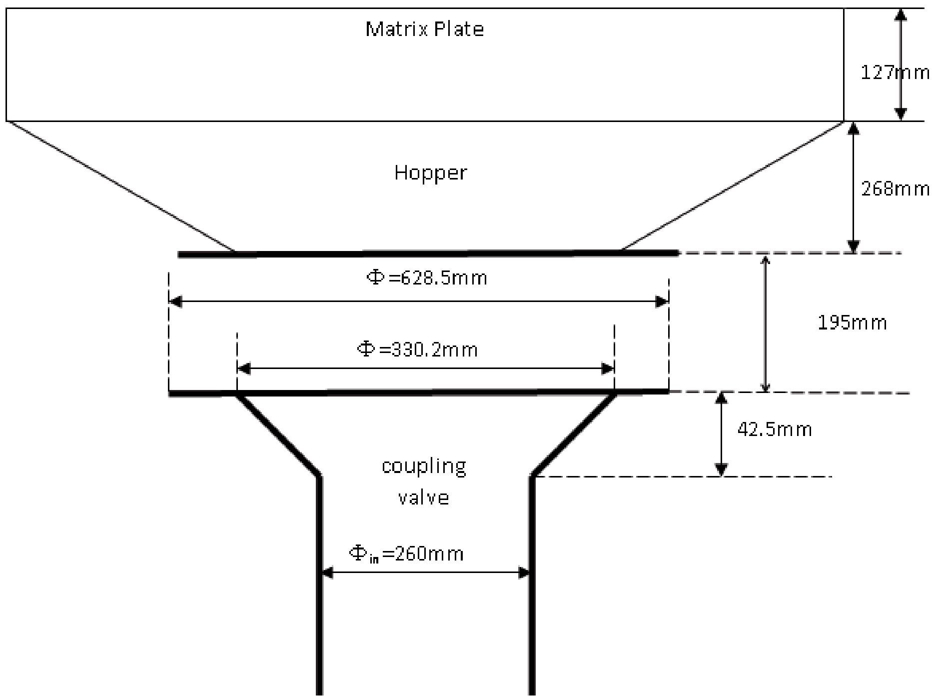


FIG. 4. Scheme of the coupling valve.

TABLE 1. APPROXIMATE LENGTH AND DIAMETER OF PIPES

Pipe	Diameter (mm)
25.4 cm (10 in) sched 5S	
$D_{internal}$	266.25
$D_{external}$	273.05
30.5 cm (12 in) sched 5S	
$D_{internal}$	315.95
$D_{external}$	323.88

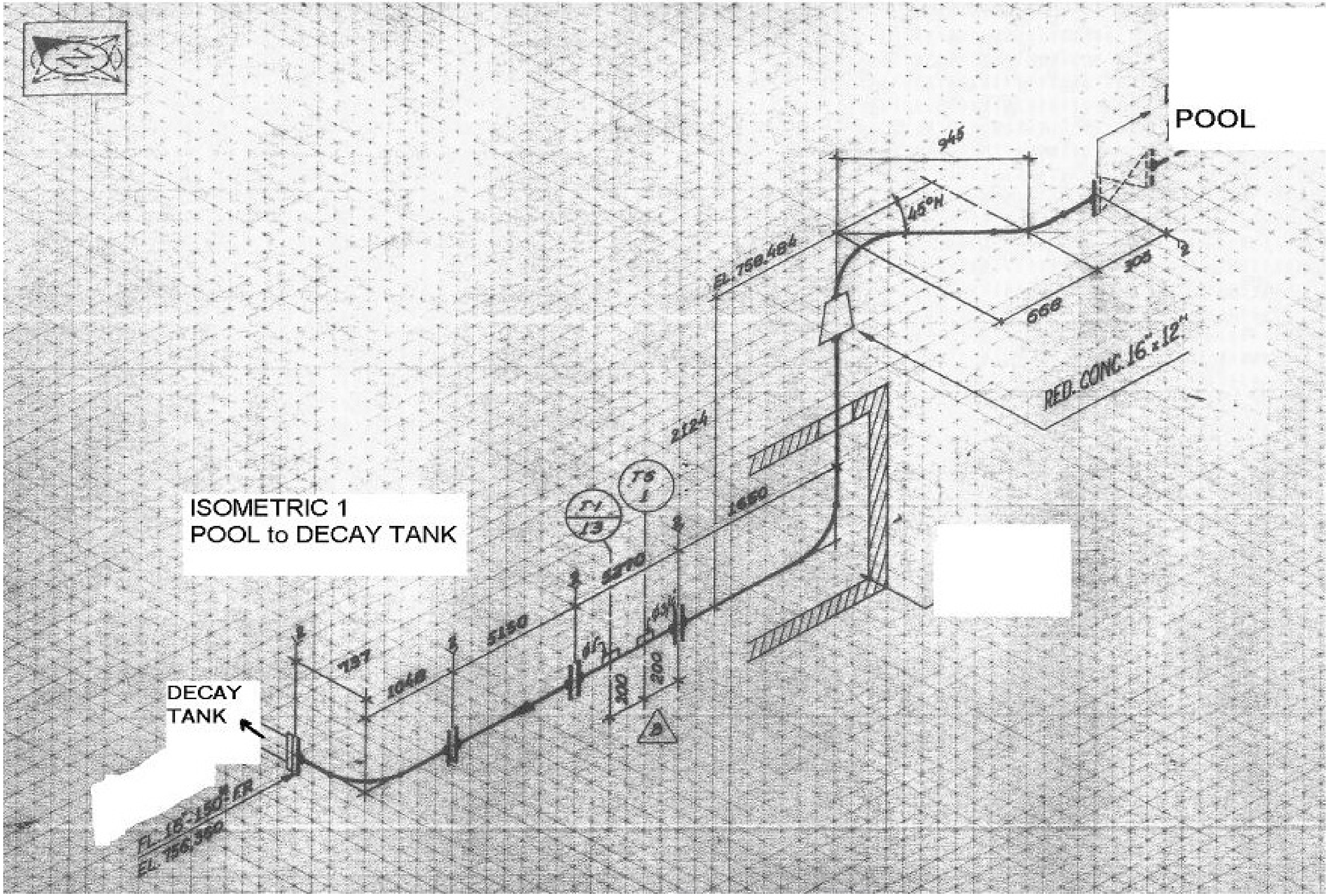


FIG. 5. Description of the pool: decay heat tank line.

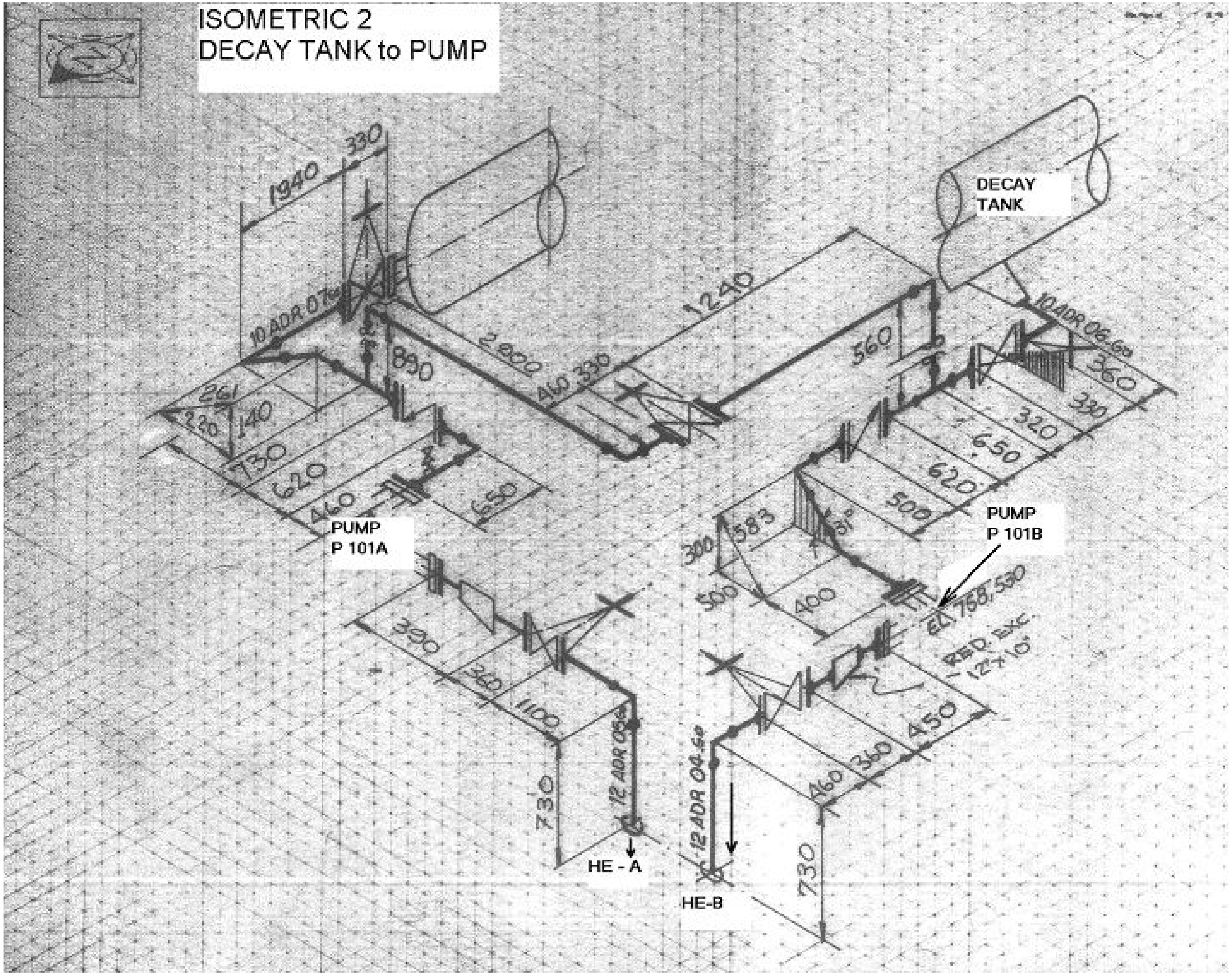


FIG. 6. Description of the decay heat tank: pump line.

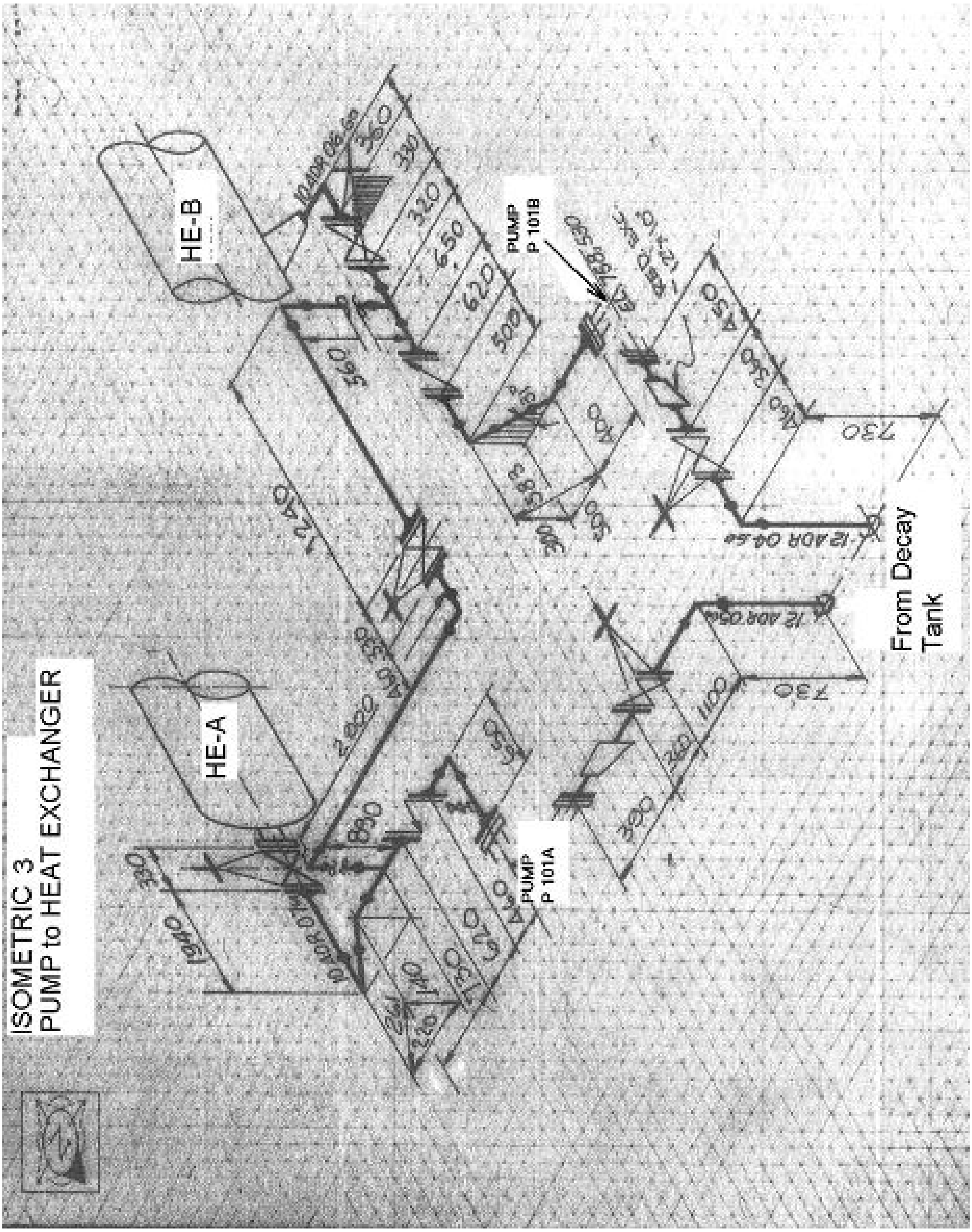


FIG. 7. Description of the decay tank: heat exchanger line.

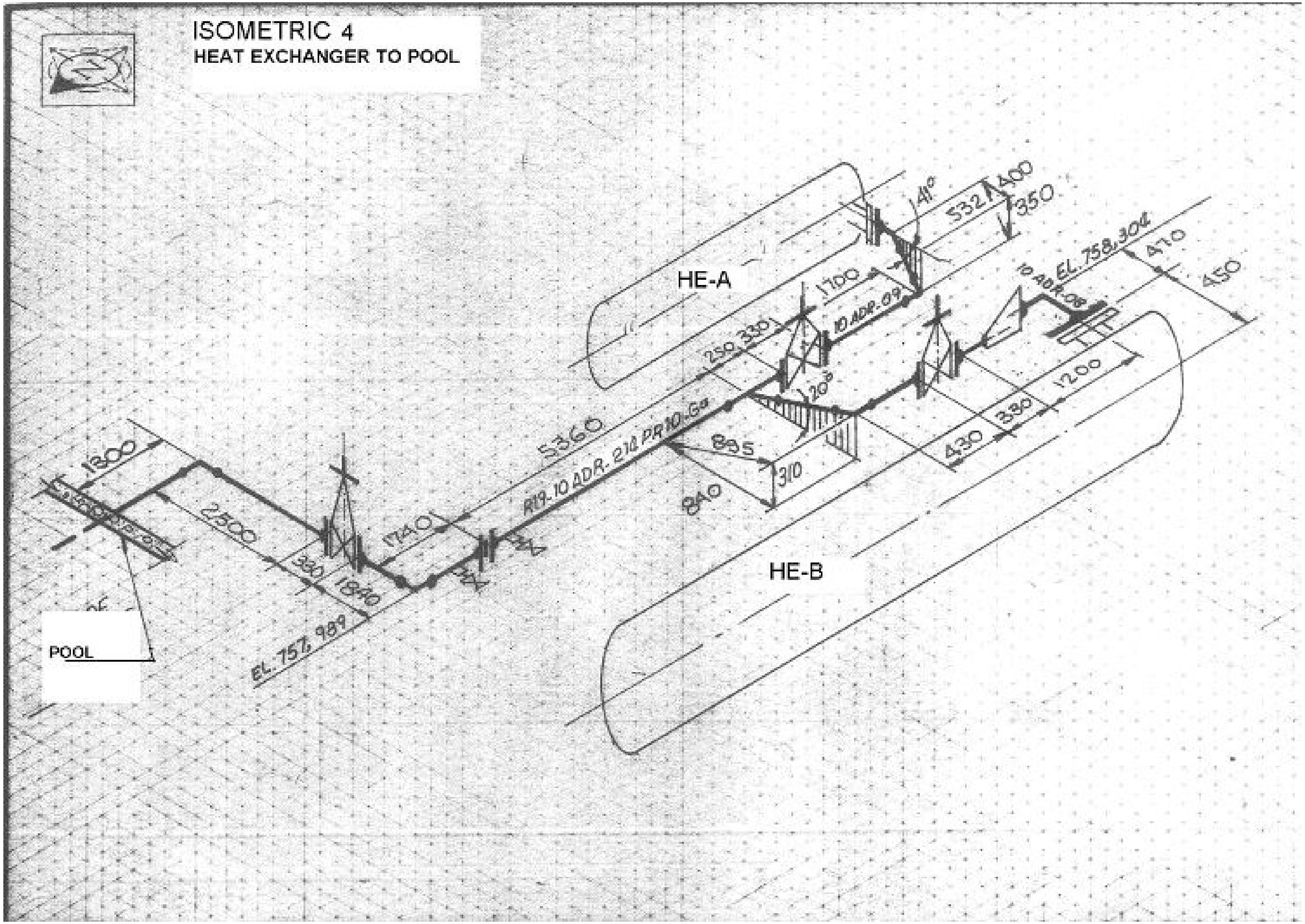


FIG. 8. Description of the heat exchanger: pool line.

Figure 9 shows a typical arrangement of the fuel assembly, beryllium reflector, control rods and irradiation positions inside the reactor core. The core comprises 20 fuel assemblies, 4 control fuel assemblies and a central device irradiation, assembled in a square matrix 5 × 5. Table 2 summarizes the main reactor design parameters.

ΔP	DP	DP	DP	DP	DP	DP	SP
SP	SP	SP	SP	SP	SP	SP	LIN
GR	SP	GR	EIRA	GR	GR	GR	GR
EIS	EIS	GR	EIRA	GR	GI	GR	CF
EIS	EIS	FE* 153	FE 168	FE 156	FE 160	FE 150	GR
GR	EIGRA I	FE 158	CFE 166	FE* 169	CFE 180	FE 171	EIF
GR	GR	FE 164	FE 161	EIBE	FE 162	FE 163	GR
GR	EIGRA II	FE 159	CFE 179	FE* 170	CFE 167	FE 154	GR
S2	GR	FE* 152	FE 155	FE 157	FE 165	FE 151	S3
GR	GR	GR	GR	GR	GR	GR	GR

Legend
ΔP — core pressure drop measurement
DP — double plug
SP — single plug
LIN — power linear channel
GR — graphite reflector
EIRA — sample irradiator
FC — fission chamber
EIS — sample irradiator
EIGRA I and II — sample irradiators
S2 and S3 — neutron detectors
FE — fuel element
FE* — DMPV=01 fuel element position
CFE — control fuel element
EIBE — sample irradiator
GI — sample irradiator
EIF — sample irradiator

FIG. 9. Typical arrangement of the fuel assembly.

TABLE 2. DESCRIPTION OF THE IEA-R1 RESEARCH REACTOR DESIGN PARAMETERS

Reactor parameter	Data	Notes
Steady state power level	5 MW	None
Fuel		
Fuel enrichment	<19.75%	None
No. of fuel assemblies in the core	24	See Fig. 9
Standard fuel assembly	20	See Fig. 9
Control fuel assembly	4	See Fig. 9
Fuel type		See Table 3
	U_3O_8-Al	Density 2.3 g/cm^3 Mass U_3O_8 per fuel assembly 1254.1 g Mass U-235 per fuel assembly 196.9 g
	U_3Si_2-Al	Density 3.0 g/cm^3 Mass U_3Si_2 per fuel assembly 1517.3 g Mass U-235 per fuel assembly 275.5 g
Max. inlet temp.	40°C	None
Difference temp. at 5 MW	5.8°C	Between inlet and outlet
No. of fuel plates in		
Standard fuel assembly	18	None
Control fuel assembly	12	None

TABLE 2. DESCRIPTION OF THE IEA-R1 RESEARCH REACTOR DESIGN PARAMETERS (cont.)

Reactor parameter	Data	Notes
Thickness of the plates		
Fuel	0.76 mm	None
Clad	0.38 mm	None
Total thickness	1.52 mm	None
Total width of the plates	67.1 mm	
Fuel meat dimensions	0.76 × 62.6 × 600	Thickness × width × height (mm)
Thickness of water channel	2.89 mm	None
Coolant flow rate		
Total	772 m^3/h	None
Fuel assembly	22.8 × 24 = 547.2 m^3/h	None
Bypass	224.8 m^3/h	None
Flow rate through the fuel assembly (normal condition)	22.8 m^3/h	Measured (average) in the peripheral channel the flow rate is 10–15% lower than the average (see Ref. [1])
Pressure drop (normal condition)	7.835 kPa	Measured
Pressure drop of primary system	400 kPa	Approximate
Control rods	Ag–In–Cd	None
Uncertainties		
Deviation in fuel loading per plate	12%	None
Fluctuation in U density	2%	None
Error in meat thickness	10%	None
Power measurement	5%	None
Power density variation	10%	None
Flow measurement	3%	None
Neutronic model	10%	None
Operational limits		
Clad temp. (max.)	95°C	None
Minimum departure from nucleate boiling ratio (MDNBR)	2.0	None
Flow instability rate	2.0	None

2. FUEL ASSEMBLY

Each standard fuel assembly has 18 fuel plates assembled on two lateral support plates, forming 17 independent flow channels (see Fig. 10). Figure 11 shows the main dimensions of the fuel assembly.

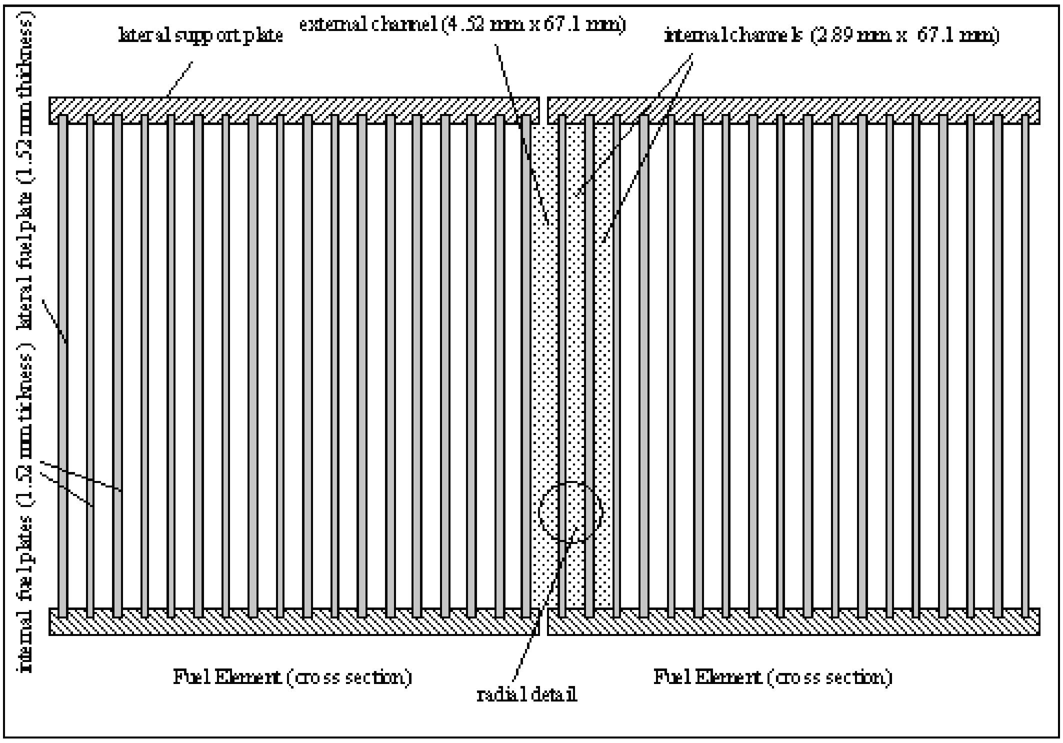
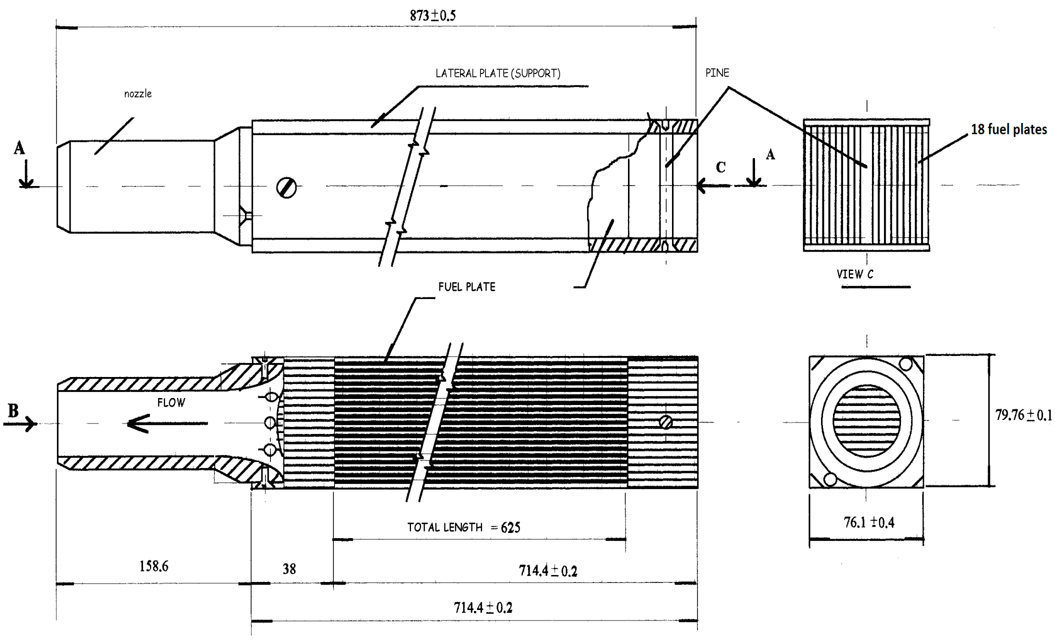


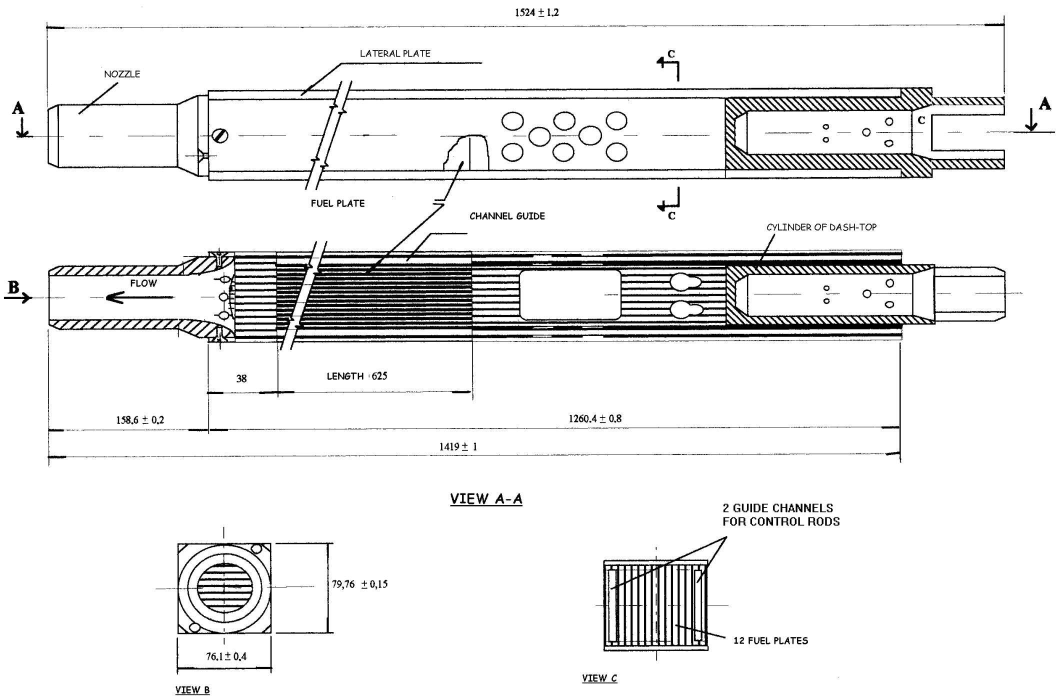
FIG. 10. Cross-section of two fuel assemblies.



Dimensions in mm.

FIG. 11. Fuel assembly (reproduced from Ref. [2]).

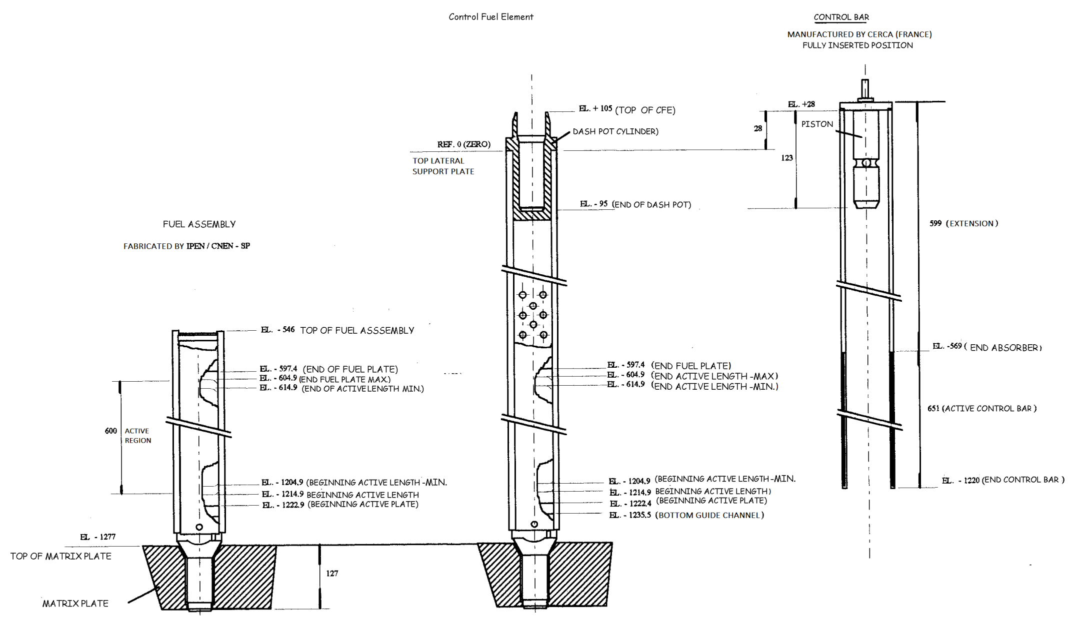
The control fuel assembly (see Fig. 12) includes only 12 plates, as there are two dummy lateral plates and 2 guide channels for the control plates.



Dimensions in mm.

FIG. 12. Control fuel assembly (reproduced from Ref. [2]).

Figure 13 shows the relative position of control fuel assembly in the matrix plate.



Dimensions in mm.

FIG. 13. Relative position of the control fuel assembly (reproduced from Ref. [2]).

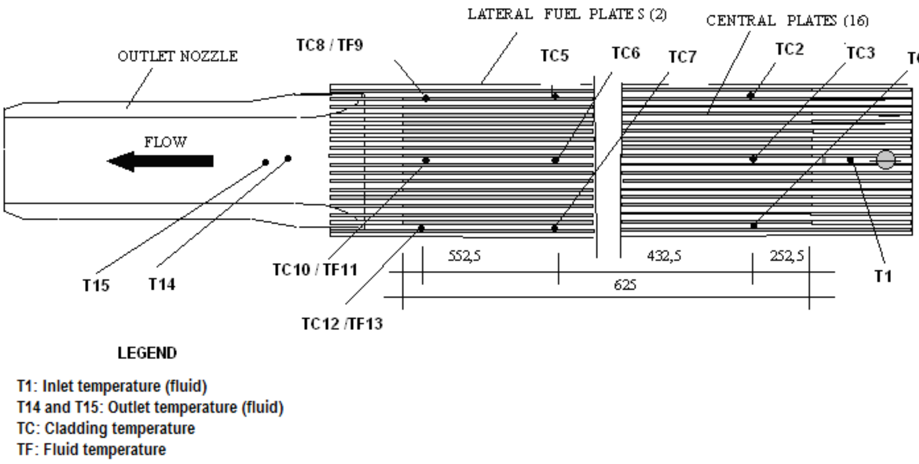
3. INSTRUMENTED FUEL ASSEMBLY

For the thermalhydraulic analysis, the estimation of the flow rate inside the fuel assembly is not enough because the lateral fuel plates need to be also cooled on the external side, which is achieved by circulating water through the fuel assemblies. The determination of the flow rate between two fuel assemblies is not easy because the channel is open, causing the redistribution of the water (cross-flow).

A relevant point is the cladding corrosion rate. After the upgrade from 2 MW to 5 MW, the corrosion rate increased in a lateral plate of one fuel assembly and some doubts appeared concerning the flow values used in the thermalhydraulic analysis. In order to clarify this and to measure the actual flow distribution through the fuel assemblies that compose the IEA-R1 active core, a dummy element (without nuclear fuel material) (DMPV-01 — pressure and flow measure device), full scale, was designed and manufactured in aluminium [1]. However, the uncertainties found for the flow rate between fuel assemblies created the need for an instrumented fuel assembly (IFA) with the following objectives:

(a) Perform more accurate safety analysis for the IEA-R1 reactor;
(b) Determine the actual cooling conditions (mainly in the outermost fuel plate);
(c) Validate computer codes used for thermalhydraulic and safety analysis of research reactors.

Figure 14 shows a sketch of the IFA, and the position of the thermocouples to measure the inlet and outlet temperatures, and along the central and lateral channels.



Dimensions in mm.

FIG. 14. Instrumented fuel assembly.

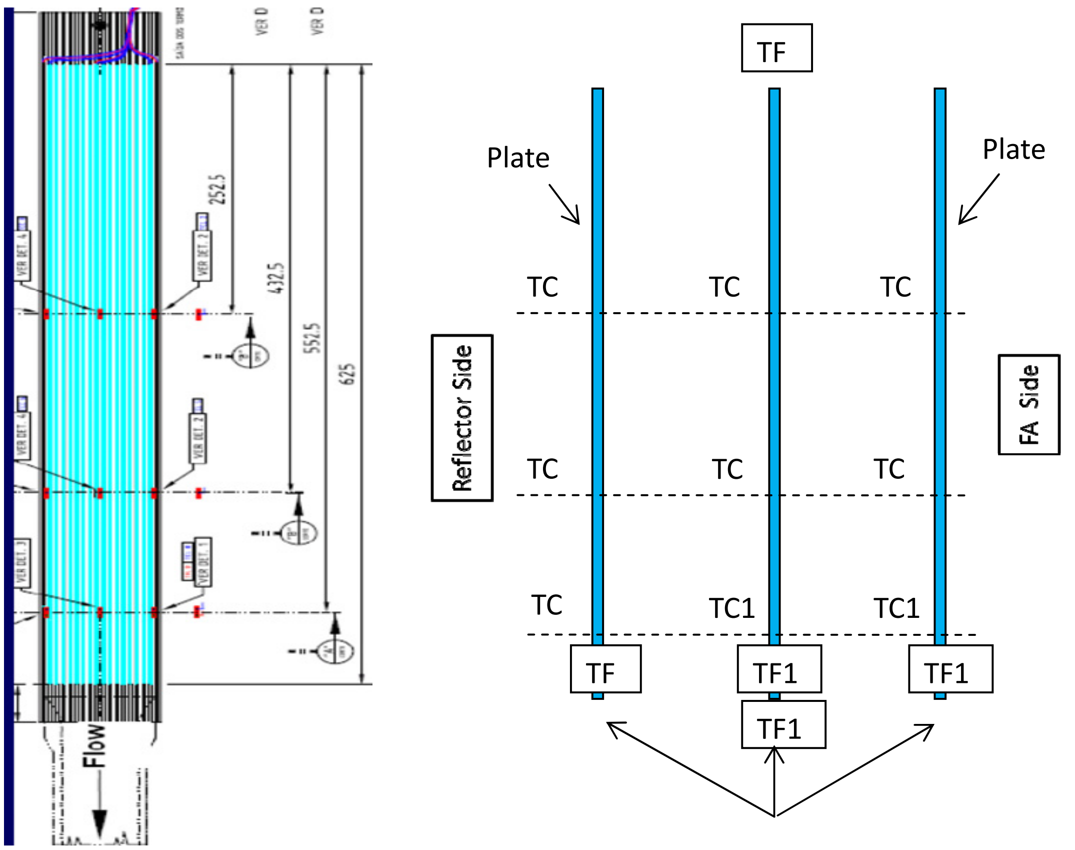
All the thermocouples are type K, 0.5 mm in diameter and calibrated. The error is less than 0.5°C for temperatures lower than 50°C, and less than 0.8°C for temperatures between 50°C and 100°C.

The impact of radial peaking factors on the thermalhydraulic behaviour of the fuel element is measured by locating the IFA at two different locations corresponding to core configurations 243 and 246, which are described in Section 4.

3.1. DESCRIPTION AND LOCATION OF THE THERMOCOUPLES

Figure 15 presents the IFA with the assigned measurements positions, and Table 3 summarizes the assigned temperatures at the considered positions. The IFA has 14 thermocouples distributed as follows:

(a) Coolant inlet and outlet temperatures;
(b) Three channels (one central and two lateral channels), each equipped with three thermocouples for the clad and one for the fluid temperatures.



FA — fuel assembly; TC — cladding temperature; TF — fluid temperature.

FIG. 15. Vertical section of an instrumented fuel element.

TABLE 3. LOCATIONS AND SYMBOLS OF TEMPERATURE MEASUREMENTS IN THE INSTRUMENTED FUEL ASSEMBLY

	Position from channel entrance		
	Reflector side	Central	Fuel assembly side
Clad temperatures (°C)			
252.5	TC2	TC3	TC4
432.5	TC5	TC6	TC7
552.5	TC8	TC10	TC12
Fluid temperatures (°C)			
552.5	TF9	TF11	TF13

Note: TF1 — fluid inlet temperature (not in Table 3); TC2 — clad lateral plate (side reflector) 252.5 mm from the entrance of the channel; TC3 — clad central plate 252.5 mm from the entrance of the channel; TC4 — clad lateral plate (side fuel assembly) 252.5 mm from the entrance of the channel; TC5 — clad lateral plate (side reflector) 432.5 mm from the entrance of the channel; TC6 — clad central plate 432.5 mm from the entrance of the channel; TC7 — clad lateral plate (side fuel assembly) 432.5 mm from the entrance of the channel; TC8 — clad lateral plate (side reflector) 552.5 mm from the entrance of the channel; TF9 — fluid inner channel lateral plate (side reflector) 552.5 mm from the entrance of the channel; TC10 — clad central plate (central) 552.5 mm from the entrance of the channel; TF11 — fluid inner channel lateral plate 552.5 mm from the entrance of the channel; TC12 — clad lateral plate (side fuel assembly) 552.5 mm from the entrance of the channel; TF13 — fluid inner channel lateral plate (side fuel assembly) 552.5 mm from the entrance of the channel; TF14 — fluid outlet temperature (not in Table 3).

4. EXPERIMENT CORE CONFIGURATION 243

Numerical data on this core configuration are provided in the file **BR_exp_conf_243.xls**. Figure 16 shows the core configuration for experiment in configuration 243 of the reactor.

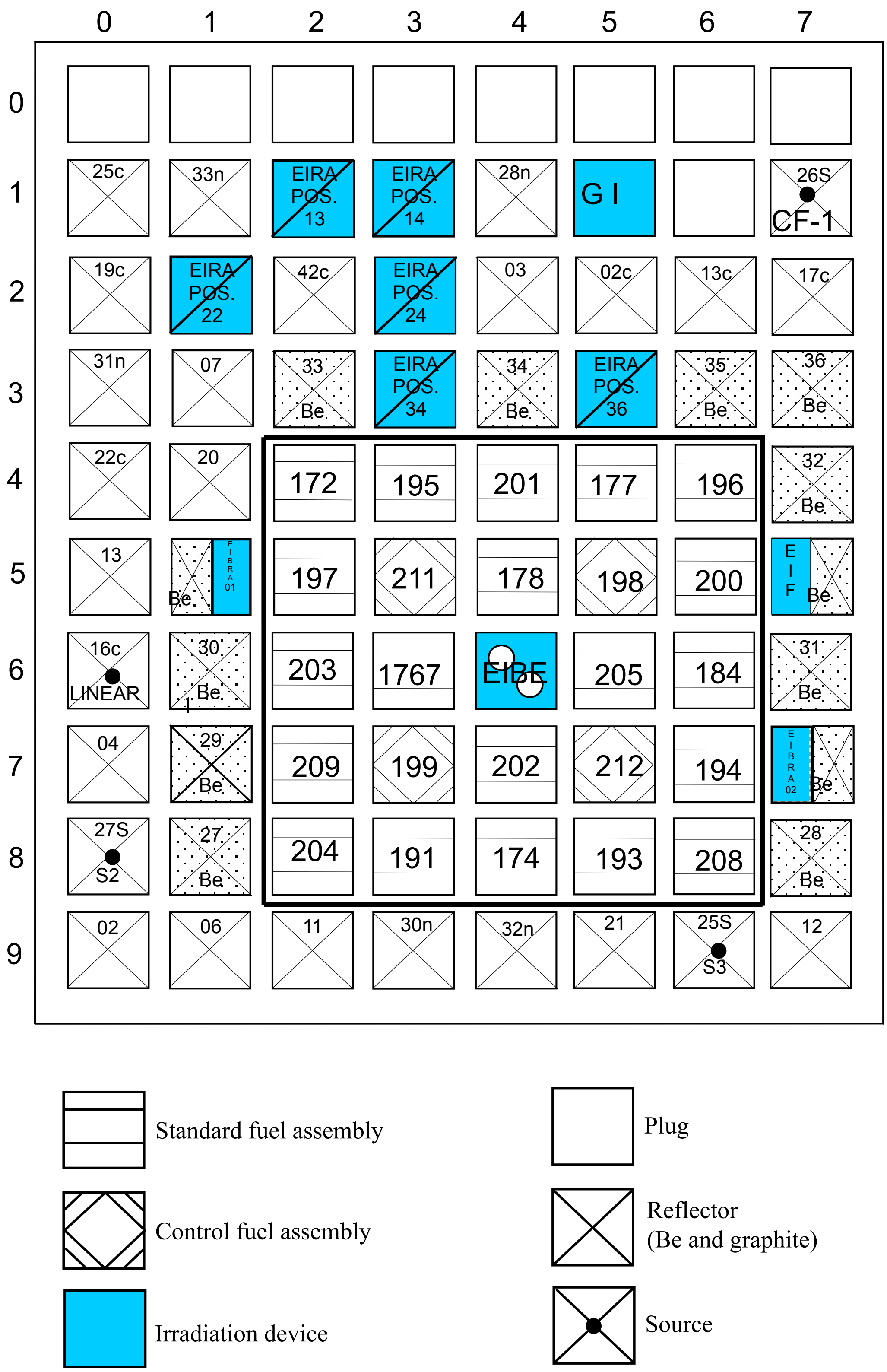


FIG. 16. Fuel element arrangement for core configuration 243.

Table 4 presents the burnup for each fuel assembly for the first experiment (core configuration 243).

TABLE 4. FUEL BURNUP FOR CONFIGURATION 243

Fuel assembly No.	Fuel assembly position in the matrix plate[a]	Fuel assembly % burnup[b]	U-235 % burnup	Type	Density of U-235 (g/cm3)	Observation
172	43	36.562	39.25	U_3Si_2Al	3.0	None
174	85	23.104	25.04	U_3Si_2Al	3.0	None
176	64	22.953	24.88	U_3Si_2Al	3.0	None
177	46	24.050	26.05	U_3Si_2Al	3.0	None
178	55	23.898	25.89	U_3Si_2Al	3.0	None
184	67	16.246	17.70	U_3Si_2Al	3.0	None
191	84	36.581	39.15	U_3O_8Al	2.3	External plate 50% U
193	86	34.246	36.71	U_3O_8Al	2.3	External plate 50% U
194	77	35.840	38.38	U_3O_8Al	2.3	External plate 50% U
195	44	35.982	38.53	U_3O_8Al	2.3	External plate 50% U
196	47	29.345	31.56	U_3O_8Al	2.3	External plate 50% U
197	53	19.392	21.00	U_3O_8Al	2.3	External plate 50% U
198	56	37.843	40.47	U_3O_8Al	2.3	Control
199	74	38.506	41.16	U_3O_8Al	2.3	Control
200	57	19.40	21.04	U_3O_8Al	2.3	External plate 50% U
201	45	2.025	2.23	U_3O_8Al	3.0	
202	75	15.929	17.3	U_3O_8Al	3.0	External plate 50% U
203	63	13.511	14.75	U_3O_8Al	2.3	External plate 50% U
204	83	6.713	7.37	U_3Si_2Al	3.0	External plate 50% U
205	66	5.107	5.62	U_3Si_2Al	3.0	External plate 50% U
208	87	0.000	0.00	U_3Si_2Al	3.0	IFA
209	73	0.000	0.00	U_3Si_2Al	3.0	New
211	54	12.205	13.29	U_3O_8Al	2.3	Control
212	76	11.977	13.04	U_3O_8Al	2.3	Control

Note: IFA – instrumented fuel assembly.

[a] The first number represents the line and the second the column in matrix plate (e.g. 43, line 4 and column 3).

[b] Including all actinides.

4.1. NEUTRONIC DATA

The neutronic analysis methodology is based on LEOPARD [3] and HAMMER-TECHNION [4, 5] codes for cross-section generation, 2DB [6] code for the core and burnup calculations in a 2-D geometry and CITATION code [7] for a 3-D analysis. The fuel cross-section is performed with LEOPARD (version modified by University of Michigan, where a plate geometry option was included) using a standard cell model (fuel, cladding and moderator) with an extra region to take into account other regions of the fuel assembly. The HAMMER-TECHNION is used to generate the cross-sections for the non-fuel regions, such as the reflector and control rods, among other things. The reactor power history is simulated with 2DB in a 2-D model. Three-dimensional calculations are finally performed with CITATION for effective multiplication factor, neutron flux and power density distributions, integral and differential control rod worth, reactivity coefficients and kinetic parameters. The ORIGEN2 [8] code was used to calculate the power decay presented in Tables 5 and 15.

4.1.1. Axial power distribution

For the normalized axial power density (see Table 5):

(1) The IFA was divided into three regions of six plates:
  (i) Fuel assembly side (average 1.0);
  (ii) Central plates (average 0.89);
  (iii) Reflector side (average 0.96).

(2) In the core, there are 408 fuel plates: 20 standard fuel assembly (with 18 fuel plates each) and four control fuel assembly (with 12 fuel plates each).

(3) For a power of 3.5 MW, the average power per fuel plate is 8.663 kW, or 155.94 kW per standard fuel assembly, and the average heat flux is 11.53 W/cm^2.

(4) The total flow rate in the IFA is 22.8 m^3/h (in 17 channels and remove the heat of 17 fuel plates, 16 internal plates and 2 half plates, one on each lateral). Hence, the calculated heat removed of the IFA (for a reactor power of 3.5 MW) is: 5.5 (plates) × 1.0 × 9.578 + 6 (plates) × 0.89 × 8.578 + 5.5 (plates) × 8.578 × 0.96 = 138.38 kW.

TABLE 5. NORMALIZED AXIAL POWER DISTRIBUTION

Active length (mm)	Av. of six lateral plates (fuel assembly side)	Av. of six central plates	Av. of six lateral plates (reflector side)
20	0.293	0.262	0.281
40	0.304	0.272	0.292
60	0.343	0.307	0.329
80	0.391	0.350	0.375
100	0.445	0.398	0.427
120	0.502	0.449	0.481
140	0.563	0.504	0.540
160	0.629	0.563	0.603
180	0.699	0.625	0.670
200	0.777	0.695	0.745
220	0.871	0.779	0.835
240	1.002	0.896	0.961
260	1.141	1.020	1.094
280	1.248	1.116	1.197
300	1.33	1.189	1.275

TABLE 5. NORMALIZED AXIAL POWER DISTRIBUTION (cont.)

Active length (mm)	Av. of six lateral plates (fuel assembly side)	Av. of six central plates	Av. of six lateral plates (reflector side)
320	1.395	1.248	1.338
340	1.443	1.291	1.384
360	1.476	1.320	1.415
380	1.493	1.335	1.432
400	1.495	1.337	1.434
420	1.48	1.324	1.419
440	1.45	1.297	1.391
460	1.405	1.257	1.347
480	1.344	1.202	1.289
500	1.271	1.137	1.219
520	1.185	1.060	1.136
540	1.091	0.976	1.046
560	0.998	0.893	0.957
580	0.928	0.830	0.890
600	0.945	0.845	0.906
Av.	1.0	0.89	0.96

Table 6 shows the calculated heat removed of the IFA by the internal flow rate for other powers operations.

TABLE 6. CALCULATED HEAT REMOVED BY INTERNAL FLOW RATE

Reactor power (MW)	1.0	2.0	3.0	3.5	4.0	4.5	5.0
Calculated heat removed from IFA (kW)	38.78	77.57	116.35	135.74	155.13	174.5	193.90

Tables 7–11 contain further data on core configuration 243.

TABLE 7. TEMPERATURE FEEDBACK COEFFICIENTS FOR FUEL

T_F (°C)	α_F (pcm/°C)
20–50	–2.120
50–100	–2.025
100–150	–1.833

[a] n.a.: not applicable.

TABLE 8. EFFECTIVE DELAYED NEUTRON FRACTIONS (β_i AND β_{eff}), CONSTANT OF DELAYED NEUTRON (λ_i) AND PROMPT NEUTRON GENERATION TIME (Λ)

Precursor group No.	β_i	λ_i (s^{-1})
1	$2.875\ 77 \times 10^{-4}$	0.012 72
2	$1.565\ 07 \times 10^{-3}$	0.031 74

TABLE 8. EFFECTIVE DELAYED NEUTRON FRACTIONS (β_i AND β_{eff}), CONSTANT OF DELAYED NEUTRON (λ_i) AND PROMPT NEUTRON GENERATION TIME (Λ) (cont.)

3	$1.429\ 32 \times 10^{-3}$	0.116
4	$3.083\ 96 \times 10^{-3}$	0.311
5	$9.787\ 83 \times 10^{-4}$	1.40
6	$2.027\ 42 \times 10^{-4}$	3.87
β_{eff}	$7.547\ 44 \times 10^{-3}$	n.a.[a]
Λ (μs)	77.063 7	n.a.[a]

[a] n.a.: not applicable.

TABLE 9. TEMPERATURE FEEDBACK COEFFICIENTS FOR THE MODERATOR

T_M (°C)	α_M (pcm/°C)	α_{DM} (pcm/°C)
20–40	–6.143	–11.307
40–60	–9.403	–13.608
60–80	–12.040	–12.210

TABLE 10. VOID COEFFICIENTS

ρ_M (g/cm^3)	% void	α_V (pcm/% void)
0.998 30–0.992 26	0.605	–202.140
0.992 26–0.983 19	0.914	–203.782
0.983 19–0.971 63	1.176	–202.268

TABLE 11. POWER VERSUS TIME

Time (s)	Power (MW)
0.0	3.500 0
4.0	3.500 0
5.0	0.269 5
5.5	0.262 5
6.0	0.255 5
8.0	0.238 0
10.0	0.224 0
12.0	0.213 5
14.0	0.206 5
19.0	0.196 0
21.0	0.193 3
22.0	0.190 9
23.0	0.189 0
24.0	0.185 5
28.0	0.181 4

TABLE 11. POWER VERSUS TIME (cont.)

Time (s)	Power (MW)
32.0	0.175 0
36.0	0.171 8
40.0	0.168 6
44.0	0.165 1
48.0	0.157 0
50.0	0.154 0
64.0	0.150 5
84.0	0.143 5
100.0	0.136 3
150.0	0.125 4
200.0	0.118 3
250.0	0.113 1
300.0	0.108 7

4.2. EXPERIMENTAL RESULTS

The following steps took place:

(1) Reactor was off when the experiment started.
(2) Pump was started with the nominal flow rate.
(3) Reactor was started up.
(4) Reactor power was stabilized at 3.5 MW, 4.0 MW, 4.5 MW and 5.0 MW for some minutes.
(5) Reactor power was reduced in steps and operated for a few minutes at the following powers: 3.5 MW, 3.0 MW, 2.0 MW and 1.0 MW.
(6) Reactor power was again increased to 3.5 MW, and after stabilization at this power, the pumps were turned off.

Figure 17 shows all the steps of this experimental operation and the experimental results are provided in the file **BR_exp_conf_243.xls**.

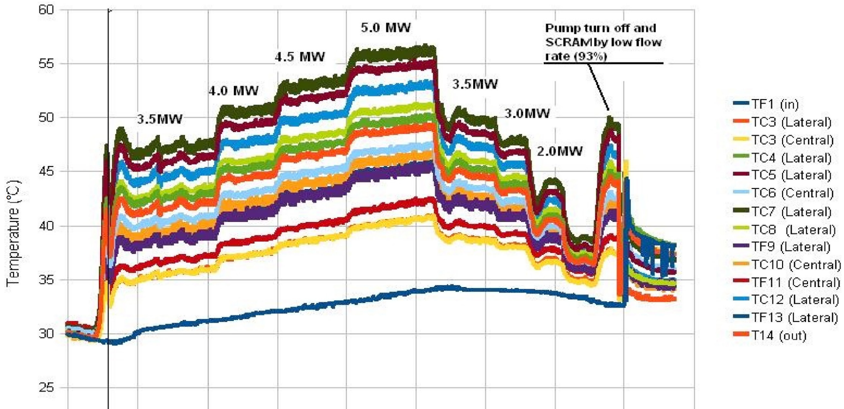


FIG. 17. Experimental result for core configuration 243.

In Table 12, the TC12 temperature measurements are inconsistent because the temperature measured by this thermocouple (TC12) for 4.5 MW (43.92°C for the range 3200–3800 s) is less than the value measured for 4.0 MW (47.64°C for the range 2300–2800 s) and 3.5 MW (44.76°C for the range 1400–1900 s). In addition, for 5.0 MW (for the range 4300–5000 s), TC12 (46.08°C) is less than the value measured for 4.0 MW (for the range 2300–2800 s), at 47.64°C.

TABLE 12. AVERAGED TEMPERATURE MEASUREMENTS OVER THE INDICATED TIME RANGE

Time range (s) and reactor power	1400–1900 (3.5 MW)	2300–2800 (4.0 MW)	3200–3800 (4.5 MW)	4300–5000 (5.0 MW)	5900–6200 (3.5 MW)	6400–6500 (3.0 MW)	6950–7100 (2.0 MW)	7430–7600 (1.0 MW)	7850–7920 (3.5 MW)
Thermocouple	Temp. (°C)								
TF1	30.95	31.67	32.53	33.59	34.09	34.03	33.71	33.19	32.69
TC2	41.95	44.41	46.47	48.72	44.47	43.23	40.42	36.76	43.84
TC3	35.77	37.28	38.68	40.30	38.64	38.04	36.63	34.70	37.52
TC4	42.76	45.27	47.29	49.57	45.13	43.80	40.90	37.05	44.49
TC5	46.14	49.25	51.78	54.56	48.45	46.77	42.96	38.07	48.01
TC6	40.74	42.95	44.87	47.01	43.41	42.32	39.88	36.69	42.55
TC7	47.31	50.51	53.02	55.85	49.61	47.79	43.90	38.83	49.16
TC8	43.34	45.99	48.22	50.69	45.86	44.48	41.40	37.40	45.19
TC10	39.85	42.02	42.81	44.90	41.88	40.94	38.84	36.07	40.90
TC12	44.76	47.64	43.92	46.08	42.65	41.61	39.25	36.17	41.70
TC2-TF1	11.00	12.74	13.94	15.13	10.39	9.19	6.70	3.57	11.15
TC3-TF1	4.83	5.61	6.16	6.71	4.56	4.00	2.92	1.51	4.83
TC4-TF1	11.81	13.61	14.77	15.98	11.04	9.77	7.19	3.86	11.81
TC5-TF1	15.20	17.59	19.26	20.96	14.36	12.74	9.25	4.89	15.32
TC6-TF1	9.79	11.28	12.35	13.42	9.32	8.28	6.17	3.50	9.86
TC7-TF1	16.37	18.84	20.49	22.26	15.52	13.76	10.19	5.65	16.47
TC8-TF1	12.40	14.32	15.70	17.10	11.77	10.45	7.68	4.21	12.51
TC10-TF1	8.90	10.36	10.29	11.31	7.79	6.91	5.12	2.88	8.22
TC12-TF1	13.81	15.97	11.39	12.49	8.56	7.57	5.54	2.99	9.01
TF14-TF1	4.84	5.61	6.17	6.75	4.63	4.14	3.07	1.74	4.88

Note: TF1 — fluid inlet temperature; TC2 — clad lateral plate (side reflector) 252.5 mm from the entrance of the channel; TC3 — clad central plate 252.5 mm from the entrance of the channel; TC4 — clad lateral plate (side fuel assembly) 252.5 mm from the entrance of the channel; TC5 — clad lateral plate (side reflector) 432.5 mm from the entrance of the channel; TC6 — clad central plate 432.5 mm from the entrance of the channel; TC7 — clad lateral plate (side fuel assembly) 432.5 mm from the entrance of the channel; TC8 — clad lateral plate (side reflector) 552.5 mm from the entrance of the channel; TC10 — clad central plate (central) 552.5 mm from the entrance of the channel; TC12 — clad lateral plate (side fuel assembly) 552.5 mm from the entrance of the channel; TF14 — fluid outlet temperature.

Table 13 presents the measured and calculated powers of the IFA for all the powers of operation of the reactor during the experiments for configuration 243.

TABLE 13. COMPARISON BETWEEN CALCULATED AND MEASURED POWER

Reactor power (MW)	3.50	4.00	4.50	5.00	3.50	3.00	2.00	1.00	3.50
Specific heat (cp) (kJ/kg °C)	4.17	4.17	4.17	4.17	4.17	4.17	4.17	4.17	4.17
Measured power of IFA = mass flow[a] × cp × (TF14-TF1) (kW)	127.35	147.61	162.39	177.73	121.96	108.94	80.75	45.92	128.60
Calculated power of IFA (KW)[b]	135.74	155.13	174.5	193.90	135.74	116.3	77.57	38.78	135.74
Relative power deviation[c]	6.2	4.85	6.93	8.34	10.15	6.33	−4.1	−18.4	5.26

Note: IFA — instrumented fuel assembly.
[a] Mass flow = (22.8 m^3/h)/3600 s × 997 kg/m^3 = 6.314 kg/s.
[b] See Table 6.
[c] RDP = ((calculated power of IFA) – (measured power of IFA))/(calculated power of IFA) × 100%.

Selected from Fig. 17, Fig. 18 shows the inlet and outlet temperature during the loss of flow accident. The sequence of events for this accident is:

(1) Pump was turned off (7925 s).
(2) Reactor turned off due to low flow (7928 s).
(3) The coupling valve opened at 7971 s, and flow inversion was observed at 8010 s.

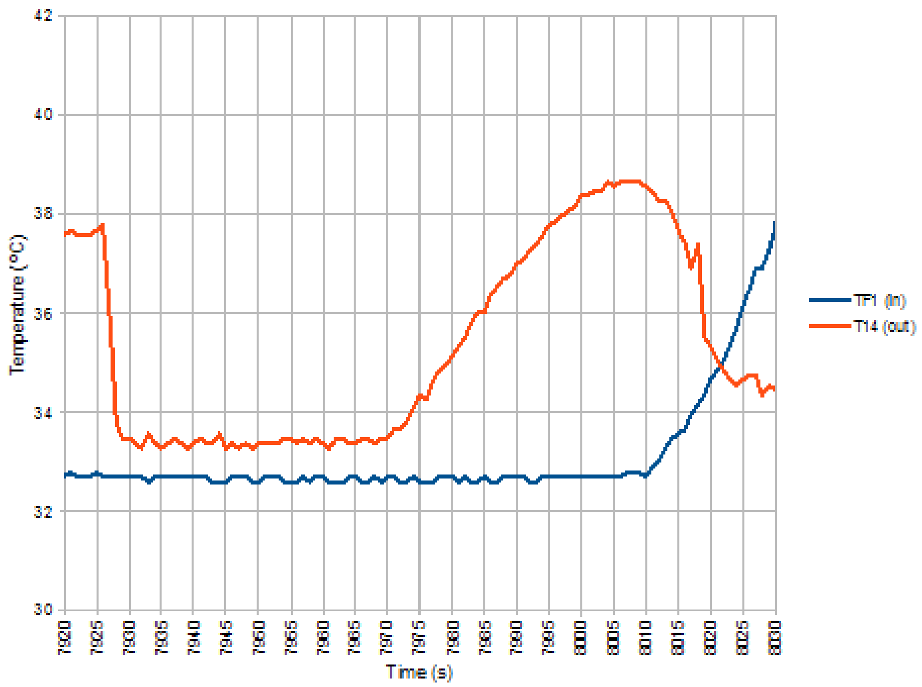


FIG. 18. Flow inversion after loss of flow accident.

4.2.1. Experimental coast down curve

Figure 19 shows the time variation of the flow rate obtained experimentally. Due to the inertial disk coupled to the rotor axis, the flow rate varies from its nominal value, 772 m^3/h, to 222 m^3/h in 46 s. After that, the header is decoupled and the fuel assembly is no longer connected to the primary pump and tubes. At this stage, the only way to refrigerate the fuel plates is through natural circulation.

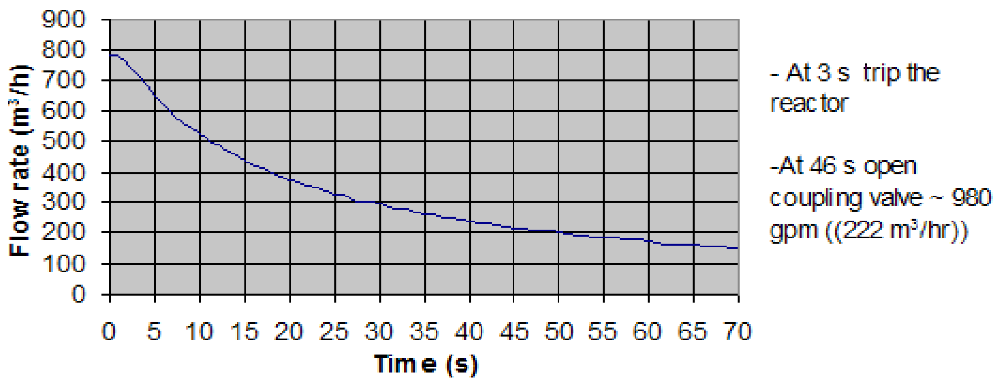


FIG. 19. Pump coast down.

5. EXPERIMENT CORE CONFIGURATION 246

Figure 20 shows the core configuration for experiment in configuration 246 of the reactor.

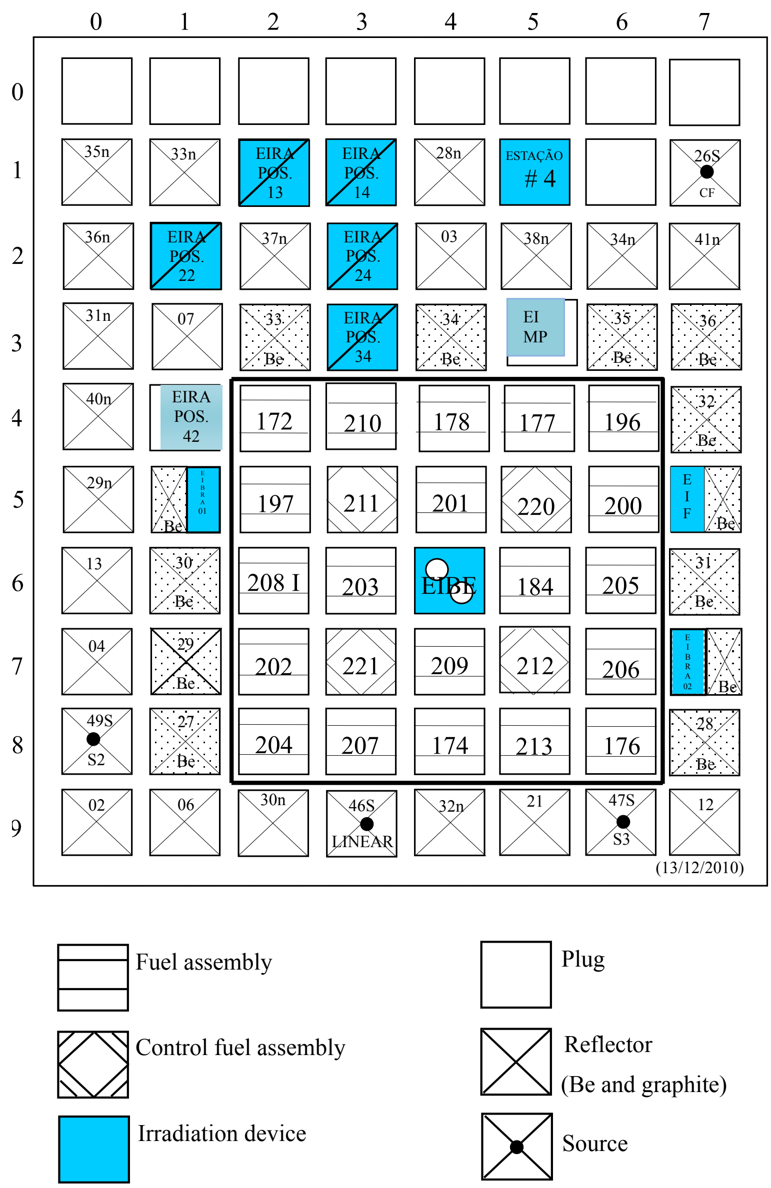


FIG. 20. Fuel element arrangement for core configuration 246.

5.1. NEUTRONIC DATA

Table 14 presents the burnup for each fuel assembly for the first experiment (configuration 246).

TABLE 14. FUEL BURNUP FOR CONFIGURATION 246

Fuel assembly No.	Fuel assembly position in the matrix plate[a]	Fuel assembly % burnup[b]	U-235% burnup	Type	U-235 density	Observation
172	43	42.549	45.49	U_3Si_2Al	3.0	None
174	85	31.917	34.37	U_3Si_2Al	3.0	None
176	87	30.695	33.09	U_3Si_2Al	3.0	None
177	46	32.771	35.27	U_3Si_2Al	3.0	None
178	45	34.321	36.90	U_3Si_2Al	3.0	None
184	66	28.318	30.58	U_3Si_2Al	3.0	None
196	47	37.448	40.06	U_3O_8Al	2.3	External plate 50% U
197	53	30.161	32.42	U_3O_8Al	2.3	External plate 50% U
200	57	30.449	32.72	U_3O_8Al	2.3	External plate 50% U
201	55	13.633	14.88	U_3Si_2Al	3.0	None
202	73	28.635	30.91	U_3Si_2Al	3.0	External plate 50% U
203	64	26.950	29.08	$U3O_8Al$	2.3	External plate 50% U
204	83	15.443	16.84	U_3Si_2Al	3.0	External plate 50% U
205	67	17.377	18.92	U_3Si_2Al	3.0	External plate 50% U
206	77	4.183	4.60	U_3Si_2Al	3.0	None
207	84	4.044	4.45	U_3Si_2Al	3.0	None
208	63	12.045	13.17	U_3Si_2Al	3.0	IFA
209	75	11.579	12.66	U_3Si_2Al	3.0	None
210	44	0.000	0.00	U_3Si_2Al	3.0	None
211	54	26.590	28.65	U_3O_8Al	2.3	Control fuel assembly
213	86	0.000	0.00	U_3Si_2Al	3.0	None
212	76	26.458	28.51	U_3O_8Al	2.3	Control fuel assembly
220	56	9.365	10.26	U_3O_8Al	2.3	Control fuel assembly
221	74	10.108	11.07	U_3O_8Al	2.3	Control fuel assembly

Note: IFA — instrumented fuel assembly.

[a] The first number represents the line and the second the column in matrix plate (e.g. 43, line 4 and column 3).

[b] Including all actinides.

5.1.1. Axial power distribution

For the normalized axial power density (see Table 15):

(1) The IFA was divided in three regions of six plates:
  (i) Fuel assembly 202 (average 1.11);
  (ii) Central plates (average 1.09);
  (iii) Fuel assembly 197 (average 1.11).

(2) In the core, there are 408 fuel plates: 20 standard fuel assembly (with 18 fuel plates each) and 4 control fuel assembly (with 12 fuel plates each).
(3) For a power of 4.0 MW, the average power per fuel pate is 9.901 kW, or 178.22 kW per standard fuel assembly, and the average heat flux is 13.18 W/cm^2.
(4) The total flow rate in the IFA is 22.8 m^3/h (in 17 channels that remove the heat of 17 fuel plates, 16 internal plates and 2 half plates, one on each lateral). Hence, the calculated heat removed of the IFA is: 5.5 (plates) × 1.11 × 9.901 + 6 (plates) × 1.09 × 9.901 + 5.5 (plates) × 9.901 × 1.11 = 185.64 kW.

TABLE 15. NORMALIZED AXIAL POWER DISTRIBUTION

Active length (mm)	Six lateral plates (side of fuel assembly 202)	Six central plates	Six lateral plates (side fuel assembly 197)
20	0.325	0.324	0.336
40	0.334	0.329	0.341
60	0.378	0.371	0.384
80	0.436	0.427	0.442
100	0.501	0.491	0.507
120	0.572	0.560	0.579
140	0.649	0.636	0.656
160	0.733	0.717	0.740
180	0.824	0.806	0.831
200	0.923	0.902	0.928
220	1.031	1.005	1.032
240	1.146	1.115	1.139
260	1.255	1.220	1.245
280	1.355	1.317	1.343
300	1.442	1.401	1.431
320	1.515	1.473	1.505
340	1.572	1.530	1.564
360	1.613	1.570	1.606
380	1.637	1.595	1.632
400	1.644	1.602	1.640
420	1.632	1.592	1.630
440	1.603	1.564	1.602
460	1.557	1.519	1.557
480	1.494	1.458	1.495
500	1.415	1.381	1.417
520	1.322	1.291	1.324
540	1.219	1.191	1.222
560	1.117	1.092	1.121
580	1.044	1.023	1.049
600	1.085	1.074	1.100
Av.	1.112	1.086	1.113

Table 16 shows the temperature feedback coefficients for the fuel.

TABLE 16. TEMPERATURE FEEDBACK COEFFICIENTS FOR FUEL

T_F (°C)	α_F (pcm/°C)
20–50	–2.015
50–100	–1.910
100–200	–1.724

Tables 17–20 contain further data on core configuration 246.

TABLE 17. EFFECTIVE DELAYED NEUTRON FRACTIONS (β_i AND β_{eff}), CONSTANT OF DELAYED NEUTRON (λ_i) AND PROMPT NEUTRON GENERATION TIME (Λ)

Precursor group number	β_i	λ_i (s^{-1})
1	$2.880\,86 \times 10^{-4}$	0.012 72
2	$1.567\,11 \times 10^{-3}$	0.031 74
3	$1.431\,16 \times 10^{-3}$	0.116
4	$3.088\,74 \times 10^{-3}$	0.311
5	$9.798\,18 \times 10^{-4}$	1.40
6	$2.027\,39 \times 10^{-4}$	3.87
β_{eff}	$7.557\,66 \times 10^{-3}$	
Λ (µs)	72.550	

TABLE 18. TEMPERATURE FEEDBACK COEFFICIENTS FOR MODERATOR

T_M (°C)	α_M (pcm/°C)	α_{DM} (pcm/°C)
20–40	–6.053	–9.608
40–60	–9.286	–11.870
60–80	–11.900	–11.822

TABLE 19. VOID COEFFICIENTS

ρ_M (g/cm^3)	% void	α_V (pcm/% void)
0.998 30–0.99 226	0.605	–199.823
0.992 26–0.983 19	0.914	–209.080
0.983 19–0.971 63	1.176	–200.004

TABLE 20. POWER VERSUS TIME

Time (s)	Power (MW)
0.0	4.000 0
4.0	4.000 0
5.0	0.308 0
5.5	0.300 0
6.0	0.292 0
8.0	0.272 0
10.0	0.256 0
12.0	0.244 0
14.0	0.236 0
19.0	0.224 0
21.0	0.220 0
22.0	0.217 0
23.0	0.216 0
24.0	0.211 2
28.0	0.206 5
32.0	0.199 2
36.0	0.195 6
40.0	0.192 0
44.0	0.188 0
48.0	0.180 0
50.0	0.176 0
64.0	0.172 0
84.0	0.164 0
100.0	0.155 8
150.0	0.143 3
200.0	0.135 2
250.0	0.129 3
300.0	0.124 3

5.2. EXPERIMENTAL RESULTS

Two series of experiments was performed for configuration 246:

(1) With a box around the core (see Fig. 21), with the objective to eliminate the cross-flow between fuel assemblies;
(2) Without a box around the core (see Fig. 22).

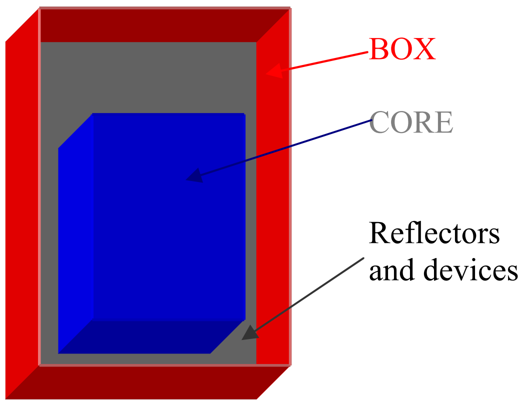


FIG. 21. Box around the core.

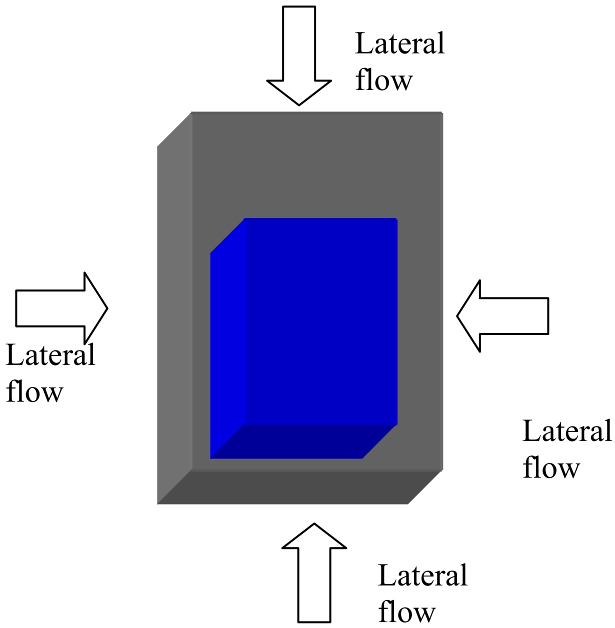


FIG. 22. Without a box around the core.

The box around the core consists of four plates with a height of the fuel assembly installed in the four lateral faces to block the entry of water from the sides. With this box, the flow of water is only downstream in the region of the fuel assemblies and reflectors.

In the first experiment performed — with a box around the core — the sequence was the following:

(a) Reactor was off when the experiment started.
(b) Pump was started with the nominal flow rate (772 m^3/h) and water pool temperature (18.3°C).
(c) Reactor was started up.
(d) Reactor power was stabilized at 4.0 MW.
(e) After some minutes, the pumps were turned off.

Figure 23 shows the temperature across time at various thermocouple positions with core configuration 246 during the loss of flow experiment with the box.

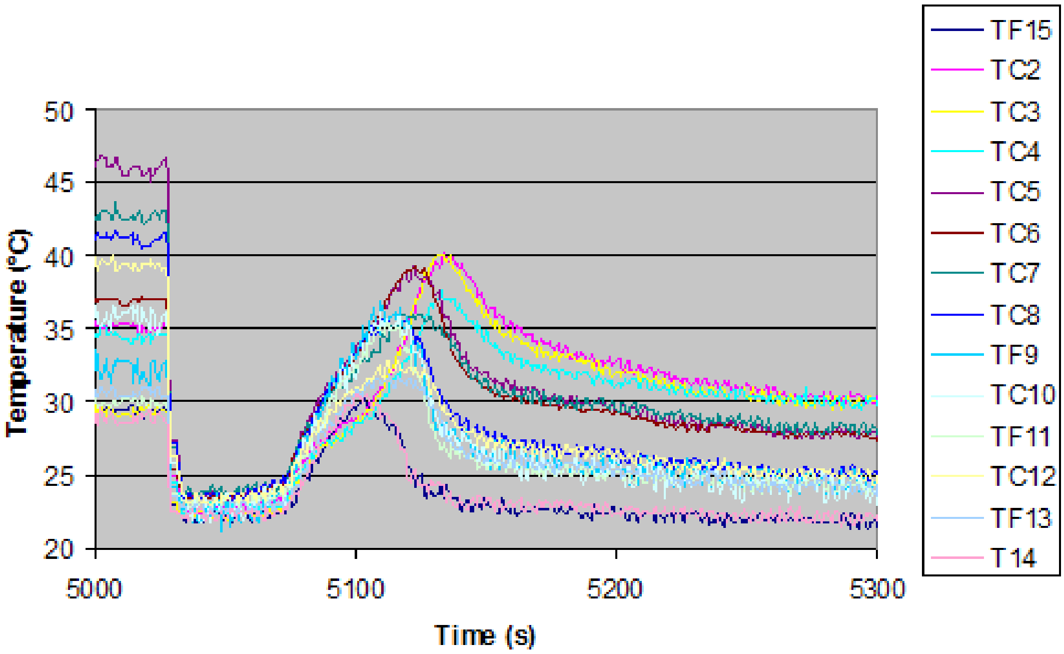


FIG. 23. Loss of flow experiment with a box around the core.

After this experiment, the box was removed and the same sequence was:

(a) Reactor was off when the experiment started.
(b) Pump was started with the nominal flow rate (772 m^3/h) and water pool temperature (20.9°C).
(c) Reactor was started up.
(d) Reactor power was stabilized at 4.0 MW.
(e) After some minutes, the pumps were turned off.

Figure 24 shows the temperature across time at various thermocouple positions with core configuration 246 during the loss of flow experiment without the box.

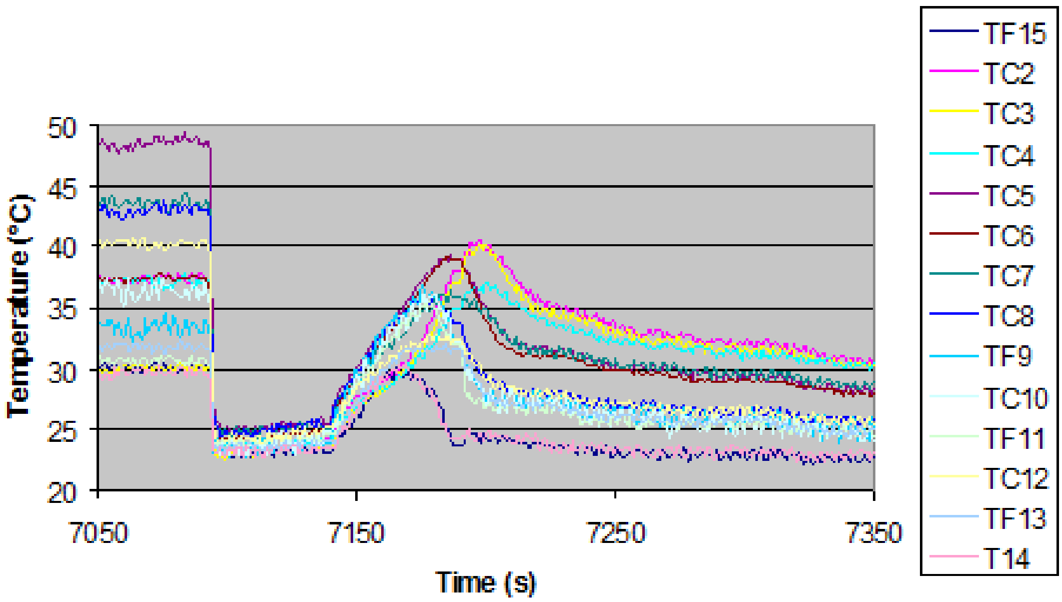


FIG. 24. Loss of flow experiment without a box around the core.

Figure 25 shows all the steps of this operation, and the experimental data and results are provided in **BR_exp_conf_246.xls**, where the curve of the pump coast down is also available.

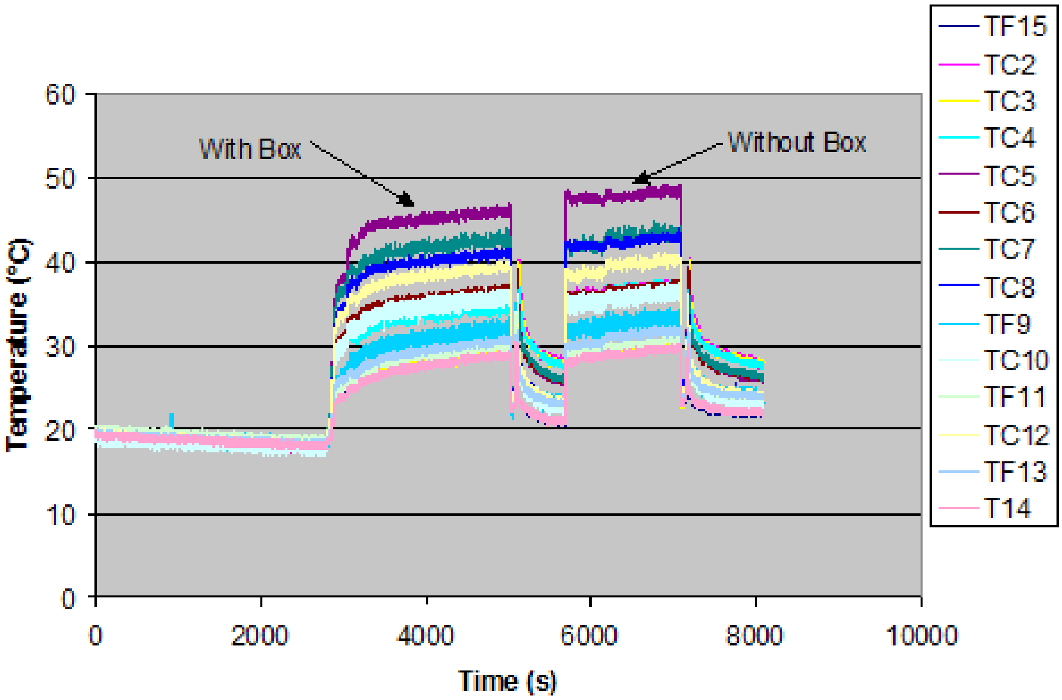


FIG. 25. Experimental results in configuration 246.

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