COMPARISON OF THERMOHYDRAULIC AND NUCLEAR ASPECTS IN A STANDARD HEU CORE AND A TYPICAL LEU CORE FOR THE HFR PETTEN A CASE STUDY

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ABSTRACT

Within the framework of the RERTR [1] program various HEU-LEU[2] core calculations have been performed by ANL[3] in a cooperative effort with ECN[4] and JRC[5] Petten. The main purpose of this work has been to gain competence in analysing HEU-LEU core conversion for high power Materials Testing Reactors and to assist in a possible HEU-LEU conversion of the HFR Petten. For reference purposes the present HFR standard core (HEU) in the "old" vessel geometry was calculated at first. As a next step the new vessel geometry and the increased fuel weights were taken into account.

Subsequently various LEU HFR core options have been analysed. Main parameters in the LEU study were the uranium loading in the meat, the fuel type, the thickness of the meat, the number of fuel plates per element and the type of burnable poison applied.

Though the study has not yet been completed, one of its striking preliminary results concerns the increased power peaking in the LEU fuel elements as compared with the HEU situation.

A preliminary analysis of the thermal characteristics of a typical LEU core as compared with a standard HEU core has been made and is presented in the paper.

A short survey of the various HEU and LEU calculations is given. The thermal safety analysis procedure for the HFR, as based on the flow instability criterion, is clarified. Finally, the thermal comparison HEU versus LEU and the resulting conclusions are presented.

1) Reduced Enrichment Research and Test Reactors Program.

2) HEU = Highly Enriched Uranium (93%), LEU = Low Enriched Uranium (20%).

3) Argonne National Laboratory (USA).

4) Energy Research Foundation, the Netherlands.

5) Joint Research Centre, Commission of the European Communities.

1. INTRODUCTION

Within the framework of the RERTR program various HEU-LEU core calculations have been performed by ANL in a cooperative effort with ECN and JRC-Petten. The main purpose of this work has been to gain competence in analysing HEU-LEU core conversions for high power Materials Testing Reactors and to assist in an eventual HEU-LEU conversion of the HFR-Petten. For reference purposes the present HFR standard core (HEU) in the "old" vessel geometry was calculated at first. As a next step the new vessel geometry and the increased fuel weights ($^{390}/270 \rightarrow {}^{420}/290$) were taken into account. Acceptable agreement was found between ECN and ANL results from these HEU analyses. Subsequently various LEU-HFR core options have been analysed. Main parameters in the LEU-study were the uranium loading in the meat, the fuel type, the thickness of the meat, the number of fuel plates per element and the type of burnable poison applied.
Main objective of the analysis was to select such a combination of above mentioned parameter-values that the resulting LEU core properties would be comparable with those of the present HEU core, taken into account certain future developments, such as a HFR power increase and a further increase in absorption effects of experimental core loading.
Though the study has not yet been completed one of its striking preliminary results concerns the increased power peaking in the LEU fuel elements as compared with the HEU situation (see figure 8).
A preliminary analysis of the thermal characteristics of a typical LEU core as compared with a standard HEU core has been made and is presented in this paper.

In chapter 3. a short survey of the various HEU and LEU calculations is given. The thermal safety analysis procedure for the HFR, as based on the flow instability criterion, is clarified in chapter 4.1.
The thermal comparion HEU versus LEU is presented in chapter 4.2.

2. GENERAL DESCRIPTION OF THE HFR

The High Flux Reactor (H.F.R.) Petten is a 45 MW Materials Testing Reactor cooled and moderated by light water. The reactor is operated under contract by the Netherlands Energy Research Foundation (ECN) for the Commission of the European Communities (CEC) at Petten Establishment, in North Holland, which is one of the four Establishments of the Joint Research Centres of the CEC.
A general layout of the reactor, the pool and the ancillory equipment is shown in figure 1.
The reactor core with its adjacent devices is contained in a closed vessel which is immersed in a pool of demineralized water, having a depth of 4.2 m above the top of the vessel, figure 2. The reactor vessel, 5.5 m high, consists of a cylindrical support, 1.6 m in diameter, a rectangular core box and an inlet plenum, figure 3.
Reactor cooling is provided by recirculating, in a closed circuit, demineralized water downwards in the reactor vessel through the core and through an external circuit with a delay tank, three main circulating pumps and the shell-side of the three water-to-water heat exchangers.
Eight horizontal beam tubes (H.B.), 17.5 cm in diameter each and two beam tubes, 25 cm in diameter each, are terminated on the East and South walls of the core box. The HB's are placed in three levels: five 17.5 cm HB's at 11.5 cm above the centre line of the reactor core, three 17.5 cm HB's at 11.5 cm below the core centre line and the two 25 cm HB's at 15 cm below the centre line. A large experimental facility, HB-11/12 (previously thermal column, later HBO), is terminated on the North wall of the core box, figure 5 (a) and 5 (b).
The core lattice is a 8 x 9 array containing 33 fuel assemblies of the MTR type, 6 control members, 17 experimental positions and 16 berylium reflector elements. The I-row with 9 berylium refelctor elements is placed adjacent to the outer core box wall, figure 6. Each assembly contains 23 vertically arranged fuel plates. Each plate consists of a layer of Al + U alloy meat 0.51 mm thick, clad with 0.38 mm thick aluminium for the inner plates and 0.57 mm thick aluminium for the outer fuel plates. The uranium is about 93% enriched in ^{235}U and each fresh fuel assembly contains 405g ^{235}U while the two flat side plates contain together 1000 mg ^{10}B.

The control of the reactor is achieved by six control members, each of which comprises a fuel section surmounted by a cadmium section. Drive mechanisms below the reactor move the control members upwards, displacing the cadmium with fuel.

Two power increases, from the original power of 20 MW, have been made so far since the reactor went first critical in 1961. Other changes and improvements have been also performed in the past, to fulfil the requirements of the clients (industries and institutes).

3. INFLUENCE OF HEU TO LEU CONVERSION ON CORE CHARACTERISTICS

3.1 HFR HEU core analysis (JRC-ECN and ANL calculations)

HFR core 526, table 1, has been selected as the HFR reference case for the ANL calculations. Flux measurements have been carried out in this core, which has a standard fuel loading pattern, while the experimental loading consists of Al-filler elements with Al/SS plugs. A special pre-irradiation program was carried out in order to obtain a representative control rod setting during the flux measurements. The nuclear and thermal characteristics of the core box have been calculated at Petten (ECN) by means of the HIP-TEDDI computer code.

Similar calculations have been performed by ANL, using the computer code DIF3D. A comparison between power production data in core 526 as calculated by ECN and ANL is given in table 2.

A good agreement of $\pm$ 5% has been found, with exception of the A-row, where differences of 10% appear. This discrepancy is probably related to the differences in the calculation model as applied by ECN (homogenized fuel assembly) and ANL (fuel region with two separate side plate ^{10}B) regions).

The following specific aspects of the HEU core configuration have been studied in detail by ANL:

- ^{10}B depletion rate in separate side plate regions
- equilibrium fuel loading pattern
- reactivity worth of control rods
- application of fuel elements/control rods with increased ^{235}U contents (405/280 and 420/290) as compared with core 526
- adaption of the calculation model to the outer core configuration of the new reactor vessel.

In table 3. the power distribution in a standard HEU core (420/290) is given for the old and new vessel respectively. These ECN results indicate that the vessel replacement will have a minor influence on the power distribution. Similar ANL calculations confirm this tendency.

3.2 HFR LEU-core analyses (ANL)

A variety of possible LEU core configurations has been studied. In table 4. several LEU core designs are compared with a standard HEU core in the new vessel configuration. From these LEU core designs core 465/375 (see table 4.) has been selected for the thermal safety analyses presented in chapter 4. The main differences of the fuel elements in this LEU-core as compared with the standard HEU fuel assemblies are:

- o Application of 20% enriched uranium (LEU) instead of ∿ 93% enriched fuel (HEU).
- o Application of U_3Si_2/U_3Si instead of UAlx as fuel type.
- o Increased uranium loading in the meat (4.1g/cc instead of 1.0g/cc).
- o Increased meat thickness (0.76 mm versus 0.51 mm).
- o Increased fuel plate thickness (1.52 mm instead of 1.27 mm).
- o Reduced number of fuel plates per fuel element/controlrod (20/17 versus 23/19).
- o Increased cooling channel gap (2.45 mm versus 2.18 mm).
- o Application of cadmium wires instead of Boron slabs as a burnable poison in the side plates (40 Cd wires; d = 0.4 mm versus 1000 mg ^{10}B per fuel element).

From the reactor performance point of view the effect of a conversion from the HEU to the LEU reference core can be summarized as follows:

- Overall decrease in the thermal flux density (< 0.683 eV) of about 15% (in core).
- Marginal deviations in fast neutron flux densities (> 1.35 MeV) ($\pm$ 5%).
- Decrease in gamma heating of about 10%.
- Minor changes in reactivity behaviour (see table 4.).
- Minor increase in hydraulic resistance of core due to reduction in coolant cross section (< 2%).
- Increase in average heat flux density in the fuel plates of about 13%.
- Increase in power peaking in fuel elements due to decrease in H/U^5 ratio (under moderated condition; see figure 8).

4. THERMAL SAFETY ANALYSIS OF HEU-TO-LEU CONVERSION

4.1. Safety assessment procedure and criteria

The thermal safety of an operational HFR core is assessed on basis of two criteria,

(I) the margin against flow-instability induced fuel plate burn-out

(II) the margin against film-boiling induced fuel plate burn-out

In all practical cases of interest the first safety criterion will be determining for the maximum allowable power. The subsequent analyses will therefore be restricted to the calculation of the flow-instability margin, which is based upon the so-called "bubble-detachment" criterium. Bubble detachment in subcooled water will only occur at a certain combination of process quantities. This condition can be represented by the following formula (see ref. 1.):

$$\eta_c \leqslant \frac{v\ (Ts - Tb)}{q} \qquad (4.1.1)$$

in which :

v = coolant speed (cm/sec)

q = local heat flux density (W/cm^2)

η_c = critical bubble detachment parameter (η_c = 32 cm^3 °C/W.sec)

Ts = saturation temperature ($Ts \approx 131$°C)

Tb = bulk water temperature

Rewriting equation (4.1) in terms of practical fuel element/reactor data, the following expression for the critical reactor power is obtained :

$$P_c = \min{}^{1)} \left[\frac{(1+\varepsilon)\ n\ .\ L_H\ .\ S_H\ .\ v\ (Ts - Ti)}{\Pi\ .\ f_h\ .\ f_q \left\{ \alpha\ .\ \eta_c + \frac{4}{\rho\ .c}\ .\ \frac{L_H}{D_H} \right\} .\ F_u\ .\ F_s} \right]_i \qquad (4.1.2)$$

in which :

ε = heat dissipation symmetry factor (for nominal case ε = 1)

n = number of fuel plates in element

L_H = heated length of plate

S_H = heated width of plate

v = coolant velocity

1) min. $[----]_i$ implies that the expression $[----]$ must be evaluated for each fuel position (i) and that the minimum value found, must be applied for the determination of Pc.

f_q = ratio of power dissipated to cooling channels to total fission power in place

Ts = saturation temperature

Ti = coolant inlet temperature

Π = power fraction of element (i)

f_h = horizontal power distribution factor in element ($f_h = \frac{P_{max}}{P_{aver.}}$)

α = vertical power peak factor

η_c = critical value of bubble detachment parameter

ρ = density of coolant

c = specific heat of coolant

D_H = heated equivalent diameter : ($D_H \approx 2$ d) in which d = coolant gap width

P_c = critical (maximum allowable) reactor power, related to the critical heat transfer conditions in a coolant channel of position (i)

F_u = total uncertainty factor ($F_u = 1.64$)

F_s = safety factor accounting for misloading errors and unscheduled power excursions (Fs = 1.20)

Since a number of parameter values from e.q. (4.1.2.) are identical for both HEU and LEU elements, the following simplified formula can be derived :

$$P_c = \min \left[\frac{C1 \cdot n \cdot v}{\Pi \cdot f_h \left\{ 1 + \frac{C2}{d} \right\}} \right]_i \qquad (4.1.3)$$

in which :

C1 = 81600

C2 = 0.7

$\Pi \cdot f_h$ = represents the peak power in a horizontal plane per element

Eq. 4.3 will be applied for thermal safety assessment of the LEU-reference core in comparison to the HEU reference core.

4.2. Safety analyses results

Based on DIF3D calculations the average and peak power densities per core position in a standard HEU core and in the selected LEU reference core have been ascertained. In tables 5 and 6 the relevant data for the thermal analyses are compiled for both core types.

Since the total hydraulic resistance of the primary circuit is marginally influenced by the difference in HEU-LEU coolant cross-section (31.7 versus 30.9 cm^2), no difference in primary coolant flow is assumed (see figure 7.). Consequently the coolant velocity will be 2% higher for the LEU case. In the right hand column of table 5 and 6 the critical reactor power for each position is presented. From these columns it can be observed that averaged over all reactor positions the allowable power is about 13% lower for the LEU core, assuming the same primary coolant inlet temperature. For the most critical positions (C4 and C6), which in this case determine the thermal limits for reactor operation, a decrease of 7% is found. For the same maximum reactor power this would mean a decrease in the maximum allowable primary inlet temperature for the LEU core of about 8 ^{0}C. The various contributions in the net 7% decrease for C4 and C6 are:

o	decrease in positional averaged power:	+ 2%
o	increase in power peak factor :	- 7%
o	decrease in heat transfer surface :	- 13%
o	increase in coolant channel gap width:	+ 9%
o	increase in coolant velocity :	+ 2%
	Total effect :	- 7%

The major negative contributions are the decrease in the number of fuel plates (23 → 20) and the increase in power peaking.
The first mentioned effect could be eliminated by the application of ultra high density fuel types ($\approx$ 5.3 g/cc) in a 23 plate assembly (d_{meat} = 0.51 mm). The second negative effect is related to the increased U-235 content of the LEU fuel elements. Since the HEU core is already in a undermoderated[1] condition ($\frac{H}{U\text{-}235} \approx 90$) an increase in U-235 contents at the same water volume will result in higher flux and power peaking effects. In this respect the application of Cd. wires in the LEU fuel elements has a drawback, since the second zone fuel elements (C4 and C6) contain only 20-30% of the original Cd contents in the side plates.

1) The ideal H/U-235 ratio for a homogeneous mixture in sphere geometry is about 450.

As compared with the ^{10}B-slab situation in the HEU core the compensation of the flux/power peaking in the side plate region will be less effective.
The major positive contribution in the combined effect of 7% is the increase in coolant channel width (0.218 → 0.245 mm), being + 9%.
Apparently the effect of the reduced number of fuel plates (- 13%) is largely compensated by the increased channel gap width (+ 9%).
An even better compensation might be obtained with a 19 plate assembly and slightly increased fuel density (4.1 → 4.2 g/cc). In this case the effects for reduced heat transfer surface, enlarged coolant channel gap width (2.66 mm) and slightly decreased coolant velocity are -17%, +16.5% and -0.8% respectively. Moreover the H/U-235 ratio is slightly improved, leading to less power peaking.

5. CONCLUSIONS

In comparing the thermal characteristics of a standard HEU core and the selected reference LEU core for the HFR, the following conclusions and remarks can be made:

- Accoring to the "bubble detachment" criterion, the core averaged positional critical reactor power for the LEU core is about 13% lower as compared with the HEU case.
- For the most critical positions (C4 and C6) a decrease in allowable reactor power of 7% is found. In practice this implies a decrease in allowable primary inlet temperature of about 8^0C in order to maintain the same safety margin at maximum power (50 MW).
- Due to the hydraulic characteristics of the total primary circuit (core, piping, pumps) a change in total coolant cross-section of the core has a minor influence on the primary flow. Consequently the change in coolant velocity in inversed proportional with a change in coolant cross section.
- According to the bubble detachment criterium a decrease in heat transfer surface (number of fuel plates) can be largely (75%) compensated by an increased coolant channel width. (see eq. 4.13).

- The increased U^{235} loading of a LEU core leads to more pronounced power peaking (7%).

Possibly a critical review of all the parameters involved in the composition of a LEU core for the HFR may result in a more balanced situation from the thermal point of view.
No accident conditions have been analysed yet, but, reffering to earlier publications in an IAEA guidebook, no significant different results are expected for the LEU case, as compared with HEU.
Apart from the thermal hydraulic aspects solutions for the expected loss in thermal flux in experimental positions have to be found.

5. REFERENCES

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Description of subroutine "CAT"
Calculation of allowable primary inlet temperature of the HFR, based on computed power data
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ADDENDUM 1

RATIO OF CRITICAL POWER

$$\frac{P_C}{P_{C,HEU,REF}} = \frac{\Pi F_{H.REF} \cdot N \cdot V}{\Pi F_H \cdot N_{REF} \cdot V_{REF} \left(0.3 + 0.7 \dfrac{D_{REF}}{D} \right)}$$

IN WHICH

REF = INDEX CURRENT HEU ELEMENT WITH 23 FUEL PLATES

ΠF_H = PEAK POWER IN HORIZONTAL PLANE IN ELEMENT

N = NUMBER OF FUEL PLATES PER ELEMENT

V = COOLANT VELOCITY

D = COOLANT GAP WIDTH

ADDENDUM 2

ELEMENT TYPE	MEAT TH.	CHANNEL WIDTH	$\frac{N}{N_{REF}}$	$\frac{V}{V_{REF}}$	$\frac{1}{0.3 + 0.7 \frac{D_{REF}}{D}}$	$\frac{P_c}{P_{c,REF}}$
23 PL HEU	0.51	2.18	1.00	1.00	1.00	1.00
23 PL LEU	0.51	2.18	1.00	1.00	1.00	1.00
22 PL LEU	0.51	2.34	0.96	0.98	1.05	0.98
21 PL LEU	0.51	2.51	0.91	0.95	1.10	0.96
20 PL LEU	0.51	2.70	0.87	0.93	1.16	0.93
19 PL LEU	0.51	2.91	0.83	0.91	1.21	0.91
23 PL LEU	0.76	1.93	1.00	1.13	0.92	1.04
22 PL LEU	0.76	2.09	0.96	1.09	0.97	1.02
21 PL LEU	0.76	2.26	0.91	1.06	1.03	0.99
20 PL LEU	0.76	2.45	0.87	1.02	1.08	0.97
19 PL LEU	0.76	2.66	0.83	0.99	1.15	0.94
18 PL LEU	0.76	2.89	0.78	0.96	1.21	0.91

	A	B	C	D	E	F	G	H	I
1	+	+	+	+	+	+	+	+	+
2	278.3 209	211.8 53	257.5 138	Al	273.1 192	Al	219.8 56	Al	+
3	217.7 44	392.0 1000	Al/SS	390.6 1000	Al/SS	389.8 1000	Al/SS	215.2 56	+
4	292.7 277	273.0 -	344.3 586	194.8 -	335.2 532	168.8 -	287.2 255	Al	+
5	225.4 66	353.5 662	Al/SS	304.3 345	Al/SS	347.3 620	Al/SS	300.9 341	+
6	294.5 177	236.5 -	339.8 574	237.7 -	340.1 567	166.9 -	289.1 250	Al	+
7	217.1 59	391.3 1000	Al/SS	390.4 1000	Al/SS	390.7 1000	Al/SS	214.1 54	+
8	278.3 198	210.9 50	258.9 154	Al	276.8 189	Al	219.2 57	Al	+
9	+	+	+	+ Pb	+ Pb	+ Pb	+	+	+

CONTROL ROD

FUEL ELEMENT
---- U-235 MASS (G)
-- B-10 MASS (MG)

FUEL ELEMENT

CONTROL MEMBER
---- U-235 MASS (G)
-- B-10 MASS (MG)

EXPERIMENT

Al / Al/SS: Al FILLER ELEMENT WITH AL OR AL/SS PLUG

REFLECTOR ELEMENT

+ (Pb) BE/BE

⊕ AL/BE

TABLE 1. HEU REFERENCE CORE 526.

	A	B	C	D	E	F	G	H	I
1	+	+	+	+	+	+	+	+	+
2	0.90	0.96	0.96	Al	0.99	Al	1.00	Al	+
3	0.92	0.97	Al/SS	1.02	Al/SS	1.02	Al/SS	1.01	+
4	0.94	0.98	1.01	1.01	1.02	1.00	1.02	Al	+
5	0.94	1.03	Al/SS	1.07	Al/SS	1.04	Al/SS	0.99	+
6	0.95	1.03	1.04	1.03	1.04	1.03	1.03	Al	+
7	0.94	0.99	Al/SS	1.05	Al/SS	1.04	Al/SS	1.03	+
8	0.91	0.98	1.22	Al	1.05	Al	1.03	Al	+
9	+	+	+	+ Pb	+ Pb	+ Pb	+	+	+

CONTROL ROD

--- POWER RATIO (ANL/ECN)

FUEL ELEMENT

--- POWER RATIO (ANL/ECN)

EXPERIMENT

Al / Al/SS: AL FILLER ELEMENT WITH AL OR AL/SS PLUG

REFLECTOR ELEMENT

+ (Pb) BE/BE

⊕ AL/BE

TABLE 2. COMPARISON OF CALCULATED POWER DATA IN HEU REF. CORE 526.

Table 3. Comparison of calculated power data in a standard HEU Core (420/490) for the "old" and "new" HFR-vessel (ECN).

			"old vessel"			"new" vessel			new/old
Core pos.	U-235 mass (g)	B-10 mass (mg)	Power perc. (Π)	Hor. peak fact. (f_h)	Peak power ($\Pi * f_h$)	Power perc. (Π)	Hor. peak fact. (f_h)	Peak power ($\Pi * f_h$)	Peak power ratio
A8	285.9	150	2.18	1.29	2.81	2.13	1.25	2.66	0.95
A7	249.7	72	2.09	1.07	2.24	2.11	1.03	2.17	0.97
A6	324.7	300	2.67	1.12	2.99	2.73	1.09	2.98	1.00
A5	249.7	72	2.36	1.06	2.50	2.43	1.04	2.53	1.01
A4	324.7	300	2.69	1.12	3.01	2.75	1.08	2.97	0.99
A3	249.7	72	2.12	1.07	2.27	2.15	1.03	2.21	0.97
A2	285.9	150	2.22	1.28	2.84	2.17	1.26	2.73	0.96
B8	221.3	40	2.24	1.19	2.67	2.26	1.19	2.69	1.01
B7	420.0	1000	3.29	1.14	3.75	3.41	1.13	3.85	1.03
B5	374.4	604	3.36	1.12	3.76	3.52	1.11	3.91	1.04
B3	374.4	1000	3.34	1.14	3.81	3.45	1.14	3.93	1.03
B2	285.9	40	2.28	1.19	2.72	2.30	1.20	2.76	1.01
C8	285.9	150	2.89	1.15	3.32	3.00	1.17	3.51	1.06
C6	374.4	604	3.44	1.11	3.82	3.59	1.10	3.95	1.03
C4	374.4	604	3.49	1.10	3.84	3.64	1.10	4.00	1.04
C2	285.9	150	2.98	1.18	3.41	3.06	1.18	3.61	1.06
D7	420.0	1000	3.33	1.09	3.63	3.45	1.09	3.76	1.04
D5	324.7	300	3.39	1.07	3.63	3.51	1.07	3.76	1.04
D3	420.0	1000	3.44	1.08	3.72	3.53	1.09	3.85	1.03
E8	285.9	150	2.47	1.20	2.96	2.61	1.27	3.31	1.12
E6	374.4	604	3.20	1.11	3.55	3.26	1.12	3.65	1.03
E4	374.4	604	3.25	1.12	3.64	3.30	1.13	3.73	1.02
E2	285.9	150	2.64	1.26	3.33	2.66	1.28	3.40	1.02
F7	420.0	1000	2.75	1.16	3.19	2.72	1.18	3.21	1.01
F5	374.4	604	3.03	1.13	3.42	2.97	1.16	3.45	1.01
F3	420.0	1000	2.82	1.15	3.24	2.75	1.18	3.25	1.00
G8	249.7	72	1.76	1.23	2.17	1.62	1.35	2.19	1.01
G6	324.7	300	2.32	1.08	2.51	2.13	1.13	2.41	0.96
G4	324.7	300	2.35	1.08	2.54	2.14	1.12	2.40	0.94
G2	249.7	72	1.83	1.28	2.34	1.66	1.36	2.26	0.97
H7	221.3	40	1.62	1.22	1.98	1.28	1.12	1.44	0.73
H5	324.7	300	2.28	1.17	2.67	1.81	1.14	2.06	0.77
H3	249.7	72	1.78	1.22	2.17	1.40	1.12	1.57	0.72
B6	253.5	-	2.16	1.15	2.48	2.28	1.14	2.60	1.05
B4	290.0	-	2.40	1.15	2.76	2.53	1.14	2.88	1.04
D6	253.5	-	2.31	1.14	2.63	2.41	1.14	2.75	1.05
D4	218.7	-	2.13	1.11	2.36	2.21	1.11	2.46	1.04
F6	185.5	-	1.54	1.14	1.76	1.53	1.16	1.77	1.00
F4	185.5	-	1.56	1.13	1.76	1.54	1.15	1.77	1.00

Table 4. Fuel cycle characteristics for Petten 26 day cycle using new tank model, ^{10}B and ^{113}Cd are treated with burn-up dependent cross sections. No control rods inserted during cycle.

^{235}U load, g/el.	420/290	499/402	499/402	465/375	559/446	559/446	536/423
#plates/element	23/19	20/17	20/17	20/17	19/16	19/16	19/16
channel gap (cm)	0.218	0.245	0.245	0.245	0.242	0.242	0.242
meat ρ_u, (g/cc)	1.016	4.396	4.396	4.096	3.939	3.939	3.763
BP	1g ^{10}B/el	40Cdw(0.4)	0.5g ^{10}B/el	40Cdw(0.4)	38Cdw(0.4)	0.5g ^{10}B/el	38Cdw(0.4)
K_{BOL}	1.10584	1.10541	1.10378	1.0974	1.09865	1.09480	1.09311
K_{BOEC}	1.05576	1.06660	1.05419	1.05791	1.05921	1.04817	1.05337
K_{MOEC}	--	1.05258	--	1.04235	1.04638	--	1.04012
K_{EOEC}	1.03028	1.04514	1.0297	1.03415	1.04003	1.02625	1.03315
$\Delta K_{(B-EOEC)}$	0.02548	0.02146	0.02447	0.02376	0.01918	0.02192	0.02022
K_{BOEC} all rods in	0.79278		0.79374	0.78969		0.7914	0.79145
rod worth, %	31.4		31.1	32.1		30.95	31.4
^{235}U load BOL, Kg	15.601	18.871	18.871	17.595	21.112	21.112	16.621
^{235}U load BOEC, Kg	11.8095	15.307	15.303	14.030	17.564	17.558	16.621
^{235}U load EOEC, Kg	10.315	13.890	13.886	12.619	16.139	16.134	15.200
Fuel meat, TK., mm	0.51	0.76	0.76	0.76	1.00	1.00	1.00
Burnable Abs.	^{10}B	^{113}Cd	^{10}B	^{113}Cd	^{113}Cd	^{10}B	^{113}Cd
g BA(BOEC)	11.50	20.07	6.06	19.610	19.85	6.39	19.540
(EOEC)	5.58	2.65	3.14	2.19	3.30	3.50	2.99
Discharge							
Burnups, STD, %	50.18	39.66	39.63	42.28	35.74	35.70	37.30
CONT, %	52.81	38.21	38.32	41.19	33.88	33.99	35.4
Fissile Pu, Kg							
BOEC	0.0196	0.332	0.335	0.324	0.361	0.363	0.353
EOEC	0.0258	0.447	0.450	0.435	0.489	0.492	0.476
^{235}U burned/MWd	1.277	1.211	1.211	1.206	1.218	1.217	1.215
Core Avg. Burnup,							
BOEC, %	24.30	18.89	18.91	20.26	16.81	16.83	17.59
EOEC, %	33.88	26.39	26.42	28.28	23.56	23.58	24.64
(BOEC) Peak Power, (w/cc)	900 C-4 951 D-3	980 (C-4)	939 (C-4)	982.7(C-4)	948(C-4)	918(C-4)	948(C-4)

Table 5. Compilation of data for the thermal analyses in a standard HEU core

Position	Power fraction (%)	Power peak factor (Hor)	Number of fuel plates	Coolant velocity (cm/s)	Coolant gap width (cm)	Positional critical reactor power (MW)
i	π	f_h	n	v	d	Pc_i (eq. 4.1.3)
A8	1.64	1.37	23	680	0.218	138
A7	1.79	1.15	23	680	0.218	150
A6	2.59	1.34	23	680	0.218	89
A5	2.34	1.17	23	680	0.218	113
A4	2.54	1.33	23	680	0.218	91
A3	1.82	1.16	23	680	0.218	146
A2	1.75	1.38	23	680	0.218	128
B8	1.84	1.12	23	680	0.218	150
B7	3.23	1.38	23	680	0.218	69
B5	3.87	1.18	23	680	0.218	68
B3	3.24	1.35	23	680	0.218	71
B2	1.97	1.12	23	680	0.218	140
C8	2.47	1.20	23	680	0.218	104
C6	3.91	1.22	23	680	0.218	65
C4	3.93	1.24	23	680	0.218	63
C2	2.53	1.20	23	680	0.218	102
D7	3.51	1.35	23	680	0.218	65
D5	3.92	1.14	23	680	0.218	69
D3	3.50	1.40	23	680	0.218	63
E8	2.23	1.28	23	680	0.218	108
E6	3.58	1.27	23	680	0.218	68
E4	3.56	1.29	23	680	0.218	67
E2	2.24	1.29	23	680	0.218	107
F7	2.73	1.45	23	680	0.218	78
F5	3.30	1.27	23	680	0.218	74
F3	2.70	1.51	23	680	0.218	76
G8	1.41	1.40	23	680	0.218	157
G6	2.14	1.38	23	680	0.218	105
G4	2.10	1.40	23	680	0.218	105
G2	1.34	1.37	23	680	0.218	168
H7	1.13	1.29	23	680	0.218	212
H5	1.61	1.31	23	680	0.218	147
H3	1.19	1.34	23	680	0.218	194
B6	2.87	1.17	19	680	0.218	76
B4	2.95	1.17	19	680	0.218	74
D6	2.94	1.17	19	680	0.218	74
D4	2.61	1.11	19	680	0.218	88
F6	1.96	1.07	19	680	0.218	122
F4	1.80	1.07	19	680	0.218	133

Table 6. Compilation of data for the thermal analyses of a typical LEU core

Position	Power fraction (%)	Power peak factor (Hor)	Number of fuel plates	Coolant velocity (cm/s)	Coolant gap width (cm)	Positional critical reactor power (MW)
i	π	f_h	n	v	d	Pc_i (eq 4.1.3)
A8	1.58	1.52	20	694	0.245	124
A7	1.78	1.31	20	694	0.245	128
A6	2.42	1.46	20	694	0.245	85
A5	2.32	1.31	20	694	0.245	89
A4	2.39	1.44	20	694	0.245	87
A3	1.81	1.31	20	694	0.245	126
A2	1.67	1.54	20	694	0.245	116
B8	1.96	1.25	20	694	0.245	122
B7	3.06	1.35	20	694	0.245	73
B5	3.65	1.24	20	694	0.245	67
B3	3.08	1.34	20	694	0.245	73
B2	2.02	1.25	20	694	0.245	118
C8	2.51	1.40	20	694	0.245	85
C6	3.83	1.30	20	694	0.245	60
C4	3.84	1.32	20	694	0.245	60
C2	2.57	1.40	20	694	0.245	84
D7	3.48	1.27	20	694	0.245	67
D5	3.91	1.19	20	694	0.245	64
D3	3.48	1.29	20	694	0.245	66
E8	2.34	1.46	20	694	0.245	88
E6	3.52	1.32	20	694	0.245	64
E4	3.52	1.36	20	694	0.245	62
E2	2.35	1.46	20	694	0.245	88
F7	2.69	1.34	20	694	0.245	83
F5	3.14	1.33	20	694	0.245	71
F3	2.68	1.37	20	694	0.245	82
G8	1.47	1.60	20	694	0.245	127
G6	2.13	1.42	20	694	0.245	99
G4	2.09	1.42	20	694	0.245	101
G2	1.44	1.58	20	694	0.245	131
H7	1.20	1.44	20	694	0.245	173
H5	1.62	1.44	20	694	0.245	128
H3	1.24	1.46	20	694	0.245	165
B6	2.92	1.24	17	694	0.245	70
B4	3.05	1.22	17	694	0.245	68
D6	3.09	1.22	17	694	0.245	67
D4	2.92	1.18	17	694	0.245	74
F6	2.20	1.23	17	694	0.245	94
F4	2.09	1.19	17	694	0.245	102

	A	B	C	D	E	F	G	H	I
1	+	+	+	+	+	+	+	+	+
2	0.91	0.84	0.82	Al	0.82	Al	0.78	Al	+
3	0.86	1.03	Al/SS	1.05	Al/SS	1.08	Al/SS	0.85	+
4	0.96	0.92	0.95	0.84	0.93	0.77	0.96	Al	+
5	0.79	0.98	Al/SS	0.93	Al/SS	0.96	Al/SS	0.87	+
6	0.96	0.92	0.92	0.91	0.94	0.77	0.94	Al	+
7	0.85	1.06	Al/SS	1.03	Al/SS	1.06	Al/SS	0.82	+
8	0.90	0.81	0.82	Al	0.81	Al	0.81	Al	+
9	+	+	+	+	+	+	+	+	+

CONTROL ROD

---- $\frac{P_{CRIT}\ (LEU)}{P_{CRIT}\ (HEU)}$

EXPERIMENT

Al / Al/SS — Al FILLER ELEMENT WITH Al OR Al/SS PLUG

FUEL ELEMENT

---- $\frac{PCRIT\ (LEU)}{PCRIT\ (HEU)}$

REFLECTOR ELEMENT

+ (Pb) BE/BE

⊕ AL/BE

(SEE EQ. 4.3. AND TABLE 5 AND 6)

TABLE 7. COMPARISON OF CRITICAL REACTOR POWER DATA IN A STANDARD HEU CORE AND A TYPICAL LEU CORE.

	A	B	C	D	E	F	G	H	I
1	+	+	+	+	+	+	+	+	+
2	0.96	1.01	1.06	Al	1.02	Al	0.97	Al	+
3	0.97	1.03	Al/SS	1.03	Al/SS	1.00	Al/SS	0.72	+
4	0.99	1.04	1.04	1.04	1.02	1.00	0.94	Al	+
5	1.01	1.04	Al/SS	1.04	Al/SS	1.01	Al/SS	0.77	+
6	1.00	1.05	1.03	1.05	1.03	1.00	0.96	Al	+
7	0.97	1.03	Al/SS	1.04	Al/SS	1.01	Al/SS	0.73	+
8	0.95	1.01	1.06	Al	1.12	Al	1.01	Al	+
9	+	+	+	+	+	+	+	+	+

CONTROL ROD

---- $\frac{P_{MAX}\ (OLD\ VESSEL)}{P_{MAX}\ (NEW\ VESSEL)}$

EXPERIMENT

Al / Al/SS — AL FILLER ELEMENT WITH Al OR Al/SS PLUG

FUEL ELEMENT

---- $\frac{P_{MAX}\ (OLD\ VESSEL)}{P_{MAX}\ (NEW\ VESSEL)}$

REFLECTOR ELEMENT

+ (Pb) BE/BE

⊕ AL/BE

STANDARD HEU CORE TYPE 420/290 (SEE TABLE 3)

TABLE 8. INFLUENCE OF THE HFR VESSEL REPLACEMENT ON THE LOCAL PEAK POWER DENSITIES.

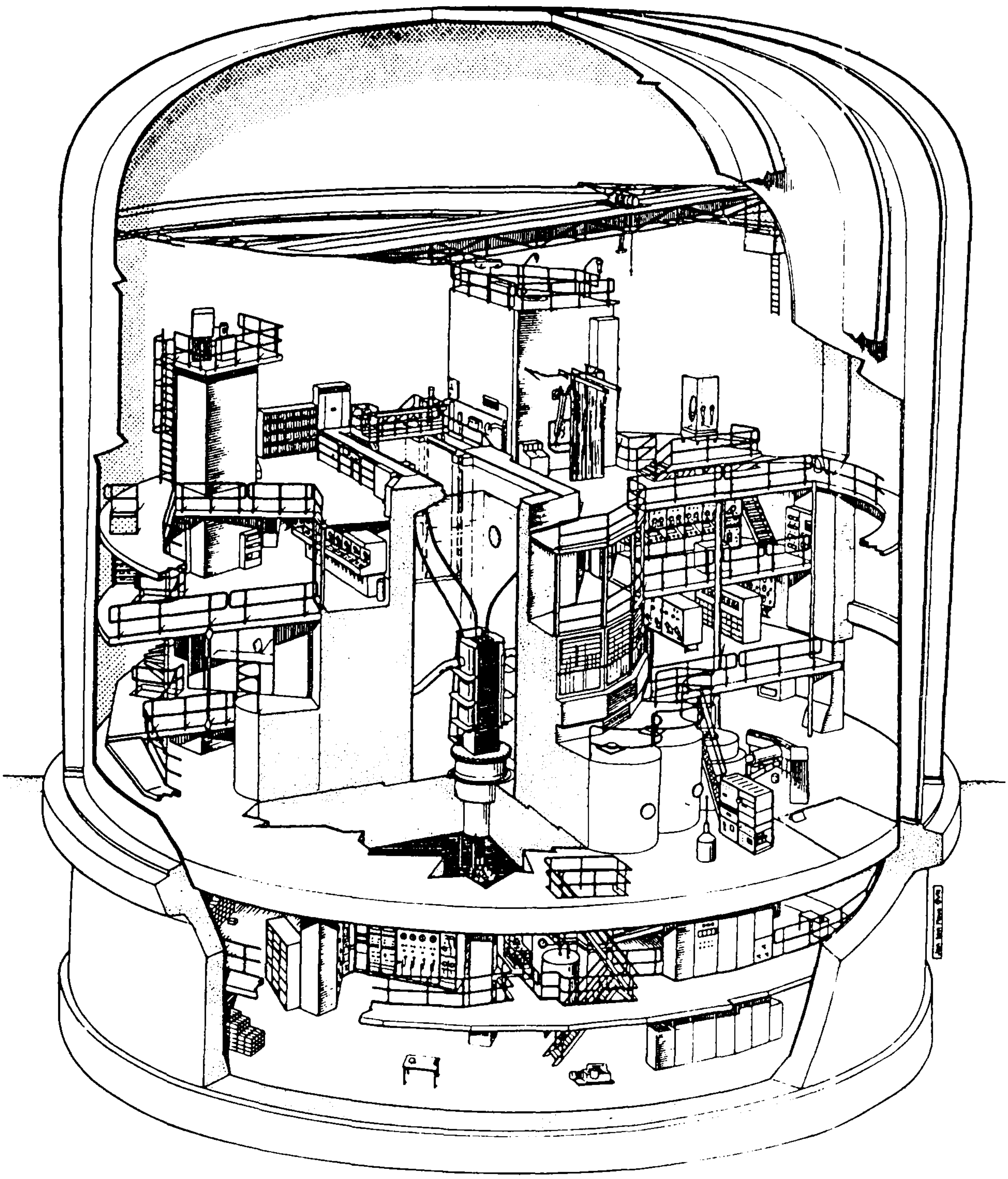

FIG. 1. HIGH FLUX MATERIALS TESTING REACTOR (HFR), PETTEN

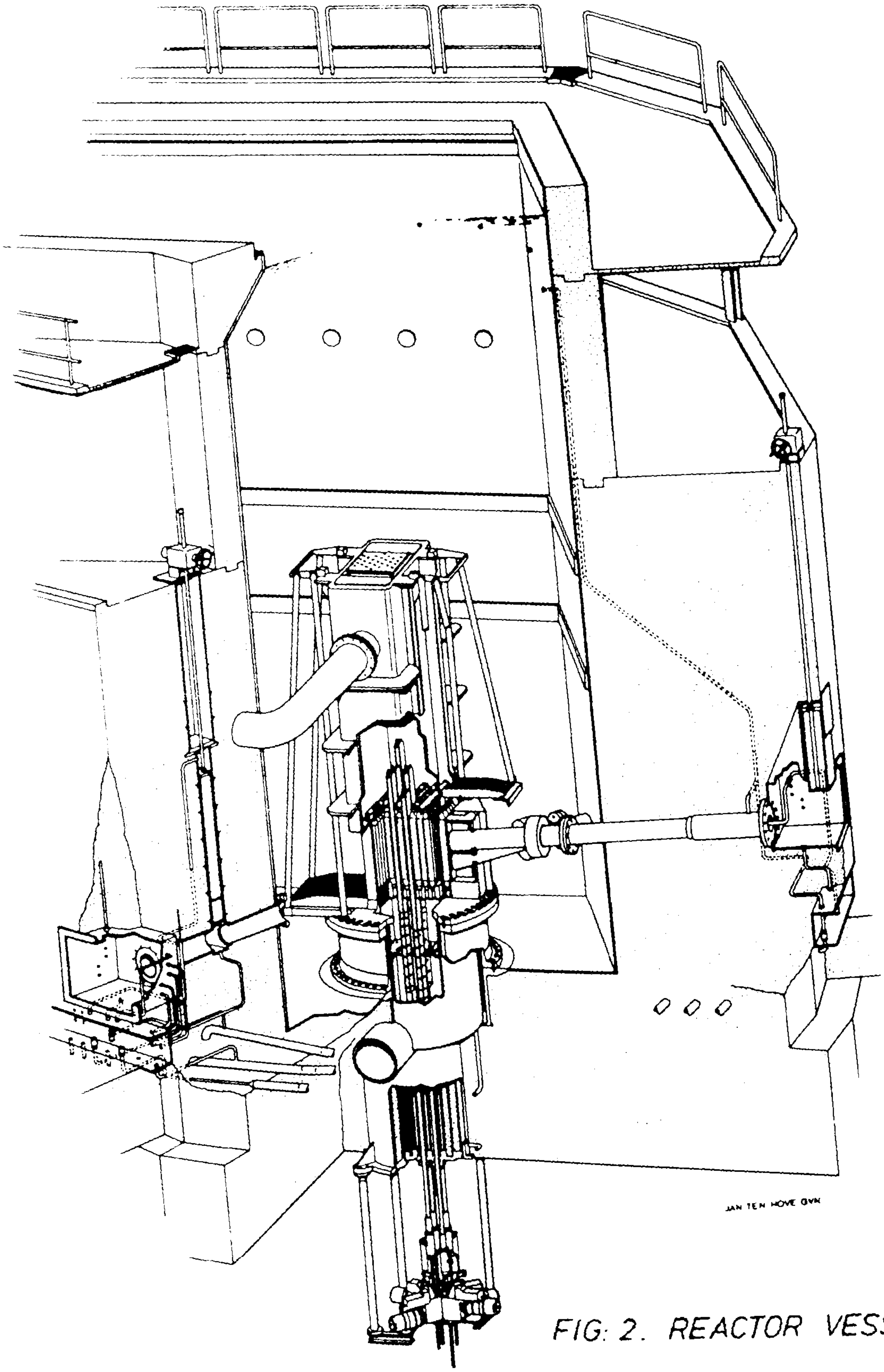

FIG: 2. REACTOR VESSEL IN POOL

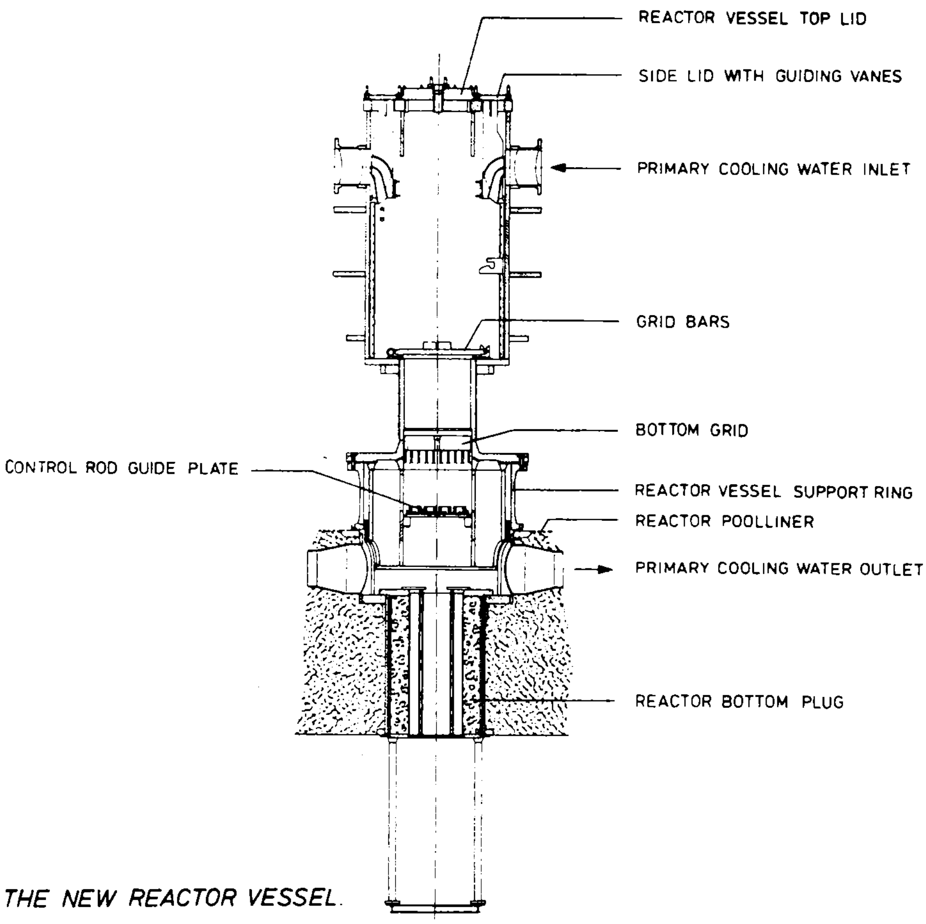

FIG.3 : THE NEW REACTOR VESSEL.

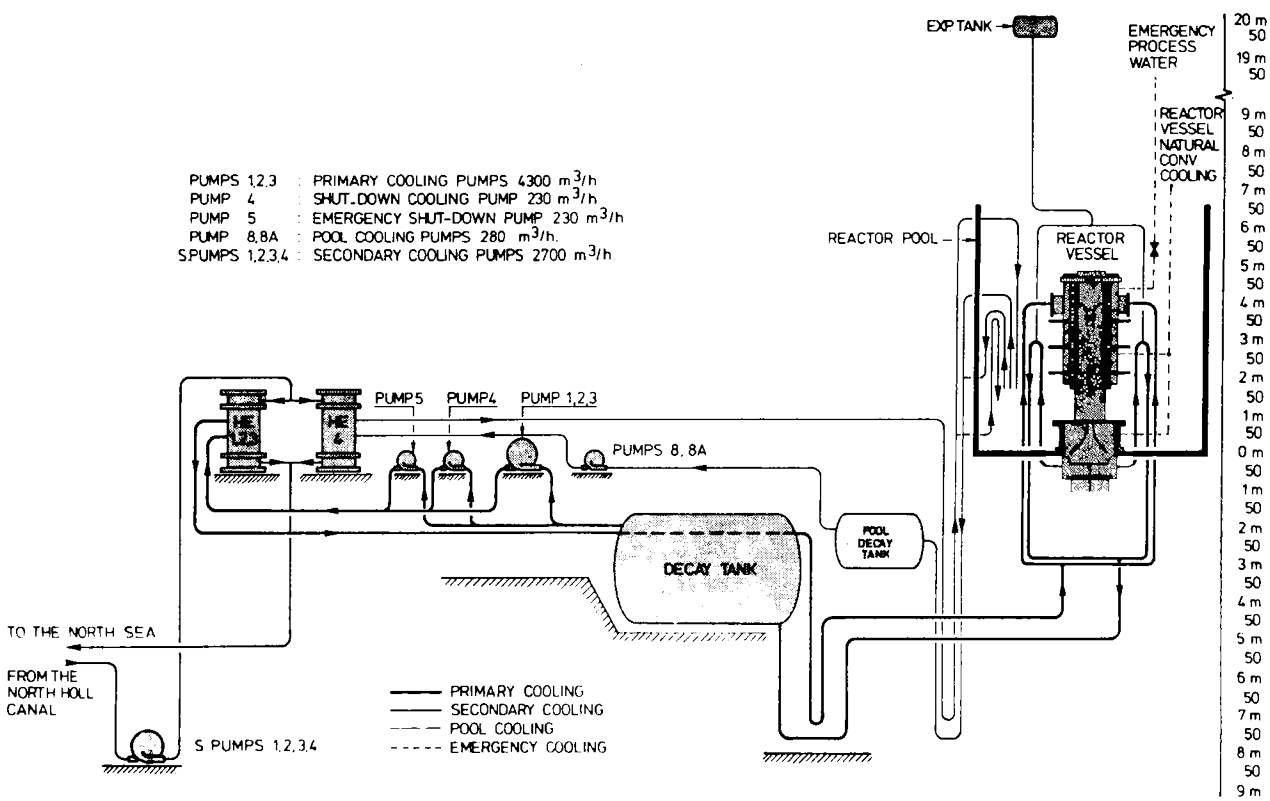

FIG. 4 FLOW DIAGRAM OF THE MAIN COOLING CIRCUITS.

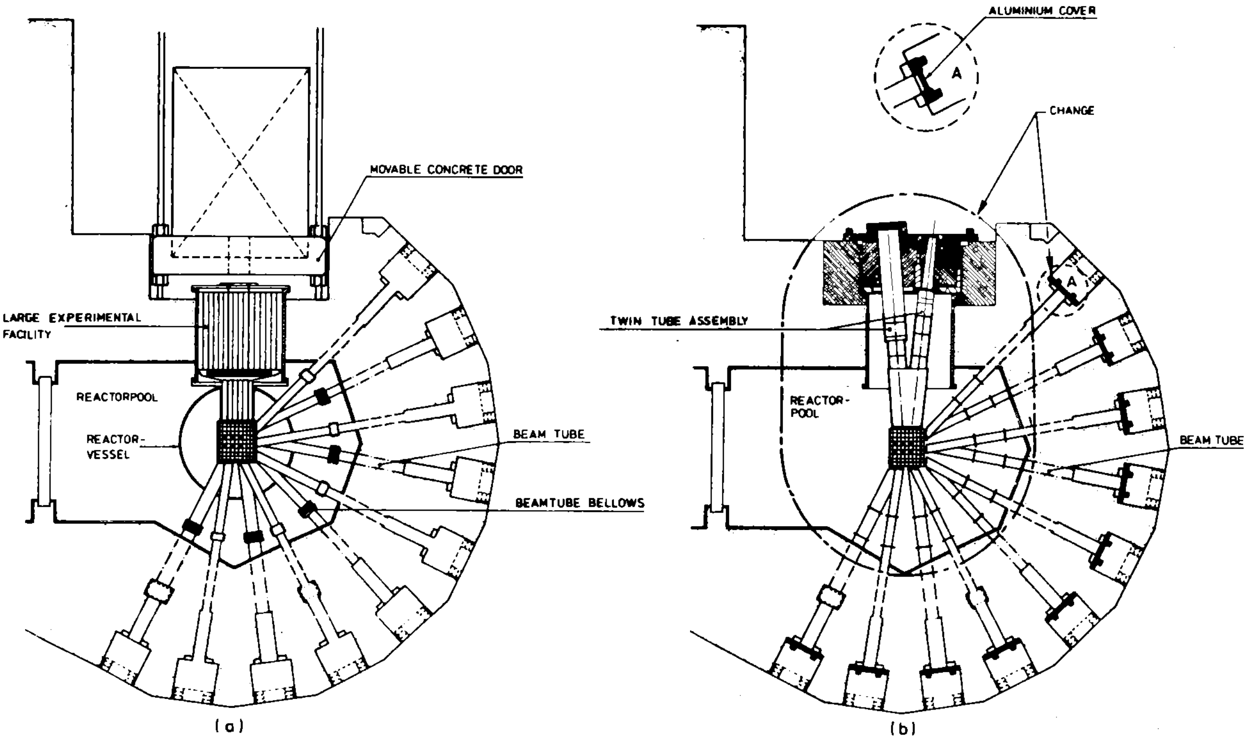

FIG 5: SCHEMATIC HORIZONTAL CROSS SECTION THROUGH REACTOR VESSEL AND BEAM TUBES.

(a) ORIGINAL DESIGN
(b) IMPROVED DESIGN

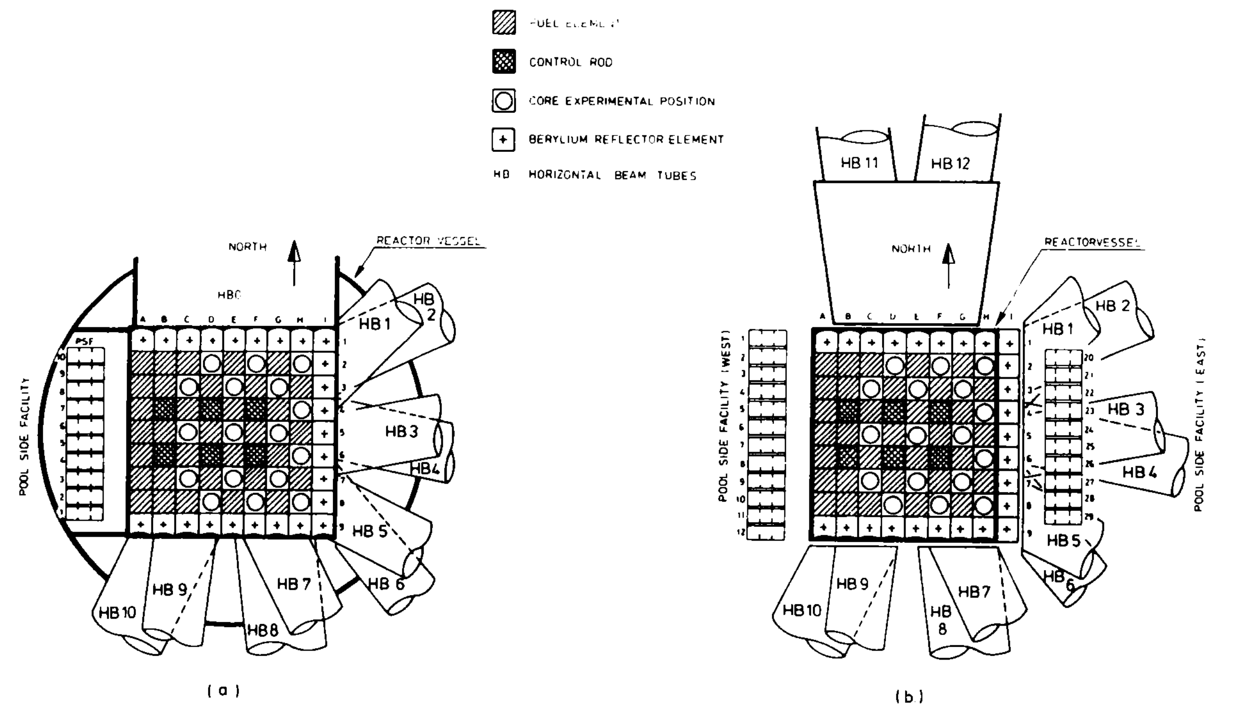

FIG 6: SCHEMATIC HORIZONTAL CROSS SECTION THROUGH REACTOR CORE AND EXPERIMENTAL FACILITIES.

(a) : ORIGINAL DESIGN
(b) : IMPROVED DESIGN

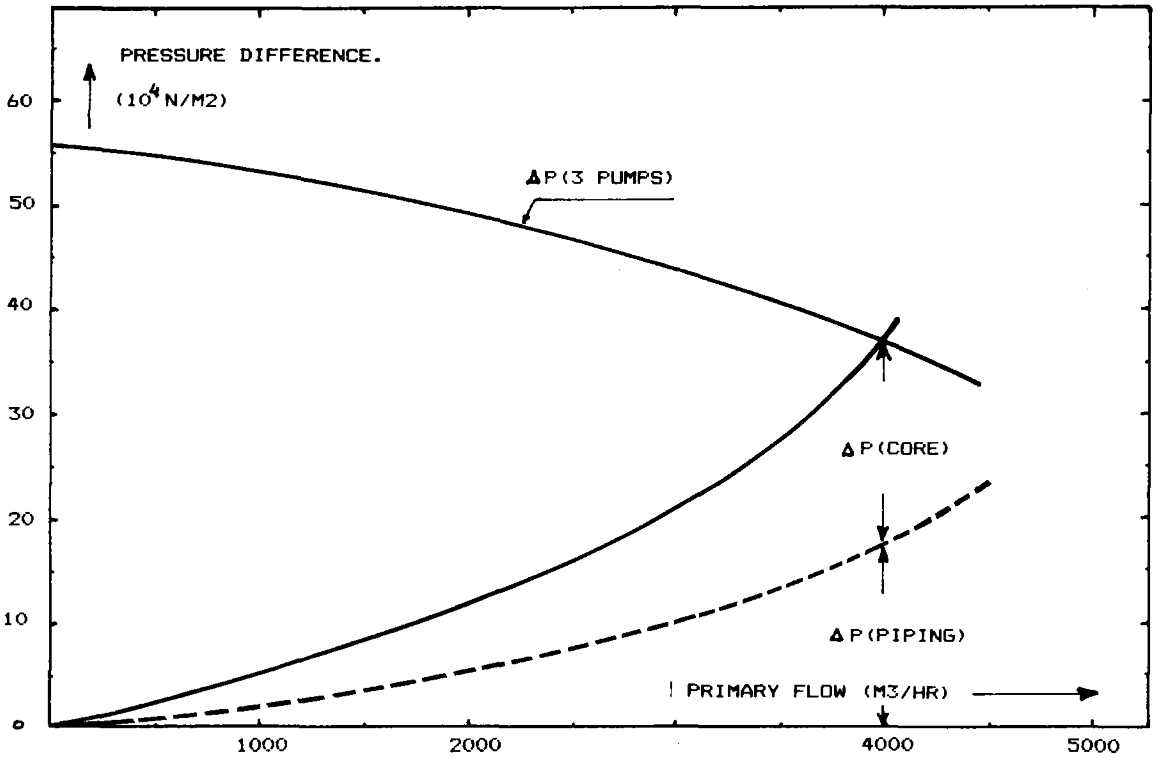

FIG 7 HYDRAULIC CHARACTERISTICS OF THE PRIMARY CIRCUIT OF THE HFR-PETTEN.

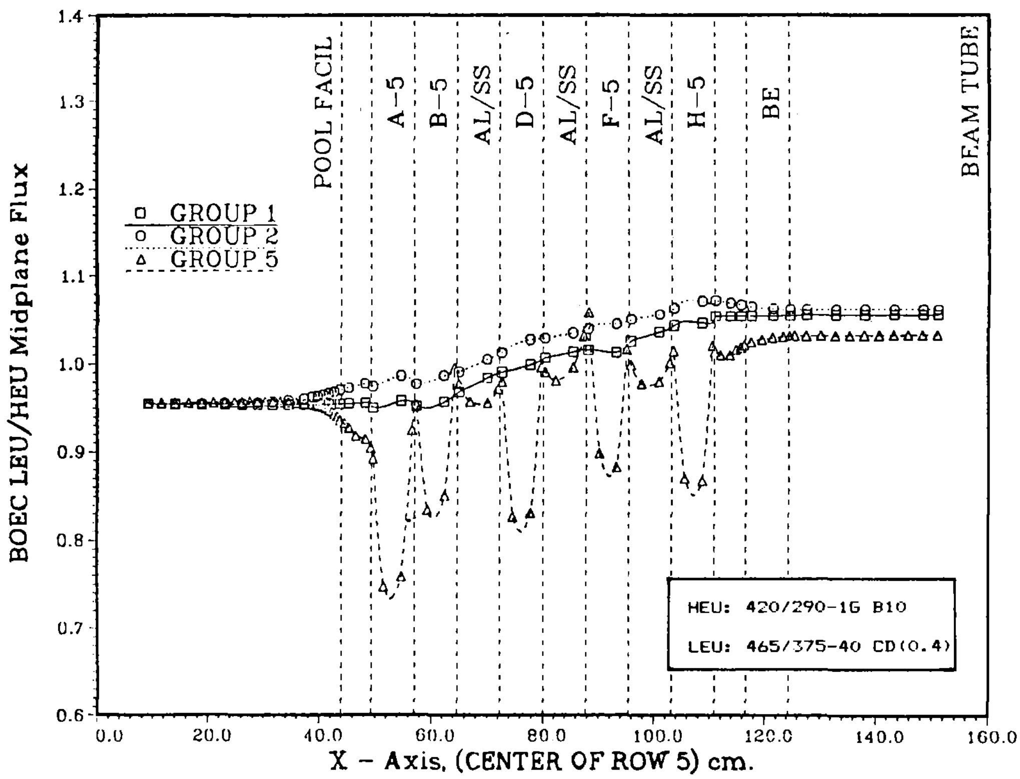

FIG.8 COMPARISON OF GROUP FLUX DENSITIES IN A HEU AND LEU CORE