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INTERNATIONAL NUCLEAR DATA COMMITTEE

FENDL NEUTRONICS BENCHMARK:

STAINLESS STEEL BULK SHIELD EXPERIMENT PERFORMED AT FRASCATI NEUTRON GENERATOR

Authors:
M. Martone, M. Angelone, P. Batistoni, M. Pillon, V. Rado
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ENEA - Ente per le Nuove tecnologie, l'Energia e l'Ambiente
C.R. Frascati - I-00044 FRASCATI (Italy)

December 1994

IAEA NUCLEAR DATA SECTION, WAGRAMERSTRASSE 5, A-1400 VIENNA

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Abstract

Following the recommendations of an IAEA Consultant's Meeting on "Preparation of Fusion Benchmarks in Electronic Format for Nuclear Data Validation Studies", the present report on benchmark experiments with 14-MeV neutrons on a stainless steel block was prepared. It complements the experimental data available on-line from the IAEA Nuclear Data Section so as to enable any user to perform transport calculations for this experiment in order to validate nuclear data libraries, such as the Fusion Evaluated Nuclear Data Library (FENDL).

December 1994

Reproduced by the IAEA in Austria
December 1994

Introduction

(written by U. von Möllendorff, Kernforschungszentrum Karlsruhe, Germany, and 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.

The present report which follows is such material. It complements the following file that is available by ftp command on-line from the IAEA:

ftp 161.5.2.2
user FENDL
cd FENDL/BENCHMARKS/ENEACEA
The file "FNG.TXT" corresponds to this document.

Revised on April 6,1994

SS Bulk Shield Benchmark Experiment at FNG
Compilation for IAEA/NDS

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ENEA - Ente per le Nuove tecnologie, l'Energia e l'Ambiente
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The present document accompanies a 3.5" floppy disk named "SS Bulk Shield Exp. at FNG/ENEA". All the files, listed below, are written with WORD for Macintosh (vers.4).

List of files	Description of contents
README.DOC	This text
INFO.DOC	General information on the benchmark experiment.
SOURCE.DATA	Source neutron spectrum averaged over a spherical cap of 60 degree aperture. Format of the data: Energy, upper group limit [MeV] I=1,172 Current spectrum [n/sr/source] I=1,173 (Total in 173') Errors (fractions) [%] I=1,172 (Error on total in 173') Spectrum normalized to one I=1,172 (Total in 173')
SOURCE.FORT	Fortran routine used by MCNP.4 to calculate the FNG 14-MeV neutron source with the proper energy/angle distribution.
GEOM.DATA	Geometrical data of the experiment given in MCNP input format.
DETEC.DATA	List of activation reactions used, foil locations inside the block and foil size. Information about detectors and measurements.
EXPE.DATA	Experimental data given for every reaction as follows: Foil position (penetration depth inside the block), [cm] E = measured reaction rate, [10^{24} ractions/source] random error on E, (fraction) [%] systematic error on E, (fraction) [%] total error on E, (quadratic sum of random and systematic error, fraction) [%]

CALC.DATA

Calculated data given for every reaction as follows:
Foil position (penetration depth inside the block), [cm]
C = calculated reaction rate [10^{24} ractions/source]
error on C due to MCNP statistics, (fraction) [%]
error on C due to uncertainty on activation cross section, (fraction) [%]
total error on C, (quadatic sum of two previous errors, fraction) [%]
C/E ratios,
total errors on C/E ratios (quadratic sum of total error on C and of total error on E, absolute)

INFO.DOC STARTS **

Measurements

The experiment consists of the irradiation of a stainless steel (SS) block by 14-MeV neutrons. The experiment is carried out at FNG (Frascati Neutron Generator) facility [1].

The absolute value of the neutron source intensity is measured by the associated particle method, by means of a SSB detector. The accuracy of the measurement improved during the experimental period from ±4.4% to ±1.6% .

The block lateral size is 100x100 cm^2, the thickness (in the z direction,where the z axis corresponds to the axis of the accelerator tube) is 70 cm. The SS is AISI316 type; the chemical composition is given in MCNP format in GEOM.DATA (M1).

The block is located at 5.3 cm distance (along the z axis) from the neutron source. Of these 5.3 cm, 5 cm are in air and the remaining 0.3 cm are due to the target support structure (1 mm Cu, 1 mm H_2O, and 1 mm SS). The assembly geometry is described in MCNP format in GEOM.DATA, the origin of coordinates is located on the neutron source. Te block is positioned over an aluminum support which is included in GEOM.DATA, as well as the bunker walls.

The current spectrum of the neutron source is given in SOURCE.DATA. This current spectrum is calculated with MCNP and represents the current averaged over a solid angle $\Delta\Omega/4\Pi=0.25$ in the z direction toward the block; this spectrum was calculated by MCNP including the target zones (TARGET ZONES only in GEOM.DATA) and using the subroutine SOURCE.FORT .

In the experiment, activation foils were located in different locations inside the block along the block central (horizontal) axis, corresponding to the z axis. Only foils of the same material were irradiated during every single irradiation. The foil location is everywhere specified (EXPE.DATA, CALC.DATA) by the penetration depth, i.e. the distance of the centre of the foil from the block surface exposed to the neutron source, in cm. The activation reaction used and the foil dimensions are given in DETEC.DATA. The nuclear data used are also given in DETEC.DATA and are taken from [2].
Soon after each irradiation, the foil activity is recorded by a set of calibrated HPGe detectors, also described in DETEC.DATA.

The measured reaction rates are given in EXPE.DATA in units of 10^{24} reactions/(source neutron). Here, the random errors include the γ-ray counting statistics and the uncertainty on HPGe calibration. Other errors (f.i. on foil mass and nuclear data) are negligible. The systematic error is the uncertainty on the neutron source intensity.

Calculations

The calculated reaction rates are given in CALC.DATA in units of 10^{24} reactions/(source neutron). These data were obtained with MCNP.4, with the geometry input given in GEOM.DATA.

The source was provided by the subroutine SOURCE.FORT, with the proper angle/energy distribution, i.e. taking into account the reaction kinematics and the slowing down of beam deuterons in the tritium/titanium target.

Transport cross sections from EFF.1 were used; the activation cross sections (for foil detectors) were taken from IRDF.90, with the exception of $^{55}Mn(n,\gamma)$ which was taken from EFF.2.

The track length tally (F4 in MCNP, flux averaged over a cell) was used to calculate the reaction rates.

The detector responses were calculated in a MCNP model including everything but "BUNKER AND WALLS" in the input GEOM.DATA. The room background contribution to the detector responses was estimated by means of a surface source at the block external boundaries. This surface source was recorded once in a dedicated MCNP run including the bunker walls in the geometry.
The finding is that the background is not negligible on Mn and Au foils located close to the block front surface and amounts to 8% at 5 cm, to 6% at 10 cm for ^{197}Au(n,γ), and to 5% at 5 cm and to 4% at 10 cm for ^{55}Mn(n,γ). This background contributions are included in the corresponding C values.

The error on reaction rate due to uncertainty on activation cross sections was calculated by variance analysis using the IRDF-90 covariance data in 175 energy group format, except for ^{55}Mn(n,γ) for which the EFF.2 covariance file has been used.
The random errors on calculated reaction rates, due to the statistics in the MCNP calculations and to uncertainty on activation cross sections, are separately given in CALC.DATA.The total error is the quadratic sum of the two.

C/E values in CALC.DATA are the ratio of calculated over measured reaction rates. The total error on C/E is the quadratic sum of the total error on C and of the total error on E.

References

[1] M. Martone, M. Angelone, M. Pillon, "The 14-MeV Frascati Neutron Generator (FNG)", ENEA Report RT/ERG/FUS/93/65
[2] J. H. Baard, W. L. Zijp, H. Nolthenius, "Nuclear Data Guide for Reactor Neutron Metrology", Kluwer Academic Publishers for the Commission of the European Community, (1989).

INFO.DOC ENDS**

SOURCE.DATA STARTS**

TARGET SOURCE SPECTRUM FOR FNG BENCHMARK ANALYSIS
UPPER LIMITS OF 172 ENERGY GROUPS (MeV)
1.0000E-07 4.1399E-07 5.3158E-07 6.8256E-07 8.7642E-07 1.1254E-06
1.4450E-06 1.8554E-06 2.3824E-06 3.0590E-06 3.9279E-06 5.0435E-06
6.4760E-06 8.3153E-06 1.0677E-05 1.3710E-05 1.7603E-05 2.2603E-05
2.9023E-05 3.7267E-05 4.7851E-05 6.1442E-05 7.8893E-05 1.0130E-04
1.3007E-04 1.6702E-04 2.1445E-04 2.7536E-04 3.5358E-04 4.5400E-04
5.8295E-04 7.4852E-04 9.6112E-04 1.2341E-03 1.5846E-03 2.0347E-03
2.2487E-03 2.4852E-03 2.6126E-03 2.7465E-03 3.0354E-03 3.3546E-03
3.7074E-03 4.3074E-03 5.5308E-03 7.1017E-03 9.1188E-03 1.0595E-02
1.1709E-02 1.5034E-02 1.9305E-02 2.1875E-02 2.3579E-02 2.4176E-02
2.4788E-02 2.6058E-02 2.7000E-02 2.8500E-02 3.1828E-02 3.4307E-02
4.0868E-02 4.6309E-02 5.2475E-02 5.6562E-02 6.7379E-02 7.2000E-02
7.9500E-02 8.2500E-02 8.6517E-02 9.8037E-02 1.1109E-01 1.1679E-01
1.2277E-01 1.2907E-01 1.3569E-01 1.4264E-01 1.4996E-01 1.5764E-01
1.6573E-01 1.7422E-01 1.8316E-01 1.9255E-01 2.0242E-01 2.1280E-01
2.2371E-01 2.3518E-01 2.4724E-01 2.7324E-01 2.8725E-01 2.9452E-01
2.9720E-01 2.9850E-01 3.0197E-01 3.3373E-01 3.6883E-01 3.8774E-01
4.0762E-01 4.5049E-01 4.9787E-01 5.2340E-01 5.5023E-01 5.7844E-01
6.0810E-01 6.3928E-01 6.7206E-01 7.0651E-01 7.4274E-01 7.8082E-01
8.2085E-01 8.6294E-01 9.0718E-01 9.6164E-01 1.0026E+00 1.1080E+00
1.1648E+00 1.2246E+00 1.2873E+00 1.3534E+00 1.4227E+00 1.4957E+00
1.5724E+00 1.6530E+00 1.7377E+00 1.8268E+00 1.9205E+00 2.0190E+00
2.1225E+00 2.2313E+00 2.3069E+00 2.3457E+00 2.3653E+00 2.3852E+00
2.4660E+00 2.5924E+00 2.7253E+00 2.8650E+00 3.0119E+00 3.1664E+00
3.3287E+00 3.6788E+00 4.0657E+00 4.4933E+00 4.7237E+00 4.9659E+00
5.2205E+00 5.4881E+00 5.7695E+00 6.0653E+00 6.3763E+00 6.5924E+00
6.7032E+00 7.0469E+00 7.4082E+00 7.7880E+00 8.1873E+00 8.6071E+00
9.0484E+00 9.5123E+00 1.0000E+01 1.0513E+01 1.1052E+01 1.1618E+01
1.2214E+01 1.2523E+01 1.2840E+01 1.3499E+01 1.3840E+01 1.4191E+01
1.4550E+01 1.4918E+01 1.5683E+01 2.0000E+01
NEUTRON CURRENT SPECTRUM IN A FORWARD CONE (60 DEGREES)
NORMALIZED TO ONE T(d,n) SOURCE NEUTRON -TOTAL IN 173'
6.5591E-06 4.4179E-06 8.0958E-07 9.1309E-07 9.1884E-07 9.6500E-07
1.0114E-06 1.1154E-06 1.2237E-06 1.3108E-06 1.4112E-06 1.4838E-06
1.5535E-06 1.7038E-06 1.8762E-06 1.9407E-06 2.1355E-06 2.3048E-06
2.4412E-06 2.8678E-06 2.8678E-06 3.1126E-06 3.4252E-06 4.0426E-06
4.2461E-06 4.4325E-06 4.7454E-06 5.4300E-06 5.4716E-06 6.1622E-06
7.0677E-06 8.1440E-06 8.7192E-06 8.9465E-06 1.0464E-05 1.1719E-05
5.4402E-06 5.4693E-06 2.7270E-06 2.6946E-06 5.4509E-06 6.4994E-06
6.7301E-06 1.0624E-05 2.0435E-05 2.3428E-05 2.3463E-05 1.7701E-05
1.1754E-05 3.5744E-05 3.8062E-05 2.3105E-05 1.3763E-05 4.4783E-06
6.0150E-06 1.3327E-05 9.8470E-06 9.9675E-06 2.2808E-05 1.9506E-05
4.7725E-05 3.5946E-05 4.4245E-05 2.8025E-05 6.8648E-05 2.9789E-05
4.7308E-05 2.1352E-05 2.3171E-05 7.3814E-05 1.0944E-04 4.6592E-05
5.5823E-05 5.7413E-05 6.5609E-05 6.4925E-05 5.6205E-05 6.6986E-05
7.2167E-05 7.5154E-05 8.2986E-05 8.6404E-05 8.8387E-05 1.0009E-04
1.0211E-04 1.0993E-04 1.1259E-04 2.5728E-04 1.3888E-04 6.5248E-05
2.6246E-05 1.1403E-05 3.3778E-05 3.1236E-04 3.3420E-04 1.9414E-04
1.9619E-04 3.9677E-04 4.6825E-04 2.5423E-04 2.5724E-04 2.7686E-04
2.8432E-04 2.7109E-04 3.0109E-04 3.0842E-04 3.2908E-04 3.5276E-04
3.4309E-04 3.5131E-04 3.6087E-04 4.4522E-04 3.1041E-04 7.4156E-04
4.0087E-04 4.2835E-04 4.3828E-04 4.2187E-04 4.4656E-04 4.4736E-04
4.2258E-04 4.2621E-04 4.2545E-04 4.4223E-04 4.2258E-04 4.2672E-04
4.1197E-04 4.3274E-04 2.8650E-04 1.4580E-04 7.5776E-05 7.8547E-05
2.9492E-04 4.2068E-04 4.1583E-04 4.1225E-04 4.2165E-04 3.9958E-04

3.9918E-04	7.6998E-04	7.3158E-04	6.9208E-04	3.0011E-04	3.1544E-04
3.0621E-04	2.9132E-04	2.6869E-04	2.9419E-04	3.0359E-04	1.9828E-04
1.0978E-04	3.1146E-04	2.8778E-04	3.2377E-04	3.0590E-04	2.8780E-04
2.8675E-04	2.9709E-04	2.4957E-04	2.4505E-04	2.3459E-04	2.4696E-04
3.0490E-04	2.8192E-04	5.1734E-04	9.3428E-04	7.6304E-04	1.3539E-03
5.8329E-02	1.4869E-01	4.0895E-02	0.0000E+00	2.7894E-01	

FRACTIONAL STANDARD DEVIATIONS

0.012	0.012	0.017	0.045	0.019	0.019
0.020	0.021	0.022	0.021	0.022	0.021
0.021	0.021	0.021	0.025	0.023	0.023
0.023	0.028	0.023	0.025	0.027	0.048
0.028	0.046	0.032	0.038	0.027	0.030
0.036	0.043	0.041	0.033	0.036	0.038
0.059	0.056	0.072	0.080	0.048	0.065
0.059	0.041	0.033	0.034	0.033	0.042
0.052	0.034	0.034	0.047	0.061	0.099
0.102	0.062	0.062	0.063	0.046	0.054
0.035	0.041	0.037	0.048	0.032	0.050
0.040	0.057	0.056	0.031	0.031	0.048
0.044	0.044	0.041	0.040	0.043	0.040
0.040	0.039	0.037	0.035	0.036	0.034
0.034	0.034	0.032	0.022	0.030	0.044
0.068	0.104	0.060	0.020	0.020	0.026
0.026	0.018	0.017	0.023	0.023	0.023
0.023	0.023	0.022	0.021	0.021	0.020
0.020	0.020	0.020	0.018	0.022	0.015
0.020	0.019	0.019	0.019	0.019	0.019
0.020	0.019	0.019	0.019	0.020	0.020
0.020	0.019	0.024	0.033	0.046	0.046
0.024	0.020	0.020	0.020	0.020	0.020
0.020	0.015	0.015	0.016	0.024	0.023
0.023	0.024	0.025	0.024	0.024	0.029
0.039	0.023	0.024	0.023	0.024	0.025
0.025	0.024	0.027	0.027	0.028	0.027
0.024	0.025	0.019	0.014	0.016	0.012
0.002	0.001	0.002	0.000	0.001	

SPECTRUM NORMALIZED TO 1.0

2.3514E-05	1.5838E-05	2.9023E-06	3.2734E-06	3.2940E-06	3.4595E-06
3.6259E-06	3.9987E-06	4.3870E-06	4.6991E-06	5.0590E-06	5.3195E-06
5.5691E-06	6.1082E-06	6.7262E-06	6.9575E-06	7.6558E-06	8.2626E-06
8.7515E-06	1.0281E-05	1.0281E-05	1.1159E-05	1.2279E-05	1.4492E-05
1.5222E-05	1.5890E-05	1.7012E-05	1.9466E-05	1.9615E-05	2.2091E-05
2.5337E-05	2.9196E-05	3.1258E-05	3.2073E-05	3.7513E-05	4.2011E-05
1.9503E-05	1.9607E-05	9.7762E-06	9.6599E-06	1.9541E-05	2.3300E-05
2.4127E-05	3.8087E-05	7.3259E-05	8.3989E-05	8.4113E-05	6.3456E-05
4.2139E-05	1.2814E-04	1.3645E-04	8.2831E-05	4.9339E-05	1.6055E-05
2.1563E-05	4.7775E-05	3.5301E-05	3.5733E-05	8.1765E-05	6.9928E-05
1.7109E-04	1.2886E-04	1.5862E-04	1.0047E-04	2.4610E-04	1.0679E-04
1.6960E-04	7.6546E-05	8.3069E-05	2.6462E-04	3.9233E-04	1.6703E-04
2.0012E-04	2.0582E-04	2.3521E-04	2.3275E-04	2.0149E-04	2.4014E-04
2.5872E-04	2.6942E-04	2.9750E-04	3.0975E-04	3.1686E-04	3.5881E-04
3.6607E-04	3.9408E-04	4.0365E-04	9.2234E-04	4.9788E-04	2.3391E-04
9.4090E-05	4.0878E-05	1.2109E-04	1.1198E-03	1.1981E-03	6.9597E-04
7.0332E-04	1.4224E-03	1.6786E-03	9.1139E-04	9.2221E-04	9.9254E-04
1.0193E-03	9.7185E-04	1.0794E-03	1.1057E-03	1.1797E-03	1.2646E-03
1.2300E-03	1.2594E-03	1.2937E-03	1.5961E-03	1.1128E-03	2.6585E-03
1.4371E-03	1.5356E-03	1.5712E-03	1.5124E-03	1.6009E-03	1.6037E-03
1.5149E-03	1.5279E-03	1.5252E-03	1.5854E-03	1.5149E-03	1.5298E-03

1.4769E-03 1.5513E-03 1.0271E-03 5.2268E-04 2.7165E-04 2.8159E-04
1.0573E-03 1.5081E-03 1.4907E-03 1.4779E-03 1.5116E-03 1.4325E-03
1.4310E-03 2.7603E-03 2.6227E-03 2.4811E-03 1.0759E-03 1.1308E-03
1.0977E-03 1.0444E-03 9.6323E-04 1.0547E-03 1.0884E-03 7.1084E-04
3.9355E-04 1.1166E-03 1.0317E-03 1.1607E-03 1.0966E-03 1.0317E-03
1.0280E-03 1.0651E-03 8.9468E-04 8.7850E-04 8.4100E-04 8.8532E-04
1.0931E-03 1.0107E-03 1.8547E-03 3.3494E-03 2.7355E-03 4.8536E-03
2.0911E-01 5.3305E-01 1.4661E-01 0.0000E+00 1.0000E+00

SOURCE.DATA ENDS***

SOURCE.FORT STARTS**

USED BY MCNP.4 TO GENERATE 14 MeV NEUTRONS WITH THE PROPER ENERGY /ANGLE DISTRIBUTION. FOR THE BENCHMARK ANALYSIS, USED INPUT PARAMETER VALUES ARE EB=0.230 (BEAM ENERGY IN MeV), XT=1.5 (TRITIUM ATOMS/TITANIUM ATOMS) AND ARE GIVEN IN THE RDUM CARD IN THE MCNP INPUT FILE

```
      SUBROUTINE SOURCE
C       DUMMY SUBROUTINE.  ABORTS JOB IF SOURCE SUBROUTINE IS MISSING.
C       IF NSR=0, SUBROUTINE SOURCE MUST BE FURNISHED BY THE USER.
C       AT ENTRANCE, A RANDOM SET OF UUU,VVV,WWW HAS BEEN DEFINED.  THE
C       FOLLOWING VARIABLES MUST BE DEFINED WITHIN THE SUBROUTINE:
C        XXX,YYY,ZZZ,ICL,JSU,ERG,WGT,TME AND POSSIBLY IPT,UUU,VVV,WWW.
C       SUBROUTINE SRCDX MAY ALSO BE NEEDED.
C
      IMPLICIT DOUBLE PRECISION (A-H,O-Z)
C
      PARAMETER
(MAXF=16,MAXI=34,MAXV=19,MAXW=3,MCPU=32,MINK=200,MIPT=3,
     1 MJSF=9,MKFT=9,MKTC=22,MLGC=100,MPB=5,MPNG=21,MRKP=100,MSEB=301,
     2
MSPARE=3,MTOP=49,MWNG=25,MXDT=20,MXDX=5,MXLV=10,NBMX=100,NDEF=14,
     3 NOVR=5,IUI=31,IUO=32,IUR=33,IUX=34,IUD=35,IUB=60,IUP=37,IUS=38,
     4 IU1=39,IU2=40,IUSW=41,IUSR=42,IUSC=43,IUC=44,IUT=45,IUZ=46,
     5 IUK=47,IU3=48,IU4=49,ZERO=0.,ONE=1.,THIRD=ONE/3.,
     6 PIE=3.1415926535898D0,AVGDN=.59703109D0,SLITE=299.7925,
     7 FSCON=137.0393)
C
C -------------------------------------------------------------------
C
C
C       VARIABLE COMMON -- VARIABLE BUT REQUIRED FOR A CONTINUE RUN.
C       ARRAYS THAT ARE BACKED UP WHEN A TRACK IS LOST.
      COMMON /VARCOM/ CPK,CTS,DBCN(20),DMP,EACC(4),FEBL(2,16),OSUM(3),
     1 OSUM2(3,3),PAX(6,16,MIPT),PRN,RANI,RANJ,RDUM(50),RIJK,RKK,
     2 RLT(2),RNR,RSUM(2),RSUM2(2,2),SMUL(3),SUMK(3),TMAV(MIPT,3),
     3 TWAC,TWSS,WCS1(MIPT),WCS2(MIPT),WGTS(2),WT0,WSSI(7),
     4 ZVARCM,
     5 IDUM(50),INIF,IST,IST0,IXAK,IXAK0,JRAD,KCSF,KCT,KCY,KNOD,
     6 KSDEF,KZKF,LOST(2),NBAL(MCPU),NBHWM,NBOV,NBT(MIPT),NBY,NCT(MIPT),
     7 NDMP,NERR,NETB(2),NFER,NPC(20),NPD,NPNM,NPP,NPPM,NPS,NPSOUT,NPSR,
     8 NQSS,NQSW,NRNH(3),NRRS,NRSW,NSA,NSA0,NSKK,NSS,NSS0,NSSI(8),NTC,
     9 NTC1,NTSS,NWER,NWSB,NWSE,NWSG(2),NWST,NWWS(2,99),NZIP,NZIX,
     1 NZIY(8,MXDX,MIPT),
     2 MVARCM
      COMMON /PBLCOM/ XXX,YYY,ZZZ,UUU,VVV,WWW,ERG,WGT,TME,VEL,DLS,
     1 DXL,DTC,ELC(MIPT),FIML(MIPT),FISMG,WTFASV,RNK,SPARE(MSPARE),
     2 ZPBLCM,
     3 XXX9(MPB),YYY9(MPB),ZZZ9(MPB),UUU9(MPB),VVV9(MPB),WWW9(MPB),
     4 ERG9(MPB),WGT9(MPB),TME9(MPB),VEL9(MPB),DLS9(MPB),DXL9(MPB),
     5 DTC9(MPB),ELC9(MPB,MIPT),FIML9(MPB,MIPT),FISMG9(MPB),
     6 WTFAS9(MPB),RNK9(MPB),SPARE9(MPB,MSPARE),
     7 ZPB9CM(MPB),
     1 NPA,ICL,JSU,IPT,IEX,NODE,IDX,NCP,JGP,LEV,III,JJJ,KKK,IAP,
     2 MPBLCM,
     3 NPA9(MPB),ICL9(MPB),JSU9(MPB),IPT9(MPB),IEX9(MPB),NODE9(MPB),
     4 IDX9(MPB),NCP9(MPB),JGP9(MPB),LEV9(MPB),III9(MPB),JJJ9(MPB),
     5 KKK9(MPB),IAP9(MPB),
```

```
  6 MPB9CM(MPB)
C
   COMMON /MARIO/ DEDX(100),EMIN,EB,XT,SML,NENE
C ********************************************************************
C      STANDARD  DATA
C ********************************************************************
   DIMENSION CDEDX(100),SIG(100),TDEDX(100),PL(100)
   DIMENSION ED(100),SUML(3601),TH(3601)
   REAL MD,MT,MN,MA
   DATA MD/2.01410219/
   DATA MT/3.01602994/
   DATA MN/1.00866544/
   DATA MA/4.00260361/
   DATA AUX/25.20734546/
   DATA Q/17.589/
C
C  INPUT DATA FOR THIS PROGRAM ARE:
C  TRITIUM ATOM PER ONE TITANIUM ATOM
C  DEUTERON / TRITONS ENERGY IN MeV, UP TO 0.5 MEV;
C ********************************************************************
C        DATA
C ********************************************************************
   DATA NPTS/50/
    DATA ED/0.010,0.020,0.030,0.040,
     *0.050,0.060,0.070,0.080,0.090,0.100,0.110,0.120,0.130,0.140,
     *0.150,0.160,0.170,0.180,0.190,0.200,0.210,0.220,0.230,0.240,0.250,
     *0.260,0.270,0.280,0.290,0.300,0.310,0.320,0.330,0.340,0.350,0.360,
     *0.370,0.380,0.390,0.400,0.410,0.420,0.430,0.440,0.450,0.460,0.470,
    *0.480,0.490,0.500,50*0.0/
C
C T(d,n) DEUTERON ON TRITIUM CROSS SECTION
C
     DATA  SIG/1.0E-4,4.3E-3,0.0196,0.0529,0.106,0.175,0.250,0.315,
     *0.367,0.394,0.399,0.387,0.367,0.339,0.317,0.286,0.262,0.236,0.215,
     *0.199,0.181,0.167,0.153,0.142,0.133,0.122,0.114,0.106,0.100,0.0952
     *,0.0911,0.0873,0.0838,0.0806,0.0775,0.0743,0.0713,0.0686,0.0659,
    *0.0635,0.0611,0.0589,0.0568,0.0549,0.0530,0.0513,0.0498,0.0483,
    *0.0469,0.0455,50*0.0/
C
C  ENERGY LOSS OF DEUTERONS IN TITANIUM AND TRITIUM
C
    DATA  CDEDX/0.142,0.1929,0.2299,0.2592,0.2835,0.3038,0.3209,
     *0.3354,0.3475,0.3577,0.3660,0.3728,0.3781,0.3821,0.3851,0.3870,
     *0.3880,0.3883,0.3879,0.3869,0.3854,0.3834,0.3811,0.3785,0.3756,
     *0.3725,0.3692,0.3658,0.3622,0.3586,0.3550,0.3513,0.3476,0.3439,
     *0.3402,0.3365,0.3329,0.3293,0.3258,0.3223,0.3189,0.3155,0.3122,
     *0.3089,0.3058,0.3027,0.2996,0.2966,0.2937,0.2909,50*0.0/
    DATA  TDEDX/0.5959,0.8014,0.9420,1.046,1.124,1.182,1.225,1.255,
     *1.275,1.287,1.291,1.290,1.285,1.275,1.262,1.247,1.231,1.212,
     *1.193,1.173,1.152,1.132,1.111,1.090,1.069,1.048,1.028,1.008,
     *0.9886,0.9695,0.9509,0.9327,0.9150,0.8978,0.8811,0.8649,0.8491,
     *0.8338,0.8190,0.8046,0.7906,0.7771,0.7640,0.7513,0.7390,0.7270,
    *0.7155,0.7042,0.6934,0.6828,50*0.0/
C
C ********************************************************************
   IF(SML.GT.0) GO TO 1111
   EMIN=0.010
C BEAM ENERGY IN MeV, UP TO 0.5 MEV;
```

```
      EB=RDUM(1)
C  TRITIUM -TITANIUM RATIO
      XT=RDUM(2)
C SOURCE COORDINATES
      XX=RDUM(3)
      YY=RDUM(4)
      ZZ=RDUM(5)
C ***********************************************************************
      DO 100 I=1,NPTS
C   STOPPING POWER
 100  DEDX(I)=CDEDX(I)*48./(48.+3.*XT)+TDEDX(I)*3.*XT/(48.+3.*XT)
      THL=-0.05
      DO 1000 L=1,3601
      THL=THL+0.05
      TH(L)=THL*3.1415927/180.
      CTH=COS(TH(L))
      NENE=0
      DO 250 I=1,NPTS
      IF(ED(I).LE.EB) NENE=I
      ET=ED(I)+Q
      B=MD*MN*ED(I)/ET/AUX
      D=MT*MA/AUX*(1.+MD*Q/(MT*ET))
      E=SQRT(D/B-1.+CTH*CTH)
      EN=ET*B*(CTH+E)**2
      G=SQRT(B*D)*E/(EN/ET)
 250  PL(I)=SIG(I)/G/DEDX(I)
      CALL XINT(ED,PL,NPTS,EMIN,EB,XR)
      SUML(L)=XR*SIN(TH(L))
 1000 CONTINUE
      CALL XINT(TH,SUML,3601,TH(1),TH(3601),TI)
      SML=TI
 1111 ICL=1
      JSU=0
      IPT=1
      XXX=XX
      YYY=YY
      ZZZ=ZZ
      TME=0.0
C 60  PMAX=1.1
 60  VVV= 2.*RANG()-1.
      I=INT(NENE*RANG())+1
      DELTA= ED(I+1)-ED(I)
      DELTE=RANG()*DELTA
      ET=ED(I)+DELTE+Q
C**** ET=ED(I)+Q
C     PMAX=PMAX*RANG()
      PMAX=RANG()
      B=MD*MN*(ED(I)+DELTE)/ET/AUX
C**** B=MD*MN*ED(I)/ET/AUX
      D=MT*MA/AUX*(1.+MD*Q/(MT*ET))
      E=SQRT(D/B-1.+VVV*VVV)
      EN=ET*B*(VVV+E)**2
      G=SQRT(B*D)*E/(EN/ET)
      XIN=ED(I)+DELTE
      CALL INTERP(ED,SIG,NPTS,XIN,SIGOUT)
      CALL INTERP(ED,DEDX,NPTS,XIN,DEDXOT)
      P=SIGOUT/G/DEDXOT
C**** P=SIG(I)/G/DEDX(I)
```

```
C**********************************************************************
C  ENEA
C     WRITE(6,*) NENE,P
C  ENEA
      IF(P.GE.1.0) WRITE(6,32)
   32 FORMAT(5X,' A T T E N Z I O N E   P .GE. 1.0')
      IF(PMAX.GT.P) GO TO 60
      ERG=EN
      ST=SQRT(1.-VVV*VVV)
   70 X1=2.*RANG()-1.
      X2=2.*RANG()-1.
      X11=X1*X1
      X22=X2*X2
      RO=X11+X22
      IF(RO.GT.1.0) GO TO 70
      SF=2*X1*X2/RO
      CF=(X11-X22)/RO
      UUU=ST*SF
      WWW=ST*CF
      WGT=1.
      RETURN
      END
C SUBROUTINE FOR INTERPOLATION
      SUBROUTINE INTERP(X,Y,NPTS,XIN,YOUT)
      IMPLICIT DOUBLE PRECISION (A-H,O-Z)
C -------------------------------------------------------------------
C
      DIMENSION X(NPTS),Y(NPTS)
      I=1
      IF(X(I)-XIN)2,10,12
    2 DO 1 I=2,NPTS
      IF(X(I)-XIN)1,10,11
    1 CONTINUE
      GO TO 12
   11 YOUT=Y(I-1)+(Y(I)-Y(I-1))/(X(I)-X(I-1))*ABS(X(I)-XIN)
      RETURN
   10 YOUT=Y(I)
      RETURN
   12 WRITE(6,100)
  100 FORMAT(20X,'ERROR  INTERPOLATION  '/)
      STOP
      END
C SUBROUTINE FOR NUMERICAL INTEGRATION
      SUBROUTINE XINT(E,F,NPTS,E1,E2,SUM)
      IMPLICIT DOUBLE PRECISION (A-H,O-Z)
C
C -------------------------------------------------------------------
C
      DIMENSION E(NPTS),F(NPTS)
      SUM=0
      L=1
      IF(E2-E1)50,55,60
   50 A=E2
      E2=E1
      E1=A
      L=-1
   60 I=1
      IF(E(I)-E1)2,10,12
```

```
 2  DO 1 I=2,NPTS
    IF(E(I)-E1)1,10,11
 1  CONTINUE
    GO TO 12
 11  CALL INTERP(E,F,NPTS,E1,FOUT)
    SUM=SUM+(E(I)-E1)*(F(I)+FOUT)*0.5
 10  KMIN=I
    J=1
    IF(E(J)-E2)3,13,12
 3  DO 4 J=2,NPTS
    IF(E(J)-E2)4,13,15
 4  CONTINUE
    GO TO 12
 15  CALL INTERP(E,F,NPTS,E2,FOUT)
    SUM=SUM+(E2-E(J-1))*(F(J-1)+FOUT)*0.5
    J=J-1
 13  KMAX=J-1
    IF(KMIN.GT.KMAX)GO TO 6
    DO 5 K=KMIN,KMAX
  5  SUM=SUM+(E(K+1)-E(K))*(F(K+1)+F(K))*0.5
 6  SUM=SUM*L
 55  RETURN
 12  WRITE(6,100)
100  FORMAT(20X,'ERROR  INTEGRAL '/)
    STOP
    END

SOURCE.FORT ENDS*************************************************************
```

```
GEOM.DATA STARTS***********************************************************

GEOMETRICAL DATA FOR "SS BULK SHIELD EXPERIMENT AT FNG"
THE NEUTRON SOURCE HAS COORDINATES X=0, Y=0, Z=0
INPUT MCNP4 (GEOMETRY)

C      TARGET ZONES
  1  3  -8.94      6  (1:7) -2  -8
  2  2  -1.0       6  8    -2  -9
  3  1  -7.954     6  9    -5  -10
  4  1  -7.954     5  -2    9  -11
  5  2  -1.0       2  -3       -11
  6  1  -7.954     3  -4       -11
  7  1  -7.954    -6  17    7  -10
  8  0            -1  17   -7
  9  1  -7.954    -6  18   10  -11
 10  1  -7.954    11 -16  12 -13
 11  1  -7.954    11 -16 -14  15
 12  2  -1.0      11 -16  13  14 -19
 13  4  4.614E-5  6  -5  10 -11
 14  4  4.614E-5 17 -18  10 -11
 15  4  4.614E-5 11 -16 (-12:-15)
 16  4  4.614E-5  4 -100 -11
 17  4  4.614E-5  6 -100  11 16 -19
 18  4  4.614E-5 -6  17  11 20  21 -19
 19  2  -1.0     -6  17  -20
 20  2  -1.0     -6  17  -21
 21  4  4.614E-5      17  -100 102 -103 104 -105 19
C    SS BLOCK
 100 1  -7.954   100 -101 102 -103 104 -105
C    DRIFT TUBE AND OTHER ACCELERATOR STRUCTURES
 22  1  -7.954   36 (22:23) -17 -19 7
 23  0           22  -17 -7
 24  1  -7.954  -22 26 24 -25
 25  0          -22 26  -24
 26  1  -0.07954 -22 26 (-23 36:-19 -36) 25
 27  7  -8.4     -26 27  -29
 28  4  4.614E-5  (-17 104:-36 -104) 27 34 -35 -105 400 (19 26:29 -26)
C    ALUMINUM SUPPORT OF THE BLOCK
 29  4  4.614E-5 17 -40 34 -35 104 -105 (-102:103:101)
 30  5  -2.7  36 -39  30 -31  41 -104
 31  4  4.614E-5 36 -40 34 -35 41 -104 (39:-30:31)
 32  5  -0.73 37 -38  32 -33  -41 +42
 33  4  4.614E-5 36 -40 34 -35 -41 +42 (-37:38:-32:33)
 34  5  -0.52 36 -40  34 -35  -42 +43
 35  5  -2.7  -43 400  -44 45
 36  4  4.614E-5 -43 400 -45
 37  5  -2.7  -43 400  -46 47
 38  4  4.614E-5 -43 400 -47
 39  5  -2.7  -43 400  -48 49
 40  4  4.614E-5 -43 400 -49
 41  5  -2.7  -43 400  -50 51
 42  4  4.614E-5 -43 400 -51
 43  4  4.614E-5 36 -40 34 -35 -43  400 44 46 48 50
C    BUNKER AND WALLS
 44  4  .00004614 300 -310 200 -210 400 -410 (40:-27:-34:35:105)
 45  6  -2.6 (-300:310:-200:210:-400:410) 309 -319 209 -219 409 -419
 46  0       -309:319:-209:219:-409:419
```

```
C    TARGET ZONES
  1  PY      0.0
  2  PY      0.1
  3  PY      0.2
  4  PY      0.32
  5  PY     -0.02
  6  PY     -1.9
  7  CY      1.5
  8  CY      1.6
  9  CY      1.7
  10  CY      1.8
  11  CY      2.4
  12   Y  -0.02  2.4   -1.45  6.4
  13   Y   .1   2.4   -1.33  6.4
  14   Y   .2   2.4    1.63  6.4
  15   Y   .32  2.4    1.75  6.4
  16 2 C/X    .15  0.0  1.6
  17  PY    -12.4
  18  PY    -4.
  19  CY     17.7
  20 2 C/Y    -16.  0.  1.6
  21 2 C/Y     16.  0.  1.6
  22  PY    -13.0
  23  CY     17.4
  24  CY      5.35
  25  CY      5.75
  26  PY    -165.
  27  PY    -255.
  29  CY     30.
 100  PY      5.3
 101  PY     71.8
 102  PX    -49.5
 103  PX    +49.5
 104  PZ    -49.2
 105  PZ    +49.2
C    ALUMINUM SUPPORT OF THE BLOCK
 30   PX  -51.5
 31   PX  +51.5
 32   PX  -57.5
 33   PX  +57.5
 34   PX  -65.0
 35   PX  +65.0
 36   PY  -42.2
 37   PY  -26.2
 38   PY  +88.8
 39   PY +122.0
 40   PY +177.8
 41   PZ  -53.2
 42   PZ -137.2
 43   PZ -173.2
 44   C/Z  -53. -7.2  11.
 45   C/Z  -53. -7.2  10.2
 46   C/Z   53. -7.2  11.
 47   C/Z   53. -7.2  10.2
 48   C/Z  -53. 142.8 11.
 49   C/Z  -53. 142.8 10.2
 50   C/Z   53. 142.8 11.
 51   C/Z   53. 142.8 10.2
```

```
C    BUNKER AND WALLS
 200 1 PX  -570
 209 1 PX  -620
 210 1 PX   570
 219 1 PX   620
 300 1 PY  -760
 309 1 PY  -810
 310 1 PY   480
 319 1 PY   530
 400   PZ  -406
 409   PZ  -456
 410   PZ   530
 419   PZ   580

MODE N
*TR1   0 0 0 45 135 90 45 45 90 90 90 0
*TR2   0 0 0 45 90 45 90 0 90 135 90 45
C      STAINLESS STEEL (AISI 316)
  M1  5010.89C -2.8E-4  5011.89C -2.87E-3  6000.89C -0.04
      14000.89C -0.41  23000.89C -0.16
     24000.89C -16.8  25055.89C -1.14 26000.91C -68.1  27059.35C -0.14
     28000.89C -10.7  42000.89C -2.12 29000.89C -0.09
C      H2O
  M2 1001.89C 2.  8016.89C 1.
C     COPPER
  M3 29000.89C 1.
C      AIR
  M4     7014.04C .788903  8016.89C .211097
C      ALUMINUM
  M5 13027.89C  1.
C     CONCRETE + 5% FE  DENS. 2.6
  M6   1001.89C -0.005358 8016.89C -0.474193 11023 -0.016312
      13027.89C -0.043491 14000.89C -0.299658 19000.35C -0.018031
     26000.91C -0.04
C      COPPER 50% IRON 50%
  M7    26000.91C .5  29000.89C .5
C       SOURCE SPECIFICATIONS
RDUM    .230  1.5  0.0   0.001 0.0

GEOM.DATA ENDS*******************************************************
```

DETEC.DATA STARTS***

I. ACTIVATION REACTIONS AND NUCLEAR DATA EMPLOYED

Reaction	Half-life	Isotopic abundance (%)	γ-ray energy (keV)	γ-ray branching (%)
$^{27}Al(n,\alpha)^{24}Na$	14.96h	100.0	1368.6	100.0
$^{56}Fe(n,p)^{56}Mn$	2.577h	91.72	846.8	98.87
$^{58}Ni(n,2n)^{57}Ni$	1.503d	68.27	1377.6	80.0
$^{58}Ni(n,p)^{58}Co$	70.92d	68.27	810.8	99.44
$^{115}In(n,n')^{115m}In$	4.486h	95.7	336.24	45.9
$^{55}Mn(n,\gamma)^{56}Mn$	2.577h	100.0	846.8	98.87
$^{197}Au(n,\gamma)^{198}Au$	2.696d	100.0	411.8	95.56

II. ACTIVATION MEASUREMENT CONDITIONS

Reaction	Foil diameter mm	Foil thickness Penetration depth < 25 cm	> 25 cm
$^{58}Ni(n,2n)^{57}Ni$	18	1 mm	2 mm
$^{27}Al(n,\alpha)^{24}Al$	18	1 mm	2 mm
$^{56}Fe(n,p)^{56}Mn$	18	1 mm	2 mm
$^{58}Ni(n,p)^{58}Co$	18	1 mm	2 mm
$^{115}In(n,n')^{115m}In$	18	1 mm	2 mm
$^{55}Mn(n,\gamma)^{56}Mn$	18	200 μm	200 μm
$^{197}Au(n,\gamma)^{198}Au$	18	50 μm	50 μm

III. GAMMA RAY MEASUREMENTS - DETECTORS AND CALIBRATION

Induced γ-rays emission was measured by three HPGe detectors. Activation reaction rates were deduced from the measured gamma-ray peak counts, using the Standard Absolute Radiometric Technique. The absolute efficiency of the HPGe detectors is determined up to 1836 keV by using standard point-like gamma-ray sources, whose intensity is known within less than ±1% for mono-gamma ray sources, and within ±2.5% for multi-gamma ray sources. The experimental data are fitted using the least square method, and the resulting uncertainty on the calibration curve is ±2% at 1σ level. A routine check of the stability of HPGe efficiency is performed using ^{137}Cs and ^{60}Co sources. Intercalibration of the detectors is also performed by comparing activated samples: the comparison typically shows an agreement among the detectors within the quoted ±2% uncertainty.

DETEC.DATA ENDS***

EXPE.DATA STARTS **

$^{27}Al(n,\alpha)^{24}Na$

Depth (cm)	E	Random error (%)	Systematic error (%)	Total error (%)
4.95	5.94E-5	3.5	4.4	5.6
10.05	1.52E-5	3.5	4.4	5.6
20.15	1.70E-6	3.8	4.4	5.8
30.30	2.29E-7	3.8	4.4	5.8
40.50	3.66E-8	3.8	4.4	5.8
50.70	6.65E-9	4.7	4.4	6.4
55.90	2.60E-9	8.9	4.4	10.0

$^{56}Fe(n,p)^{56}Mn$

Depth (cm)	E	Random error (%)	Systematic error (%)	Total error (%)
4.95	5.09E-5	3.3	4.4	5.5
10.05	1.28E-5	3.3	4.4	5.5
20.15	1.48E-6	4.5	4.4	6.3
30.30	1.93E-7	5.1	4.4	6.7
40.50	3.24E-8	4.5	4.4	6.3
50.70	5.61E-9	5.6	4.4	7.1
60.90	1.21E-9	15.4	4.4	16.0

$^{115}In(n,n')^{115m}In$

Depth (cm)	E	Random error (%)	Systematic error (%)	Total error (%)
4.95	1.67E-4	2.2	2.5	3.3
10.05	6.81E-5	2.2	2.5	3.3
20.15	1.35E-5	2.3	2.5	3.4
30.30	2.86E-6	3.0	2.5	3.9
40.50	6.81E-7	3.7	2.5	4.5
50.60	1.57E-7	3.9	2.5	4.6
60.90	4.02E-8	4.1	2.5	4.8

$^{197}Au(n,\gamma)^{197}Au$

Depth (cm)	E	Random error (%)	Systematic error (%)	Total error (%)
5.0	7.79E-4	1.9	4.4	4.8
10.0	8.88E-4	2.4	4.4	5.0
20.0	9.29E-4	2.4	4.4	5.0
30.0	7.76E-4	2.4	4.4	5.0
40.0	5.33E-4	3.3	4.4	5.5
50.0	3.11E-4	3.3	4.4	5.5
60.0	1.17E-4	3.3	4.4	5.5

$^{55}Mn(n,\gamma)^{56}Mn$

Depth (cm)	E	Random error (%)	Systematic error (%)	Total error (%)
5.0	2.66E-5	2.4	4.4	5.0
10.0	3.24E-5	2.4	4.4	5.0
20.0	3.33E-5	2.4	4.4	5.0
30.0	2.40E-5	2.4	4.4	5.0
40.0	1.58E-5	2.4	4.4	5.0
50.0	1.00E-5	2.4	4.4	5.0
60.0	3.89E-6	2.4	4.4	5.0

$^{58}Ni(n,p)^{58}Co$

Depth (cm)	E	Random error (%)	Systematic error (%)	Total error (%)
4.95	2.17E-4	2.8	1.6	3.2
10.05	6.66E-5	2.8	1.6	3.2
20.15	9.21E-6	2.9	1.6	3.3
30.30	1.41E-6	3.3	1.6	3.7
40.50	2.45E-7	4.1	1.6	4.4
50.70	3.85E-8	5.3	1.6	5.5
60.90	7.10E-9	6.3	1.6	6.5

$^{58}Ni(n,2n)^{57}Ni$

Depth (cm)	E	Random error (%)	Systematic error (%)	Total error (%)
4.95	1.78E-5	3.1	1.6	3.5
10.05	4.32E-6	3.7	1.6	4.0
20.15	4.42E-7	3.9	1.6	4.2
30.30	5.98E-8	6.2	1.6	6.4

EXPE.DATA ENDS **

CALC.DATA STARTS***

$^{27}Al(n,\alpha)^{24}Na$

Depth (cm)	C	MCNP statistics %	Error due to activ. react.%	Tot. error on C %	C/E	Total error on C/E
4.95	5.71E-5	2.0	0.6	2.1	0.96	0.06
10.05	1.40E-5	2.0	0.6	2.1	0.92	0.06
20.15	1.48E-6	2.0	0.6	2.1	0.87	0.06
30.30	2.09E-7	2.0	0.6	2.1	0.91	0.06
40.50	3.40E-8	2.0	0.6	2.1	0.93	0.06
50.70	5.93E-9	2.0	0.6	2.1	0.89	0.06
55.90	2.53E-9	2.0	0.6	2.1	0.97	0.10

$^{56}Fe(n,p)^{56}Mn$

Depth (cm)	C	MCNP statistics %	Error due to activ. react.%	Tot. error on C %	C/E	Total error on C/E
4.95	5.39E-5	2.0	1.3	2.4	1.06	0.06
10.05	1.29E-5	2.0	1.3	2.4	1.01	0.06
20.15	1.38E-6	2.0	1.3	2.4	0.93	0.07
30.30	1.94E-7	2.0	1.3	2.4	1.01	0.07
40.50	3.07E-8	2.0	1.3	2.4	0.95	0.07
50.70	5.07E-9	2.0	1.3	2.4	0.90	0.07
60.90	8.94E-10	2.0	1.3	2.4	0.74	0.12

$^{115}In(n,n')^{115m}In$

Depth (cm)	C	MCNP statistics %	Error due to activ. react.%	Tot. error on C %	C/E	Total error on C/E
4.95	1.64E-4	2.0	2.0	2.8	0.98	0.04
10.05	6.31E-5	1.0	2.6	2.8	0.93	0.04
20.15	1.20E-5	1.0	2.7	2.9	0.89	0.04
30.30	2.53E-6	1.0	2.7	2.9	0.88	0.04
40.50	5.36E-7	1.0	2.7	2.9	0.79	0.04
50.60	1.27E-7	2.0	2.2	3.0	0.81	0.04
60.90	2.78E-8	2.0	2.2	3.0	0.69	0.04

$^{197}Au(n,\gamma)^{197}Au$

Depth (cm)	C	MCNP statistics %	Error due to activ. react.%	Tot. error on C %	C/E	Total error on C/E
5.0	8.18E-4	3.0	0.2	3.0	1.05	0.06
10.0	9.26E-4	3.0	0.2	3.0	1.04	0.06
20.0	9.97E-4	2.0	0.2	2.0	1.07	0.05
30.0	8.65E-4	2.0	0.2	2.0	1.11	0.05
40.0	5.66E-4	2.0	0.2	2.0	1.06	0.06
50.0	3.51E-4	2.0	0.2	2.0	1.13	0.07
60.0	1.44E-4	2.0	0.2	2.0	1.23	0.07

$^{55}Mn(n,\gamma)^{56}Mn$

Depth (cm)	C	MCNP statistics %	Error due to activ. react.%	Tot. error on C %	C/E	Total error on C/E
5.0	2.62E-5	3.0	11.4	11.8	0.98	0.13
10.0	3.33E-5	2.0	10.6	10.8	1.03	0.12
20.0	3.02E-5	2.0	9.8	10.0	0.91	0.11
30.0	2.52E-5	1.0	9.7	9.8	1.05	0.11
40.0	1.76E-5	1.0	9.1	9.2	1.11	0.10
50.0	9.92E-6	1.0	9.0	9.1	0.99	0.10
60.0	4.11E-6	1.0	9.3	9.4	1.06	0.11

$^{58}Ni(n,p)^{58}Co$

Depth (cm)	C	MCNP statistics %	Error due to activ. react.%	Tot. error on C %	C/E	Total error on C/E
4.95	2.29E-4	2.0	9.8	10.0	1.06	0.10
10.05	6.64E-5	2.0	8.1	8.3	1.00	0.09
20.15	8.74E-6	2.0	6.6	6.9	0.95	0.07
30.30	1.42E-6	1.0	6.1	6.2	1.01	0.08
40.50	2.46E-7	1.0	5.9	6.0	1.00	0.08
50.70	4.20E-8	1.0	5.7	5.8	1.09	0.08
60.90	7.83E-9	1.0	5.5	5.6	1.10	0.09

$^{58}Ni(n,2n)^{57}Ni$

Depth (cm)	C	MCNP statistics %	Error due to activ. react.%	Tot. error on C %	C/E	Total error on C/E
4.95	1.77E-5	2.0	2.0	2.8	0.99	0.05
10.05	4.14E-6	2.0	2.0	2.8	0.96	0.05
20.15	4.06E-7	2.0	1.7	2.6	0.92	0.05
30.30	5.31E-8	2.0	1.7	2.6	0.89	0.06

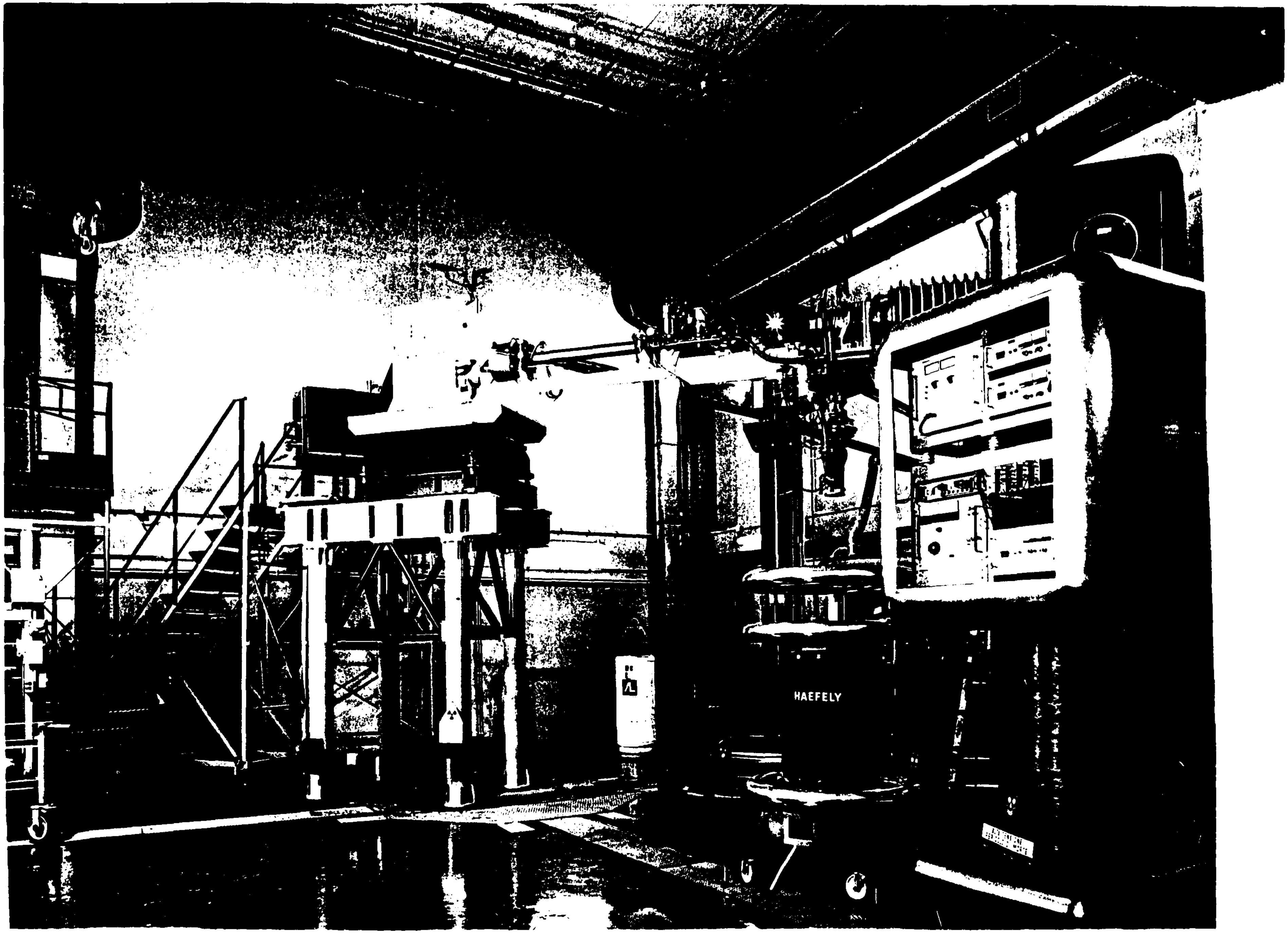
CALC.DATA ENDS***

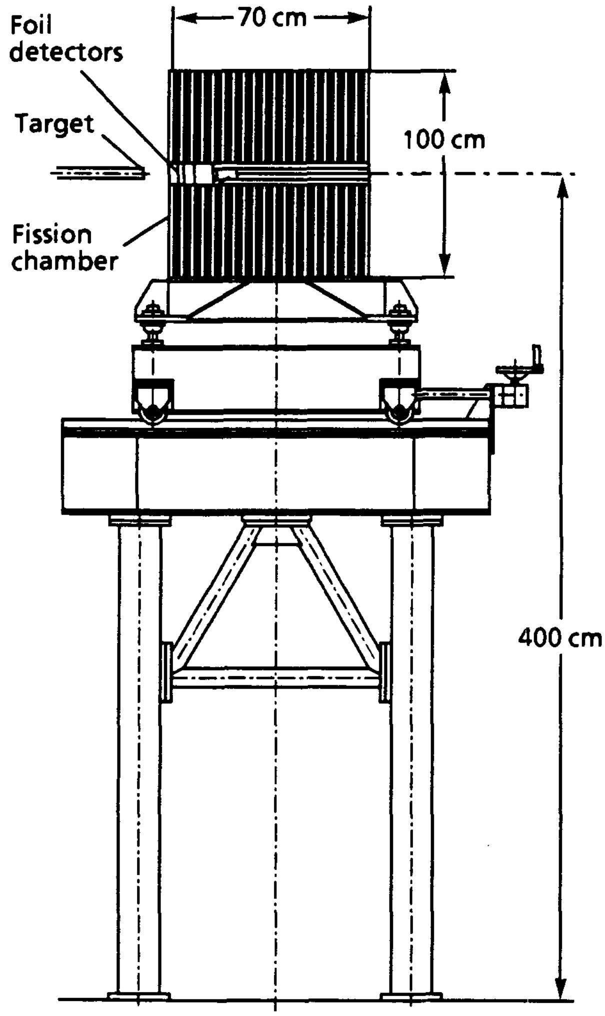
Figures

1. Photograph of the Benchmark Experiment set up in the FNG bunker hall, showing the FNG generator, the SS block in front of the target, and the aluminum movable tower (yellow) on which the block is located.

2. The SS block with the central channel used to introduce the foil detectors at various penetration depths. The channel is aligned along the beam propagation direction, and the voids between foils are filled with stainless steel plugs.

3. Planimetry of the FNG bunker hall and building.


HAEFELY



Foil
detectors
Target
Fission
chamber
70 cm
100 cm
400 cm

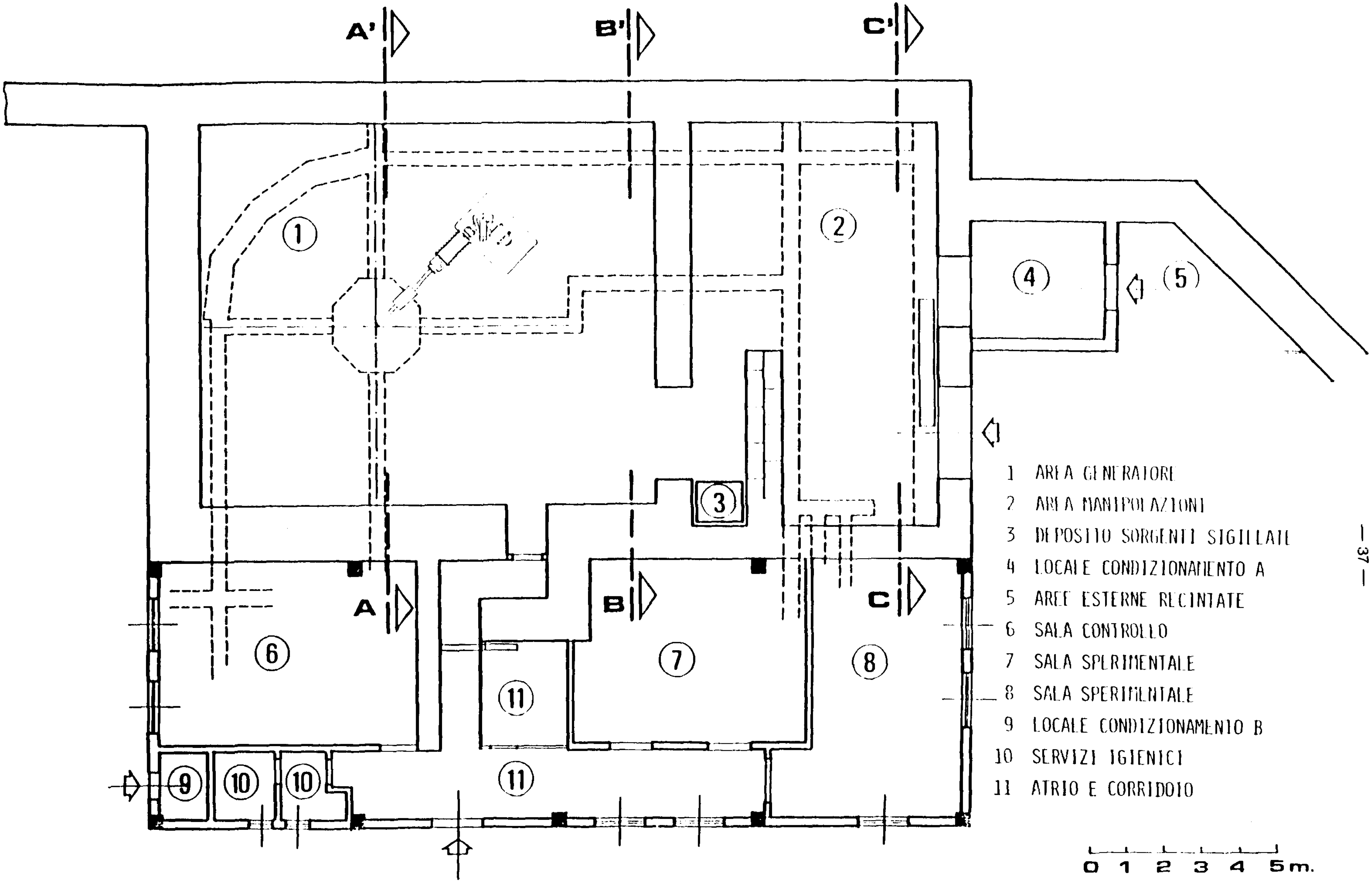


Fig.2.4. - Planimetria della sala del generatore e dei locali attigui con