Power Transient in a CANDU Reactor

by

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PART B: INDUSTRIAL PROJECT

A project report submitted in partial fulfillment of the requirements for the degree of Master of Engineering Physics

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ATOMIC ENERGY OF CANADA LIMITED

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ABSTRACT

In this report, the effectiveness of a proposed Shutoff Rod System a CANDU Reactor was investigated. A full core simulation was done, to study the neutronic power transient following a change in coolant conditions.

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1. INTRODUCTION

Bruce-B is a Canadian nuclear reactor of the CANDU-PHW type. Namely, it is a pressurized heavy-water-cooled and moderated reactor, using natural uranium as fuel.

The proposed shutoff rod configuration for Bruce-B consists of 32 rods. According to a conservative philosophy, two out of these 32 rods must be assumed unavailable, at any time (due to incidental malfunction, testing, etc.).

Two sets of the two most important shutoff rods were suggested[1]. The important rods are defined such that when they are not available, the static negative reactivity of the remaining rods is a minimum.

With reference to Figure 1, the recommended sets are:

(a) Set 1

This results in a minimum static reactivity worth for the remaining 30 shutoff rods of -50.61 mk. The reactivity worth of all 32 shutoff rods is -68.59 mk.

(b) Set 2

This does not result in a minimum reactivity worth for the remaining 30 rods. However, there are two important reasons for which they could be considered as the two most important rods

(i) These two rods lie in the adjuster flattend region

(ii) For equilibrium fuelling conditions the hottest bundle powers occur along the traverse of these rods.

The purpose of this report is to investigate the effectiveness of the shutoff rod shutdown system in Bruce-B, with Set 1 unavailable.

For the case where the Set 2 is unavailable, a study was done by Ontario Hydro.

The effectiveness of the remaining 30 rods, with Set 1 unavailable, was investigated during the following assumed accident:

(i) The poison injection system unavailable

(ii) Loss-of-Coolant accident caused by a 40% break in the reactor inlet header with the reactor at equilibrium fuel conditions.

The above accident is considered as a severe one, and the reasons why are given in Section 2.

The reactor simulation model and the lattice parameters used in the analysis are given in Section 3.

A summary of the sumulation technique is given in Section 4. The CRNL CPC 6600 computer was used for the analysis.

The results of the study are given in Section 5, and the conclusions in Section 6.

2. CHOICE OF THE LIMITING ACCIDENT

Bruce-B will have two shutdown systems. The first system is the shutoff rods called SDS-1 for short. The second is the poison injection system, called SDS-2. They are both designated to shut the reactor down safely in the event of severe failures of major components, including those of the heat transport system.

According to a conservative[2] safety philosophy, "any one of the two shutdown safety systems will fail to perform its required function when called upon to counteract any single process system failure".

One consequence of the above restriction is that, in case of loss-of-coolant, due to a pipe rupture in the heat transport system, only one shutdown system must be efficient to ensure that fission products will not be released from the containment. To ensure this, it is sufficient to maintain fuel channel integrity. This in turn is assured, if fuel breakup is prevented.

The assumption of the unavailability of one of the two Bruce-B shutdown systems following a pipe break in the heat transport system defines a dual accident.[3]

Following a loss-of-coolant accident, the heat content of UO_2 fuel is raised rapidly to a certain level where the fuel breaks up. Consequently, the fuel sheath breaks up too. The fuel rod integrity is guaranteed by limiting the radially averaged stored specific energy at any axial location in the fuel during the power pulse below a conservative value[3] of 200 cal/gr of UO_2.

In the present study the poison injection system is assumed unavailable. The shutoff rods must keep the short-term energy input below the fuel breakup threshold. The effectiveness of the shutoff rod system was analyzed, for equilibrium fuel conditions where the assumed margin to fuel breakup is the smallest. The energy generated during the power pulse is primarily a function of the void holdup, the voiding transient, the trip signal delays, and the shutdown system characteristics. The void holdup depends on the fuel irradiation, and the boron concentration in the moderator. The holdup decreases as the fuel is irradiated and the boron concentration is decreased. A higher void holdup leads to larger energy generation during the power pulse. On the other hand, the larger delayed neutron fraction and negative fuel temperature feedback of fresh fuel compensate for the higher void holdup. This is the reason why the equilibrium fuel conditions were chosen, because they are the most stringent test of the shutoff rod systems effectiveness. The total void holdup was assumed to be 13.34 mk, including an uncertainty[4] of 1.4 mk. The atomic percentage of D_2O in the coolant was assumed to be 97.15%.

The voiding transient following a loss-of-coolant accident depends on the break location, break size, and reactor condition. The 100% break is defined as twice the flow area of the broken pipe. Breaks in the reactor inlet header lead to the fastest core voiding rates during the initial period of blowdown. The highest stored specific energy in the fuel results from a 40% break in the reactor inlet header with equilibrium fuel conditions[3].

In the case where the shutoff rod system is unavailable when a LOCA occurs, the poison injection system must limit the short term energy input below the fuel breakup threshold. Although the shutoff rod system trips sooner than the poison injection system, the faster shutdown rate of the poison results in a smaller energy input to the fuel.

From all the above, the conclusion is that the highest stored specific energy in the fuel results from a 40% break in the reactor inlet header, with equilibrium fuel and the poison injection system assumed unavailable.

3. REACTOR MODEL AND LATTICE PARAMETERS

The reactor core was simulated by a full core model, shown in Figure 1 and 2. The analysis was performed using the three-dimensional computer programme CERBERUS. The reactor core has two burn-up regions, an inner one of 208 channels, and an outer one of 272 channels. The average irradiations for the two regions are 1.85 m/kb and 1.55 m/kb respectively. The material properties for reflector, outer core and inner core are given in Table 1, as calculated by the POWDERPUFS (v) lattice parameter code.

The adjuster and shutoff rods were simulated by smearing their incremental parameters over a parallelpiped of cross-sectional area 28.575 x 49.53 cm^2. The incremental parameters were obtained by the SUPERCELL method, and they are given in Table 2.

In the case of the zone controllers, the absorber was assumed to be uniformly distributed throughout the length of the controller. The simulation was done by smearing their effect uniformly over the whole length of the compartment. The incremental SUPERCELL parameters used, are given in Table 2.

Six group delayed neutron fractions and their respective delay constants, average fast group and slow group velocities and effective prompt neutron lifetime, are given in Table 3, as calculated using the DN program[5]. The fast neutron average velocity was taken to be equal to $V_1 = 8.189 \times 10^6$ cm/sec. The fast flux is assumed to follow the 1/E behavior and the cutoff energy between the two groups is assumed to be 0.625 ev.

The 3-D full core simulation model contained a number of approximations concerning the exact locations of the fuel channels and shutoff rods. This was done in order to minimize computer cost and turn around time. The approximations were:

(i) Channels B6, B19, D4, D21, F02, F23, X6, X19, V4, V21, T2, T23, have been eliminated and replaced by channels A7, A18, G1, G24, H1, H24, Y7, Y18, R1, R24, S1, S24.

(ii) The shutoff rods at distance 38.0 cm were shifted by 2.5 cm away from the centre line.

(iii) The shutoff rods at distance 161.0 cm were shifted by 3.765 cm towards the centre line.

(iv) The shutoff rods at distance 43.0 cm were shifted by 2.6 cm towards the centre line.

(v) The shutoff rods at distance 203.0 cm were shifted by 3.765 cm away from the centre line.

The effects of these approximations were compared against an exact model, for steady state conditions, and the results showed that the approximations do not affect the calculations significantly. Table 4 contains a comparison of the data.

The atomic percentage of D_2O in the coolant was assumed to be 97.15%.

The total time delay from the moment of the break to the initiation of shutoff rod insertion was assumed to be 0.4 sec. This delay consists of time required to reach the trip level together with its associated signal and the time required to de-energize the clutch that releases the shutoff rods. For the shutoff rod insertion, the average performance curve was used, including 1-σ correction[6]. This is conservative, because the average curve is slow. The shutoff rod insertion vs time is shown in Figure 3.

The coolant voiding rate was based upon a 40% reactor inlet header break[7]. The coolant void fraction vs time curve, of Figure 4, was obtained by using data from RODFLOW (runs 21-R1H-25-ICF-37-100 and 21-RIH-40-ICF-37-100). The total void holdup was assumed to be 13.4 mk including an uncertainty of 1.4 mk.

4. SUMMARY OF THE SIMULATION TECHNIQUE

The study of the power excursion following the break in the reactor inlet header, was done at small time steps ΔT (see Table 5), with the dynamic code CERBERUS, for a total of 2.50 sec. This is the time needed for the shutoff rods to be fully inserted. After this time, the power excursion is almost terminated and the study was continued with bigger time steps ΔT until 300 sec. At this time, even the longest lived group of delayed neutrons has decayed away.

The preliminary data which is required before the simulation can proceed are:

(a) Reactor Core Simulation Model

Accurate representation of the reactor core and the reactivity devices has a determining effect on the results of the analysis, and the computational cost involved.

(b) Lattice Parameters

The lattice parameters are calculated for the homogeneous basic fuel types by the lattice physics code POWDERPUFS (P.P.V.). The incremental properties of the reactivity devices are obtained from the SUPERCELL code.

(c) Coolant Voiding Transient

For a given break of the P.H.T. system, the coolant voiding transient is obtained from the RODFLOW code. For each time step ΔT of the simulation, the new material properties are calculated with PPV, corresponding to the respective coolant density at this time.

(d) The Total Voiding Holdup, and the Coolant Purity

(e) The Time Delay of the Reactivity Mechanism

This delay time consists of two components:

(i) Time after the break until the required trip level is reached (with its associated signal).

(ii) Time required to de-energize the clutch that releases the shutoff rods.

(f) The Reactivity Service Insertion Rate

(g) Delayed Neutron Data

Using the above preliminary data, the computer simulations technique proceeds as follows:

Step 1: Corresponding to the steady-state coolant density the lattice parameters are calculated using the lattice code POWDERPUFS [OUTPUT = TAPE 21].

Step 2: The computer programme MATMAP using TAPE 21 as input, calculates the material properties of the reactivity devices, for the core geometry at this time [OUTPUT = TAPE 19].

Step 3: Using TAPE 19 as input to the dynamic code CERBERUS, a steady-state run is done which simulates the condition of the nuclear reactor, before the accident.

Step 4: This is the beginning of the power transient calculations. Step 1 is repeated for a coolant density p(ΔT) corresponding to a time ΔT after the accident.

Step 5: Corresponding to the new locations of the reactivity mechanisms, Step 2 is repeated.

Step 6: Using CERBERUS, a first flux shape calculation is done.

Step 7: The reactor point kinetics code KREND calculates the flux amplitude at time ΔT.

Stop 8: The final flux shape calculations are done, with CERBERUS.

For each new time step ΔT, STEP 4 to STEP 8 are being repeated.

5. RESULTS AND DISCUSSIONS

The results of the power transient are given in Tables 5, 6 and 7.

Table 5 gives, as a function of time, the values of the coolant void fraction, neutronic power amplitude, dynamic system reactivity, overall form factor, and the maximum bundle powers.

The total delayed neutron fraction was 5.68×10^{-3} (Table 3). The maximum value of dynamic reactivity during the power transient occured at 0.90 sec after the accident and it was 5.18 mk, so the transient was sub-prompt critical by about half a mk.

The steady state bundle power of Table 5 includes decay power, whereas, the variations during the transient refer to neutronic power. In order to obtain the total power transient, the decay power should be subtracted from the initial bundle power and the decay power transient then added to the neutronic power transient. The variations of neutronic power amplitude and dynamic reactivity are shown graphically in Figure 5 and 6 respectively, as a function of time, and shutoff rod insertion percentage.

The neutronic power amplitude values given in Table 5 represent average neutronic bundle power variations during the transient. For fuel break-up considerations the integrated energy generation is of importance. The time of integration is from 0 to 2.5 sec. This is the time needed for the shutoff rods to be fully inserted. After this time, the power excursion is almost terminated. The obtained value was:

$$E\ (2.5) = \int_{0}^{2.5} P dt = 5.21 \text{ f.p.s.}$$

The unit f.p.s. is the energy produced by the hottest bundle in one second at steady state full power.

Table 6 gives the time history of power in those fuel bundles which produced maximum power at some time during the transient. The integrated energy values for these bundles are given in Table 7. The integrated energy generation for the hottest bundle (Table 7) is:

> E (2.5) in the hottest bundle (V12/7) = 7.49 f.p.s. where the unit f.p.s. is the energy produced in one second in the hottest fuel bundle at steady state full reactor power. The variations of power in the hottest bundle are shown in Figure 5. The energy generation, as a function of time and shutoff rod movement is shown in Figure 7. Approximately 70% of the energy generated in the hottest bundle is produced before the shutoff rods reach the centre of the reactor core.

6. CONCLUSIONS

Three main conclusions can be drawn from the analysis of the power transient, in BRUCE-B during the dual-accident of loss-of-coolant from a 40% break in the reactor inlet header and the second shutdown system being unavailable.

(1) If one pair of the two most important shutoff rods of the proposed SDS-1 system is unavailable then the

Average Bundle Integrated Power is 5.21 f.p.s.
Hottest Bundle Integrated Power is 7.49 f.p.s.

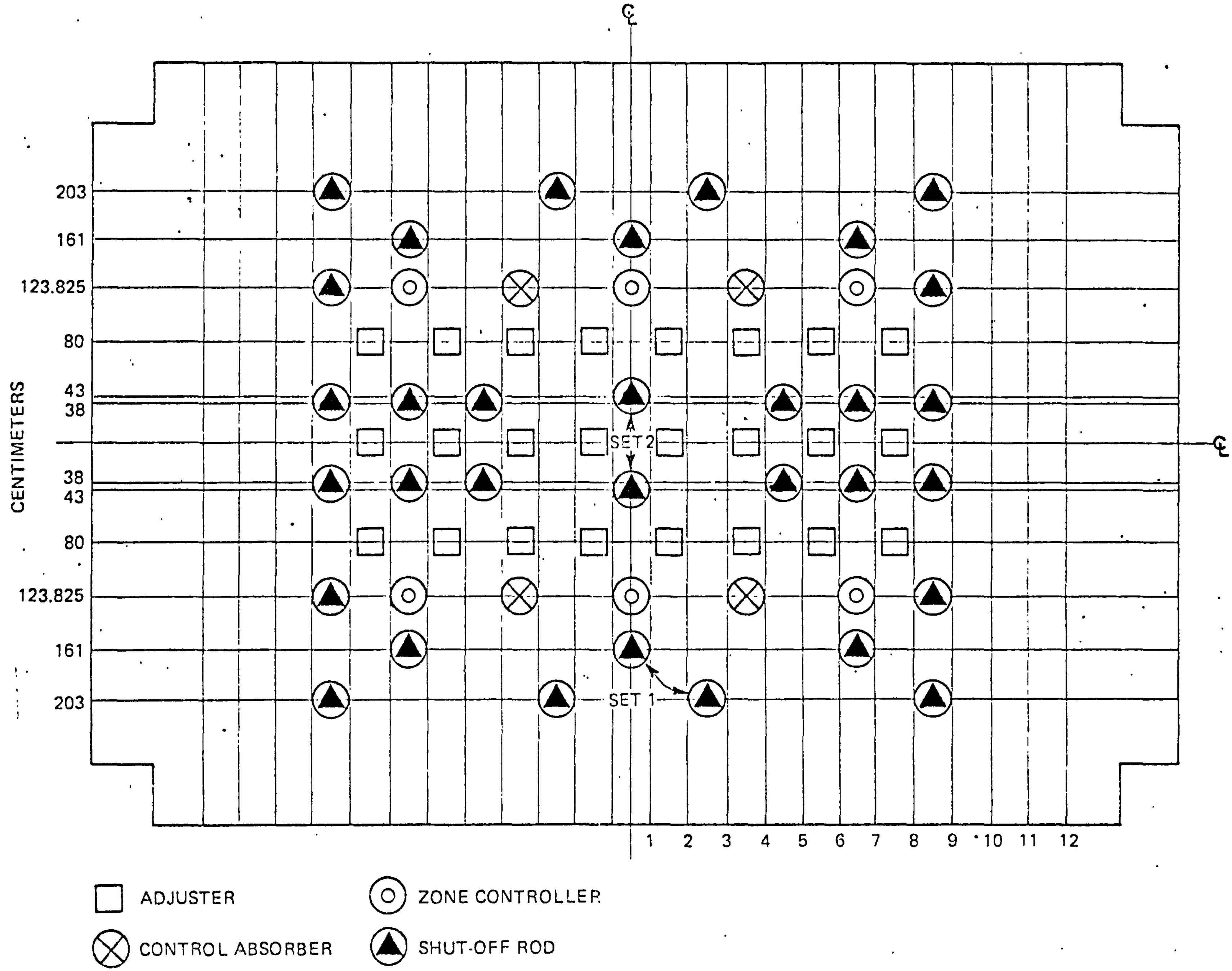
(2) The hottest bundles are located at the bottom of the reactor core, where the shutoff rods reach last.

(3) The main factors influencing the power excursion are the void holdup, the coolant voiding transient, the trip signal delay timer and the shutdown system characteristics.

For the postulated accident, the hottest and the average bundle powers, can be reduced, by reducing the shutoff rod total delay time, or by increasing the reactivity depth of the shutoff rod system.

7. REFERENCES

References 1 to 7, O.A. Trojan (AECL), private communication.



℄
CENTIMETERS
203
161
123.825
80
43
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38
43
80
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161
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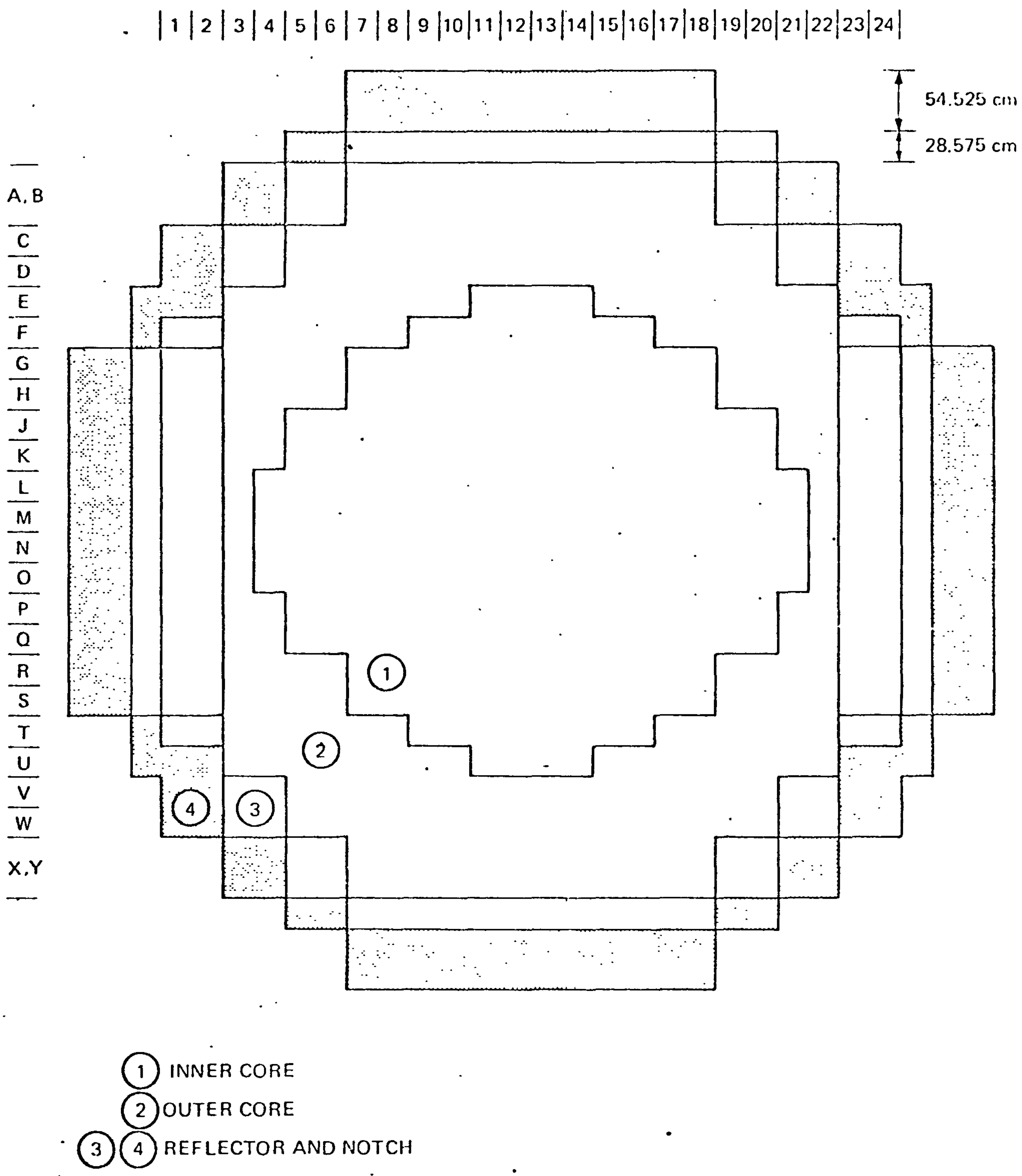
SET 2
SET 1
1
2
3
4
5
6
7
8
9
10
11
12
ADJUSTER
ZONE CONTROLLER
CONTROL ABSORBER
SHUT-OFF ROD

FIGURE 2 BRUCE B REACTOR CORE MODEL



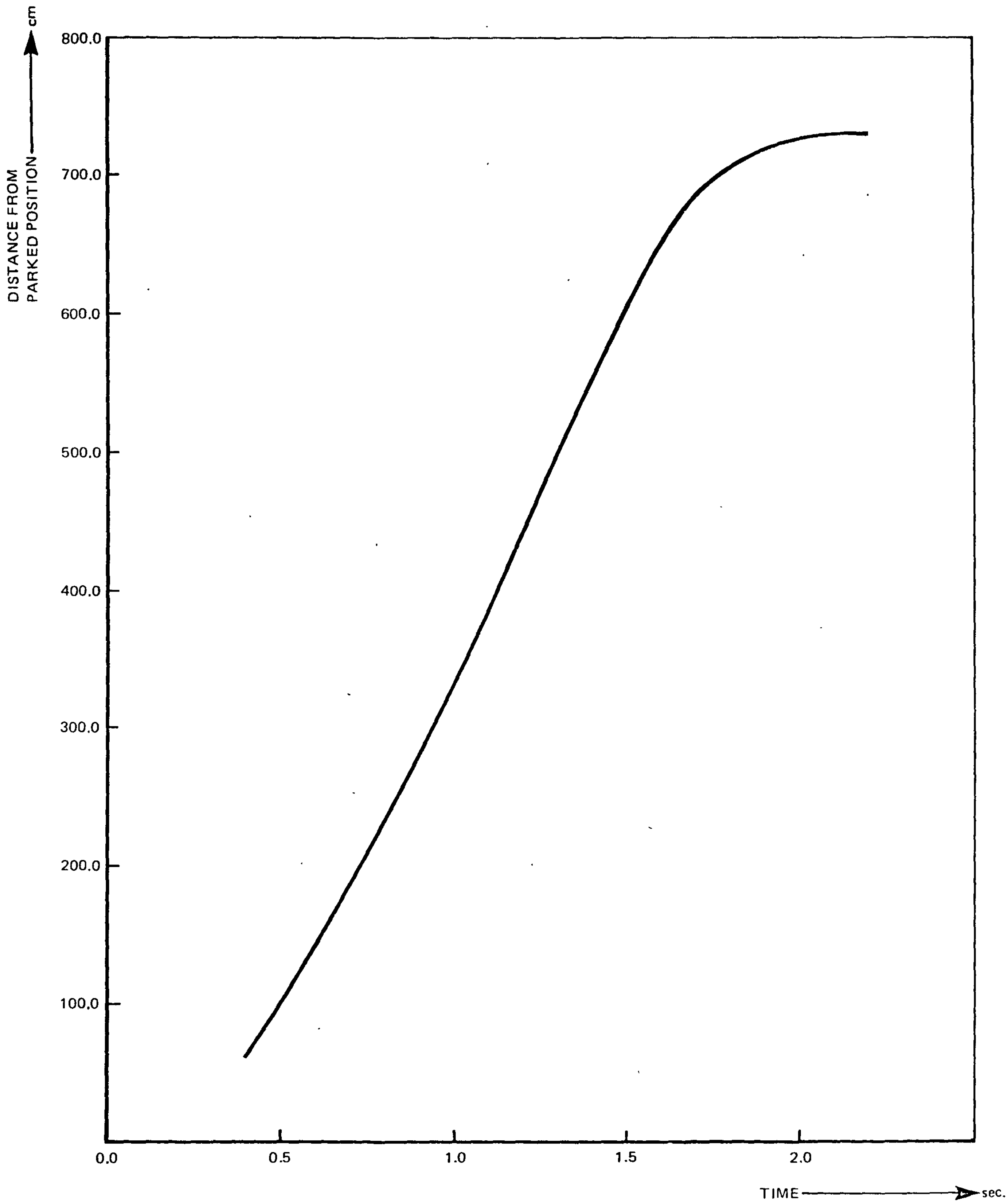


FIGURE 3 MOVEMENT OF S.O.R. VS TIME

FIGURE 4 COOLANT VOID FRACTION VS. TIME

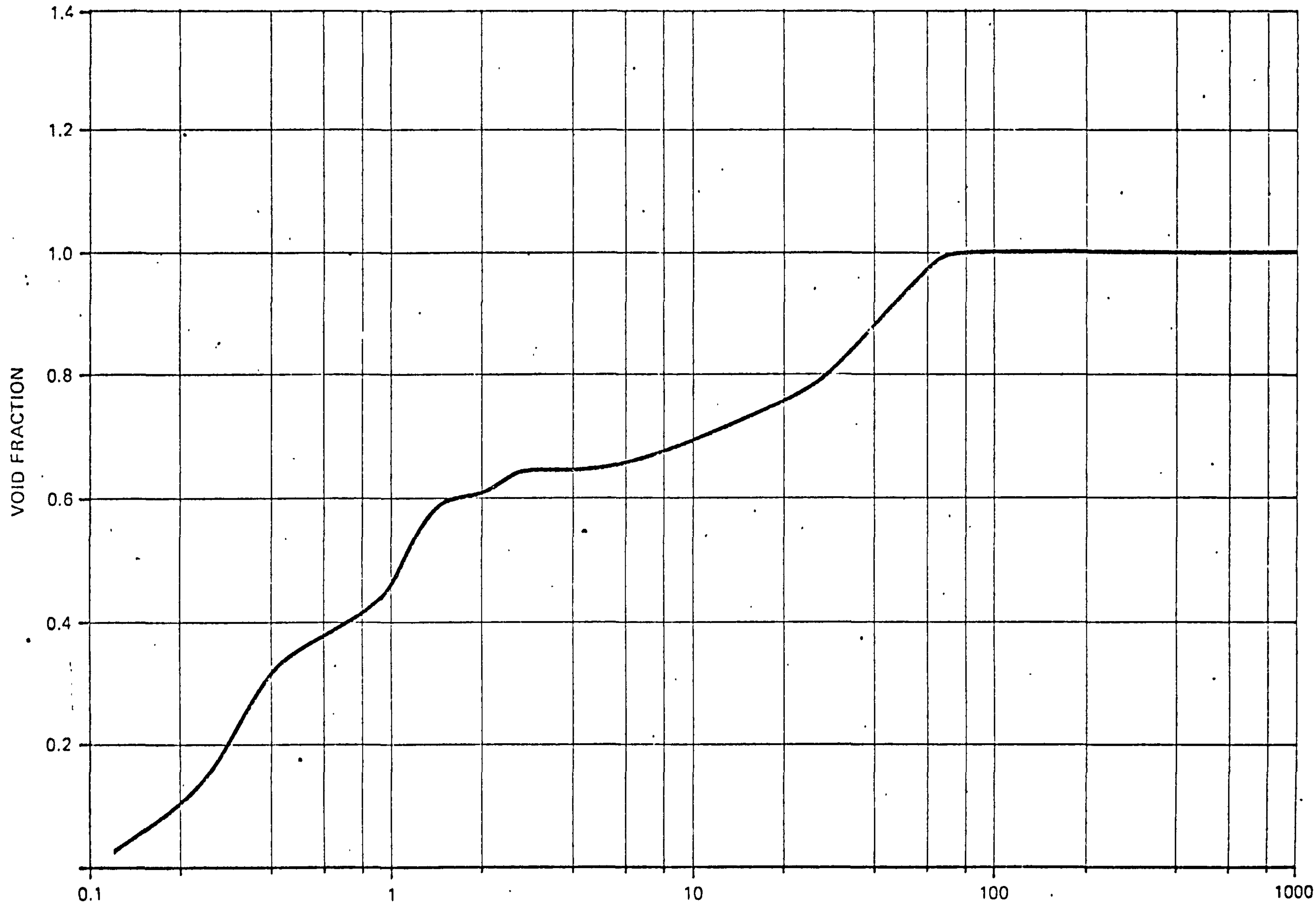


FIGURE 5 NEUTRONIC POWER AMPLITUDE VS. TIME

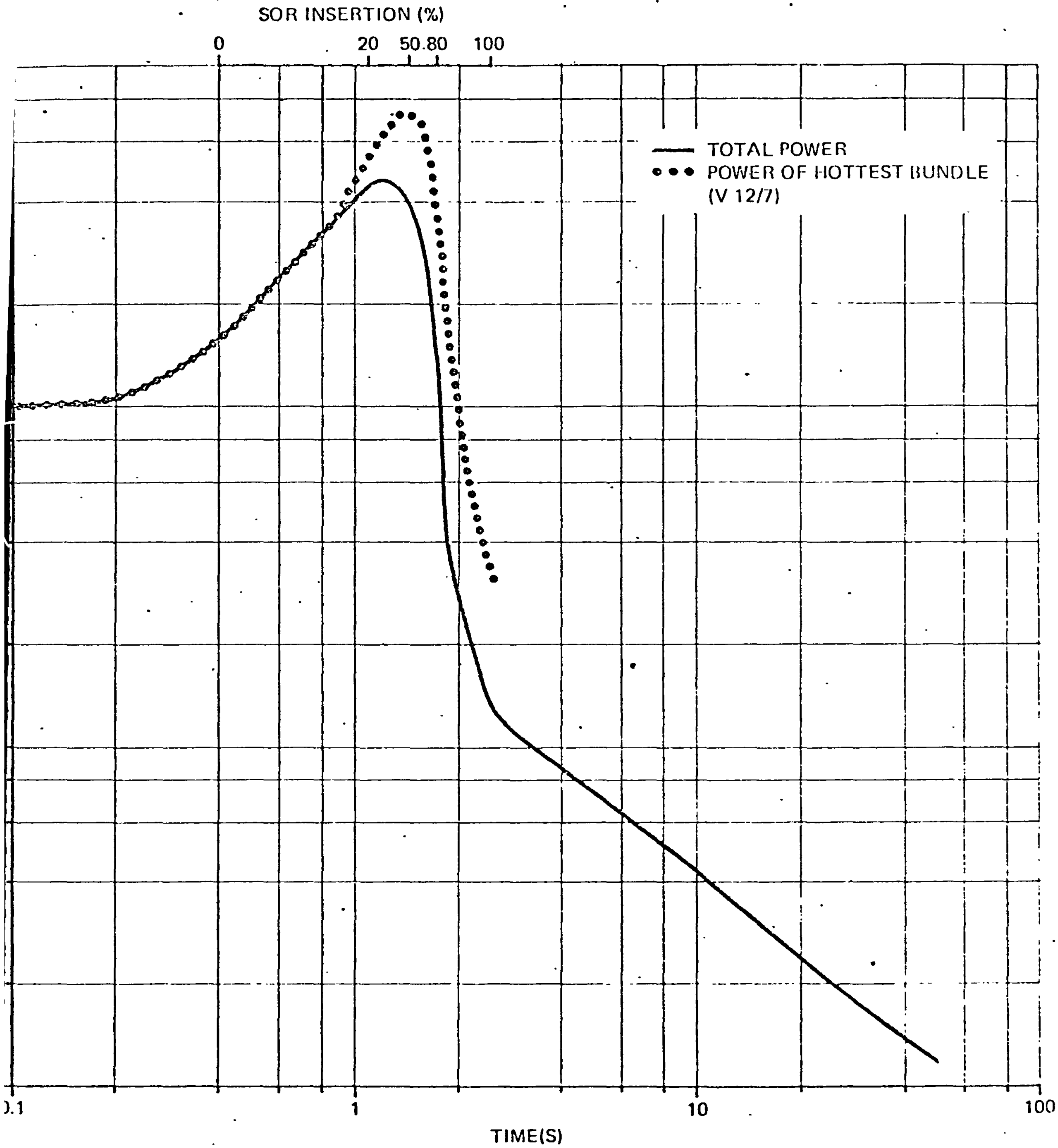


FIGURE 6 DYNAMIC REACTIVITY VS. TIME

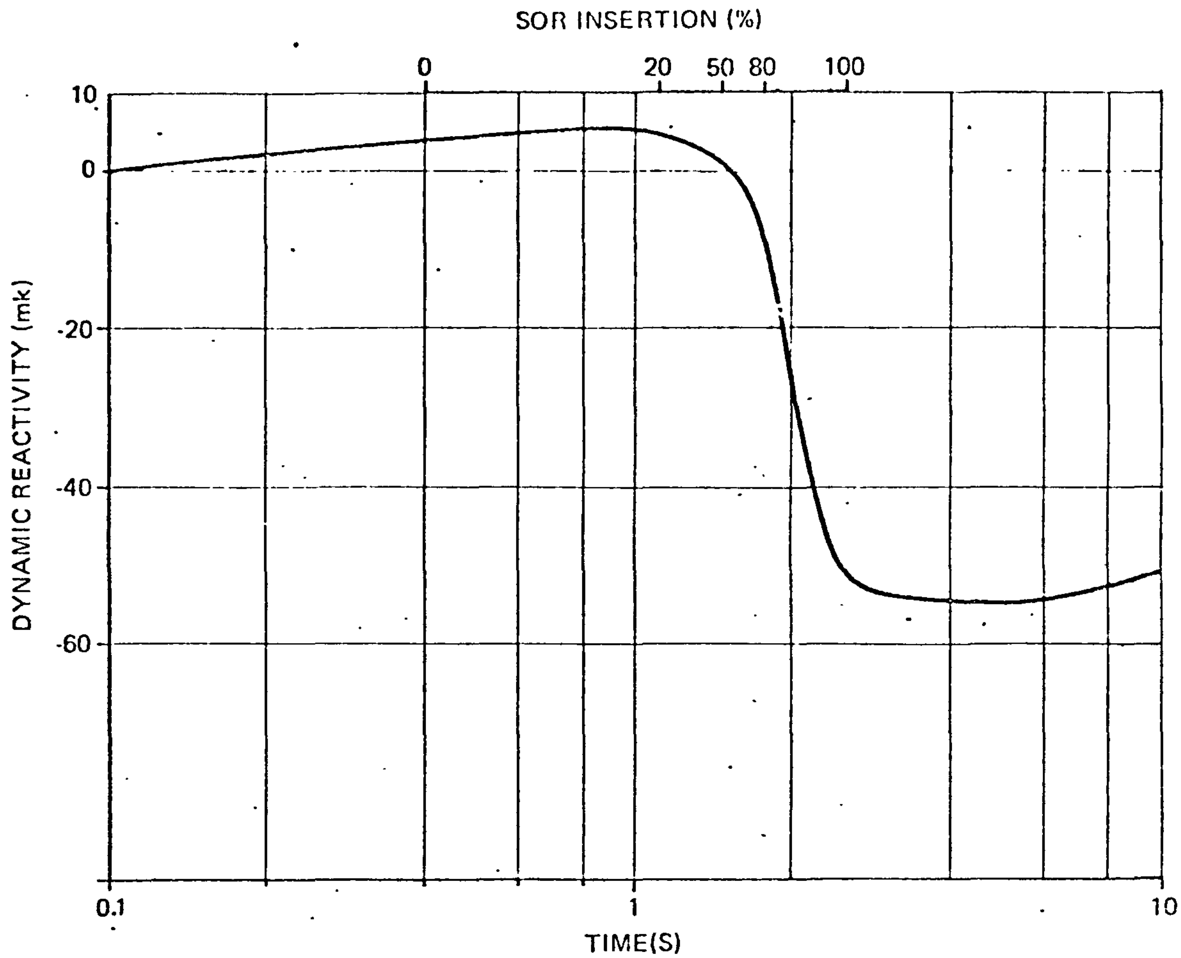
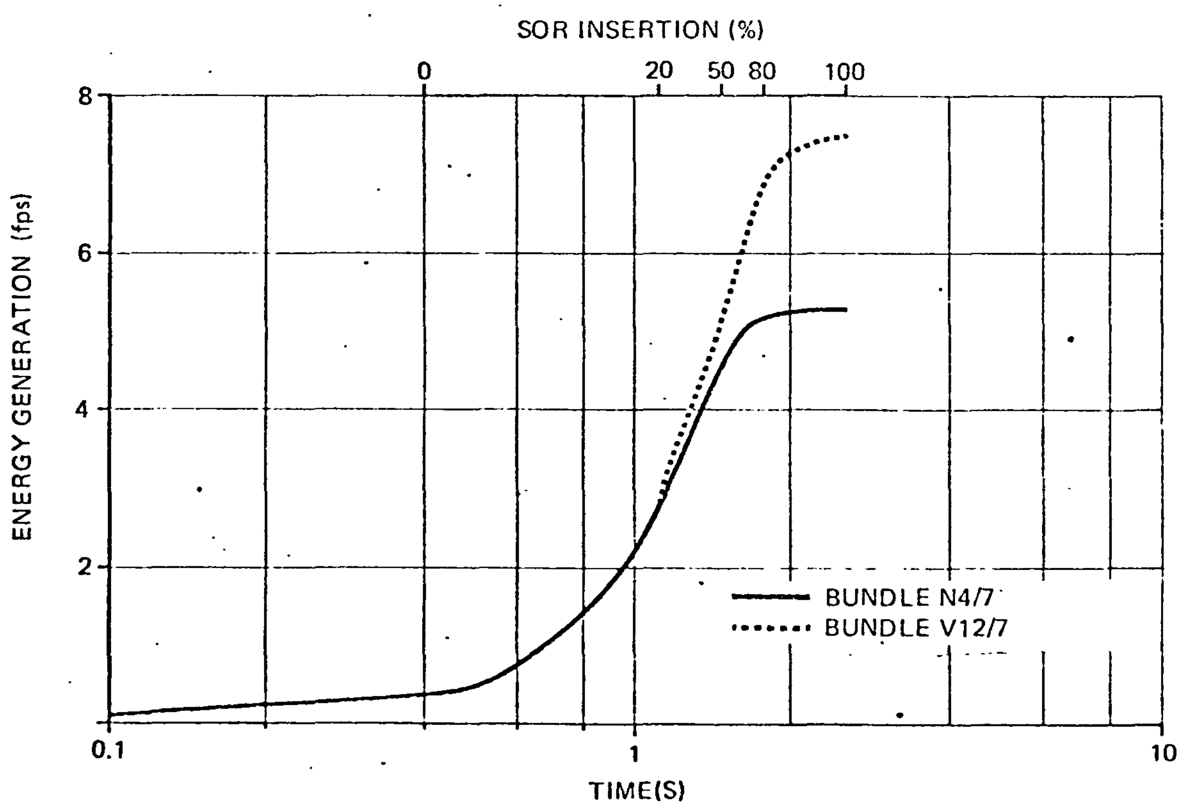


FIGURE 7 GENERATION IN FUEL BUNDLES VS. TIME



<u>TABLE 1</u>

Material Properties for Bruce B							
Material Type	D_1 cm	D_2 cm	Σ_a^1 cm^{-1}	Σ_a^2 cm^{-1}	$\nu\Sigma_f^2$ cm^{-1}	Σ_R^1 cm^{-1}	H
Reflector	1.32282	0.88554	1.0-11	8.22253-5	0.0	1.00801-2	0.0
Outer Core 1.55 n/kb	1.27895	0.949198	7.63646-4	3.99594-3	4.64384-3	7.36032-3	2.40309-1
Inner Core 1.85 n/kb	1.27895	0.949171	7.63235-4	4.02027-3	4.62367-3	7.36073-3	2.37863-1

TABLE 2

Incremental "SUPERCELL" Properties For Reactivity Devices

Reactivity Device Type	$\Delta\Sigma_{tR}^{1}$ cm^{-1}	$\Delta\Sigma_{tR}^{1}$ cm^{-1}	$\Delta\Sigma_{a}^{1}$ cm^{-1}	$\Delta\Sigma_{a}^{2}$ cm^{-1}	$\Delta(r\Sigma_{t})^{2}$ cm^{-1}	$\Delta\Sigma_{R}^{1}$ cm^{-1}	ΔH
Adjuster Rod A	0.0	-1.05-4	0.27-4	0.66-3	0.485-4	0.0	0.0
Adjuster Rod B	0.0	-1.25-4	0.3013-4	0.72-3	0.580-4	0.0	0.0
Adjuster Rod C	0.0	-0.79-4	0.2237-4	0.576-3	0.380-4	0.0	0.0
Adjuster Rod D	0.0	-0.68-4	0.2063-4	0.542-3	0.330-4	0.0	0.0
Zone Controllers	-0.67484-2	1.7856-2	0.1205-4	0.3329-3	0.049-3	0.1128-3	0.0
Shutoff Rod	0.0	-0.000686	5.96215-4	4.29302-3	0.5168-3	0.0	0.0

TABLE 3

Delayed Neutron Parameters

Group	β	
1	0.00024654	0.0009059
2	0.00116961	0.03155520
3	0.00103556	0.12146920
4	0.00226109	0.31880969
5	0.00077333	1.38876030
6	0.00019347	3.78138520

Total	β	0.0056796
	ℓ^*	0.89302 ms
	V_1	8.189×10^6 cm/s
	V_2	2.7522×10^5 cm/s

TABLE 4

Comparison of the Reactor Model Used With "Exact" Model at Steady State

	This Model*	"Exact" Model*
Maximum Bundle Power (KW)	712	732
Maximum Channel Power (MW)	6.44	6.45
Axial Form Factor	0.753	0.735
Radial Form Factor	0.872	0.870
Overall Form Factor	0.657	0.640
Reactivity Worth of all SOR (mk)	67.59	68.59
Reactivity Worth of SOR With Set 1 missing (mk)	49.83	50.61
Reactivity Worth of SOR with Set 2 missing (mk)	51.45	52.04
Reactivity Worth of all adjuster rods (mk)	17.66	17.66

* The material properties used correspond to a coolant purity of 99.722%.

Variations of Void Fraction, Neutronic Power Amplitude, Dynamic Reactivity, Form Factor and Maximum Bundle Powers

Time (sec)	Void Fraction	Neutronic Power Amplitude	Dynamic Reactivity (mk)	Overall F-Factor	Max. Bundle Power (kw)	
					Value	Location
0.07	0.0	1.000	0.0	0.6367	734.9	N4/7
0.12	0.023	1.012	0.306	0.6367	743.9	N4/7
0.22	0.118	1.090	1.459	0.6367	848.5	N4/7
0.35	0.275	1.368	3.475	0.6369	1005.4	N4/7
0.40	0.319	1.540	4.059	0.6370	1131.4	N4/7
0.90	0.428	3.709	5.177	0.6231	2785.9	V12/7
1.03	0.460	4.337	4.624	0.5748	3538.1	V12/7
1.10	0.490	4.563	4.321	0.5363	3981.5	V12/7
1.16	0.516	4.668	4.003	0.4988	4379.9	V12/7
1.22	0.540	4.700	3.741	0.4625	4755.5	V12/7
1.28	0.560	4.659	3.411	0.4289	5083.9	V12/7
1.34	0.574	4.534	2.952	0.3978	5334.7	V12/7
1.39	0.584	4.332	2.305	0.3706	5470.4	V12/7
1.45	0.592	3.981	1.628	0.3423	5442.9	V12/7
1.50	0.595	3.600	0.620	0.3177	5304.2	V12/7
1.55	0.597	3.131	-0.547	0.2935	4992.6	V12/7
1.60	0.599	2.617	-1.971	0.2704	4529.9	V12/7
1.65	0.599	2.096	-3.770	0.2489	3940.6	V12/7
1.70	0.599	1.604	-6.050	0.2281	3290.4	W12/7
1.75	0.599	1.176	-9.013	0.2095	2625.8	W12/7
1.81	0.599	0.792	-12.841	0.1945	1906.2	W12/7
1.86	0.599	0.59	-18.435	0.1844	1419.4	W12/7
1.99	0.605	0.275	-31.409	0.1911	672.9	W12/7
2.05	0.612	0.202	-43.480	0.2141	441.8	W13/8
2.50	0.640	0.125	-54.760	0.3014	194.5	V14/9
5.0	0.656	0.069	-56.740	0.3132	103.3	V14/9
10.0	0.687	0.040	-52.980	0.3282	57.4	V14/9
30.0	0.812	0.016	-53.630	0.3410	21.8	V14/9
50.0	0.937	0.011	-51.230	0.3478	14.4	V14/9
70.0	1.0	0.008	-49.750	0.3519	10.8	V14/9
100.0	1.0	0.006	-49.650	0.3527	7.9	V14/9
200.0	1.0	0.004	-49.630	0.3555	5.8	V14/9
300.0	1.0	0.003	-49.690	0.3565	5.2	V14/9

TABLE 6

Time History of Power in Those Bundles which
Produced Maximum Power at Some Time During the Transient

T	Bundle Powers Normalized to Maximum Bundle Powers At Steady State, At Locations				
(sec)	N4/7	V12/7	W12/7	W13/8	V14/9
0.07	1.00*	0.99	0.97	0.92	0.89
0.12	1.02	1.00	0.98	0.93	0.89
0.22	1.15	1.15	1.12	1.06	1.02
0.35	1.37	1.36	1.32	1.26	1.21
0.40	1.54	1.53	1.49	1.42	1.36
0.90	3.77	3.79	3.68	3.49	3.36
1.03	4.63	4.81	4.67	4.45	4.27
1.10	5.01	5.42	5.27	5.00	4.80
1.16	5.27	5.96	5.80	5.52	5.28
1.22	5.41	6.47	6.31	5.99	5.72
1.28	5.43	6.92	6.76	6.42	6.11
1.34	5.30	7.26	7.11	6.75	6.40
1.39	5.02	7.44	7.30	6.94	6.56
1.45	4.49	7.41	7.29	6.92	6.52
1.50	3.87	7.22	7.12	6.76	6.34
1.55	3.07	6.79	6.73	6.39	5.96
1.60	2.17	6.16	6.13	5.82	5.40
1.65	1.15	5.36	5.36	5.08	4.69
1.70	0.68	4.45	4.48	4.25	3.24
1.75	0.41	3.53	3.57	3.39	3.09
1.81	0.25	2.54	2.59	2.47	2.23
1.86	0.18	1.72	1.93	1.85	1.66
1.99	0.11	0.85	0.92	0.89	0.82
2.05	0.10	0.52	0.59	0.60	0.56
2.50	0.08	0.15	0.17	0.22	0.26

*This is the hottest bundle at steady state, full power. The energy production in this bundle in one second is one fps, as defined in Table 7.

Integrated Energy Generation in Various Fuel Bundles and SOR Insertion
at Various Times During the Initial 2.50 Sec.

Time (sec)	Energy Production in Bundles at Locations (fps)* (4,13,7) N4-7	(12,21,7) V12-7	(12,22,7) W12-7	(13,22,8) W13-8	(14,21,9) V14-9	Shutoff Rod Insertion: Distance from Horizontal Axis (Lattice Pitches)	Percentage of SOR's Within Reactor Core
0	0	0	0	0	0	Parked 15.24 cm	0
0.07	0.07	0.07	0.07	0.06	0.06	Above the	0
0.12	0.12	0.12	0.12	0.11	0.11	Calandria	0
0.22	0.23	0.23	0.22	0.21	0.20		0
0.35	0.39	0.39	0.38	0.36	0.35		0
0.40	0.47	0.46	0.45	0.43	0.41		0
0.90	1.79	1.79	1.74	1.66	1.59	12	5
1.03	2.34	2.35	2.28	2.17	2.09	10	15
1.10	2.68	2.71	2.63	2.50	2.41	9	20
1.16	2.98	3.05	2.97	2.82	2.71	8	25
1.22	3.30	3.43	3.33	3.16	3.04	7	30
1.28	3.63	3.83	3.72	3.54	3.40	6	35
1.34	3.95	4.25	4.14	3.93	3.77	5	40
1.39	4.21	4.62	4.50	4.27	4.10	4	45
1.45	4.50	5.07	4.94	4.69	4.49	3	50
1.50	4.70	5.43	5.30	5.03	4.81	2	55
1.55	4.88	5.78	5.64	5.36	5.12	1	60
1.60	5.01	6.11	5.96	5.66	5.40	0	65
1.65	5.09	6.39	6.25	5.94	5.65	-1	70
1.70	5.14	6.64	6.50	6.17	5.85	-2	75
1.75	5.17	6.84	6.70	6.36	6.01	-3	80
1.81	5.19	7.02	6.88	6.54	6.17	-4	85
1.86	5.20	7.13	6.99	6.65	6.27	-5	90
1.99	5.21	7.30	7.18	6.82	6.43	-7	100
2.05	5.22	7.34	7.23	6.87	6.47	-8	100
2.50	5.26	7.49	7.40	7.05	6.66	-10	100

*The unit fps here is defined as the energy produced by the hottest bundle in one second at steady state full power.