REACTIVITY ACCIDENT ANALYSIS IN MTR CORES

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ABSTRACT

The purpose of the present work is the analysis of reactivity transients in MTR cores with LEU and HEU fuels.

The analysis includes the following aspects:

The phenomenology of the principal events of the accident that takes place, when a reactivity of more than 1$ is inserted in a critical core, in less than 1 second.

The description of the accident that happened in the RA-2 critical facility in September 1983.

The evaluation of the accident from differents points of view:

Theoretical and qualitative analysis,
Paret Code calculations,
Comparison with Spert I and Cabri experiments,
and with post-accident inspections.

Differences between LEU and HEU RA-2 cores.

INTRODUCTION

The analysis of a reactivity accident is an important part of a safety study in every nuclear reactor. By means of it, the total released energy, the maximum power, the maximum fuel, clad and coolant temperatures, the eventual partial core melting (and consequently, the fission product liberation to air), etc., may be predicted for every probable case considered.

In this work, some calculational tools are presented, which were used in the RA-2 accident of September 1983, and may help for this type of study.

Nevertheless, we want to remark the difference between a safety analysis, and the detailed calculation of an accident that actually ocurred. In this last case, there are uncertainties in some of the parameters which are determinant for the values reached by the main variables characterizing the accident, so that careful limits for those parameters are needed, while in a safety analysis such parameters have postulated values.

PHENOMENOLOGY OF THE PRINCIPAL EVENTS IN A REACTIVITY ACCIDENT

Description

In this analysis, a fast reactivity insertion transient due to a ramp of more than 1$ in less than 1 second is considered. The accident is supposed to happen in a critical assembly, with MTR fuels, moderated and cooled with light water, initially at very low power (less than 1 watt) and without considering scram during the first seconds.

The variation of the characteristic parameters of the transient is shown in Fig. 1..

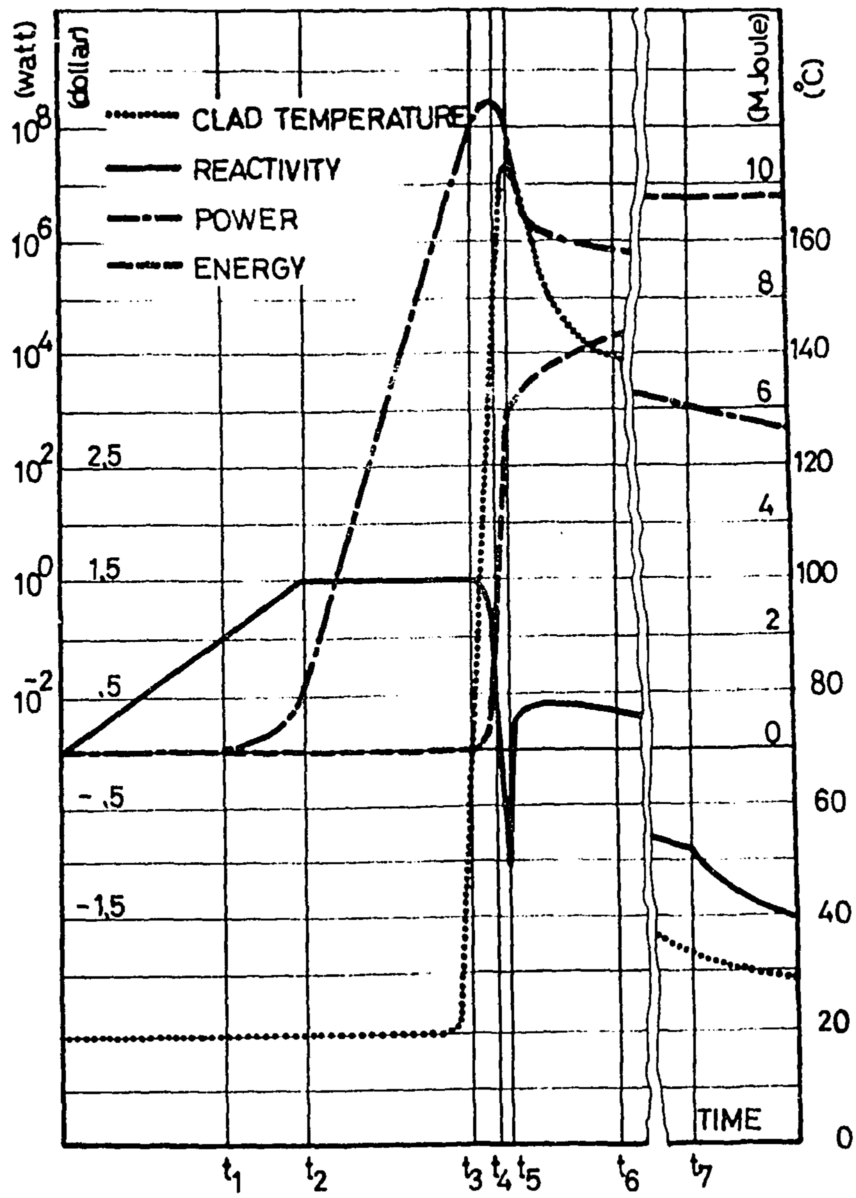

Fig. 1. Variation of Characteristic Parameters

The prompt-critical condition is reached at t_1, while the maximum reactivity insertion occurs at t_2. The feedback mechanisms are active from t_3 on. The power peak is reached at t_4, being t_5 the instant of maximum clad temperature. The power change is very slow at t_6 (quasi-equilibrium power) and the scram mechanism becomes active at t_7.

Between t_3 and t_5, fuel, clad, and coolant temperatures (T_f, T_c, T_r) increase; boiling appears and the reactivity feedback mechanism develops. If the flux and the clad temperature are higher than their critical values during this period, flow instabilities or DNB occur for these channel conditions, and their clad temperature are predicted to exceed melting temperature.

Feedback Mechanisms

The most important feedback mechanisms above mentioned are:

- Doppler effect in the fuel,
- Uniform heating of core,
- Fuel plate thermal expansion,
- Moderator thermal expansion,
- Steam formation in the channels,
- Geometric change in the configuration.

The fuel Doppler effect follows energy generation instantaneously. It is the more important, the less enriched are the fuel elements (FE).

The global core heating produces a reactivity feedback estimated by the reactivity temperature coefficient, α_T.

The fuel plate and moderator expansion, as well the steam formation, produce a reactivity feedback estimated by the void coefficient, α_V.

In MTR fuels, some plates may get out of their original position, due to pressures created during the transient. This geometrical change also affects reactivity.

RA-2 ACCIDENT

Characteristics of the critical facility

The RA-2 is a pool type critical assembly of variable configuration used for experimentation, with MTR FE, 90 % enriched. The nominal power is 0.1 watt. The moderator is demineralized light water at atmospheric pressure.

It has two safety systems, one of them, the fast one, producing the falling down of neutron absorbing rods, the slow one, lets the water drain from the tank.

An illustration of a fuel plate is shown in Fig. 2. The fuel plate contained Uranium-Aluminium alloy, cladded in Al.

In a standard fuel element (SFE), containing 148 g U-235, there were 19 plates inserted regularly between two Aluminium lateral supports.

In a control fuel element (CFE), containing 117 g U-235, there were 15 plates, and in the place of plates 3-4, 16-17 of SFE, 2 SS cladded Cadmium plates could be moved.

Some views of SFE are also shown in Fig. 2. The FE are placed on a grid plate, with a 8.06 cm pitch in one direction and a 7.72 cm pitch in the other direction. Usually water reflector is used, but Graphite, and Berylium, Al cladded reflectors, with the same external dimensions as SFE could be used.

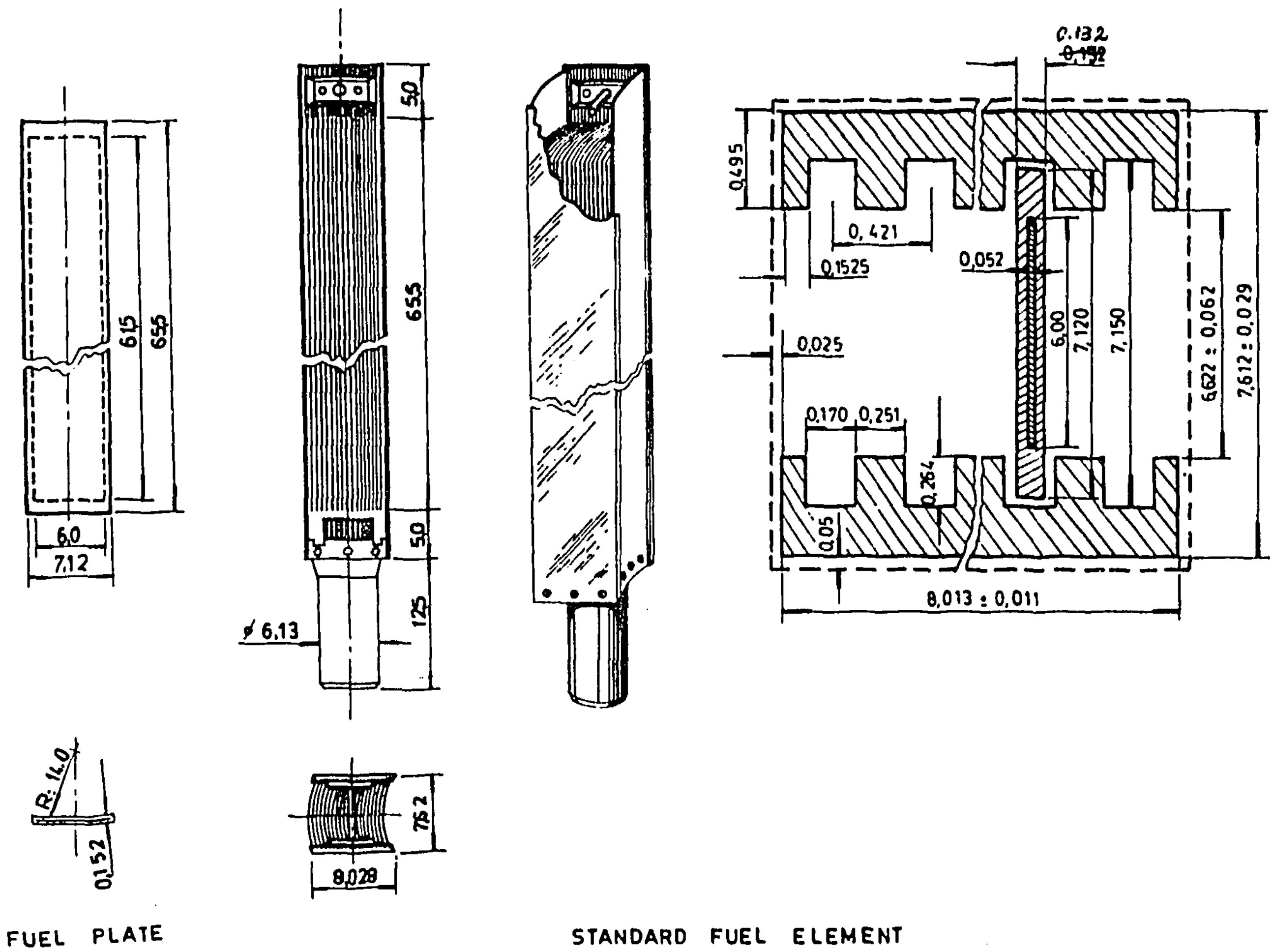

Fig. 2. Views of Standard Fuel Element

Description of the accident sequence

Although the actual sequence cannot be exacty confirmed, the most probable development of events ocurred as follows:

The pool water level was, as the operator said, slightly below the top of active zone of fuel plates.

The core configuration before any change was the one shown in Fig. 3 as A (-2.7$), and the required configuration was the G (= -3.1$). After some changes, subcritical configurations B, C, and D, the operator reaches the configuration E. In this situation, and being a configuration not necessarily subcritical, a CFE without the absorbing plates inside it, was almost completely inserted in the C3 position (Configuration F). This event meant a reactivity excess greater than 1$, producing a prompt critical condition.

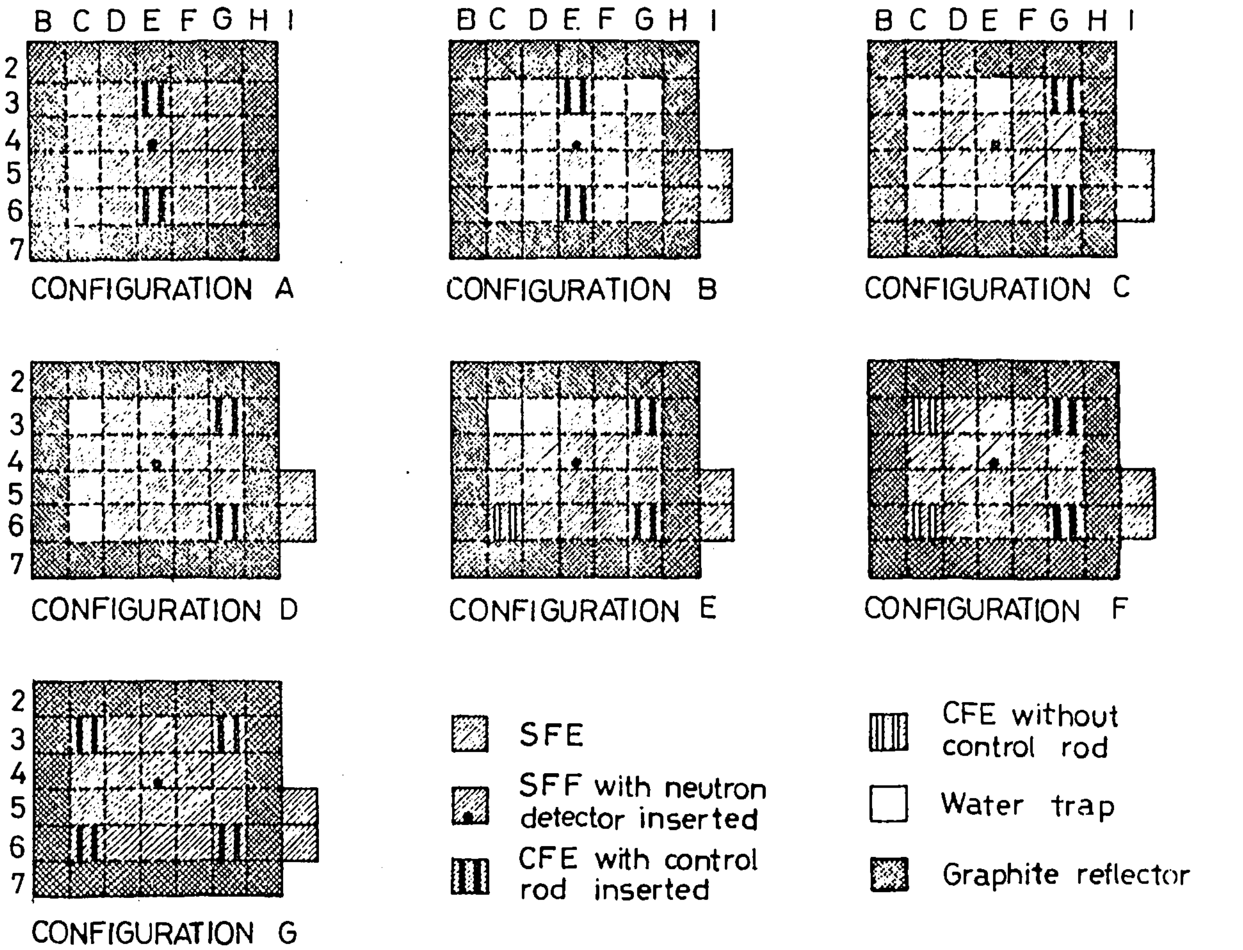

Fig. 3. RA-2 Accident Probable Configuration Sequence

EVALUATION OF THE ACCIDENT

Some different methods were considered in order to obtain the results that predict the transient best.

Theoretical Analysis

An elemental neutronic model was used in order to know the principal parameters describing the transient and to estimate trends.

The model is based in the point kinetics, neglecting delayed neutrons and external source; the feedback reactivity depends on the energy released by fission without delay time:

$$\frac{dP(t)}{dt} = \frac{(\$(t) - 1) \quad P(t)}{\Lambda^*} \tag{1}$$

$$\$(t) = \$_0 - \gamma E^n(t) \tag{2}$$

initial conditions: $P(0) = P_0$, $\$(0) = \$_0$, $E(0) = 0$

where:

$P(t)$: reactor power

$E(t) = \int_0^t P(t)\, dt$: energy released

$\Lambda^* = \Lambda/\beta$ · reduced prompt neutron generation time

γ: energy reactivity coefficient

Both parameters γ and n could be obtained either by adjusting measured values [1], or, by analytical models giving the contribution of different feedback mechanisms produced by the energy released [2].

As an illustration γ depends on α_V strongly, while n is associated with the peak asymmetry (n=1, symmetric peak).

With the following definitions:

$$\alpha_0 = \frac{\$_0 - 1}{\Lambda^*}$$: reciprocal initial period

$$E_T = \frac{(n+1)^{1/n} (\$_0 - 1)^{1/n}}{\gamma^{1/n}} \quad : \text{ Total energy released}$$

$$A = \frac{(E_T)^n}{(P_0/\alpha_0)^n}$$

The terms for the energy and power are:

$$\frac{E_T^n - E^n(t)}{(E(t) + P_0/\alpha_0)^n} = A\, e^{-n\alpha_0 t} \tag{3}$$

$$P(t) = \frac{\alpha_0 (E_T^n - E^n(t)) (E(t) + P_0/\alpha_0)}{E_T^n + E^{n-1}(t) (P_0/\alpha_0)} \tag{4}$$

(Valid for cases n=1,2 and 3)[3].

If $E_T \gg P_o/\alpha_o$, for the peak instant, t_p, they are reduced to:

$$t_p = \frac{1}{n\alpha_o} \ln (A/n) \tag{5}$$

$$E(t_p) = \frac{(\$_0 - 1)^{1/n}}{\gamma^{1/n}} \tag{6}$$

$$P(t_p) = \frac{n}{n+1} \frac{(\$_0 - 1) E(t_p)}{\Lambda^*} \tag{7}$$

As delayed neutrons are not taken into account, this model predicts a pronounced diminishing power that is not real.

Eq. (6) relates the energy released up to the instant the power peak is reached, with the reactivity excess and the energy coefficient. If a scram takes place, this value is the most important contribution to the total released energy. It is independent of Λ^*.

The peak power depends on the reactivity excess strongly, being inversely proportional to Λ^*, as eq. (7) shows it.

According to the preceding arguments, it may be concluded that in LEU cores (smaller Λ^*) with similar γ , the power peaks produced are higher and narrower than the corresponding peaks in HEU cores.

Paret Code Calculations

Paret, an ANL code, was used in order to calculate the transient. It is a coupled thermalhydraulic-neutronic code with a continuous reactivity feedback [4].

A nodal point kinetics up to 4 regions is used for the neutronic model. The heat transfer in each channel is computed on the basis of a one-dimensional conduction solution for each of up to a maximum of 20 axial sections.

The heat transfer correlations chosen (as suggested by ANL), were:

simple phase	Sieder - Tate
two phase	Mc. Adams
nucleate boiling	Bergles - Rohsenow
critical heat flux	Tong
void model	Zuber

The core was represented by two regions, one of them having average channels and the other, the hottest channels of the core.

Input Data:

Inserted Reactivity: A lineal insertion of 1.5$ in 0.5 sec..

This reactivity value is the result of neutronic calculations corrected with two experimental terms, one of them representing the value of the in-core detector anti-reactivity, and the other, due to the change of water level respect to the infinitely reflected condition.

The cross sections of the different materials of the core, were checked by means of a calculation-experiment correlation obtained with more than 40 different configurations [5].

The insertion time may be changed up to 2.5 sec. without affecting results.

β, Λ: 0.0078 and 78 μsec., respectively.

By means of the pulsed neutron source and neutron noise techniques, a value for the ratio β/Λ of 100 sec^{-1} was obtained[6].

By using a 5 energy group perturbation theory, separate values for β and Λ were calculated[7], being the agreement between calculated and "measured" ratios, reasonably good, so that these values were adopted.

Void coefficient: -0.31\$/(%void).

As an extrapolation, this value for the void coefficient was obtained from an indirect measurement of α_V in another configuration, at room temperature, with the moderator partially replaced by Aluminium. The correction due to dispersion of Al was calculated with perturbation theory[8].

Temperature coefficient: -0.022\$/°C at 50 °C.

This coefficient was calculated with neutronic codes in a 5 energy group structure with R-Z geometry and without control-rods inserted[7].

Initial power: 10^{-3} watt.

This is an uncertainty but this error does not introduce changes in the results.

Inlet temperature: 20 °C.

U-Al alloy density: 3.12 g/cm^3.

Total peaking factor: 2.52

Fuel caloric capacity: 0.75 J/g°C at 40 °C

Collapse time of bubbles: 5×10^{-4} sec. (Zuber model, Nucleate boiling)

Expansion coefficient of fuel plates: 23.8×10^{-6} cm/cm°C

Fuel thermal conductivity: 1.65 watt/cm °C

RESULTS AND DISCUSSION

The results of the Paret calculation of the accident are shown in Fig. 4, 5, 6, and 7. The principal results are given in table 1.

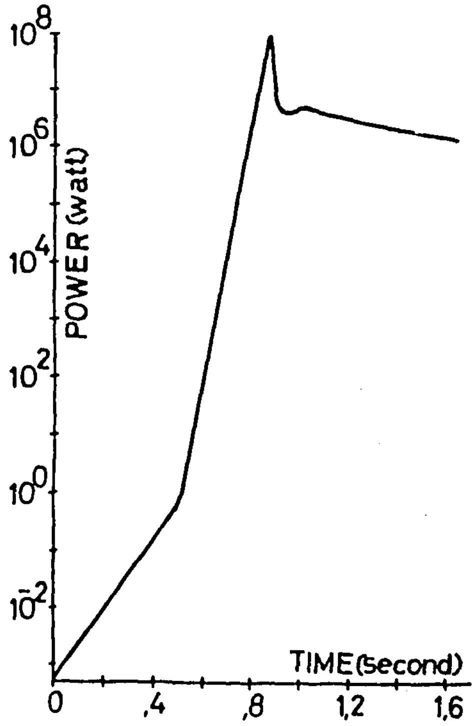

Fig. 4. Power vs. Time

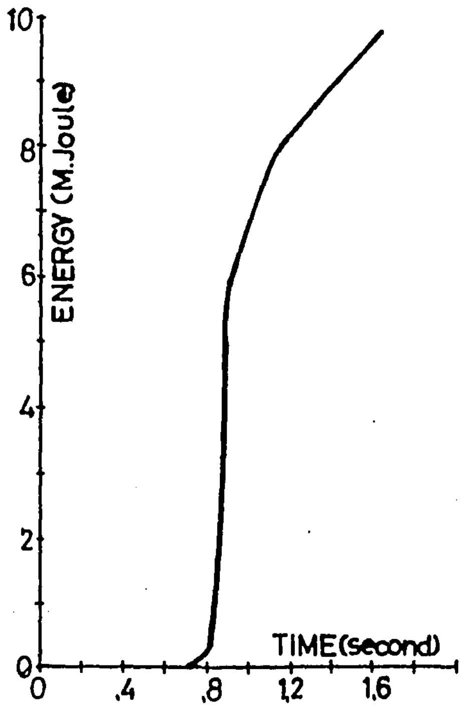

Fig. 5. Energy vs. Time

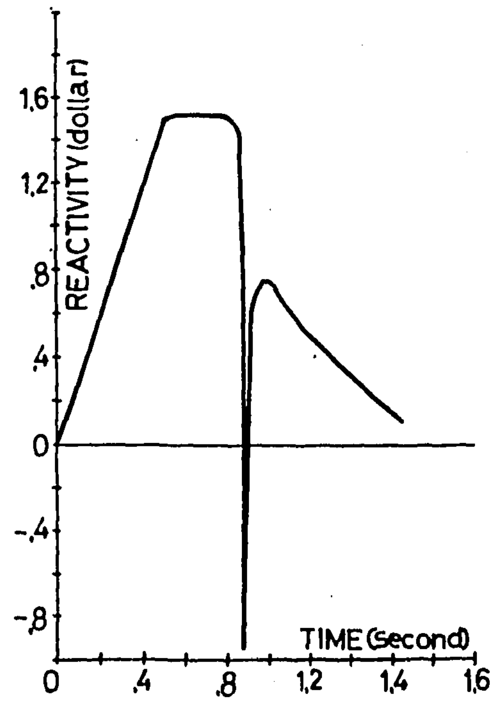

Fig. 6. Reactivity vs. Time

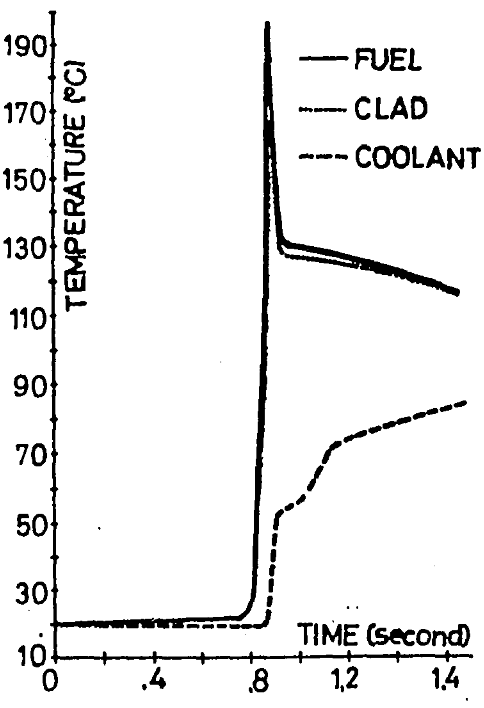

Fig. 7. Fuel, Clad and Coolant Temperature vs. Time

Table 1. Results obtained with Paret code of RA-2 Accident

Parameter	Maximum value	Values at t=1.65 sec.
Power (Mwatt)	198	2.34
Energy (MJoule)	4.8 ($t=t_p$)	9.82
Fuel temperature (°C)	197	116.8
Clad temperature (°C)	168	116.2
Coolant temperature (°C)		84.1

It can be seen that the peak power occured before the transient first second, and nucleate boiling is predicted in the hottest channels shortly after.

The peak temperature of center fuel and clad were far below melting point.

It is very probable that a delay time may have elapsed before the draining of water from the tank. If this was the case, the energy produced after the power peak may have been as important as the one produced before, or even greater than it.

Figure 8 shows a good agreement between Paret results and those obtained from the analytical model for n=3.

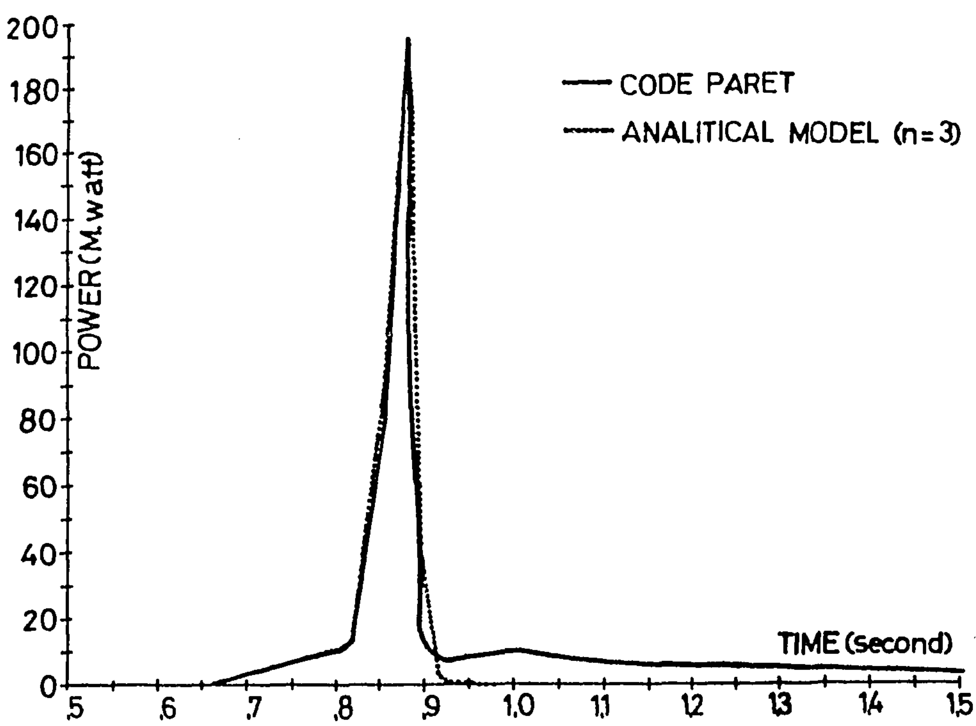

Fig. 8. Comparison between Paret and Analytical Model

This code was also used to predict the influence of transient on uncertanties in parameter values. The results are summarized in Table 2.

Table 2. Influence of transient on uncertainties in parameters

Parameter	Change	Relative change (%)					
		$P(t_p)$	$E(t_p)$	$T_f(t_p)$	$T_c(t_p)$	P(1.65 sec.)	E(1.65 sec.)
P_0	5(3)[a] w	0	0	0	0	----	----
	1(5) w	-48.	-42.	-20.	-20.	----	----
Λ/β	+10%	-23.	-2.0	-4.0	-1.0	----	----
α_V	+10%	-1.2	-0.5	-2.0	-2.0	----	----
α_T	+200%	-12.	0	0	0	-200	-200
$\$_0$	1.3	-52.	-7.	-20.	----	----	----
	1.7	+73.	+17.	+5.	----	----	----
Insertion reactivity time	2.5 s	0	0	0	0	-----	----
Bubble collapse time	+1000%	-19.	-19.	-38.	-24.	----	----
Fuel caloric capacity	+20%	+2.0	+5.3	+6.9	+5.4	----	----
Clad expansion coefficient	+400%	-11.	-2.3	-5.4	-5.5	----	----

[a]Read : 5×10^3

Only was considered self-limited transients.

The most important uncertainties are those in $\$_0$, and Λ^*. The α_T affects the quasi-equilibrium power.

Table 2 reproduces trends given by the analytical model.

COMPARISON WITH SPERT AND CABRI EXPERIMENTS AND WITH POST-ACCIDENT INSPECTIONS

Table 3 shows a comparison between neutronic parameters of the RA-2 accident configuration with the A-17/28 of SPERT I reactor and Si/23 of CABRI reactor.

Table 3. Comparison between neutronic parameters

Parameter	SPERT I[a] (A-17/28)	CABRI[b] (Si/23)	RA-2 (Conf. F)
β	7.6 (-3)*	7.8 (-3)*	7.8 (-3)*
Λ (sec.)	53.2 (-6)*	58.5 (-6)*	78.0 (-6)*
Λ/β (sec.)	7.0 (-3)*	7.5 (-3)*	10.0 (-3)
Channel Thickness (mm)	2.87	2.12	2.89
Metal/water ratio	0.79	0.79	0.605
Atomic ratio H/U	320	-----	370
Temperature coeficient (c/°C)	-0.670 (20 °C) -2.70 (95 °C)	-0.40 (30 °C) -1.10 (50 °C) -2.10 (70 °C)	-2.20 (50 °C) -4.40 (95 °C)
Global void coefficient ($/cm^3)	-4.6 (-4)	-2.9 (-3)	-4.2 (-4)
Maximum void coefficient ($/cm^3)	-9.3 (-4)	-4.1 (-3)	-10.0 (-4)

*calculated values   [a]From reference 9   [b]From reference 10

In the RA-2 reactor the neutron generation time is greater due to the existence of a graphite reflector. As the moderating ratio is better, the void coefficient is somewhat smaller. So that, for the analytical model, for the same reactivity insertion, there would be more energy released at peak instant. On the other hand, for peak power value, both effects are of opposite sign.

Table 4 shows a comparison between calculated (RA-2) and measured (SPERT I) values for the same reactivity insertion.

Table 4. Comparison between RA-2 and SPERT I values for the same reactivity insertion

Parameter	SPERT I[a] (A-17/28)	RA-2 (conf. F)
$P(t_p)$ (Mwatt)	400	197
$E(t_p)$ (MJoule)	8.0	4.8
T_c(max.) (°C)	180	168

[a]From reference 11

As has been explained before, from extrapolation of the SPERT I experiments, greater values than those calculated for the accident main parameters would have been expected. It is not a simple task to explain this discrepancy.

For instance, it is well known that for periods longer than 30 milisec., power peaks would not exist or would be low, while for periods shorter than 10 milisec., some FE would be partially destroyed, which has not been observed in the RA-2 transient[12].

The existence of a power peak follows from assertions of the operator and people standing near the reactor.

The calculated reactivity excess (1.5$) corresponds to an intermediate period value of 19 milisec..

DIFFERENCES BETWEEN LEU AND HEU CORES

Figure 9 and table 5 show the comparison between the calculation of RA-2 core with LEU and HEU fuels.

The calculations show that the general behavior of LEU and HEU cores during fast transients are quite similar. Reference [13] shows similar conclusions.

The LEU core shows a lower and narrower peak. The prompt Doppler feedback seems to play an important role. Aditionally, a higher void coefficient is expected due to a harder spectrum in LEU fuels, and this effect would give a smaller peak power and lower energy release. The difference in T_c is due to the change in the fuel caloric capacity, and thermal conductivity.

Table 5. Comparison between RA-2 core with HEU and LEU fuels

Parameter	HEU	LEU
$P(t_p)$ (Mwatt)	197	157
$E(t_p)$ (MJoule)	4.8	3.9
$T_c(max)$ (°C)	168	130

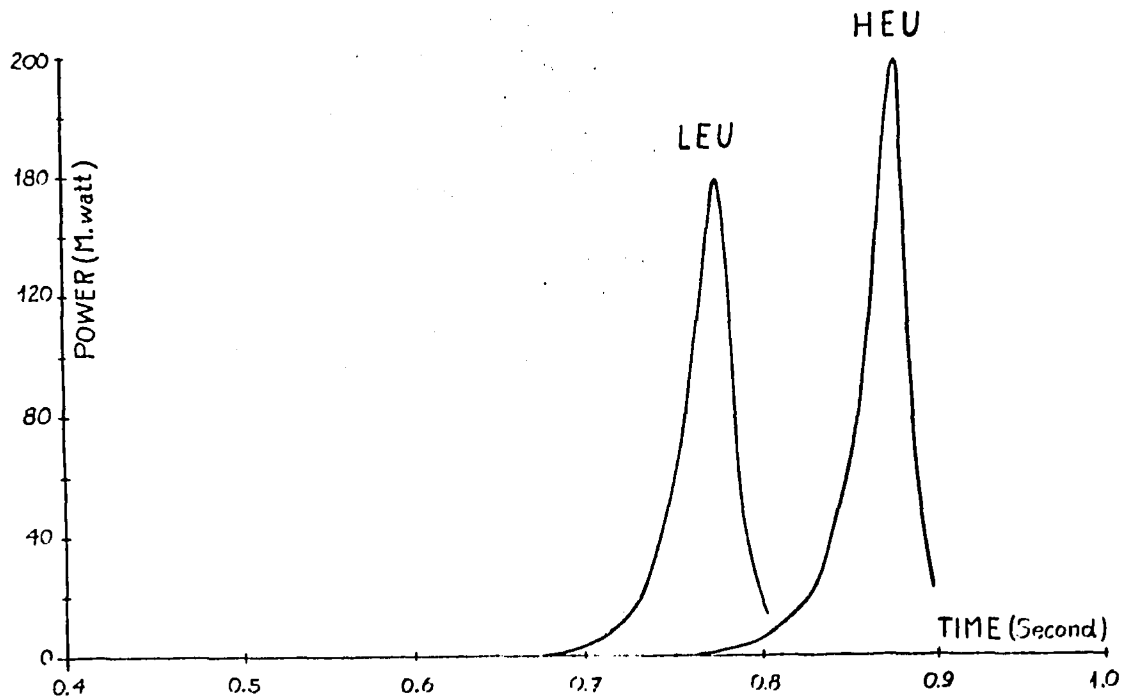

Fig. 9. Power vs. Time HEU and LEU RA-2 fuels

CONCLUSIONS

The results obtained through the present study are summarized as follows:

(1) The energy generated in the RA-2 accident, until the peak power instant ranges 5 and 9 MJoule. After the peak is reached due to the quasi-equilibrium power, its value may be increased in a similar quantity, so that during all the transient, the Total energy released may be estimated in a value between 10 and 15 MJoule.

(2) No fuel plate melting ocurred. The calculations performed validate this assertion.

(3) The Paret results agree with the analytical model trends.

(4) The consequences of a reactivity accident in a LEU core are similar or lower than those for a HEU core.

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