NEACRP-L-151

OECD NUCLEAR ENERGY AGENCY

COMMITTEE ON REACTOR PHYSICS

REACTOR SHIELDING BENCHMARK NO. 2
for a Pressurised Water Reactor

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A. Introduction

At the Paris Meeting on Sensitivity Studies and Shielding Benchmarks (7-10 October 1975), the general feeling was that it could be constructive to embark upon some collaborative work in the field of sensitivity calculations. More precisely it appeared to be desirable to have two theoretical sensitivity benchmarks which could be collected and distributed by Dr. Nicks of EURATOM. The different shielding groups involved in this common exercise would then submit their results to the Vienna meeting (in Autumn 1976), which should reach some conclusions on data needs. The object of these theoretical sensitivity reference studies is to demonstrate the accuracy of the different data sets employed by the different countries and organisations.

The two benchmarks suggested concerned a fast reactor shielding configuration (1D) and a PWR which would essentially be a 2-dimensional problem. As far as the fast reactor is concerned, a suggestion was made to integrate the shielding problem into the existing proposal (which has been made through the NEACRP) for a fast reactor core benchmark.

It was also agreed to specify the PWR as a 2D problem also amenable to a 1D model. Dr. Hehn, assisted by Dr. Koban of KWU, kindly accepted to specify this second problem.

B. Benchmark name and type

PWR - 1300 MW, generic reactor shield design of a standard power reactor of PWR type.

C. System description

Fig. 1 gives a cross-sectional view of the reactor core, core baffle, core barrel (thermal shield), and reactor pressure vessel. The core contains 193 fuel assemblies with 103 t of uranium 3.2% enriched and 61 control rods. Water serves as a coolant and as a moderator at a pressure of 158 bar with 291°C inlet temperature and 326°C outlet temperature. The active height of the core is 3.9 m and the radial diameter

3.45 m along a main axis. The thermal shield is a stainless steel cylinder with an inner diameter of 4.21 m and 8 cm thick. The reactor pressure vessel with inner austenitic cladding has a diameter of 5 m and a total thickness of 25.6 cm. The pressure vessel is surrounded by thermal insulation and the primary concrete shield normally designed in two separate layers with a total thickness of 2 m.

D. Model description

1. One-dimensional model

The radial dimensions of a one-dimensional cylindrical configuration are given in Table 1 and the material composition in Table 2. Table 3 shows the power distribution in the outer core region containing three rows of fuel assemblies as indicated in Fig. 1. This outer core region gives a good representation of the radiation sources needed for shielding calculation. In Table 4 a proposal is made for the local mesh distribution along a radial main axis. For having identical normalization of the neutron and gamma sources in the core, the absolute power density in W/cm^3 should be used and the assumption of 200 MeV/fission. For the EURLIB multigroup structure with 100 neutron groups the neutron fission spectrum are given in Table 5 according to

$$\chi_N(E) = 0.7415 \sqrt{E}. e^{-E/\theta}$$

with θ = 1.323 MeV for thermal fission of U-35

E in MeV

$$\int_0^\infty \chi_N(E) d_E = 1$$

The neutron yield per fission reaction is then

$$Y_N(E) = \nu . \chi(E)$$

with ν = 2.419 for thermal fission of U-35.

The total gamma yield per fission reaction of prompt and delayed gamma rays are given in Table 6 for the standard group structure of 20 gamma groups in EURLIB. The fission yield of neutrons are taken from ENDF/B-IV and the gamma yield from the POPOP library.

2. Two-dimensional model

Since the calculation effort needed for the two-dimensional model is relatively high, the problem should be reconsidered with the results of the one-dimensional model available.

E. Aims of the calculation

1. Calculation of integral target quantities of interest in reactor shielding

1.1 Production of neutron damage (production rate of displacements per atom) in the pressure vessel at interval number 120.

1.2 Gamma heating rate in W/cm^3 in the concrete shield at interval number 147.

1.3 Biological dose rate in rem/h at the outer side of concrete shield at interval number 202.

2. Calculation of cross section sensitivities for damage, heating and biological dose as stated above

2.1 Determination of the importance of the cross sections for different nuclides. Total sensitivity to neutron cross sections and total sensitivity to gamma cross sections of the different nuclides present summed over energy groups and spatial zones.

2.2 Determination of the importance of angular moments. Sensitivity to Legendre moments of neutron and gamma cross sections (percent change to P_3 approximation) for the different nuclides present summed over energy groups and spatial zones.

2.3 Determination of important energy regions. Cross section sensitivities per unit lethargy (energy profiles) for the different nuclides present summed over spatial zones.

Table 1 Radial dimensions along a main axis with simplified concrete shield.

Zone	Zone radius [cm]	Zone thickness [cm]
Reactor core	172.5	-
Core baffle	175.0	2.5
1. Water layer	210.5	35.5
Core barrel	218.5	8.0
2. Waterlayer	250.0	31.5
Austenitic cladding	250.6	0.6
Pressure vessel	275.6	25.0
Concrete shield	475.6	200.0

Table 2 Material composition $[10^{24} \quad cm^{-3}]$

Zone / Nuclide	Reactor core	Core baffle core barrel aust. cladding	Pressure vessel	1. and 2. water layer	Concrete shield
H	2.8226E-02	-	-	5.0556E-2	4.4126E-3
O	2.7154E-02	-	-	2.5278E-2	4.7751E-2
Al	-	-	-	-	2.4553E-3
Si	-	-	-	-	9.4350E-3
Ca	-	-	-	-	6.6115E-3
Cr	-	1.6913E-2	-	-	-
Mn	-	1.1187E-3	-	-	-
Fe	9.8235E-4	6.4478E-2	8.465E-2	-	-
Ni	-	8.5357E-3	-	-	-
Zr	4.3246E-3	-	-	-	-
^{235}U	2.1040E-4	-	-	-	-
^{238}U	6.3087E-3	-	-	-	-

Table 3 Power distribution in outer core region (3 rows of fuel assemblies) along a radial main axis

Interval number	Radius [cm]	Interval centre	Power distribution factor
1	103.5	105.8	1.03
2	108.1	110.4	0.965
3	112.7	115.0	0.92
4	117.3	119.6	0.88
5	121.9	124.2	0.855
6	126.5	128.5	0.878
7	130.5	132.5	0.842
8	134.5	136.0	0.818
9	137.5	139.0	0.797
10	140.5	142.0	0.783
11	143.5	145.0	0.79
12	146.5	148.0	0.81
13	149.5	150.5	0.998
14	151.5	152.5	0.995
15	153.5	154.5	0.99
16	155.5	156.5	0.98
17	157.5	158.5	0.95
18	159.5	160.25	0.916
19	161.0	161.75	0.881
20	162.5	163.25	0.842
21	164.0	164.5	0.805
22	165.0	165.85	0.756
23	166.7	167.2	0.697
24	167.7	168.2	0.65
25	168.7	169.2	0.595
26	169.7	170.1	0.537
27	170.5	170.9	0.478
28	171.3	171.55	0.427
29	171.8	172.0	0.388
30	172.2	172.35	0.357
	172.5		

Power distribution factor 1 $\hat{=}$ 92.23 $\frac{W}{cm^3}$

Table 4 Proposal of a local mesh distribution along a radial main axis

Zone and total number of intervals	Number of intervals	Thickness of interval [cm]	Radius [cm]
Reactor core	-	-	103.5
	5	4.6	126.5
	2	4.0	134.5
	5	3.0	149.5
	5	2.0	159.5
	4	1.5	165.5
	1	1.2	166.7
	3	1.0	169.7
	2	0.8	171.3
	1	0.5	171.8
	1	0.4	172.2
30	1	0.3	172.5
Core baffle			
5	5	0.5	175.0
1. water layer			
	2	0.4	175.8
	2	0.5	176.8
	2	0.6	178.0
	8	1.0	186.0
	9	1.5	199.5
	8	1.0	207.5
	2	0.6	208.7
	2	0.5	209.7
37	2	0.4	210.5
Core barrel			
	2	0.4	211.3
	2	0.5	212.3
	1	0.7	213.0
	3	1.0	216.0
	1	0.7	216.7
	2	0.5	217.7
13	2	0.4	218.5

cont.

Table 4 (continuing)

Zone and total number of intervals	Number of intervals	Thickness of interval [cm]	Radius [cm]
2. water layer			
	2	0.4	219.3
	2	0.6	220.5
	2	0.8	222.1
	5	1.0	227.1
	1	1.15	228.25
	8	1.5	240.25
	1	1.15	241.40
	5	1.0	246.4
	2	0.8	248.0
	2	0.6	249.2
32	2	0.4	250.0
Cladding			
2	2	0.3	250.6
Pressure vessel			
	1	0.3	250.9
	1	0.45	251.35
	1	0.6	251.95
	3	0.8	254.35
	5	1.0	259.35
	5	1.5	266.85
	5	1.0	271.85
	3	0.8	274.25
	1	0.6	274.85
	1	0.45	275.3
27	1	0.3	275.6
Concrete shield			
	1	0.4	276.0
	2	0.5	277.0
	3	0.8	279.4
	5	1.0	284.4
	5	1.5	291.9
	5	2.0	301.9
	5	2.5	314.4
	5	3.0	329.4
	5	4.0	349.4
	5	5.0	374.4
	5	6.0	404.4
	9	7.0	467.4
56	1	8.2	475.6

IM = 202

Table 5: Neutron fission spectrum of U-235 for 100 neutron groups (EURLIB)

Group g	χ_g	Group g	χ_g	Group g	χ_g
1	8.97261E-5	31	3.61562E-2	61	1.36893E-4
2	2.13026E-4	32	3.35911E-2	62	1.87240E-4
3	4.58930E-4	33	3.09640E-2	63	2.69199E-4
4	9.07590E-4	34	2.83579E-2	64	4.09377E-4
5	1.65783E-3	35	2.58209E-2	65	2.82063E-4
6	2.82088E-3	36	2.33865E-2	66	1.94347E-4
7	4.49934E-3	37	2.10762E-2	67	1.33795E-4
8	3.06231E-3	38	1.89111E-2	68	9.20799E-5
9	3.70767E-3	39	1.69032E-2	69	6.33517E-5
10	4.43038E-3	40	1.50532E-2	70	4.35775E-5
11	5.23040E-3	41	1.33624E-2	71	2.99680E-5
12	1.31411E-2	42	1.18273E-2	72	2.06081E-5
13	1.71130E-2	43	1.04405E-2	73	1.41702E-5
14	1.01627E-2	44	9.19165E-3	74	9.74090E-6
15	1.12714E-2	45	8.07803E-3	75	0.0
16	2.59018E-2	46	7.08551E-3	76	0.0
17	3.03093E-2	47	6.20250E-3	77	0.0
18	3.44500E-2	48	5.41990E-3		
19	3.81366E-2	49	4.73089E-3		
20	4.12106E-2	50	4.12301E-3		
21	4.35819E-2	51	3.58943E-3		
22	2.24567E-2	52	3.12326E-3		
23	2.27593E-2	53	2.71087E-3		
24	4.60585E-2	54	5.31127E-3		
25	4.62095E-2	55	3.71142E-3		
26	4.56394E-2	56	2.58380E-3		
27	4.45264E-2	57	1.79371E-3		
28	4.29072E-2	58	1.24255E-3		
29	4.09266E-2	59	7.11865E-4	99	0.0
30	3.86661E-2	60	1.47301E-4	100	0.0

Table 6: Total gamma yield per fission (prompt + delayed) for 20 gamma groups (EURLIB)

Group g	Energy (MeV)	Y_g
	14.0	
1	12.0	0.0
2	10.0	0.0
3	8.0	3.85499E-4
4	6.5	4.61582E-3
5	5.0	3.55212E-2
6	4.0	1.14826E-1
7	3.0	3.48427E-1
8	2.5	3.91799E-1
9	2.0	7.08001E-1
10	1.66	7.60580E-1
11	1.33	9.93915E-1
12	1.0	1.71500
13	0.8	1.48385
14	0.6	1.92847
15	0.4	3.15475
16	0.3	1.86547
17	0.2	1.86547
18	0.1	2.39756
19	0.05	6.84255E-1
20	0.02	5.47356E-1

Table 7 : EURLIB Group Structure

Group	Energy Range	Group	Energy Range
1	1.3499E 07 – 1.4918E 07	51	1.3569E 05 – 1.4996E 05
2	1.2214E 07 – 1.3499E 07	52	1.2277E 05 – 1.3569E 05
3	1.1052E 07 – 1.2214E 07	53	1.1109E 05 – 1.2277E 05
4	1.0000E 07 – 1.1052E 07	54	8.6517E 04 – 1.1109E 05
5	9.0484E 06 – 1.0000E 07	55	6.7379E 04 – 8.6517E 04
6	8.1873E 06 – 9.0484E 06	56	5.2475E 04 – 6.7379E 04
7	7.4082E 06 – 8.1873E 06	57	4.0868E 04 – 5.2475E 04
8	7.0469E 06 – 7.4082E 06	58	3.1828E 04 – 4.0868E 04
9	6.7032E 06 – 7.0469E 06	59	2.6050E 04 – 3.1828E 04
10	6.3763E 06 – 6.7032E 06	60	2.4788E 04 – 2.6050E 04
11	6.0653E 06 – 6.3763E 06	61	2.3570E 04 – 2.4788E 04
12	5.4881E 06 – 6.0653E 06	62	2.1870E 04 – 2.3570E 04
13	4.9659E 06 – 5.4881E 06	63	1.9305E 04 – 2.1870E 04
14	4.7240E 06 – 4.9659E 06	64	1.5034E 04 – 1.9305E 04
15	4.4933E 06 – 4.7240E 06	65	1.1709E 04 – 1.5034E 04
16	4.0657E 06 – 4.4933E 06	66	9.1188E 03 – 1.1709E 04
17	3.6788E 06 – 4.0657E 06	67	7.1017E 03 – 9.1188E 03
18	3.3287E 06 – 3.6788E 06	68	5.5308E 03 – 7.1017E 03
19	3.0112E 06 – 3.3287E 06	69	4.3074E 03 – 5.5308E 03
20	2.7253E 06 – 3.0112E 06	70	3.3546E 03 – 4.3074E 03
21	2.4660E 06 – 2.7253E 06	71	2.6126E 03 – 3.3546E 03
22	2.3460E 06 – 2.4660E 06	72	2.0347E 03 – 2.6126E 03
23	2.2313E 06 – 2.3460E 06	73	1.5846E 03 – 2.0347E 03
24	2.0190E 06 – 2.2313E 06	74	1.2341E 03 – 1.5846E 03
25	1.8268E 06 – 2.0190E 06	75	9.6112E 02 – 1.2341E 03
26	1.6530E 06 – 1.8268E 06	76	7.4852E 02 – 9.6112E 02
27	1.4957E 06 – 1.6530E 06	77	5.8295E 02 – 7.4852E 02
28	1.3534E 06 – 1.4957E 06	78	4.5400E 02 – 5.8295E 02
29	1.2246E 06 – 1.3524E 06	79	3.5357E 02 – 4.5400E 02
30	1.1080E 06 – 1.2246E 06	80	2.7536E 02 – 3.5357E 02
31	1.0026E 06 – 1.1080E 06	81	2.1445E 02 – 2.7536E 02
32	9.0718E 05 – 1.0026E 06	82	1.6702E 02 – 2.1445E 02
33	8.2085E 05 – 9.0718E 05	83	1.3007E 02 – 1.6702E 02
34	7.4274E 05 – 8.2085E 05	84	1.0130E 02 – 1.3007E 02
35	6.7206E 05 – 7.4274E 05	85	7.8893E 01 – 1.0130E 02
36	6.0810E 05 – 6.7206E 05	86	6.1442E 01 – 7.8893E 01
37	5.5023E 05 – 6.0810E 05	87	4.7851E 01 – 6.1442E 01
38	4.9787E 05 – 5.5023E 05	88	3.7267E 01 – 4.7851E 01
39	4.5049E 05 – 4.9787E 05	89	2.9023E 01 – 3.7267E 01
40	4.0762E 05 – 4.5049E 05	90	2.2603E 01 – 2.9023E 01
41	3.6883E 05 – 4.0762E 05	91	1.7603E 01 – 2.2603E 01
42	3.3373E 05 – 3.6883E 05	92	1.0677E 01 – 1.7603E 01
43	3.0197E 05 – 3.3373E 05	93	8.3153E 00 – 1.0677E 01
44	2.7324E 05 – 3.0197E 05	94	5.0435E 00 – 8.3153E 00
45	2.4724E 05 – 2.7324E 05	95	3.0592E 00 – 5.0435E 00
46	2.2371E 05 – 2.4724E 05	96	1.8554E 00 – 3.0592E 00
47	2.0242E 05 – 2.2371E 05	97	1.1254E 00 – 1.8554E 00
48	1.8316E 05 – 2.0242E 05	98	6.2500E–01 – 1.1254E 00
49	1.6573E 05 – 1.8316E 05	99	4.1399E–01 – 6.2500E–01
50	1.4996E 05 – 1.6573E 05		

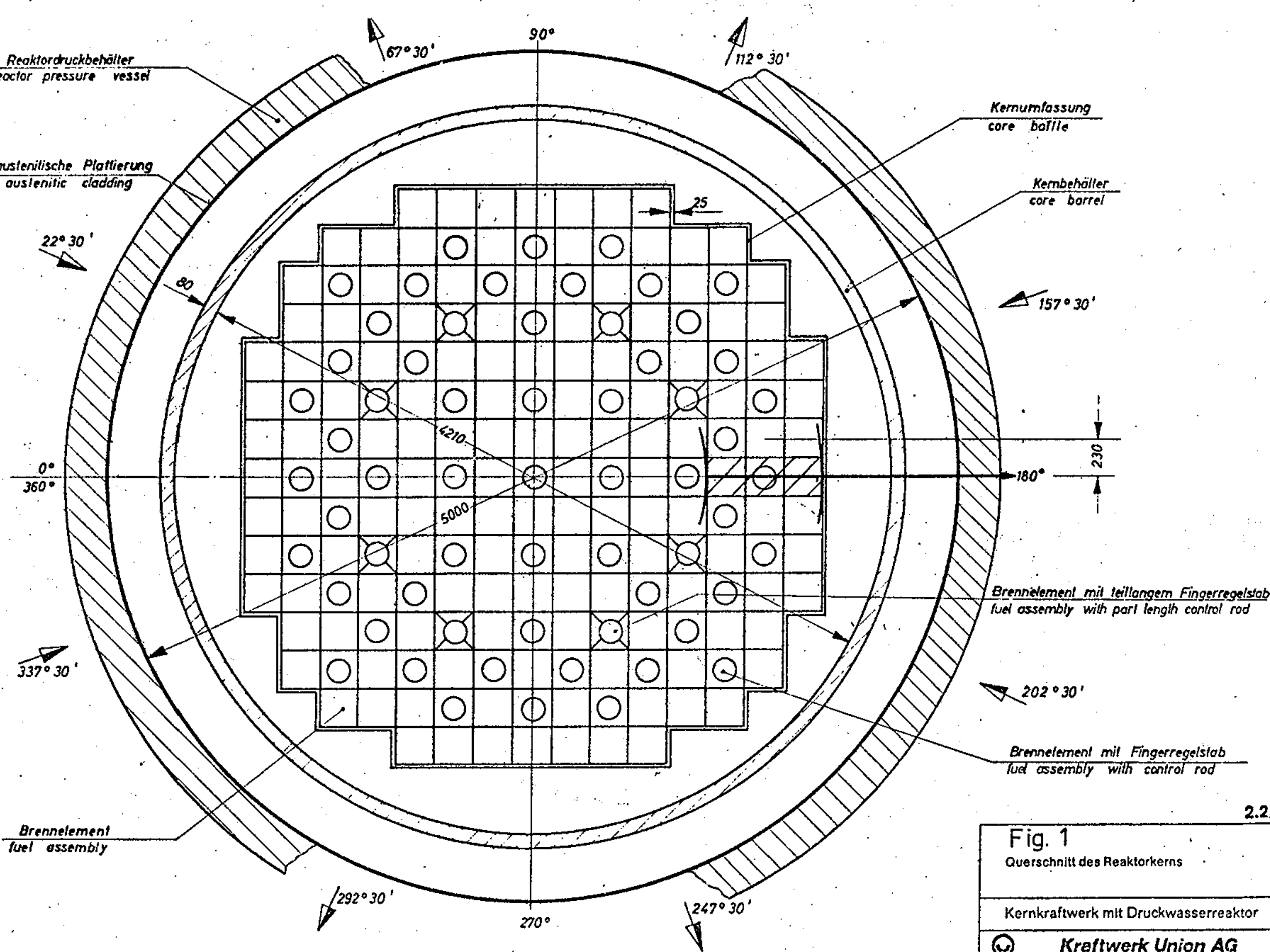

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Fig. 1
Querschnitt des Reaktorkerns

Kernkraftwerk mit Druckwasserreaktor

Kraftwerk Union AG

7100070096