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BURN-UP ASSESSMENTS FOR A LAND BASED POWER REACTOR

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by

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1. INTRODUCTION

One of the main purposes of the Dragon Reactor Experiment is to test fuel elements suitable for a land based power reactor with the same principle features as the Dragon Reactor itself. In order to incorporate some fuel elements into the first charge which are realistic simulations of a power reactor fuel element, the performance of a 1000 MW(thermal) power reactor as a function of power density, fuel loading and thorium to uranium ratio was studied.

This report deals only with batch loadings. A comparison with other investigations which include recycling and reshuffling is difficult.

2. MAIN FEATURES OF PROGRAMME USED (FEVER)

FEVER [1] is a burn-up code written by General Atomic for the IBM 7090. The source programme is written in FORTRAN II. The programme allows max. 150 mesh points in up to 20 regions with max. 4 energy groups. Slowing down may occur only to the next adjacent group. The fission source is in the highest energy group only.

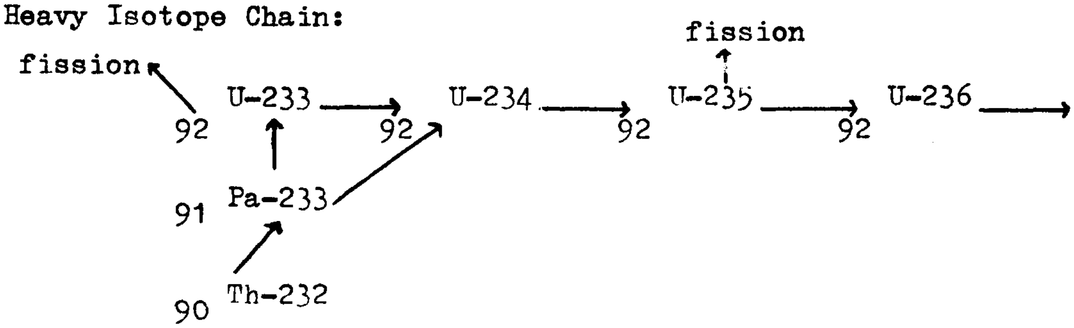
The programme (object deck) is read into the machine followed by the library data and input data for certain cases. Then the actual calculation (execution of programme) is performed. Fig. 1 shows the flow diagram in a simplified version. Some options for additional routines like control poison adjustments, xenon overriding, hot max. and cold shutdown k_{eff} are not shown on the diagram. A short description of the important block follows.

2.1 Library Data

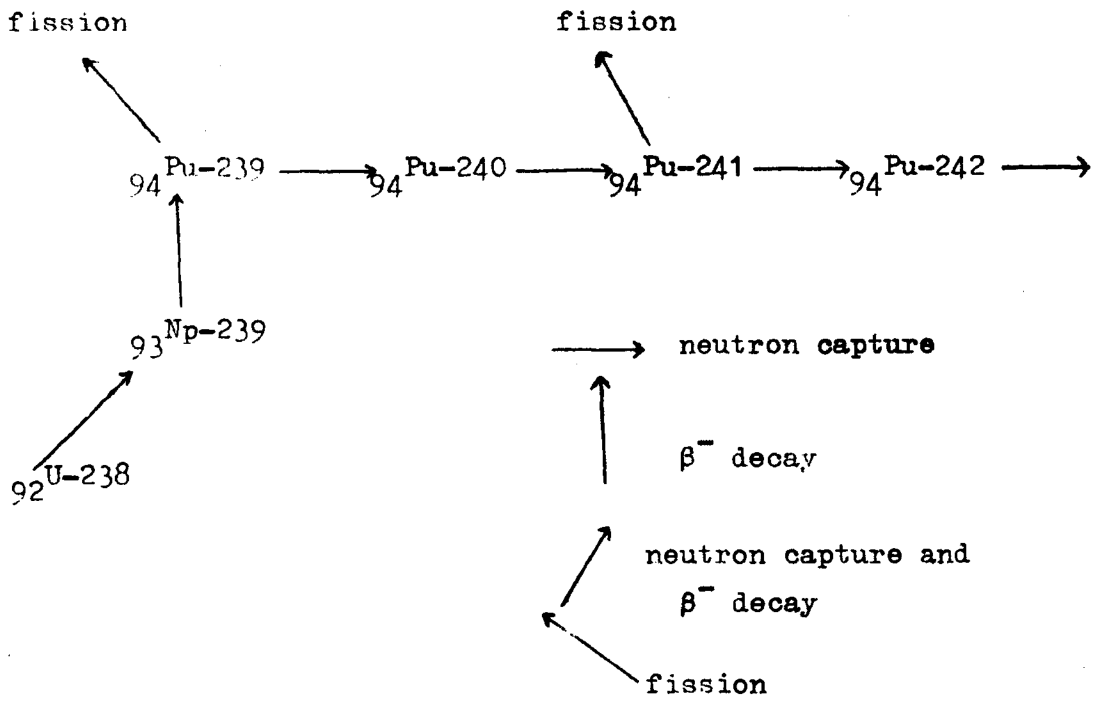
The library contains few group nuclear data for a set of materials (maximum total 25).

Materials which are Incorporated:

(i) 1st Heavy Isotope Chain:



(ii) 2nd Heavy Isotope Chain:



(iii) Non-saturating fission product aggregate

(iv) Xe-135 and Sm-149

(v) special materials: other saturating fission products or aggregate and homogeneously distributed burnable poison. (Max. 10 elements)

(vi) Lumped poisons: Nuclides with a concentration dependent self-shielding factor (max. 3 elements)

(vii) Control poisons

(viii) Non-burnable materials: moderator, coolant, structural materials (max. 10).

<u>Following Data are given:</u>

	$\nu \cdot \sigma_{fission}$	(microscopic fission cross section)
	σ_{tr}	(transport cross section)
	σ_a	(absorption cross section)
	σ_{out}	(scattering out from group $i \longrightarrow i + 1$)
	ν	(Neutrons per fission)

Densities

fission yields for the heavy isotope chains resulting from U-233, U-235, Pu-239 and Pu-241 and capture factors.

In addition some convergence limits and other control numbers are in the library.

2.2 Input Data:

For each case one has to specify:

(i) Control Numbers, e.g.:

number of regions, trigger for slab or cylindrical geometry, option for by-passing the burn-up branch, xenon overriding, control poison adjustment, computation of hot max. and cold shutdown k_{eff}, option to read in new library data and punching card of selected results, geometric dimension of reactor, mesh point.

(ii) Few group albedos for left and right side.

(iii) Region and Material Characterising Numbers:

Geometric information (length, co-ordinate), transverse few group buckling number and type of materials, their concentrations, self-shielding factors and block numbers, (identification number of nuclear data in library). For lumped poisons the coefficient of the power serie for the self-shielding factor and geometrical data are supplied. Fission yields for Xe-135, Sm-149, non-saturating fission product and special materials have to be specified. When control poison is used information about the concentration limits and adjustment scheme must be stated. (Since FEVER is only one-dimensional the withdrawal and insertion of the rods is simulated by varying the atomic concentration.)

(iv) Depletion Data:

Power of reactor
Large burn-up time step (time after new diffusion calculation is performed)
Small time step (for calculating the depletion)
Xenon removal constant
Fission per Watts/sec
Criterion for stopping calculation (e.g., when $k_{eff} < 1$).

2.3 Few Group and Diffusion Calculation:

With this information (library, input data) and some internal constants which are supplied in the programme (Atom number, π, Avogadro's number, decay constants) the programme is ready to calculate. This block is also entered after each large time step.

Few group constants are calculated by:

$$\sum_i(t) = N(t) \cdot \sigma_i \qquad i....1 \text{ up to } 4$$

The few group microscopic cross section are given in the library and do not change with burn-up. Therefore the flux spectrum change during burn-up is not taken into account.

The system of one-dimensional few group diffusion equations is solved by an iteration process which takes into account:

<u>Losses:</u>

Leakage	$(\mathrm{div}\ D_i\ \mathrm{grad}\ \phi_i)$
Group absorption	$(\Sigma_{ai}\ \phi_i)$
Transversal leakage	$(D_i\ B_i^2\ \phi_i)$
Scattering out losses from group i	$(\Sigma_{si}\ \phi_i)$.

<u>Gains:</u>

Fission neutrons (only highest energy) caused by fissions by all group neutrons:

$$\frac{1}{\lambda} \sum_{i=1}^{4} \nu\, \Sigma_{fi}\ \phi_i$$

Neutrons slowed down from the next adjacent (higher) energy groups

$$\left(\Sigma_{s_{i-1}}\ \phi_{i-1}\right)$$

Denotations used:

i=1	highest energy group		
i=4	lowest energy group		
ϕ_i	group flux		
D_i	diffusion coefficient		
B_i^2	transversal buckling		
Σ	macroscopic cross section for	a	absorption
		s	scattering out (removal)
		f	fission
ν	neutrons per fission		

As a result of this block one gets the relative flux, power and effective multiplication factor.

The next blocks compute and print out the following results:

(i) Atomic densities of each region and material with few group self-shielding factors.

(ii) Actual weight of each material of the two heavy isotope chains for each region and the total reactor.

(iii) Regional few group constants in the diffusion equation above (D, B, Σ).

(iv) K_{eff}.

(v) Power and group fluxes at each mesh point.

(vi) Absolute few group fluxes for each region and for the whole core.

Neutron Balance:

Neutron balance per source neutron and fractional absorption and productions over the core is calculated.

Burn-up:

The depletion build-up for each region during a large time step is calculated. Only the amplitude but not the spectrum is changed during this step. This large step is sub-divided into smaller time steps in which the actual change in the regional material concentrations is calculated. The amplitude of the flux is readjusted after each small step.

(i) Heavy Isotope Chain:

The equation for each region is a time dependent differential equation which can be very easily derived from the chain diagram. They are represented by Finite Difference Equations, where Δt is a small time step.

For Pa-233, Np-239 the analytical solution is taken.

(ii) Xe-135 and Sm-149:

For these materials the standard equilibrium concentration is calculated because the saturation is searched very quickly compared to the usual time steps.

(iii) Special Materials:

The rate of concentration change = production by fission + capture in preceding nuclide - absorption.

Analytical solutions can be found assuming that the concentrations, fluxes do not change during this small time step.

(iv) Lumped Poisons:

Only absorption but with concentration dependent self-shielding factor, is taken into account.

This block calculates the conversion ratio which is defined by:

$$C = \frac{\text{Rate of production of fissionable material}}{\text{Rate of destruction of fissionable material}}$$

This is done in each region and also averaged over the reactor. In addition the relative contribution of fission caused by U-233, U-235, Pu-239, Pu-241 is calculated.

Final Procedure:

After this block the reactor time is increased by one large time step and the above mentioned data are printed out. New material concentrations are now available and the machine control is handed over to the block for calculating few group constants. Then a spectrum calculation is performed and this goes on until a criterion for termination is fulfilled (e.g., $k_{eff} < 1$).

Then a new case can be started or the machine terminates the run completely.

3. 1000 MWth POWER REACTOR

To investigate the burn-up, lifetime and other long time reactor data about 50 cases were treated with following specifications.

3.1 Reactor Data:

Thermal total power	1000 MW	
Average core power densities:	5 MW/m^3	(23 cases)
	10 MW/m^3	(6 cases)
	11.5 MW/m^3	(12 cases)
	15 MW/m^3	(32 cases)

Geometry:

A cylindrical reactor is assumed with no axial dependence of flux and other properties (one-dimensional treatment).

Core Dimensions			
Power Density (MW/m^3)	Height (m)	Radius (m)	$\frac{\text{Radius}}{\text{Height}}$
5	4.9	3.6	0.735
10	3.89	2.86	0.735
11.5	3.4	2.86	0.845
15	3.4	2.50	0.735

Because the burn-up is calculated with fluxes and concentrations averaged over regions it was advisable to sub-divide the core into regions through the initial loading of uranium, thorium and carbon are the same throughout all core regions.

Mesh Point Distribution					
Region	Part of Reactor	Number of Intervals	Position (cm)		
			5 MW/m^3	10, 11.5 MW/m^3	15 MW/m^3
1	↑	5	80	60	50
2		5	160	120	100
3	Core	5	240	180	150
4		5	320	240	200
5	↓	10	360	286	250
6	↑ Reflector	6	390	316	280
7	↓	5	440	366	330

3.2 Nuclear Data:

No control poison adjustment is calculated. The fuel elements are assumed to be fully retaining with no Xe-removal.

Materials Incorporated:

Two heavy isotope chains

Non-saturating fission product aggregate

Xe-135, Sm-149

Short lived fission product aggregate

B-10, Rh-103, Sm-151, Eu-153

Natural boron and carbon

No lumped poison is considered.

In the hot case 1200° Kelvin is assumed.

The energy groups are as follows:

Group 1	10^7-9.8 . 10^4 eV
Group 2	9.8 . 10^4-19.9 eV
Group 3	19.9-2.154 eV
Group 4	2.154-0.002154 eV.

3.3 Burn-up Data:

The large time steps within which the depletion is calculated without correcting the spectrum is specified to be 400 days.

In between, 10 small time steps are chosen for calculating the depletion with flux amplitude and concentration corrections.

3.4 Loading and Reactor Life:

Each case is specified by the initial loading:

Weight of Uranium-235 in core

$\frac{\text{Thorium-232}}{\text{Uranium-235}}$ atomic ratio (N-value)

The graphite concentration in the core is slightly less than in the reflector:

($\frac{\text{Graphite}}{\text{Uranium-235}}$ atomic ratio = S-value).

Then the whole core is depleted until no excess reactivity (k_{eff} = 1) is available. This is called a batch loading. The programme of course does not search for k_{eff} = 1. But this can be found very easily by interpolation.

3.5 Results

Tables 1-3 show the treated cases with selected results.

Figs. 2-4 are plots of selected results versus N-values for different initial Uranium-235 loadings and core power densities.

4. TABLE NOTATIONS

UL: total weight of Uranium-235 in core at start-up of depletion

Rating: power per unit weight of initial Uranium-235

N: $\frac{\text{Thorium-232}}{\text{Uranium-235}}$ atomic ratio at start-up

S: $\frac{\text{Graphite}}{\text{Uranium-235}}$ atomic ratio at start-up

T: Lifetime of reactor.

The batch containing thorium, uranium and graphite is depleted. The excess reactivity decreases due to the build-up of fission products and burn-out of uranium-235. The gain of U-233 from Th-232 only delays the burn-up. When $k_{eff} = 1$ the batch has to be removed. The time from start-up till $k_{eff} = 1$ is called the lifetime.

Specific burn-up: The total power output is P • T assuming a constant burn-up at a certain power level P (e.g., 1000 MW). The specific burn-up is $\frac{P \cdot T}{\text{initial fissionable material}}$ and is a measure for the economic investment of Uranium-235.

FIFA: Number of fissions per initial atom of fissionable material

$$FIFA = \frac{P \cdot T}{UL \cdot \Lambda} \qquad \Lambda = 0.95 \text{ MWd per 1 g fissioned U-235}$$

Th used / U-233 gained: give an indication how much fissionable material has been gained from the Th cycle

U-235 used: Amount of U-235 burnt

(U-235-U-233): balance between fissionable material used and gained expressed in equivalent energy (1 Giga = 10^3 Mega). This is the difference between the potential energy of the system of beginning and end of life. 1 MWd was equated to 1.25 g U-235 or 1.15 g U-233.

Conversion factor: $\dfrac{\text{rate of production of fissionable material}}{\text{rate of destruction of fissionable material}}$ at the end of lifetime

v: a similar expression as FIFA and shows the efficiency in using the thorium cycle

$$v = \frac{\text{total thermal energy produced during core life}}{\text{energy equivalent balance between fissionable material used and gained}}$$

$k_{eff}^{initial}$: effective multiplication factor at beginning of core life

$\rho_{initial}$: $\rho = \dfrac{k_{eff} - 1}{k_{eff}}$ at beginning of core life

This number gives the control requirements.

5. DISCUSSION

Initial Reactivity

(i) By increasing the N-value the $k_{eff}^{initial}$ decreases due to the higher absorption in thorium-232.

(ii) By increasing the start-up uranium loading the reactivity goes down because of the worse moderation (assuming constant graphite concentration).

(iii) The $k_{eff}^{initial}$ increases with decreasing power density (bigger core) due to the smaller leakage.

The Lifetime of the Core

(i) Dependence in Respect of N:

The lifetime is governed by two opposite effects:

1st effect: By increasing N the start-up reactivity falls,

2nd effect: By increasing N the rate of the reactivity loss decreases due to the gain of U-233 by the thorium cycle.

In this way the lifetime will go through a maximum with increasing N (other parameters constant).

(ii) As a general trend the lifetime increases with larger uranium loading which is quite obvious.

Table 1

1000 MWth Power Reactor 5 MW/m^3

UL (kg)	Rating (MW/kg U-235)	N	S	T (days)	Specific Burn-up (MWd/kg U-235)	FIFA	Th used (kg)	U-233 gained (kg)	U-235 used (kg)	Conversion factor at k_{eff}=1	(U-235-U-233) Equivalent GWd	v	$k_{eff}^{initial}$	$\rho_{initial}$ %
800	1.25	20	7320	457	571	0.6009	600	290	610	0.81	235.7	1.938	1.14	12.28
1000	1.0	20	5845	1000	1000	1.0525	1000	400	770	0.81	268.0	3.731	1.12	10.71
1200	0.83	12	4884	1250	1038	1.0925	900	310	1030	0.75	554.3	2.255	1.28	21.88
		14	4879	1270	1054	1.1093	1100	370	1020	0.78	494.1	2.570	1.22	18.03
		16	4873	1300	1079	1.1356	1200	420	1000	0.81	434.6	2.991	1.17	14.53
		20	4860	1200	996	1.0483	1200	530	930	0.85	282.9	4.242	1.09	8.26
1500	0.67	10	3906	1600	1072	1.1283	1140	350	1330	0.75	759.5	2.107	1.30	23.08
		12	3900	1650	1105	1.1631	1300	430	1300	0.78	666.0	2.477	1.24	19.35
		14	3894	1680	1126	1.1852	1460	510	1280	0.81	580.3	2.895	1.19	15.96
		16	3890	1750	1172	1.2336	1680	590	1260	0.83	494.7	3.537	1.14	12.28
		20	3876	1600	1072	1.1284	1720	740	1180	0.87	300.2	5.329	1.05	4.76
1800	0.56	8	3256	1850	1036	1.0904	1200	350	1613	0.68	986.0	1.88	1.34	25.36
		10	3250	2000	1120	1.1788	1500	450	1610	0.77	896.5	2.231	1.28	21.88
		12	3244	2000	1120	1.1788	1700	560	1560	0.79	760.8	2.628	1.21	17.36
		14	3222	2000	1120	1.1788	1800	670	1520	0.81	633.1	3.159	1.16	13.79
		16	3232	2000	1120	1.1788	2000	770	1480	0.84	514.1	3.893	1.11	9.91
		20	3219	1600	896	0.9430	1800	890	1280	0.88	249.7	6.408	1.03	2.91
2200	0.45	8	2659	2333	1050	1.1051	1600	470	1970	0.73	1167.1	1.999	1.31	23.66
		12	2647	2533	1140	1.1999	2200	760	1900	0.80	858.8	2.949	1.18	15.25
		16	2635	2400	1080	1.1367	2500	1020	1770	0.86	528.6	4.540	1.08	7.41
2600	0.38	8	2246	2900	1102	1.1599	2100	600	2400	0.79	1398.0	2.074	1.28	21.88
		12	2234	3000	1140	1.1999	2700	970	2220	0.82	932.1	3.216	1.15	13.04
		16	2222	2400	912	0.9599	2500	1230	1920	0.86	465.9	5.151	1.05	4.76

Table 2

1000 MWth Power Reactor 10 and 11.5 MW/m^3 respectively

UL (kg)	Rating (MW/kg U-235)	N	S	T (days)	Specific Burn-up (MWd/kg U-235)	FIFA	Th used (kg)	U-233 gained (kg)	U-235 used (kg)	Conversion factor at $k_{eff}=1$	(U-235-U-233) Equivalent GWd	v	$k_{eff}^{initial}$	$\rho_{initial}$ %
10 MW/m^3														
1000	1.0	8	2828	1000	1000	1.053	700	210	860	0.77	505.3	1.979	1.30	23.08
		12	2816	1000	1000	1.053	1000	325	810	0.78	365.2	2.738	1.17	14.52
1200	0.83	8	2352	1200	1000	1.053	870	280	1040	0.73	588.4	2.039	1.27	21.25
		12	2340	1200	1000	1.053	1100	430	970	0.80	401.9	2.985	1.14	12.28
1500	0.67	8	1877	1530	1025	1.079	1200	390	1280	0.75	684.7	2.234	1.23	18.69
		10	1871	1400	938	0.987	1300	490	1250	0.78	573.7	2.440	1.16	13.79
11.5 MW/m^3														
350	2.86	4	7121	260	744	0.783	120	16	210	0.53	154.1	1.687	1.52	34.21
		8	7109	312	892	0.939	200	46	250	0.63	160.0	1.950	1.39	28.06
		12	7097	320	915	0.963	256	67	262	0.66	151.3	2.115	1.28	21.88
		16	7085	320	915	0.963	300	88	240	0.67	115.4	2.772	1.20	16.66
		20	7065	267	764	0.804	260	93	200	0.58	79.1	3.375	1.12	10.71
525	1.9	4	4743	500	950	1.000	240	45	448	0.60	319.3	1.565	1.51	33.77
		8	4731	533	1013	1.066	360	88	438	0.72	273.8	1.946	1.36	26.47
		12	4719	533	1013	1.066	220	130	423	0.77	225.3	2.365	1.24	19.35
		16	4707	472	897	0.944	420	160	394	0.79	176.0	2.682	1.17	14.52
700	1.43	4	3554	614	878	0.924	270	62	560	0.64	394.1	1.557	1.49	32.88
		8	3542	700	1000	1.053	480	14	600	0.73	467.9	1.496	1.33	24.81
		16	3517	600	858	0.903	600	25	500	0.81	378.3	1.586	1.10	9.10

Table 3

1000 MWth Power Reactor 15 MW/m^3

UL (kg)	Rating (MW/kg U-235)	N	S	T (days)	Specific Burn-up (MWd/kg U-235)	FIFA	Th used (kg)	U-233 gained (kg)	U-235 used (kg)	Conversion factor at $k_{eff}=1$	(U-235-U-233) Equivalent GWd	v	$k_{eff}^{initial}$	$\rho_{initial}$ %
400	2.5	4	4634	338	845	0.8894	160	33	313	0.60	221.7	1.5245	1.50	33.33
		8	4622	380	945	0.9947	260	76	330	0.71	197.9	1.9202	1.34	25.37
		12	4610	365	915	0.9631	310	100	307	0.73	158.6	2.3014	1.22	18.03
		16	4598	320	800	0.8421	310	120	258	0.67	102.1	3.1342	1.12	10.71
		20	4585	180	444	0.4674	200	80	150	0.39	50.4	3.5714	1.04	3.85
600	1.67	4	3084	570	949	0.9989	290	58	540	0.65	381.5	1.4941	1.46	31.51
		8	3072	600	1002	1.0547	440	120	490	0.74	287.6	2.0862	1.29	22.48
		12	3060	560	935	0.9842	490	180	460	0.77	211.5	2.6477	1.17	14.53
		16	3048	350	584	0.6147	370	180	340	0.70	115.5	3.0303	1.07	6.54
800	1.25	4	2309	728	910	0.9579	360	90	700	0.65	481.7	1.5113	1.42	29.58
		6	2304	777	971	1.0221	480	140	700	0.70	438.3	1.7728	1.33	24.81
		8	2298	800	1000	1.0526	600	190	690	0.74	386.8	2.0682	1.25	20.00
		10	2292	756	945	0.9947	650	230	650	0.76	320.0	2.3625	1.18	15.25
		12	2286	667	834	0.8779	630	270	560	0.77	213.1	3.1299	1.12	10.71
		14	2277	400	500	0.5263	400	240	430	0.76	135.3	2.9564	1.07	6.54
		16	2274	200	250	0.2632	200	130	210	0.41	55.0	3.6364	1.02	1.96
1000	1.0	0	1838	680	680	0.7158	0	0	870	0	696.0	0.9770	1.73	42.19
		2	1851	848	848	0.8926	300	65	900	0.54	663.5	1.2781	1.52	34.21
		4	1845	933	933	0.9821	490	130	890	0.66	599.0	1.5575	1.39	28.06
		6	1839	967	967	1.0178	530	190	880	0.70	538.7	1.7951	1.29	22.48
		8	1833	970	970	1.0210	810	260	860	0.74	461.8	2.1005	1.21	17.36
		12	1821	685	685	0.7210	700	350	650	0.78	215.5	3.1786	1.08	7.41
		14	1815	300	300	0.3158	300	200	340	0.59	98.0	3.0612	1.03	2.91
		15	1812	0	0	0	0	0	0	0	0	0	0	0
1500	0.67	6	1219	1400	933	0.9821	1080	360	1260	0.71	694.8	2.0140	1.22	18.03
		8	1213	1275	850	0.8947	1300	460	1170	0.74	535.8	2.3790	1.14	12.28
		10	1207	875	583	0.6137	830	470	910	0.74	319.1	2.7421	1.07	6.54
		12	1201	150	100	0.1053	160	125	300	0.295	131.3	1.1424	1.01	1.00
1800	0.556	4		1650	917	0.9652	990	300	1580	0.65	1003.0	1.6451	1.29	22.48
		6		1600	890	0.9368	1210	470	1470	0.70	767.3	2.0853	1.19	15.97
		8		1350	750	0.7894	1180	570	1300	0.73	544.1	2.4810	1.11	9.91
		10		400	222	0.2337	400	300	520	0.71	155.1	2.5790	1.03	2.91

(iii) At small power densities the leakage is less due to the larger core and hence a longer lifetime can be expected.

FIFA:

(i) N-Dependence:

For the same reasons as above the curve in respect to N for constant power density and uranium loading has a maximum which is shifted to the higher N-values with decreasing UL. Increasing the power density the maximum is shifted to smaller N-values.

The curve of the maximum value has also a maximum for a certain UL.

v-Value:

This value which gives an indication how much more energy is gained than invested by using the conversion effects of the thorium cycle (see Table Notations).

General Trend:

(i) Larger reactors give higher v-values.

(ii) Increasing the N-value v gets higher, of course the disadvantage is the shorter lifetime which needs more reprocessing.

(iii) Higher uranium loading increases also the v-values, but the uranium investment is larger.

6. ACKNOWLEDGMENTS

The authors wish to express their gratitude to Dr. E. Schröder for his great interest and advice in preparing this report.

7. REFERENCES

[1] FEVER: "A One-dimensional Few-Group Depletion Programme for Reactor Analysis", F. Todt.
General Atomic GA.2749, November 1962.

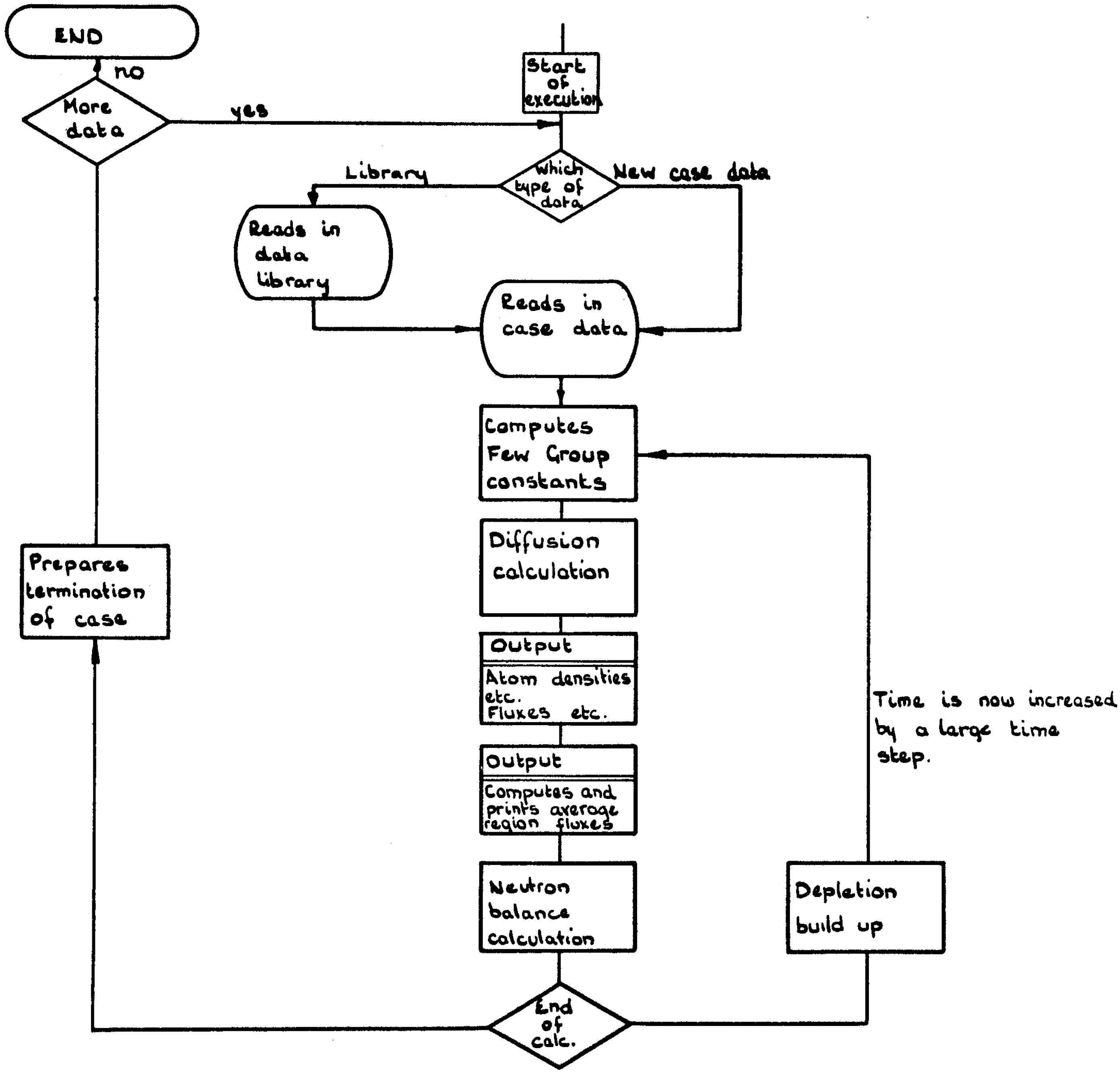
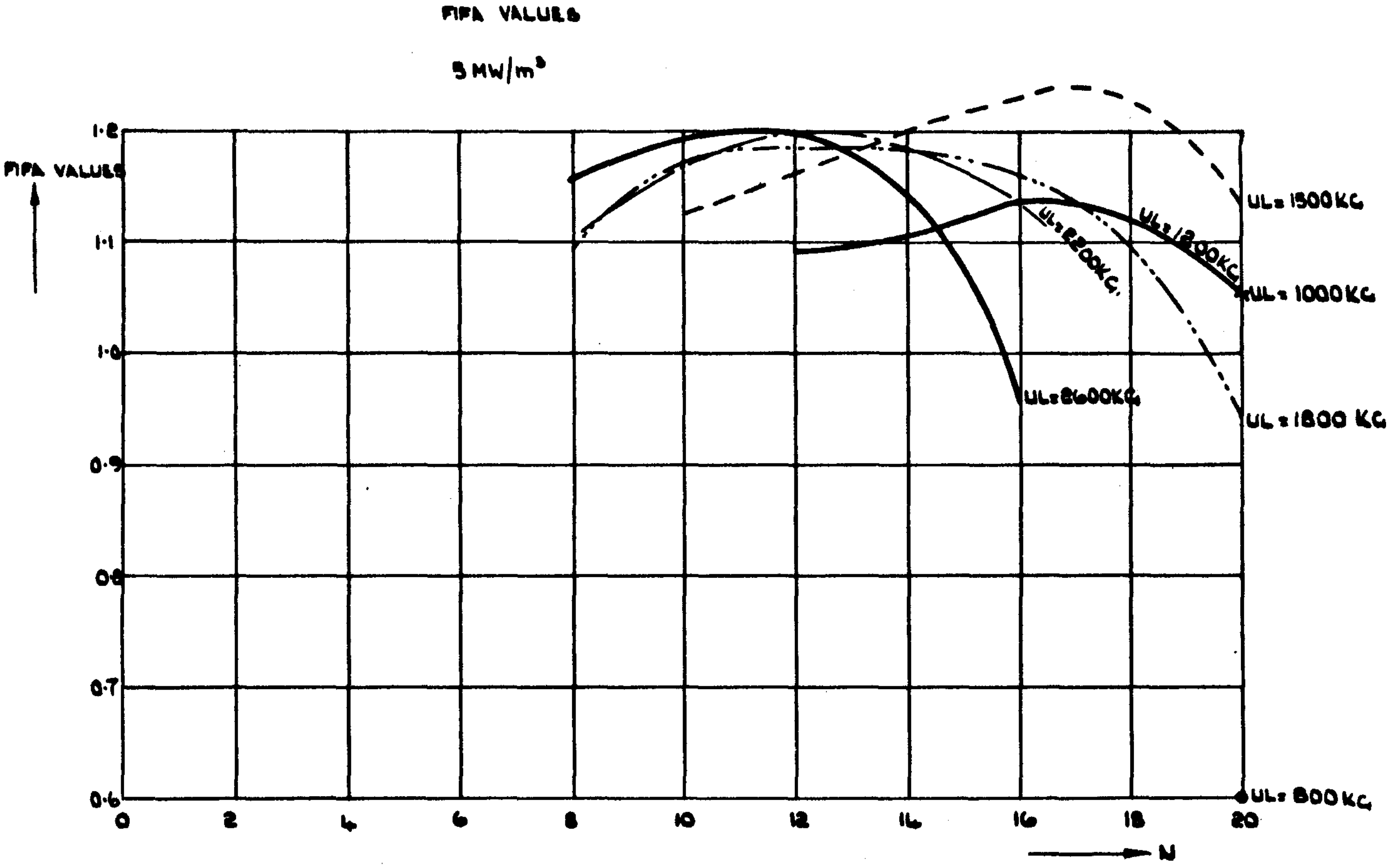


FIG. 1 FLOW DIAGRAM OF ONE DIMENSIONAL BURNUP CODE FEVER.



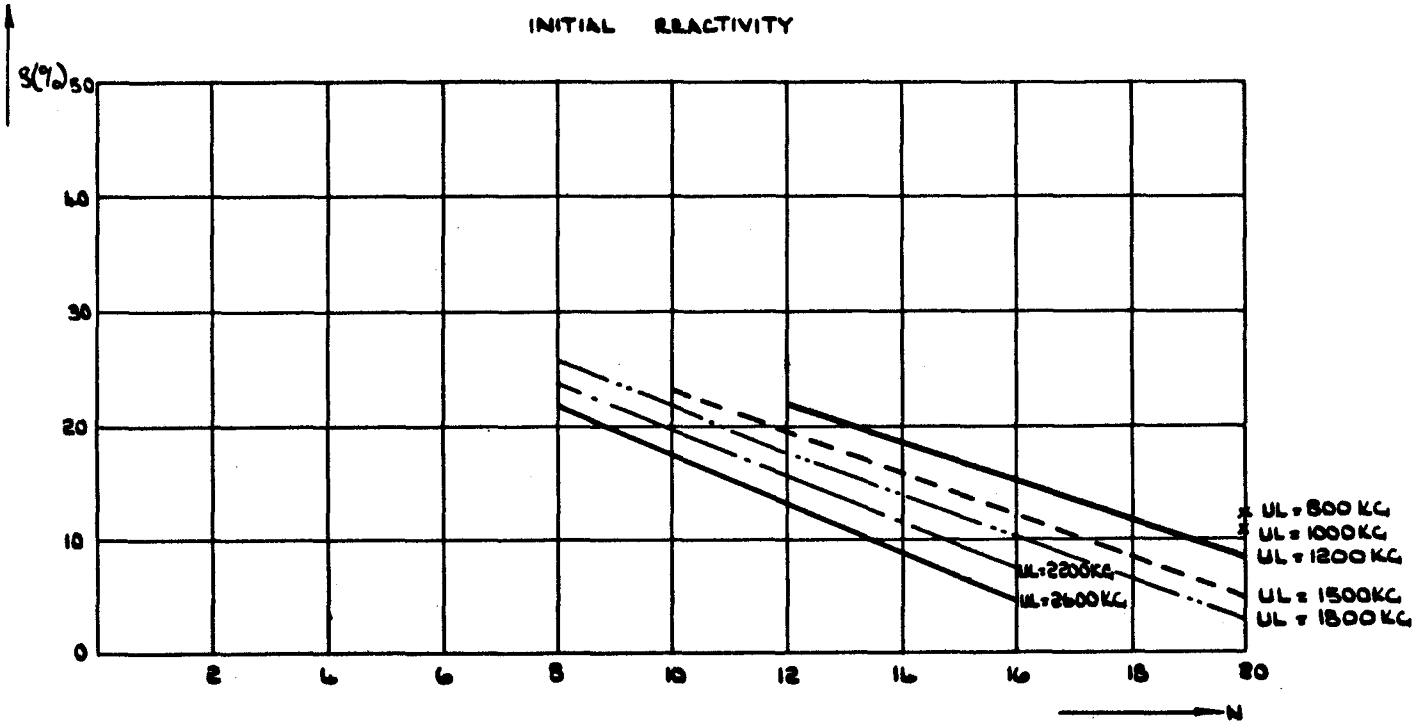


FIG. 2 1000 MW_{th} POWER REACTOR 5 MW/m^3

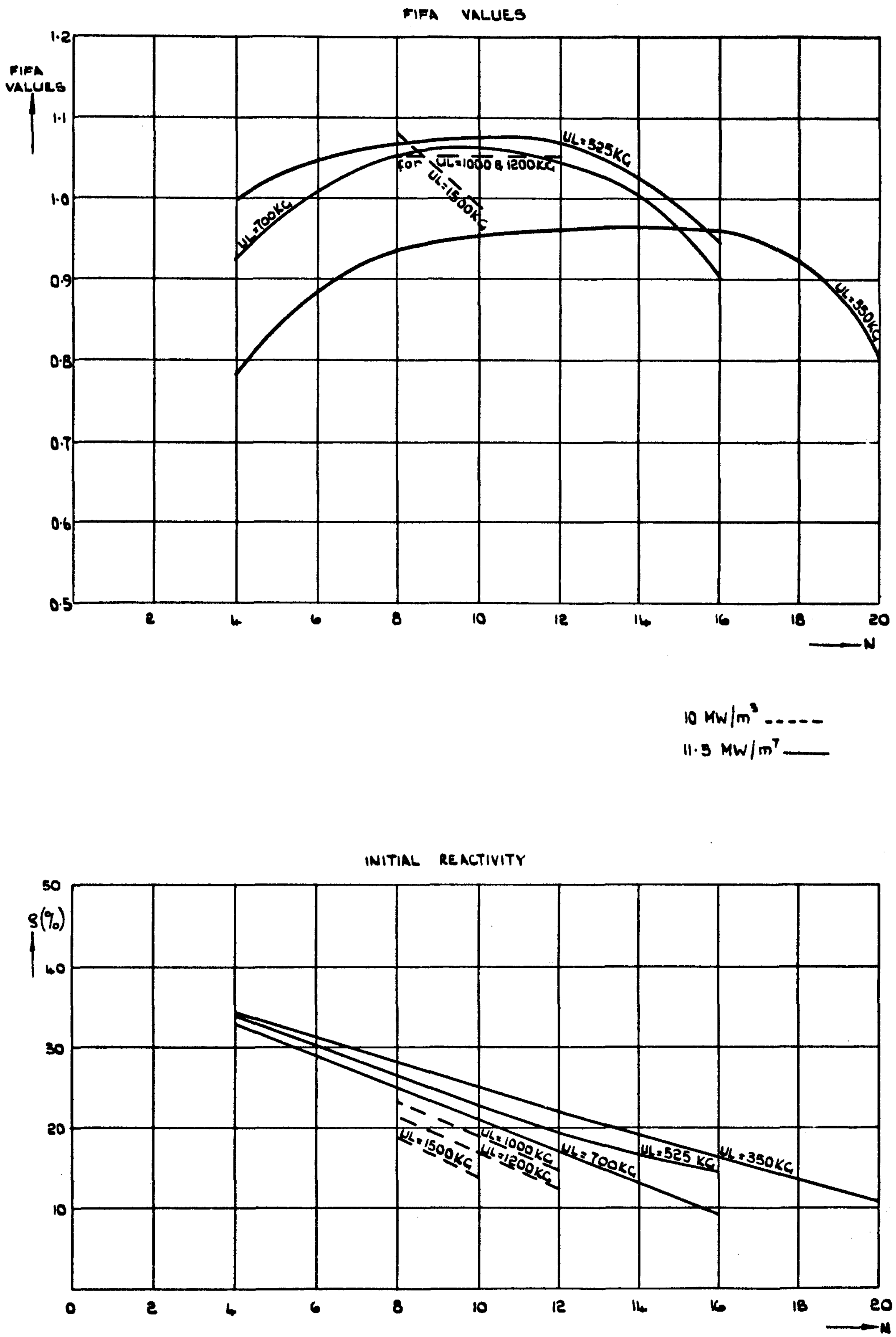
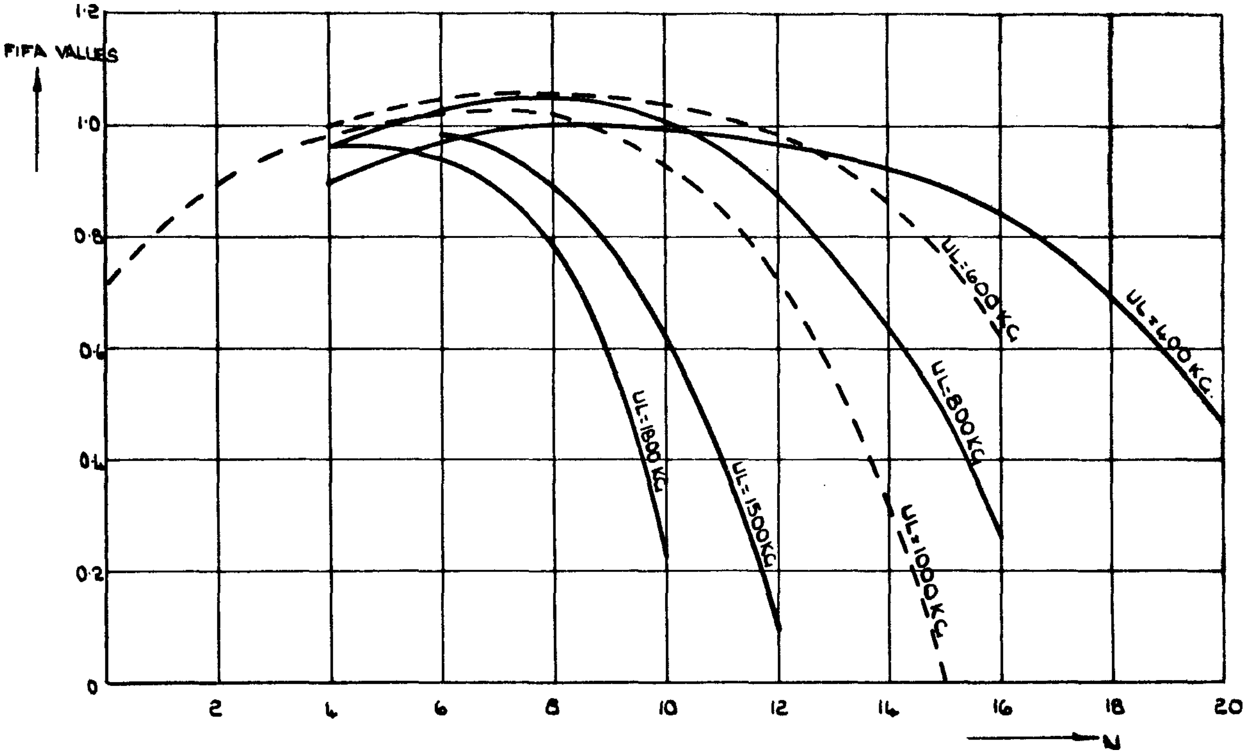


FIG. 3 1000 MW_{th} POWER REACTOR 10 & 11·5 MW/m^3 RESPECTIVELY.

15 MW/m^3

FIFA VALUES



INITIAL REACTIVITY

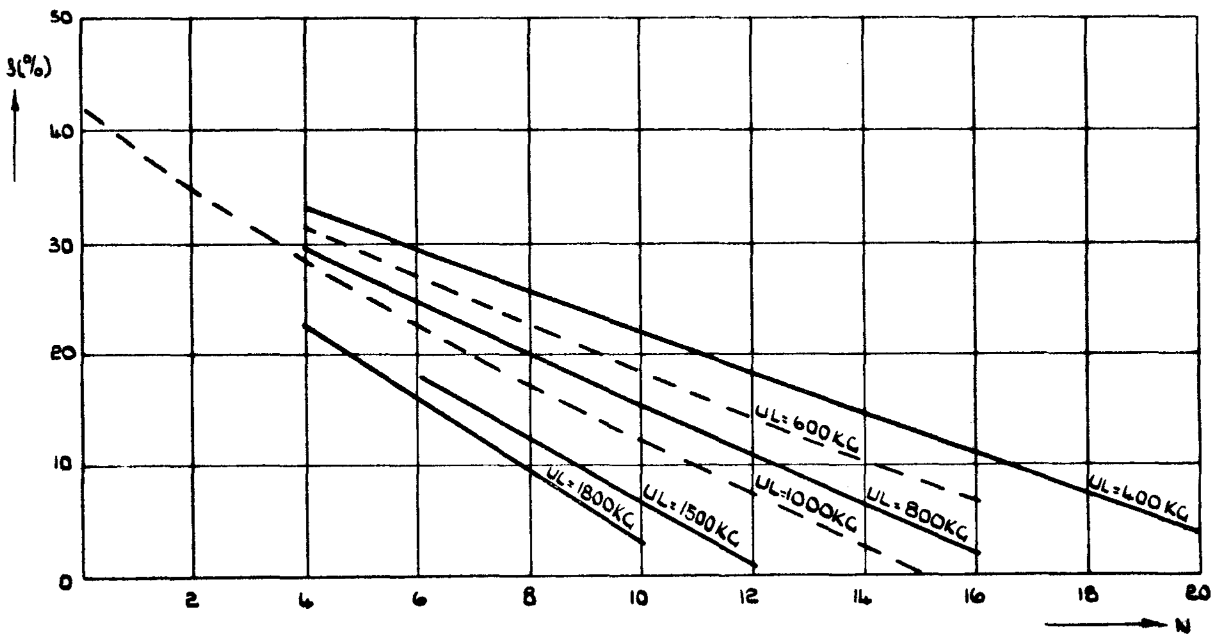


FIG. 4 1000 MW$_{th}$ POWER REACTOR 15 MW/m^3