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FENDL ACTIVATION BENCHMARK:

SPECIFICATIONS FOR THE CALCULATIONAL ACTIVATION BENCHMARK

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with an introduction by
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Vienna, Austria

December 1994

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Abstract

During the IAEA Advisory Group Meeting on "Improved Evaluations and Integral Data Testing for FENDL" held in Garching near Munich, Germany in the period 12-16 September 1994, the Working Group IV recommended that a calculational activation benchmark representative of the ITER design should be developed. This report gives detailed description and specifications for the calculational benchmark as well as the neutron flux and adjoint gamma flux.

December 1994

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December 1994

Introduction

(written by S. Ganesan, IAEA Nuclear Data Section, Vienna, Austria)

The IAEA Nuclear Data Section has implemented a computerized collection of data from those integral neutronic experiments that are suitable to test libraries of evaluated fusion relevant nuclear data ("benchmark experiments"). In particular, the Fusion Evaluated Nuclear Data Library (FENDL), the reference library for the International Thermonuclear Experimental Reactor (ITER) project, should be validated using these experimental data. An IAEA Consultant's Meeting on "Preparation of Fusion Benchmarks in Electronic Format for Nuclear Data Validation Studies has given detailed recommendations for submissions of experimental data and parameters for this collection (see summary report INDC(NDS)-298, March 1994). It was pointed out that, in addition to numerical data, explanatory hard-copy material in the form of text and figures is indispensable to enable calculations to be made.

An extract reproduced below from the recommendations of the IAEA Advisory Group Meeting on "Improved Evaluations and Integral Data Testing for FENDL", organized by the International Atomic Energy Agency in co-operation with the Max-Planck-Institut für Plasmaphysik and held at the Max-Planck-Institut für Plasmaphysik, Garching near Munich, Germany, 12 to 16 September 1994 (Ref. summary report: INDC(NDS)-312, November 1994) in the Working Group (IV) on Activation (F.M. Mann (U.S.A.) Chairman, R.A. Forrest (United Kingdom) Secretary of the Working Group) follows:

There was a proposal for a further calculational benchmark focused on the ITER design activity. This was agreed and the following procedure was adopted:

a. M.E. Sawan will provide data corresponding to a series of zones in a 1D representation of an ITER design. These data are:

(i) A set of neutron fluxes for each zone
(ii) Irradiation history
(iii) Material compositions
(iv) Adjoint g fluxes corresponding to a point at the rear of the shield to enable contact g does to be calculated (that include g transport).

b. Participants will use FENDL-A1.1 cross sections and FENDL decay data library to do the calculation.

c. Reported data to include the inventory of the 'top ten' contributing nuclides and the decay heat in each zone and the point g dose rate; all of these at the various cooling times defined.

d. Results will be collated, inter-compared, and conclusions drawn, with R. Roussin as the contact point.

M.E. Sawan agreed to finish the design of the benchmark and distribute to the IAEA-Nuclear Data Section who will make it available to interested parties for comments by 31 October 1994. When agreed the data will be made available by the IAEA-Nuclear Data Section.

Preliminary interested participants are: M.E. Sawan, M.Z. Youssef, H. Attaya, E.T. Cheng and R.A. Forrest.

Following this recommendation, the neutronics and shielding calculational benchmark has been compiled and placed into the Nuclear Data Section online system. This document complements the online information available by ftp command from the IAEA:

ftp 161.5.2.2
user FENDL
cd FENDL/BENCHMARKS/WISCONSIN

The files "ACTIVATION.CALC", "GAMMA-ADJ" and "NFLUX.ACT" in this directory correspond to this document.

SPECIFICATIONS FOR THE CALCULATIONAL ACTIVATION BENCHMARK

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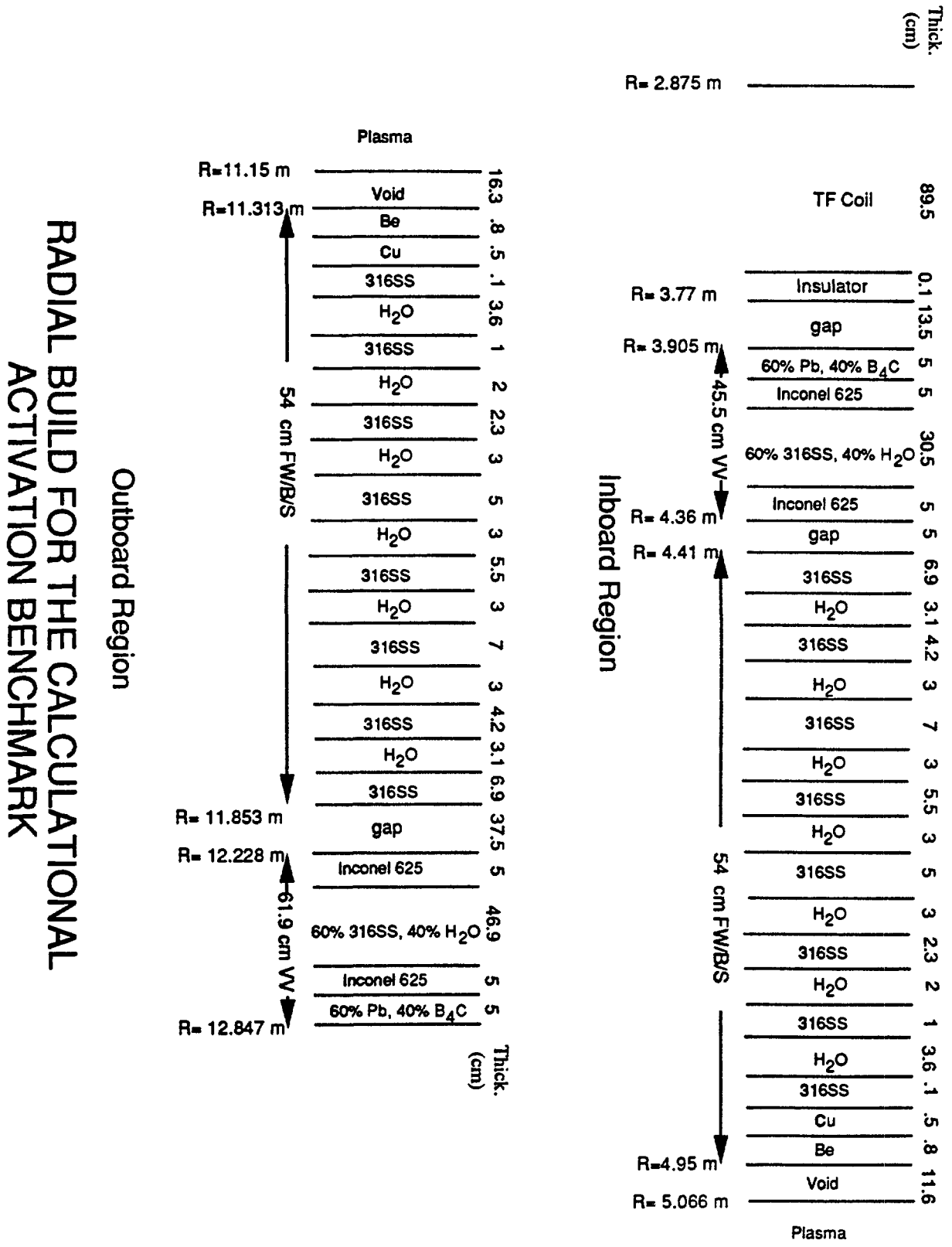
26 October 1994

Calculational Activation Benchmark Description

- The benchmark represents the reference steel/water shielding blanket design in the ITER outline design.
- The first wall is 14 mm thick consisting of 8 mm thick Be coating and 5 mm Cu attached to 1 mm thick SS.
- The shielding blanket is 526 mm thick with alternating layers of 316 SS and water.
- A double wall Inconel 625 vacuum vessel is used with single size water cooled 316 SS balls. The VV walls are 50 mm thick. A 50 mm thick back shield zone made of lead and boron carbide is used at the back of the VV. The total VV thickness is 455 mm in inboard region and 619 mm in outboard region.
- 1-D toroidal cylindrical model with inboard and outboard regions modeled simultaneously is used.
- The model includes 51 zones divided into 468 intervals.
- A maximum fine mesh interval width of 1 cm is used in the model except in plasma and void zones.
- A uniform 14.1 MeV isotropic neutron source in the plasma zone.
- The source in the plasma zone is normalized to 6.1E17 n/cm.s yielding inboard and outboard neutron wall loadings of 1 and 1.5 MW/m^2, respectively.
- Calculations are to be performed for two irradiation histories:
  1- Continuous irradiation for 3 years (9.45×10^7 s).
  2- Uniform pulsed operation with 94500 pulses each is 1000 s wide with dwell time of 1200 s between pulses.

Neutron Flux and Adjoint Gamma Flux Files

- The neutron flux is provided at all fine mesh intervals in the VITAMIN-J 175 energy group structure.
- The flux was calculated using the ONEDANT code with the processed FENDL multi-group cross section data library FENDL/MG.
- An ASCII file containing the neutron flux can be accessed from the IAEA NDS online system. The file name is nflux.act. The neutron flux in the 175 energy groups is listed for each of the 468 mesh intervals in 6(1PE12.5) format. A header card (format A3,I6) with the interval number precedes the spectrum for each interval.
- The adjoint gamma flux is provided at all fine mesh intervals in the VITAMIN-J 42 energy group structure.
- The adjoint flux was determined using the ONEDANT code with the FENDL/MG data. The source spectrum in the adjoint calculation represents the tissue kerma factor (flux-to-dose conversion factor) and is located at the outer surface of the outboard B_4C/Pb shield (interval # 468).
- An ASCII file containing the gamma adjoint flux can be accessed from the IAEA NDS online system. The file name is gamma.adj. The adjoint gamma flux in the 42 energy groups is listed for each of the 468 mesh intervals in 6(1PE12.5) format. A header card (format A3,I6) with the interval number precedes the spectrum for each interval.
- The biological dose rate (micro Sv per hr) at the back of shield can be calculated at any time following shutdown by performing energy and volume integration for the product of the adjoint flux and the decay gamma source ($\gamma/cm^2.s$) and dividing the result by the volume of interval # 468.


RADIAL BUILD FOR THE CALCULATIONAL ACTIVATION BENCHMARK
Outboard Region
Plasma
R=11.15 m
Void
R=11.313 m
Be
Cu
316SS
H2O
316SS
H2O
316SS
H2O
316SS
H2O
316SS
H2O
316SS
H2O
316SS
gap
R= 11.853 m
R= 12.228 m
Inconel 625
60% 316SS, 40% H2O
Inconel 625
60% Pb, 40% B4C
R= 12.847 m
54 cm FW/B/S
61.9 cm VV
16.3 .8 .5 .1 3.6 1 2 2.3 3 5 3 5.5 3 7 3 4.2 3.1 6.9 37.5 5 46.9 5 5
Thick. (cm)
Inboard Region
R= 2.875 m
TF Coil
Insulator
R= 3.77 m
gap
R= 3.905 m
60% Pb, 40% B4C
Inconel 625
60% 316SS, 40% H2O
Inconel 625
R= 4.36 m
gap
R= 4.41 m
316SS
H2O
316SS
H2O
316SS
H2O
316SS
H2O
316SS
H2O
316SS
H2O
316SS
Cu
Be
R=4.95 m
Void
R= 5.066 m
45.5 cm VV
54 cm FW/B/S
89.5 0.1 13.5 5 5 30.5 5 5 6.9 3.1 4.2 3 7 3 5.5 3 5 3 2.3 2 1 3.6 .1 .5 .8 11.6
Thick. (cm)
Plasma

Zone Specifications

Zone Number	Material	Thickness (cm)	Number of Intervals
Central Zone			
1	void	287.5	29
Inboard Magnet			
2	47% 316SS 12% Cu 17.2% Liq. He 13.3% R-glass epoxy 3% Nb_3Sn 7.5% bronze	89.4	90
3	R-glass epoxy	0.1	1
Gap			
4	void	13.5	14
Inboard VV			
5	60% Pb 40% B_4C	5	5
6	Inconel 625	5	5
7	60% 316SS 40% H_2O	30.5	31
8	Inconel 625	5	5
Gap			
9	void	5	5
Inboard Blanket			
10	316SS	6.9	7
11	H_2O	3.1	4
12	316SS	4.2	5
13	H_2O	3	3
14	316SS	7	7
15	H_2O	3	3
16	316SS	5.5	6
17	H_2O	3	3

18	316SS	5	5
19	H_2O	3	3
20	316SS	2.3	3
21	H_2O	2	2
22	316SS	1	1
23	H_2O	3.6	4
Inboard FW			
24	316SS	0.1	1
25	Cu-Be-Ni	0.5	1
26	Be	0.8	1
Inboard Scrapeoff			
27	void	11.6	2
Plasma			
28	void	608.4	61
Outboard Scrapeoff			
29	void	16.3	2
Outboard FW			
30	Be	0.8	1
31	Cu-Be-Ni	0.5	1
32	316SS	0.1	1
Outboard Blanket			
33	H_2O	3.6	4
34	316SS	1	1
35	H_2O	2	2
36	316SS	2.3	3
37	H_2O	3	3
38	316SS	5	5
39	H_2O	3	3
40	316SS	5.5	6
41	H_2O	3	3
42	316SS	7	7
43	H_2O	3	3
44	316SS	4.2	5
45	H_2O	3.1	4
46	316SS	6.9	7

Gap			
47	void	37.5	38
Outboard VV			
48	Inconel 625	5	5
49	60% 316SS 40% H_2O	46.9	47
50	Inconel 625	5	5
51	60% Pb 40% B_4C	5	5

Material Composition

Elemental composition is given. Natural abundances of the isotopes in each element should be used to calculate the isotopic composition.

Material	Constituent Element	Nuclide Density (nuclei/b.cm)
R-glass epoxy	H	2.16300E-02
	C	1.89200E-02
	N	2.06000E-03
	O	2.70600E-02
	Mg	1.19000E-03
	Al	3.93000E-03
	Si	8.00000E-03
	S	5.10000E-04
	Cu	9.10000E-04
Cu	Cu	8.29204E-02
Nb_3Sn	Nb	4.09117E-02
	Sn	1.36372E-02
Liq. He	He	1.83643E-02
Pb	Pb	3.29558E-02
B_4C	B	1.09849E-01
	C	2.74621E-02
H_2O	H	6.68560E-02
	O	3.34280E-02
Cu-Be-Ni	Be	2.97000E-03
	Ni	1.82000E-03
	Cu	8.20000E-02
Be	Be	1.23619E-01
Bronze	Cu	7.67230E-02
	Sn	3.57200E-03

SS316	B	4.37546E-06
	C	7.08895E-05
	N	2.36402E-04
	O	5.91306E-06
	Al	5.25950E-04
	Si	7.74757E-04
	P	3.97072E-05
	S	1.47526E-05
	K	6.04926E-07
	Ti	3.95194E-05
	V	3.71431E-06
	Cr	1.55566E-02
	Mn	1.46375E-03
	Fe	5.45732E-02
	Co	2.40797E-05
	Ni	1.06384E-02
	Cu	7.44397E-05
	As	3.15700E-07
	Zr	1.03900E-06
	Nb	1.01830E-06
	Mo	1.23274E-03
	Ag	8.77100E-08
	Cd	8.41500E-08
	Sn	7.96952E-07
	Sb	1.94200E-07
	Ba	1.72200E-07
	Tb	1.48800E-07
	Ta	1.30709E-07
	W	2.57300E-07
	Ir	1.23000E-07
	Pb	1.82623E-07
	Bi	1.81082E-07

Inconel 625	C	1.68800E-04
	Al	2.25000E-04
	Si	3.42000E-04
	S	1.27000E-05
	Ti	2.01000E-04
	Cr	2.16000E-02
	Mn	3.68000E-05
	Fe	2.32000E-03
	Co	4.30000E-05
	Ni	5.30000E-02
	Cu	2.39000E-05
	Nb	1.66800E-03
	Mo	4.90000E-03
	Ta	2.15500E-04

Isotopic Natural Abundances for Elements Used in the Calculational Activation Benchmark (From Chart of the Nuclides, 14th Edition, April 1988)

Element	Z	A	% Abundance
H	1	1	99.9850000
		2	0.0150000
He	2	3	0.0001380
		4	99.9998620
Be	4	9	100.0000000
B	5	10	19.9000000
		11	80.1000000
C	6	12	98.9000000
		13	1.1000000
N	7	14	99.6300000
		15	0.3700000
O	8	16	99.7600000
		17	0.0400000
		18	0.2000000
Na	11	23	100.0000000
Mg	12	24	78.9900000
		25	10.0000000
		26	11.0100000
Al	13	27	100.0000000
Si	14	28	92.2300000
		29	4.6700000
		30	3.1000000

P	15	31	100.0000000
S	16	32	95.0200000
		33	0.7500000
		34	4.2100000
		36	0.0200000
K	19	39	93.2581000
		40	0.0117000
		41	6.7302000
Ca	20	40	96.9410000
		42	0.6470000
		43	0.1350000
		44	2.0860000
		46	0.0040000
		48	0.1870000
Ti	22	46	8.0000000
		47	7.3000000
		48	73.8000000
		49	5.5000000
		50	5.4000000
V	23	50	0.2500000
		51	99.7500000
Cr	24	50	4.3450000
		52	83.7900000
		53	9.5000000
		54	2.3650000
Mn	25	55	100.0000000
Fe	26	54	5.9000000
		56	91.7200000
		57	2.1000000
		58	0.2800000
Co	27	59	100.0000000

Ni	28	58	68.2700000
		60	26.1000000
		61	1.1300000
		62	3.5900000
		64	0.9100000
Cu	29	63	69.1700000
		65	30.8300000
As	33	75	100.0000000
Zr	40	90	51.4500000
		91	11.2200000
		92	17.1500000
		94	17.3800000
		96	2.8000000
Nb	41	93	100.0000000
Mo	42	92	14.8400000
		94	9.2500000
		95	15.9200000
		96	16.6800000
		97	9.5500000
		98	24.1300000
		100	9.6300000
Ag	47	107	51.8390000
		109	48.1610000
Cd	48	106	1.2500000
		108	0.8900000
		110	12.4900000
		111	12.8000000
		112	24.1300000
		113	12.2200000
		114	28.7300000
		116	7.4900000

Sn	50	112	0.9700000
		114	0.6500000
		115	0.3600000
		116	14.5300000
		117	7.6800000
		118	24.2200000
		119	8.5800000
		120	32.5900000
		122	4.6300000
		124	5.7900000
Sb	51	121	57.4000000
		123	42.6000000
Ba	56	130	0.1060000
		132	0.1010000
		134	2.4200000
		135	6.5930000
		136	7.8500000
		137	11.2300000
		138	71.7000000
Tb	65	159	100.0000000
Ta	73	180	0.0120000
		181	99.9880000
W	74	180	0.1200000
		182	26.3000000
		183	14.2800000
		184	30.7000000
		186	28.6000000
Ir	77	191	37.3000000
		193	62.7000000
Pb	82	204	1.4000000
		206	24.1000000
		207	22.1000000
		208	52.4000000
Bi	83	209	100.0000000

Information Requested

Codes and Data:

1- Activation code used.
2- Activation data evaluation used.
3- Processing codes used to generate the activation library.
4- Energy group structure.
5- Weight function used to generate multi-group data.
6- Decay data library used.
7- Computing facility and CPU time used.

Calculation Results:

1- The specific activity (Bq/m^3) in the non-void zones at cooling times of 0, 1 hour, 1 day, 1 week, 1 month, 1 year, and 100 years after end of full reactor operation. The ten major contributing nuclides should be identified and their contributions provided.

2- The specific decay heat (W/m^3) in the non-void zones at cooling times of 0, 1 hour, 1 day, 1 week, 1 month, 1 year, and 100 years after end of full reactor operation. The ten major contributing nuclides should be identified and their contributions provided.

3- The biological dose rate (micro Sv/hr) at the back of the outboard shield at cooling times of 0, 1 hour, 1 day, 1 week, and 1 month after end of full reactor operation.