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PART. II

PWR SHIELDING BENCHMARK

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PWR SHIELDING BENCHMARK

by G. Hehn

Abstract

The results of the PWR shielding benchmark exercise, initiated by NEACRP in 1980, are summarized and analysed by a status report on nuclear data accuracies in LWR shielding. Seven organisations have participated in the exercise. In the first part shielding target quantities of primary interest have been calculated using the same transport code ANISN but eight different widely used versions of multigroup libraries for coupled neutron and gamma calculations. Whereas much progress has been made in recent years in calculating fast neutron fluxes needed for prediction of neutron damage, there are still large discrepancies in calculating thermal neutron fluxes with considerable consequences for the neutron dose and the gamma sources hereof. In the second stage of the exercise detailed cross-section sensitivity studies have been performed, showing the importance of partial cross-sections to the target quantities considered. The final aim of the benchmark exercise was to show, how we meet the target accuracies required for reactor safety and radiation protection. Therefore for the first time a complete uncertainty analysis has been prepared for a typical shield of light water reactors. The results can serve as basic reference in shield design work, where uncertainties are needed and have to be stated properly.

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1. AIMS OF THE BENCHMARK

In the last years the status of shielding data libraries has been improved considerably. New informations like cross-section errors and correlation matrices have been evaluated. Several advanced multigroup data libraries had been processed and are in use. Since the first PWR shield benchmark of NEACRP, which had been presented and discussed on a Specialists' Meeting in Vienna 1976, large progress has been achieved in analysing uncertainties.[1] Therefore a second intercomparison of PWR shield calculations had been initiated in 1980 together with a similar activity on fast breeder shielding.[2] The specification of this PWR shielding benchmark is given in Annex I. In a common effort several widely used group data libraries should be compared and detailed sensitivity studies should be performed. But the final aim was to produce a shield benchmark with complete error analysis.

In recent years the target accuracies in radiation shielding were improved to meet the advanced requirements of reactor safety and radiation protection. For target quantities of main importance an error analysis is requested. But this couldn't be performed rigorously. Many assumptions had to be made, resulting in highly uncertain values of the errors quoted. Today safety factors are still applied in shielding, which lead to rather expensive shields. In this aspect the common effort on shield design benchmarks can help to quantify the uncertainties for practical shields of power reactors.

The PWR shield benchmark had been specified for the radial shield configuration of a 1300 MW_{el} power plant, where measurements are performed and might be available for comparison with calculations. The calculations have been made along the midplane of the active core in one-dimensional cylindrical geometry for a simplified representation as shown in figure 1.

For the second NEA PWR shield benchmark the following three parts of common work were agreed upon:
- At first an intercomparison should be made of integral target quantities like activation rate, neutron damage rate, gamma heating rate, and biological dose rate calculated with different in-house shielding libraries but otherwise identical input of the transport code ANISN.[11]
- Then sensitivity studies were requested of the above quantities to all cross-sections, radiation sources and conversion data, showing the details of data requirements for shielding of light water reactors.
- Finally an uncertainty analysis should be provided of the above quantities to all errors of cross-sections, radiation sources and conversion data resulting in a benchmark with a complete error analysis needed urgently as basic reference in practical design calculations.

2. CALCULATION OF TARGET QUANTITIES FOR SHIELDING

Seven contributions were presented from AAEC-Lucas Heights, AERE-Winfrith, CEA-Saclay, CEN/SCK-Mol, EIR-Würenlingen, MAPI-Tokyo, and IKE-Stuttgart.

The results obtained have been related to the IKE values and are presented in table 1 for neutrons. Eight different libraries have been compared. The calculations of the fast neutron flux show good agreement up to the pressure vessel (interval 107). But outside the concrete shield (interval 189) the fast neutron dose rate deviates already by a factor of two. As has been known, the representation of thermal neutrons isn't good in coupled neutron-gamma libraries. Therefore a special effort has been undertaken in the IKE results to consider upscattering below 3 eV in 33 energy groups. By intercomparing all results presented in table 1 the following conclusions can be drawn:

- The ENDF/B-5 fission spectrum produces higher values of the target quantities of +4 % compared to the NBS standard.[7]
- The Cranberg fission spectrum gives lower values of approximately -10 % compared to the NBS standard.[6]
- All adjoint calculations, which aren't shown in table 1, give higher values ranging up to 13 %.[6, 7, 10] It can be assumed that the forward calculations are more precise.
- For target quantities concerning the fast neutron flux good agreement is achieved up to the pressure vessel (interval 107) for all libraries applied.
- The dose rate of fast neutrons outside the concrete shield shows deviations by a factor of two.
- The correct treatment of thermal neutrons in coupled neutron-gamma libraries is still a problem producing large uncertainties in neutron dose, thermal activation and gamma production.
- Both methods of resonance treatment applied in shielding, e.g. self-shielding factor or exact $1/\sigma_T$ -weighting of important components like iron and stainless steel components, are equally in use. For the target quantities considered both methods are well applicable.
- The multigroup in-house shielding libraries compared in the benchmark exercise concentrate at the 100 group level of EURLIB or RADHEAT or at 171 groups of VITAMIN-C.
- Since it can be assumed, that nearly all group data used are based on the same evaluated point data bank like ENDF/B-4 with eventually some recent improvements, most of the discrepancies observed have their origen in differences of group data processing, especially in flux weighting, resonance treatment and energy group structure. Some effects are due to numerical errors of the integration procedures in the processing codes. The standard deviations shown are a good measure of all uncertainties introduced by group data processing.

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For gamma calculations the participating organizations are listed in table 2. The results compared are again normalized to the IKE values. For gamma heatng the values vary between -41 % and +55 %. The gamma dose rate outside the concrete shield ranges from -30 % to +62 %. The main conclusions are:

- The gamma field is influenced strongly by uncertainties of the neutron fluxes and predominantly of thermal neutrons.
- The results depend on the weighting fluxes applied to the low energy gamma cross-sections, if the group number is small.
- The largest differences are caused by gamma production data. Therefore updating of gamma production data is very important.

3. SENSITIVITY AND UNCERTAINTY ANALYSIS

Sensitivity studies were performed by AERE-Winfrith, CEA-Saclay, CEN/SCK-Mol, MAPI-Tokyo, and IKE-Stuttgart.

Inspite of differences in cross-sections used the sensitivities calculated are similar. In table 3 the total sensitivities of the responses are given for the most important nuclides separated into neutron and gamma cross-sections.[6] For the sensitivities of the neutron dose to oxygen and iron cross-sections the values given are upper limits, see also fig. 34 - 39.[10] The sensitivities of each response to the Legendre expansion of the neutron cross-sections are shown in table 4 for the same most important nuclides.[6] Finally a series of figures from 2 to 33 is given, which represents the energy profiles for the neutron responses in the energy details of VITAMIN-C.[7] The conclusions of the cross-section sensitivity studies can be summarized in the following:

- The cross-sections of the elements H, O, and Fe are the most important ones for all shielding target quantities. Of secondary importance are the additional elements of stainless steel like Ni and Cr, the additional elements of concrete like Si and Ca and finally U-38 of the core materials.
- For neutron cross-sections the sensitivity profiles concentrate in the energy range between 2 MeV and 10 MeV. For the neutron dose rate there is a strong sensitivity peak at the oxygen minimum of 2.3 MeV, resulting in a demand of high accuracy of all cross-sections at this energy.
- For gamma cross-sections the sensitivity profiles change more strongly with the shielding target quantity considered. For gamma heating in the baffle near the core there is a broad maximum in the upper keV and lower MeV energy range. In the case of gamma heating in the pressure vessel the sensitivity maximum is shifted to the lower MeV energy range. And finally for the gamma dose rate outside the concrete shield the energy range of importance lies between 6 MeV and 10 MeV.
- For both kinds of radiation the expansion of the scattering cross-sections is needed up to the P_3-moment at least.

Mainly because of the quadratic addition rule of errors, the biggest contributions dominate over all smaller portions drastically, so that the error analysis of nuclear data for shielding target quantities can concentrate on the main contributing parts, which simplifices the work considerably. A complete uncertainty analysis has been performed by IKE-Stuttgart.[10] The codes ANISN, SWANLAKE and SENSIT were applied in EURLIB group structure.[11, 12,13,14.] The 30 group COVFILS covariance information, based on ENDF/B-5, was extended to 100 groups. According to the uncertainties given for the partial cross-sections as well as for the energy distribution of secondaries (SED) we obtain the results shown for neutron target quantities in table 5 and for gamma target quantities in table 6.

The error contributions of the three most important nuclides H, O and Fe were considered only. Compared with previous results based on older covariance informations we get an appeciable reduction of the uncertainties especially for the contribution of iron. The highest target accuracies are needed for radiation damage calculations. For damage further improvements are possible by reducing the uncertainties of inelastic scattering in iron and of elastic scattering in hydrogen. The uncertainty related to the energy distribution of secondaries (SED) shows clearly the need for accurate gamma production cross-sections of iron, which determine the data error of all gamma target quantities essentially. The contribution of the uncertainties in the fission source and detector data are approximately the same for activation and damage as given in table 7.

Finally the total nuclear data uncertainty is summarized in table 8. Compared with the uncertainties produced by different effects in group data processing we can conclude, that for target quantities with high requirements on precision like neutron damage in the pressure vessel or activation of detectors in surveillance capsules the dominating uncertainties originate form nuclear data and not from processing problems. For calculation of gamma heating the processing error gets larger, so that improvements in group structure and nuclear data processing are needed. But the latter is of extreme importance in determining the neutron and gamma dose outside the reactor shield. All multigroup data libraries processed directly from point data into a group structure of approximately 100 neutron groups underestimate the fast neutron dose outside the reactor shield by a factor of two. The total uncertainty given in table 8 for the neutron dose does not include the errors of thermal neutrons. The deficiencies of the neutron group data observed are restricted to dose calculations outside the biological reactor shield or to the determination of activated products within the concrete shield. To describe the penetration of neutrons through 2 meters of concrete we have to improve the flux weighting for fast neutrons as well as that for the thermal energy range.

One can conclude, that the common effort on the PWR shield benchmark within NEACRP has shown us clearly, where we meet the target accuracies in practical shield design and where we don't. For some target quantities like neutron damage and activation reactions in the pressure vessel and surveillance capsules we have to improve the data basis, e.g. by adjustment of the iron inelastic cross-section and its covariance matrix to integral experiments. For other target quantities like dose or gamma heating we have recognized deficiencies in our libraries, which come from group structures and group data processing. Improvements of this kind can be performed easily.

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Table 1: Neutron Target Quantities, Calculated with 8 Different Libraries, Related to the IKE Values. 4,5,6,7,8,9,10

No	Participating Organization	Pressure Vessel Damage Rate ($dpa\ s^{-1}$) Int.107	Barrel Neutron Activation ($cm^{-3}s^{-1}$) Int.105	Barrel Neutron Activation ($cm^{-3}s^{-1}$) Int.72	Concrete Shield Neutron Dose ($Sv.h^{-1}$) Fast Int.189	Concrete Shield Neutron Dose ($Sv.h^{-1}$) Thermal Int.189	Remarks: Fission Spectrum	Remarks: Library Groups	Remarks: Resonance Treatment
1	IKE Stuttgart	1.69 -12*	6.57 +5	1.53 +7	1.23 -5	0.92 -5	NBS	$161 n_f$ $33 n_{th}$	$\sigma_{Fe}:1/\sigma_T$-w.
2	AAEC Lucas Heights	92 %	108 %	102 %	112 %	193 %	NBS	40n	
3	AERE Winfrith	97 % 109 %	98 % 110 %	97 % 105 %	450 % 395 %	total total	NBS	EUR-3 UKNDL 100	$\sigma_{Fe}:1/\sigma_T$-w. $\sigma_{Fe}:1/\sigma_T$-w.
4	CEA Saclay + +	81 % 82 % 91 %	88 % 89 % 101 %	90 % 90 % 99 %	43 % 44 % 50 %	70 % 825 % 914 %	Cranberg Cranberg B-5	VIT-C EUR-4 EUR-4	f-factor $\sigma_{Fe}:1/\sigma_T$-w.
5	CEN/SCK Mol +	100 % 104 %	101 % 105 %	98 % 102 %	100 % 104 %	304 % 314 %	NBS B-5	VIC-C VIC-C	f-factor
6	EIR Würenlingen	96 %	98 %	96 %	46 %	58 %	NBS	EUR-4-IKE	$\sigma_{Fe}:1/\sigma_T$-w.
7	MAPI Tokyo	98 %	96 %	95 %	56 %	87 %	NBS	RADHEAT 100n	f-factor
	Average Val. Stand. Dev.	1.65 -12 6.4 %	6.64 +5 5.2 %	1.51 +7 3.6 %	0.94 -5 36 %	2.54 -5 325 %			

* read $1.69\ 10^{-12}$ + not included in averaging

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Table 2: Gamma Target Quantities, Calculated with 7 Different Libraries, Related to the IKE Values. 4,5,6,8,9,10

No	Partizipating Organization	Baffle	Barrel	Pr.Vessel	Concr.Sh.	Concr. Shield	Remarks: N-Fission	Remarks: Library
		Gamma Heating (Wcm^{-3})				Gamma Dose ($Sv.h^{-1}$)	Spectrum	Groups
		Int.18	Int.60	Int.105	Int.134	Int.189		
1	IKE Stuttgart	4.21	6.52 -1	2.10 -2	9.88 -6	1.54 -4	NBS	$161n_f/33n_{th}/36\gamma$
2	AAEC Lucas Heights	–	–	–	–	130 %	NBS	$40n/?\ \gamma$
3	AERE Winfrith	59 %	127 %	98 %	–	152 %	NBS	EUR-3
4	CEA Saclay +	97 % 97 % 97 %	79 % 153 % 155 %	78 % 105 % 107 %	85 % 97 % 100 %	79 % 150 % 162 %	Cranberg Cranberg B-5	VIT-C EUR-4 EUR-4
5	EIR Würenlingen	99 %	98 %	93 %	88 %	70 %	NBS	EUR-4-IKE
6	MAPI Tokyo	84 %	101 %	99 %	101 %	113 %	NBS	RADHEAT $100n/13\gamma$
	Average Val.	4.00	7.58 -1	2.09 -2	9.61 -6	1.87 -4		
	Stand. Dev.	19 %	25 %	5.0 %	6.2 %	34 %		

+ not included in averaging

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Table 3: Sensitivity of the Responses to the Cross-Sections of each Nuclide.
Sum over all Groups and Regions

Nuclide \ Response		Damage rate (int. 107)	Activation (int. 72)	Neutron dose (int. 189)	Gamma dose (int. 189)	Heating rate (int. 18)	Heating rate (int. 105)
H	n	- 5.20	- 3.11	- 8.24	- 3.97	- 0.08	- 0.48
	γ				- 0.16	- 0.02	- 0.33
O	n	- 1.84	- 1.09	- 7.79	- 2.01	- 0.03	- 0.15
	γ				- 5.20	- 0.13	- 1.44
Fe	n	- 1.37	- 1.30	- 4.70	- 1.42	+ 0.03	+ 0.01
	γ				- 3.78	- 0.14	- 1.88
Ni	n	- 0.21	- 0.20	- 0.24	- 0.06		+ 0.02
	γ				- 0.16	- 0.03	- 0.30
Si	n			- 2.33	- 0.26		
	γ				- 1.79		
Ca	n			- 2.78	- 0.38		
	γ				- 1.96		
Cr	n	- 0.38	- 0.35	- 0.40	- 0.15	+ 0.02	+ 0.03
	γ				- 0.22	- 0.04	- 0.48
U^{238}	n	- 0.31	- 0.31	- 0.32	- 0.33	- 0.05	- 0.15
	γ				- 0.03	- 0.74	- 0.35

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Table 4: Sensitivity of the Responses (in %) to the Order of Legendre Expansion for each Nuclide. - Sum over all Groups and Regions

Nuclide (Response)		Damage rate (int. 107)	Activation (int. 72)	Neutron dose (int. 189)	Gamma dose (int. 189)	Heating rate (int. 18)	Heating rate (int. 105)
H	P_0	- 99.5	- 38.4	- 149.7	- 81.8	- 0.31	- 17.2
	P_1	- 21.4	- 12.5	- 25.4	- 14.4	0.03	- 0.40
	P_2	- 1.32	- 1.92	- 3.53	- 1.40		0.47
O	P_0	- 43.4	- 24.8	- 113.0	- 97.1	- 0.16	- 28.5
	P_1	- 8.20	- 5.73	- 19.9	- 26.5	- 0.15	1.21
	P_2	- 0.58	- 0.89	- 3.28	- 3.95	- 0.02	2.03
Fe	P_0	- 48.7	- 52.4	-209.7	- 75.0	3.23	-20.7
	P_1	- 12.5	1.00	- 58.3	- 20.7	0.29	- 6.51
	P_2	- 2.74	3.13	- 8.90	- 4.84	0.02	- 1.82
Ni	P_0	- 9.42	- 7.69	- 11.5	- 5.88	0.47	- 4.50
	P_1	- 2.10	0.17	- 3.33	- 1.56	0.04	- 0.88
	P_2	- 0.17	0.47	- 0.49	- 0.23		- 0.11
Si	P_0			- 43.3	- 24.4		
	P_1			- 6.77	- 7.15		
	P_2			- 0.63	- 1.16		
Ca	P_0			- 45.2	- 24.1		
	P_1			- 8.64	- 7.18		
	P_2			- 0.87	- 1.16		
Cr	P_0	- 18.3	- 15.5	- 19.9	- 10.9	0.86	- 7.83
	P_1	- 4.08	0.31	- 5.78	- 2.85	0.08	- 1.49
	P_2	- 0.32	0.91	- 0.83	- 0.40		- 0.17
U^{238}	P_0	- 6.63	- 6.77	- 6.61	- 4.48	- 4.12	- 2.64
	P_1	0.96	0.88	1.04	0.49	- 0.03	0.15
	P_2	0.31	0.26	0.34	0.12	- 0.09	0.02

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Table 5: Relative Standard Deviation (1σ) for Neutron Quantities due to Partial Cross-Section and SED Uncertainties

Cross-Section Type	Fe-Activation (Barrel, Int. 72)			Radiation Damage (RPV, Int. 107)			Neutron Dose (Concrete, Int. 189)		
	H-1	O-16	Fe	H-1	O-16	Fe	H-1	O-16	Fe
σ_n	2.54	1.3	0.98	4.14	2.26	1.24	4.95	7.71	3.33
SED (n,n)	-	-	-	2.28	1.58	-	5.93	7.79	-
$\sigma_{n'}$	-	1.35	4.49	-	2.45	4.36	-	2.78	7.56
SED (n,n')	-	-	1.47	-	0.06	2.46	-	0.09	5.58
$\sigma_{n,p}$	-	0.95	-	-	1.57	-	-	0.04	-
Cross-Section Error	2.54	1.43	4.03	4.14	2.31	3.89	4.95	7.71	6.9
SED Error	-	-	1.47	2.28	1.58	2.46	5.93	7.79	5.58
Partial Sums	2.54	1.43	4.29	4.73	2.8	4.60	7.72	10.96	8.87
Total Sum %		5.19			7.17			16.1	

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Table 6: Relative Standard Deviation (1σ) for Gamma Quantities due to Partial Cross-Section and SED Uncertainties

Cross-Section Type	Gamma Heating (Barrel, Int. 18)			Gamma Heating (Cladding, Int. 105)			Gamma Dose (Concrete, Int. 189)		
	H-1	O-16	Fe	H-1	O-16	Fe	H-1	O-16	Fe
Neutron									
σ_n	0.07	0.03	0.05	0.31	0.18	0.10	3.03	2.83	1.90
SED (n,n)	0.98	1.10	-	1.85	1.69	-	5.97	5.31	-
$\sigma_{n'}$	-	0.0	0.01	-	0.05	0.01	-	1.49	4.10
SED (n,n')	-	-	0.11	-	-	0.30	-	0.05	3.79
$\sigma_{n,\gamma}$	0.01	-	0.71	0.07	-	0.64	0.18	-	1.65
N-Gamma-Prod. SED (n,γ)	-	0.0	3.36	-	0.06	7.32	-	0.03	7.95
Gamma									
σ_γ Uncorr.	0.01	0.04	0.04	0.08	0.36	0.54	-	-	-
σ_γ (+1)-Corr.	0.01	0.13	0.15	0.32	1.49	2.09	0.15	4.58	3.53
SED (γ,γ)	0.05	0.35	0.60	1.51	6.13	3.89	0.10	4.95	0.99
Total Error (+1)-Corr.									
Cross-Sec. Error	0.07	0.14	0.73	0.45	1.50	2.19	3.04	5.38	5.37
SED Error	0.98	1.15	3.41	2.39	6.35	8.29	5.97	7.25	8.86
Partial Sums	0.98	1.16	3.49	2.43	6.53	8.58	6.70	9.04	10.36
Total Sum %		3.81			11.05			15.30	

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Table 7: Relative Standard Deviation from Source and Detector Uncertainties

		Fission Neutron	Source Gamma	Detector Data
Activation	Int. 72	4.64	-	4.22
	Int. 105	4.70	-	4.15
Damage	Int. 107	4.73	-	3.06+
Heating	Int. 18	2.31	2.98	-
	Int. 105	3.03	1.54	-
Neutron Dose	Int. 189	4.69	-	-
Gamma Dose	Int. 189	4.26	0.11	-

+*Determined with regard of the error correlation between the* elastic and inelastic part of the iron displacement cross section. (Error contribution from the elastic part: 3,21 % and from the inelastic: 1,13 %)

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Table 8: Total Uncertainty of Shielding Target Quantities from Nuclear Data and Data Processing Errors for the NEA PWR Benchmark

	Fe-Activat. (Barrel)	Rad.Damage (RPV)	Neutron Dose (Concrete)	Gamma Dose (Concrete)	Gamma Heating (Baffle)	Gamma Heating (Cladding)
Cross-Section Error	4.97	6.13+	11.47	8.19	0.75	2.60
SED Error	1.47	3.71	11.27	12.91	3.73	10.71
Source Uncertainty	4.64	4.73	4.69	4.26	3.77	3.40
Detector Uncertainty	4.22	3.06+	–	–	–	–
Total Nuclear Data Error	8.14	7.70	16.75	15.87	5.36	11.55
Group Data Processing Error	3.6	6.4	36	34	19	5
Total Uncertainty	8.9	10.0	40	38	20	13

+ (-1) -correlated (4.81 %)

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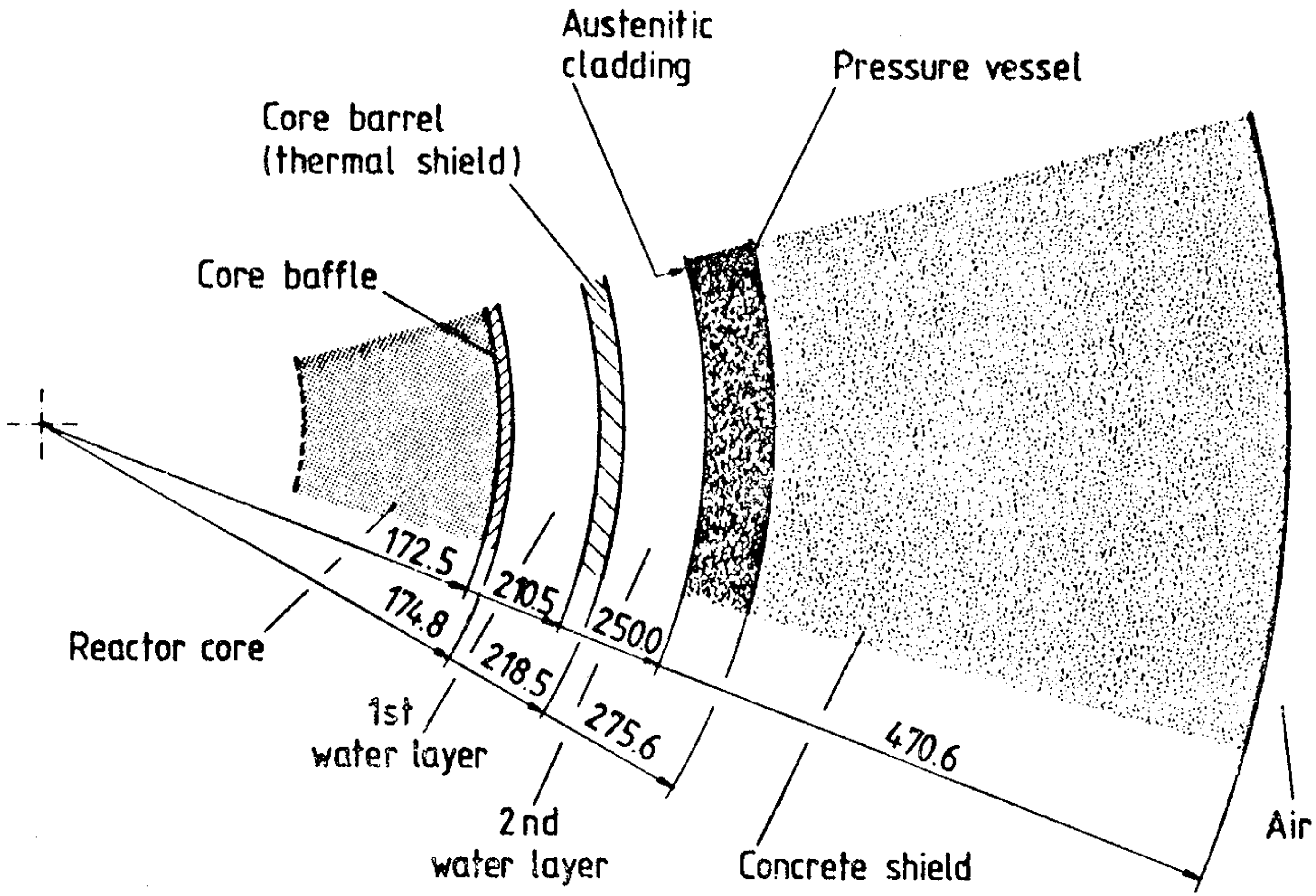

(distances in cm)

Fig. 1: Geometrical Model of the Radial Shield Layers Considered in the NEA Benchmark for a PWR, 1300 MW_{el}

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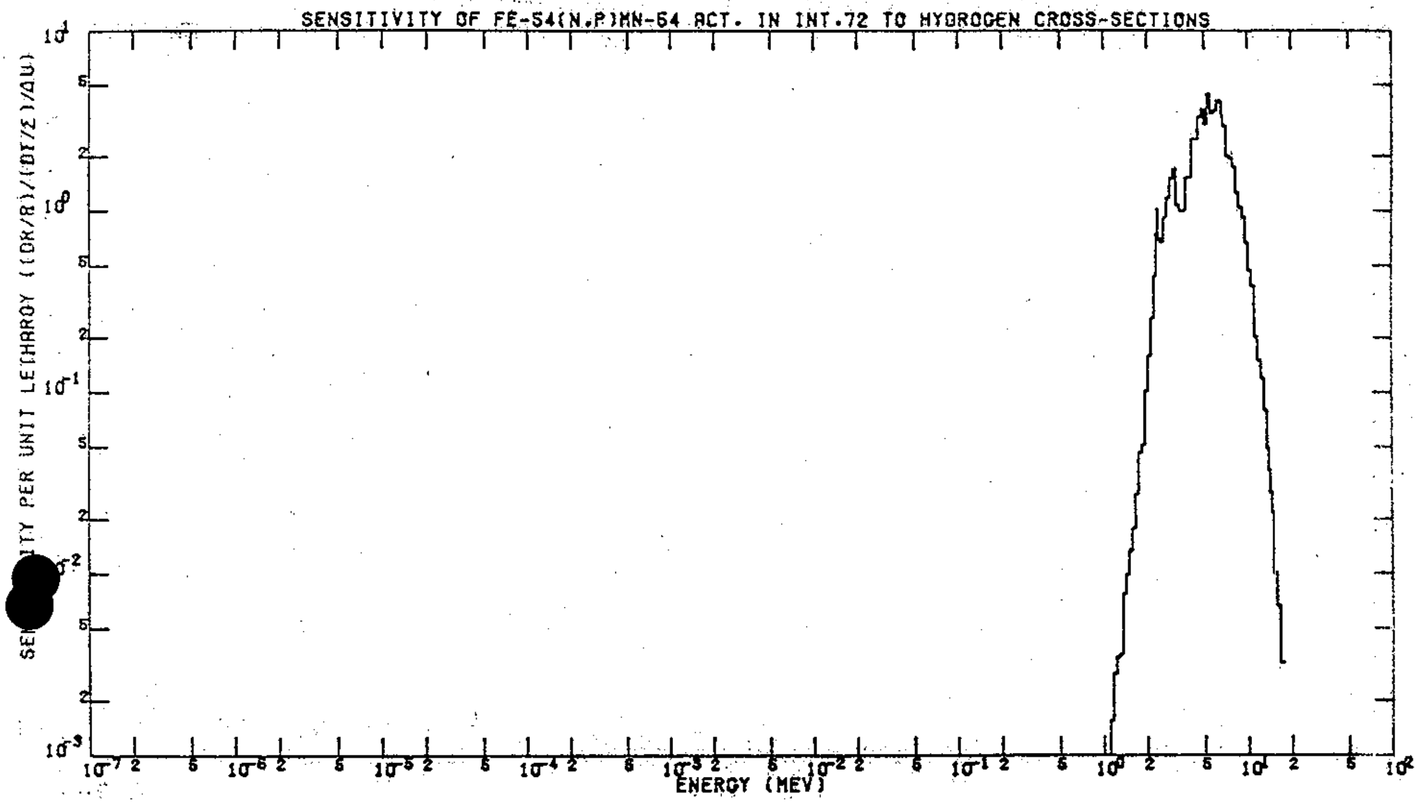

SENSITIVITY OF FE-54(N,P)MN-54 ACT. IN INT.72 TO HYDROGEN CROSS-SECTIONS
ENERGY (MEV)

Fig. 2

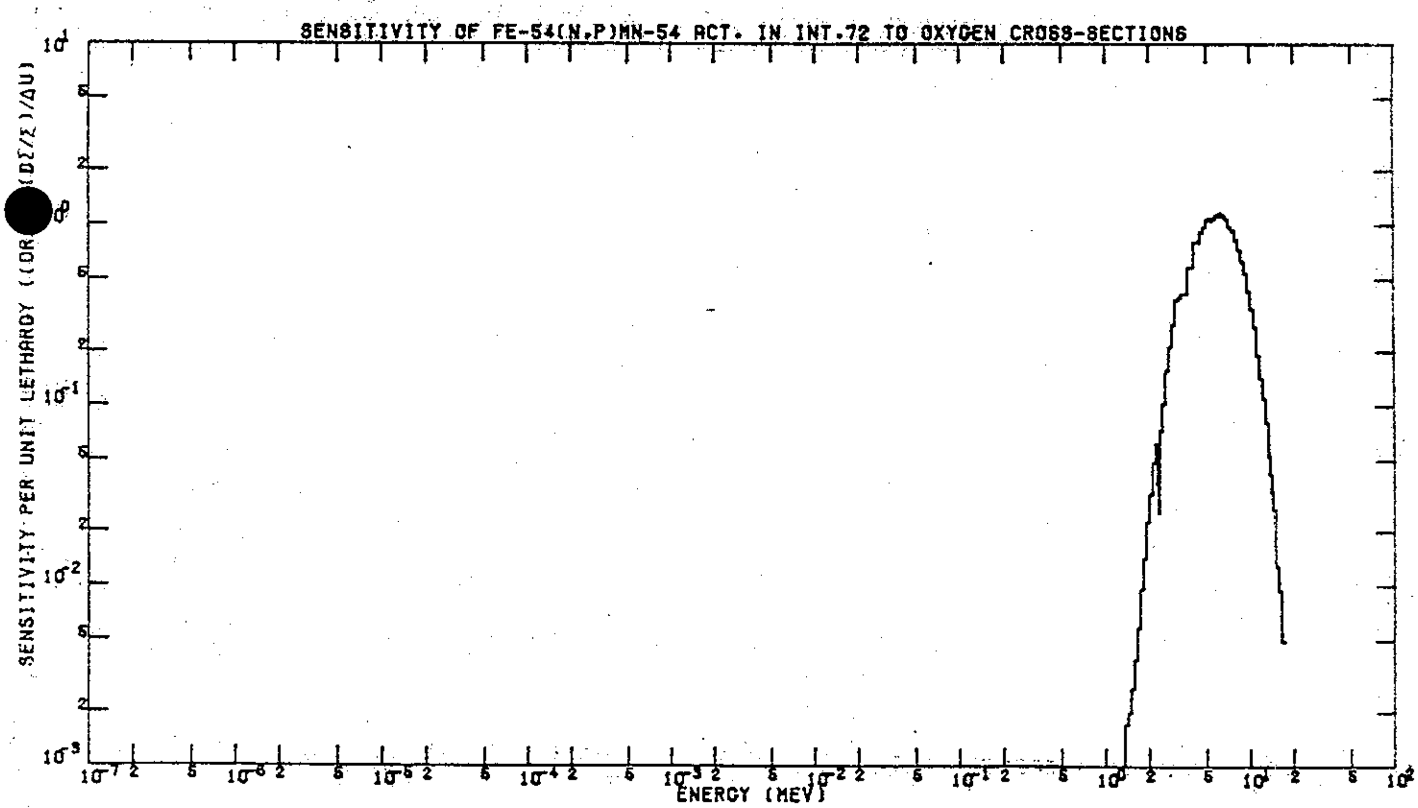

SENSITIVITY OF FE-54(N,P)MN-54 ACT. IN INT.72 TO OXYGEN CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY
ENERGY (MEV)

Fig. 3

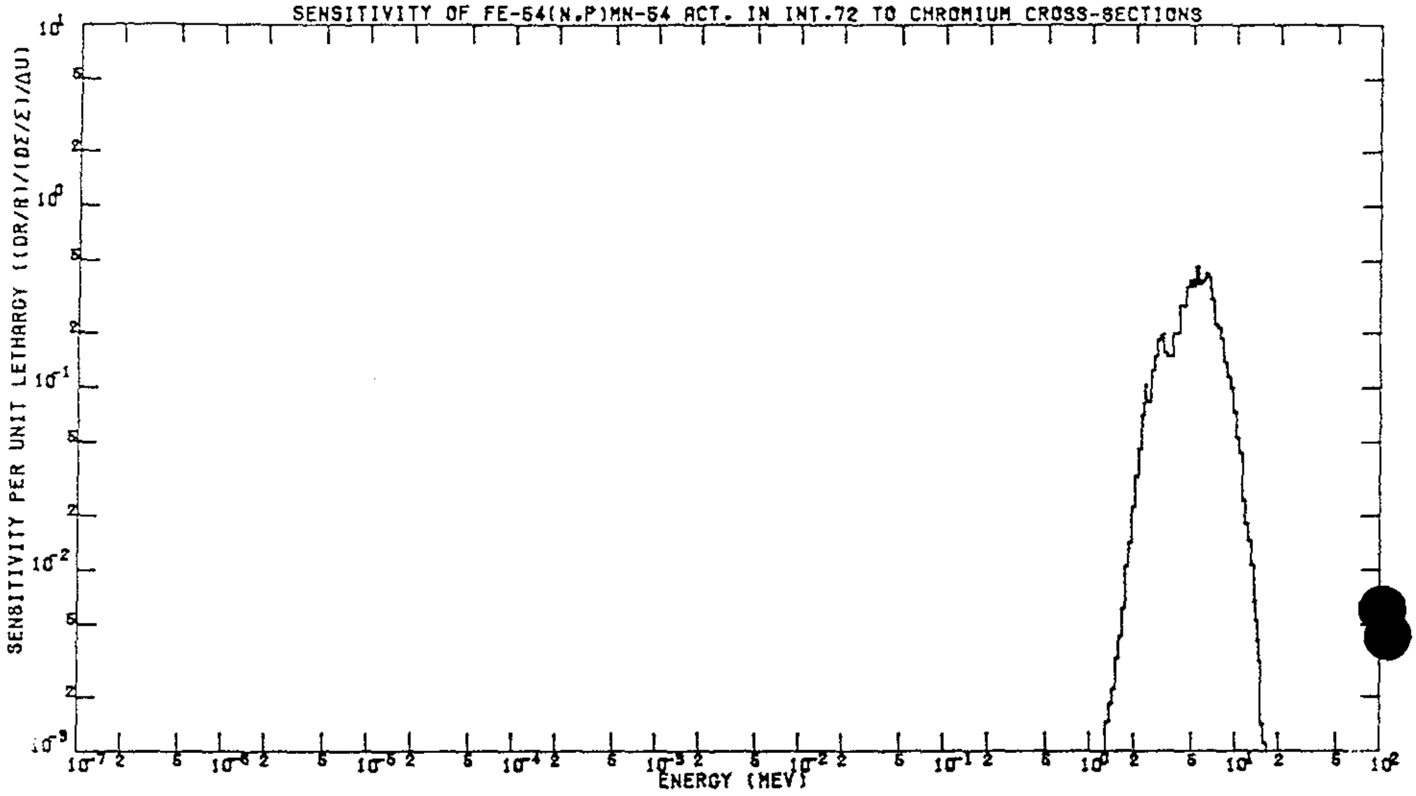

SENSITIVITY OF FE-54(N,P)MN-54 ACT. IN INT.72 TO CHROMIUM CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 4

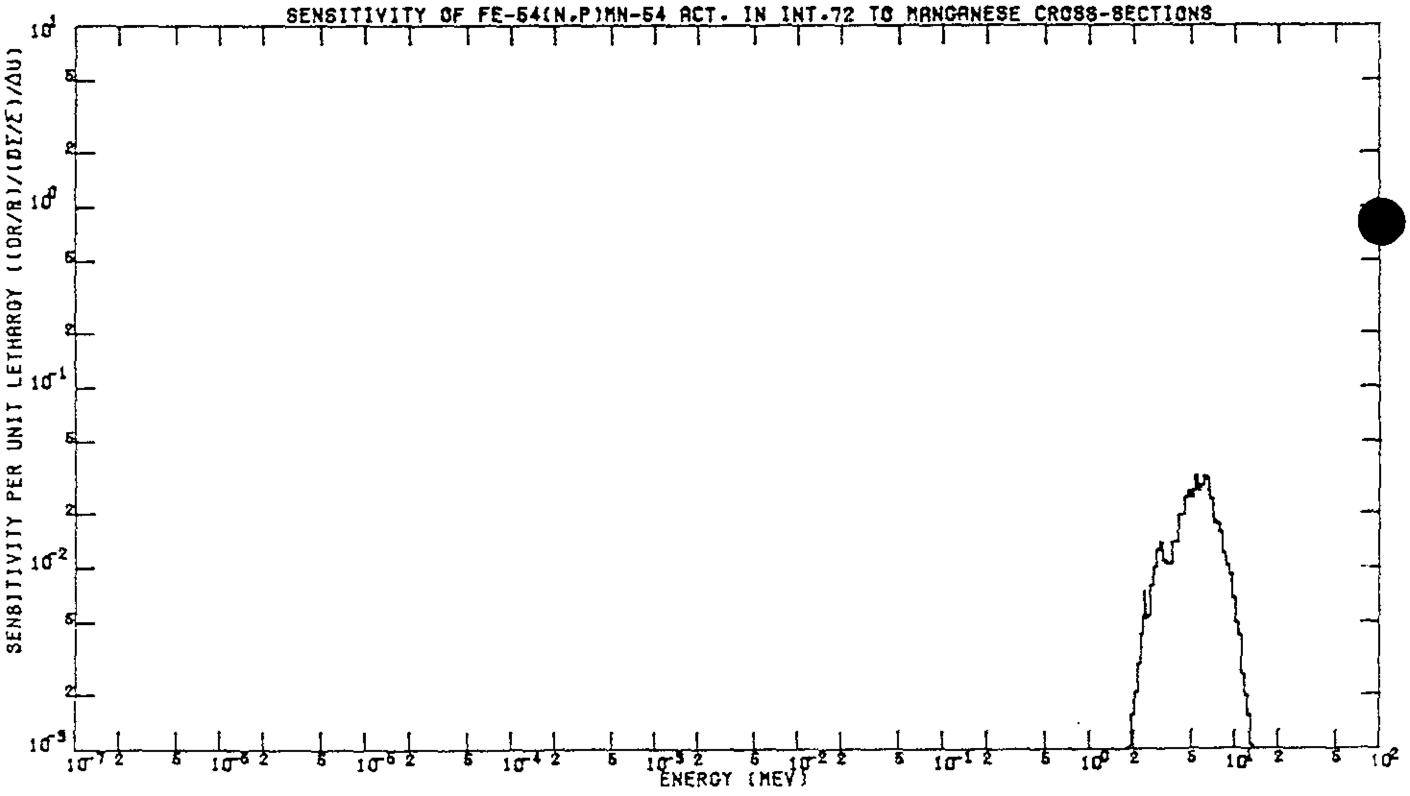

SENSITIVITY OF FE-54(N,P)MN-54 ACT. IN INT.72 TO MANGANESE CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 5

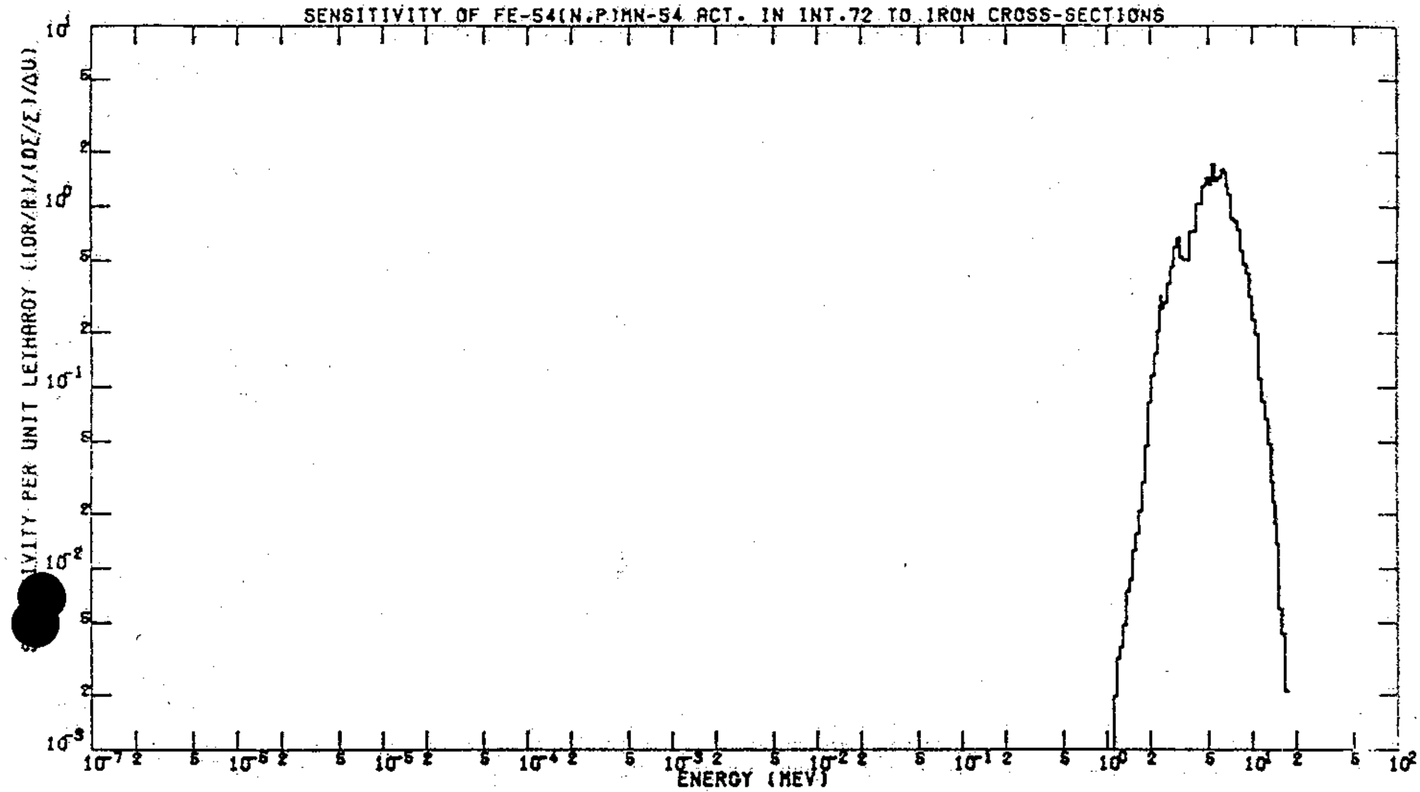

SENSITIVITY OF FE-54(N,P)MN-54 ACT. IN INT.72 TO IRON CROSS-SECTIONS
ENERGY (MEV)

Fig. 6

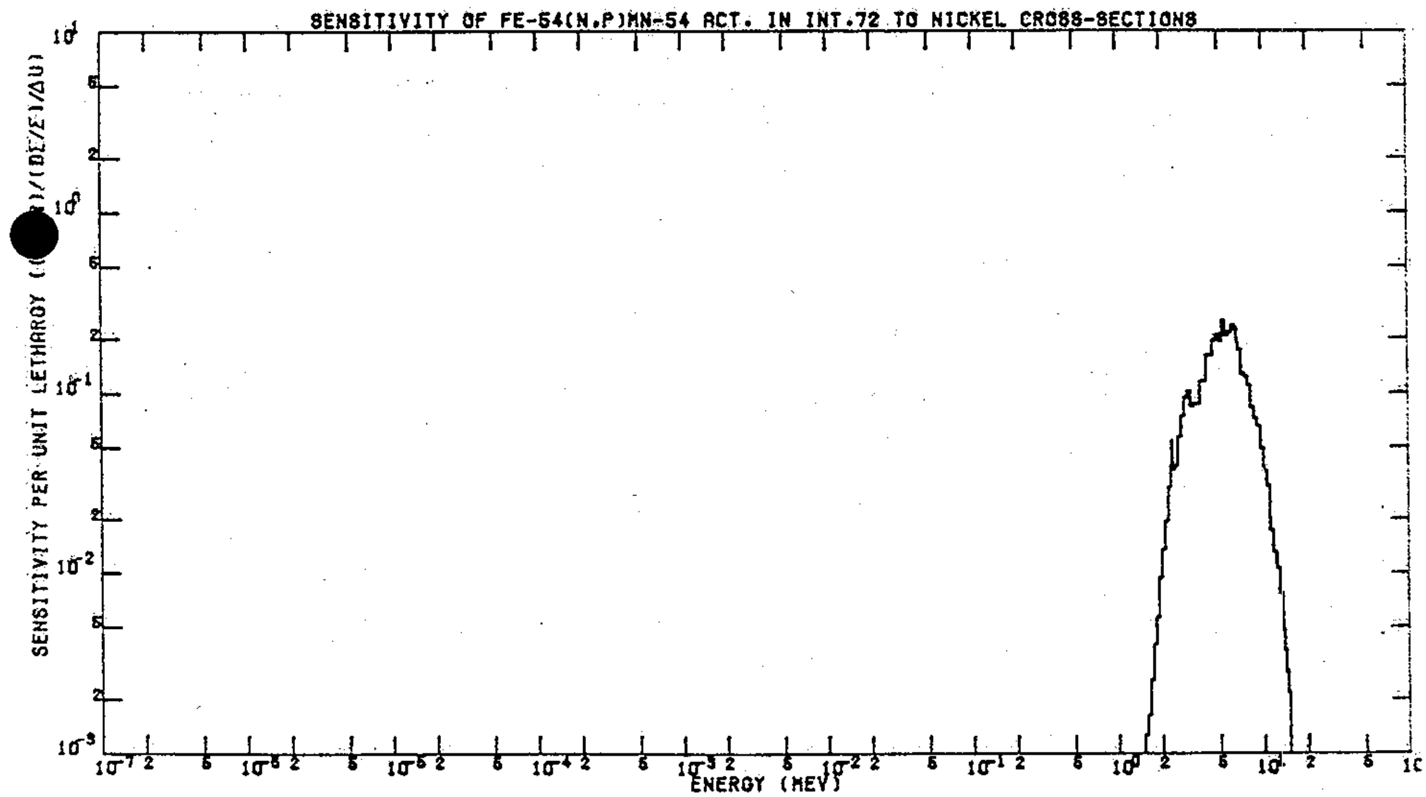

SENSITIVITY OF FE-54(N,P)MN-54 ACT. IN INT.72 TO NICKEL CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY
ENERGY (MEV)

Fig. 7

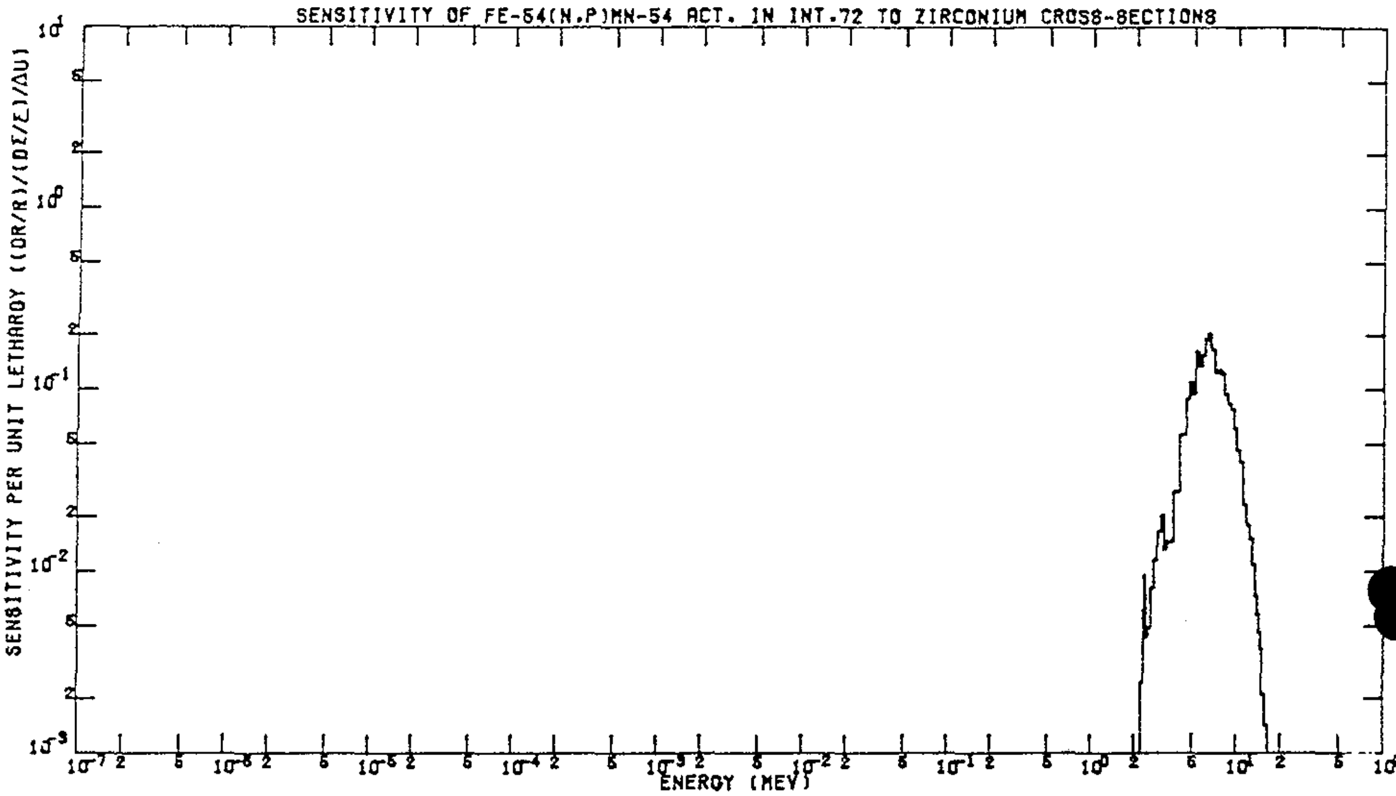

SENSITIVITY OF FE-54(N,P)MN-54 ACT. IN INT.72 TO ZIRCONIUM CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 8

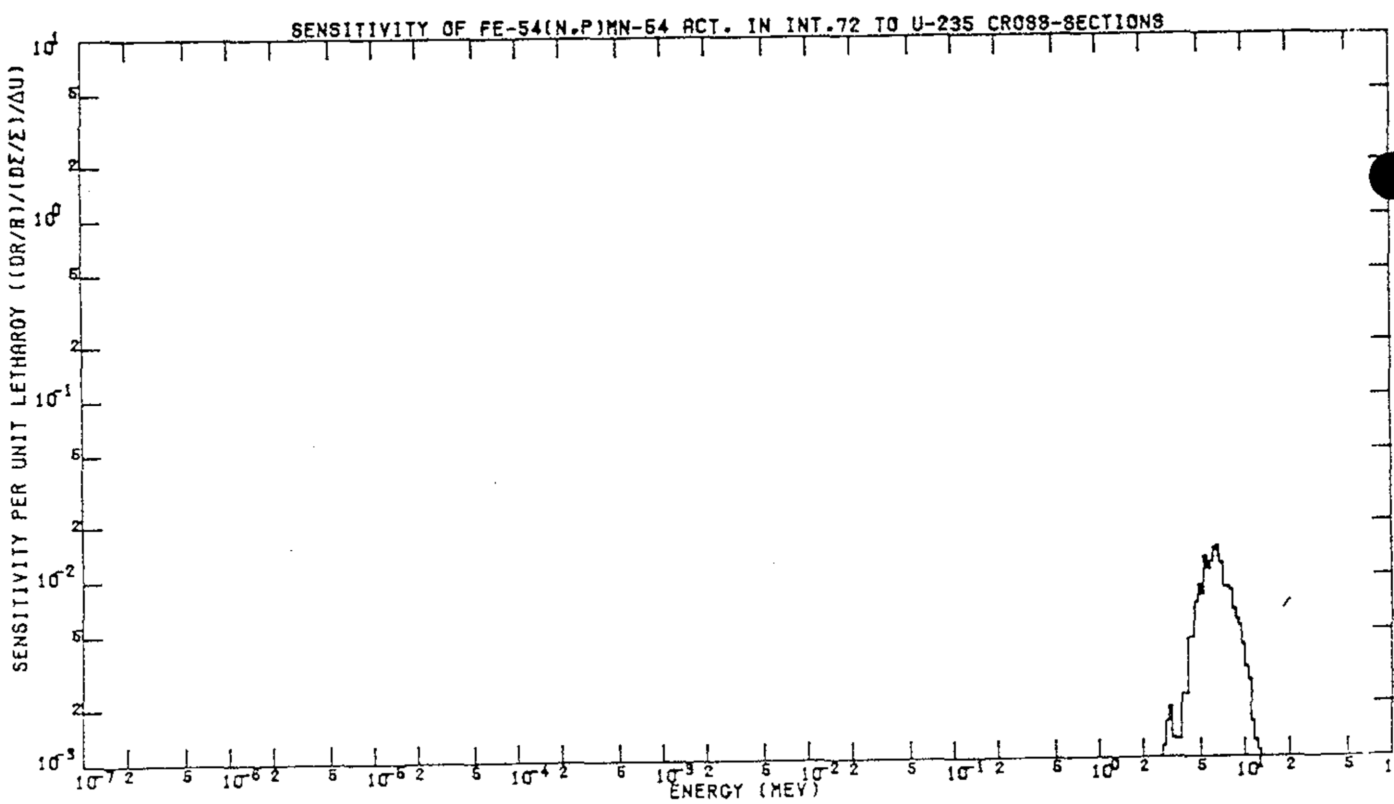
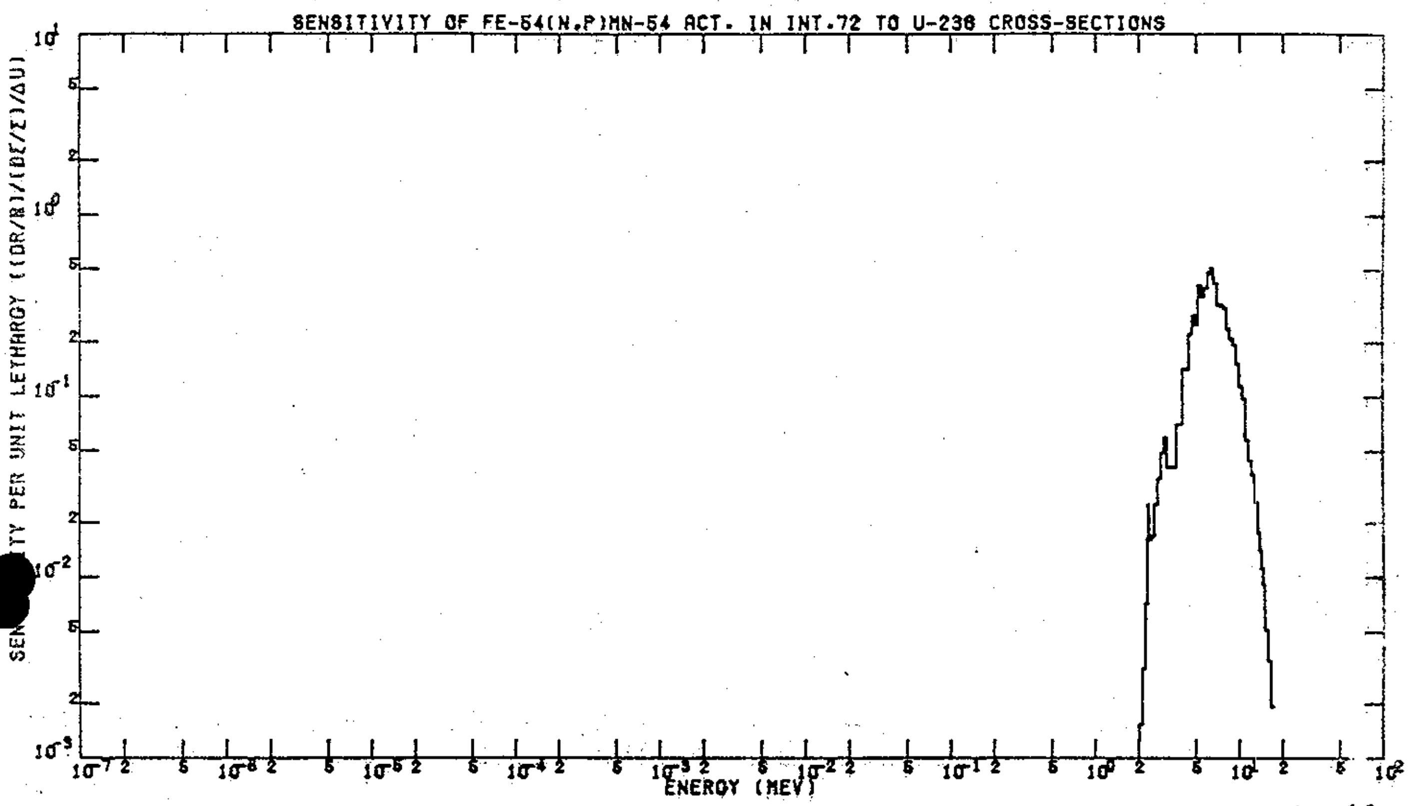
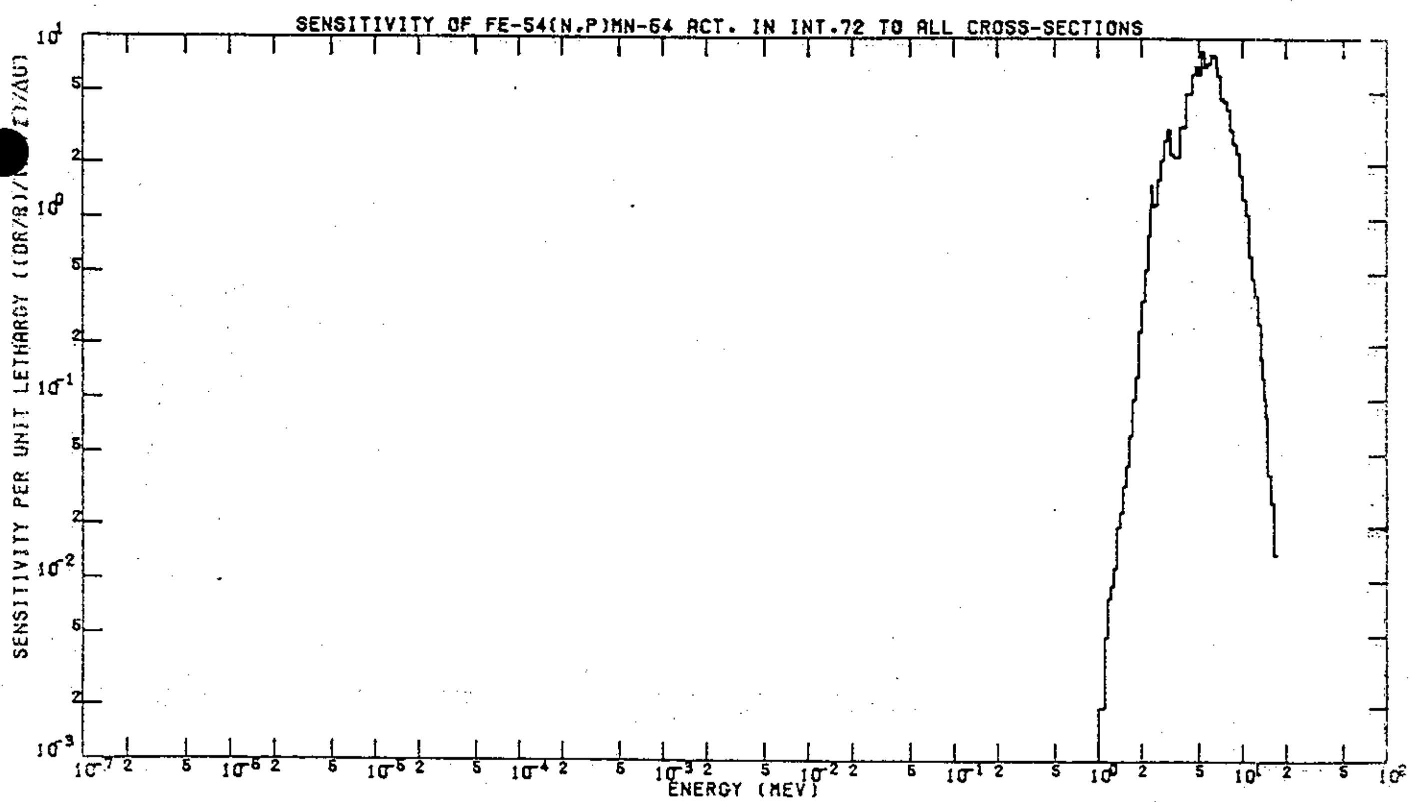

SENSITIVITY OF FE-54(N,P)MN-54 ACT. IN INT.72 TO U-235 CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 9



SENSITIVITY OF FE-54(N,P)MN-54 ACT. IN INT.72 TO U-238 CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 10



SENSITIVITY OF FE-54(N,P)MN-54 ACT. IN INT.72 TO ALL CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 11

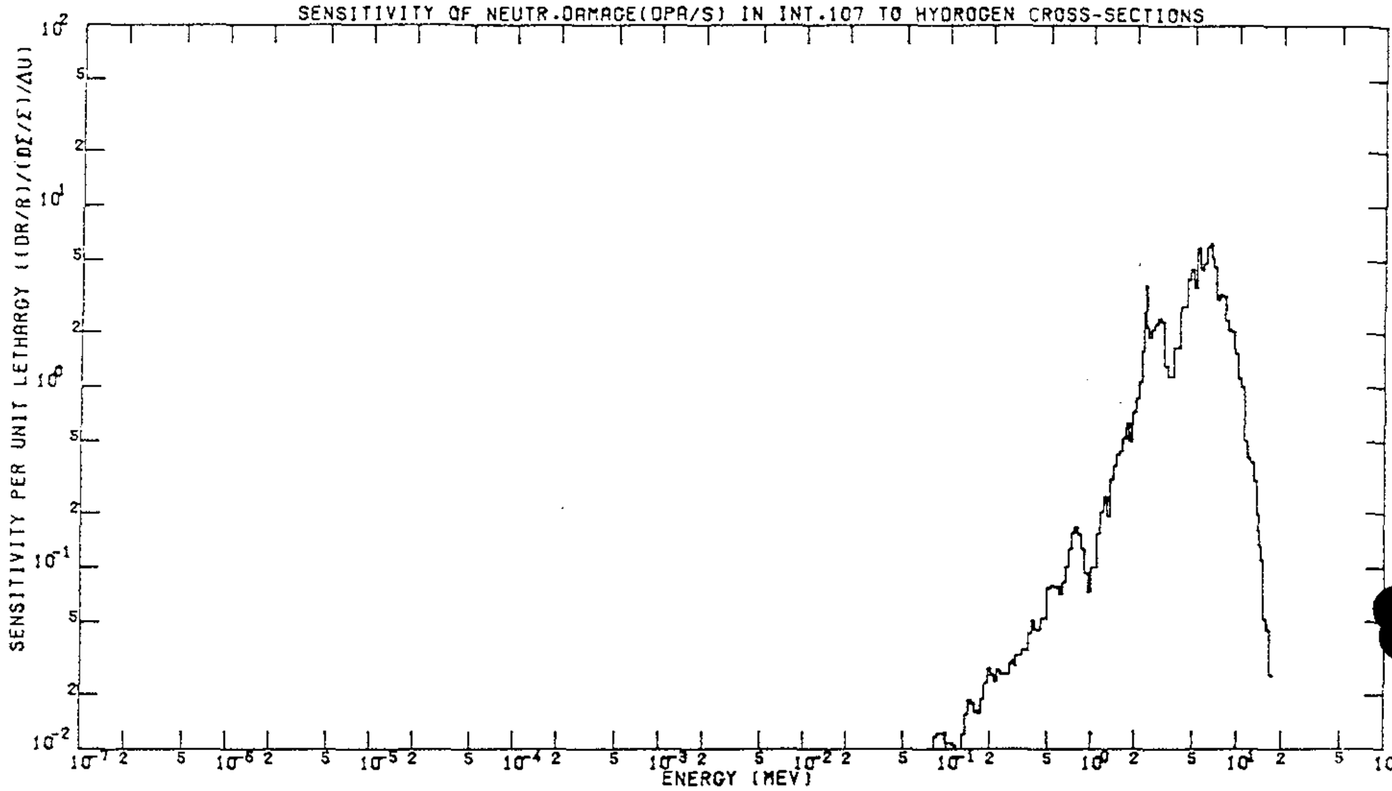

SENSITIVITY OF NEUTR.DAMAGE(DPA/S) IN INT.107 TO HYDROGEN CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 12

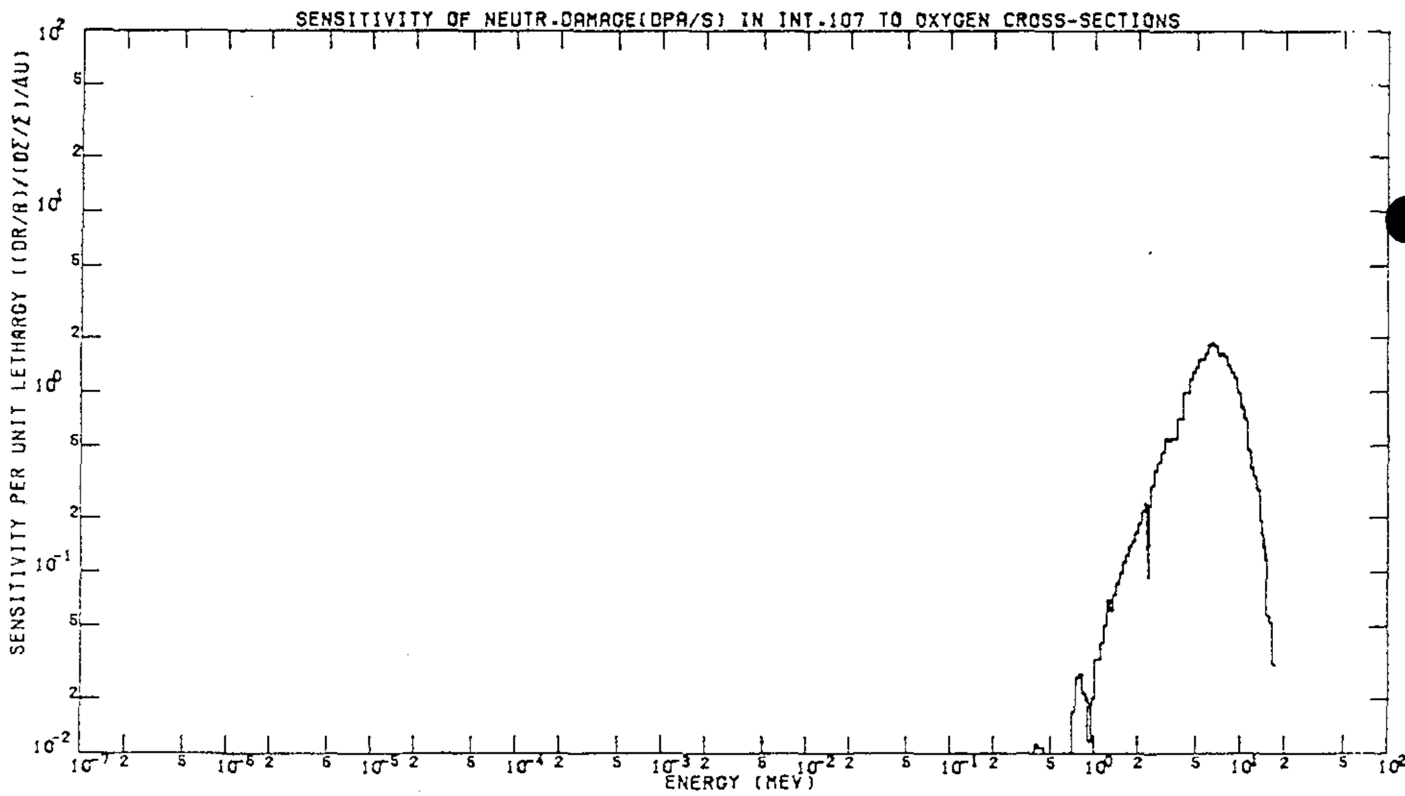

SENSITIVITY OF NEUTR.DAMAGE(DPA/S) IN INT.107 TO OXYGEN CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 13

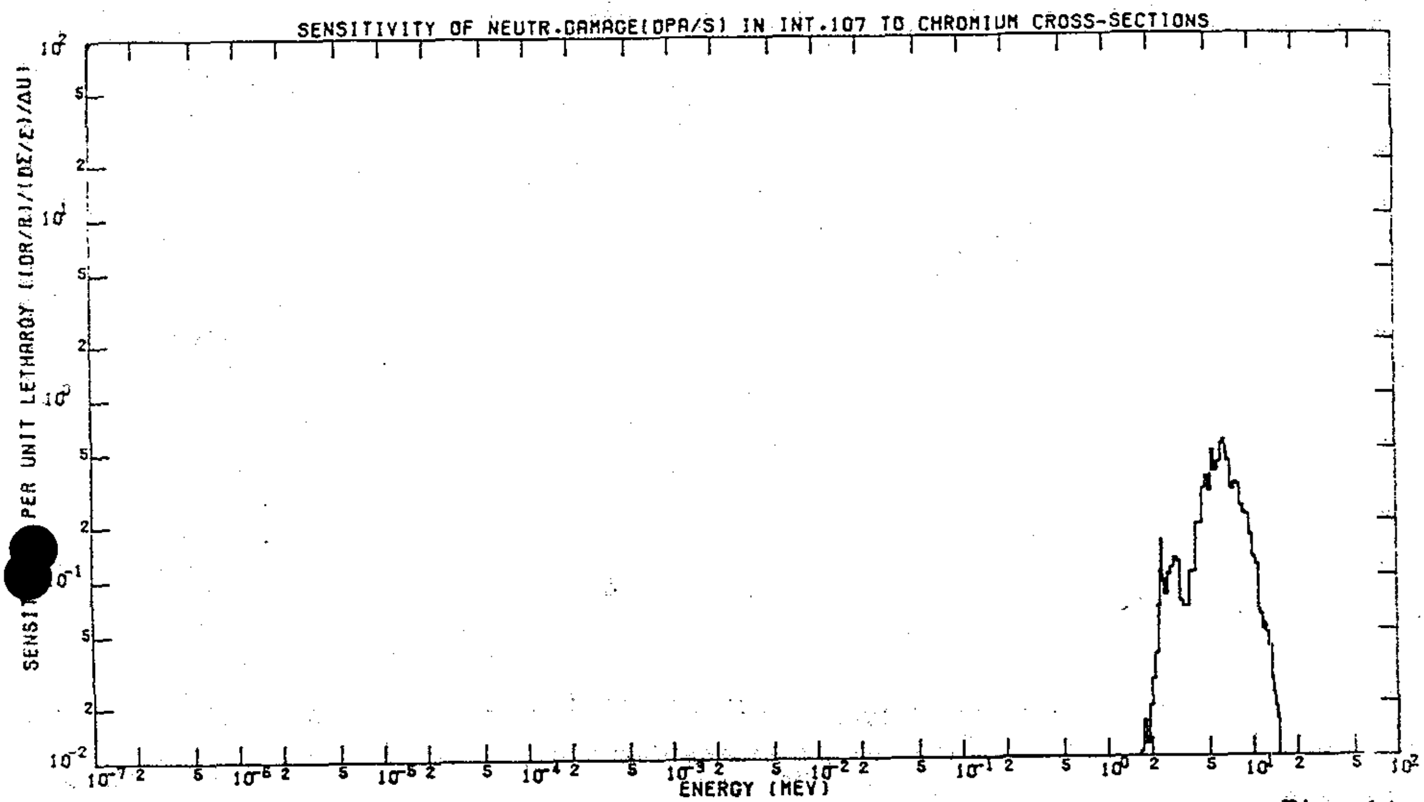

SENSITIVITY OF NEUTR.DAMAGE(DPA/S) IN INT.107 TO CHROMIUM CROSS-SECTIONS
ENERGY (MEV)

Fig. 14

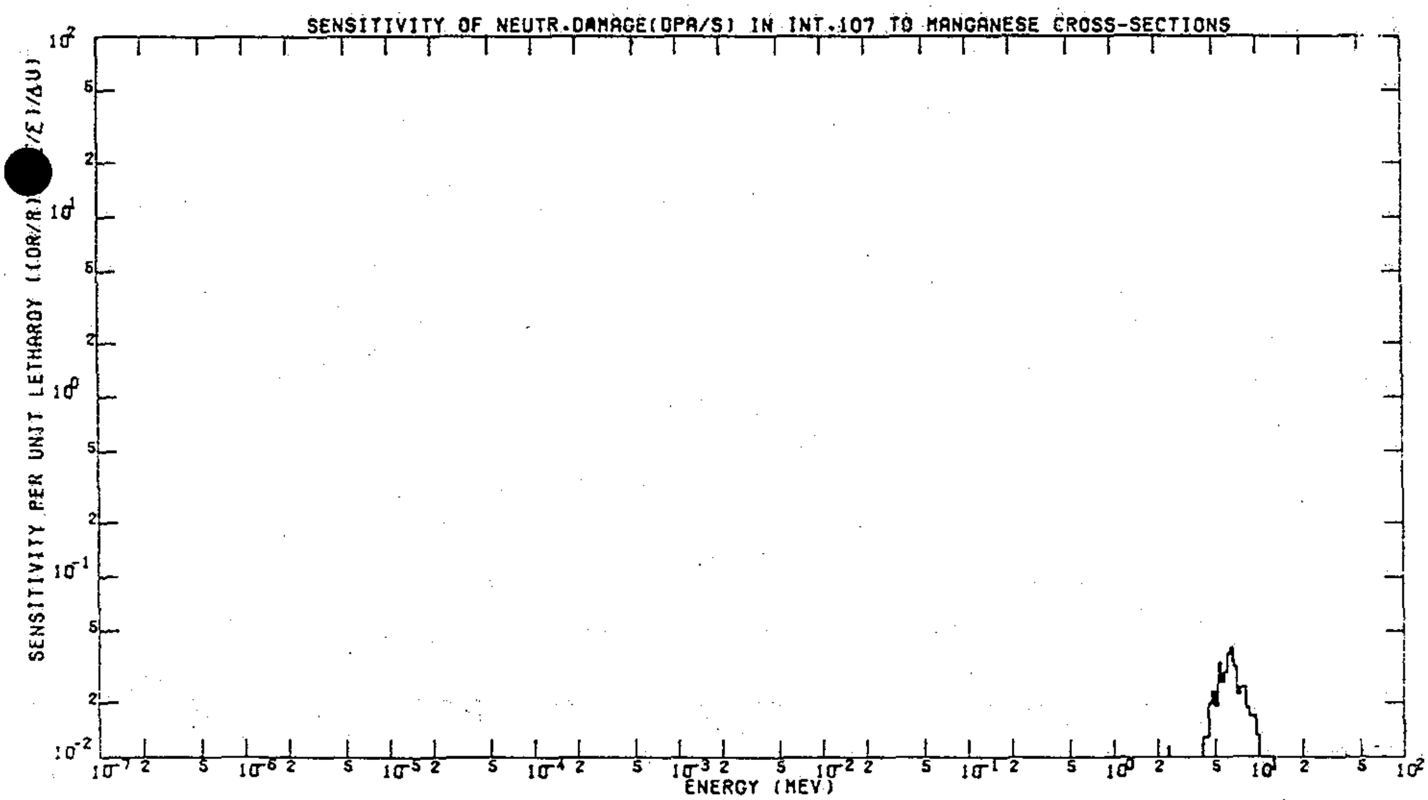

SENSITIVITY OF NEUTR.DAMAGE(DPA/S) IN INT.107 TO MANGANESE CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 15

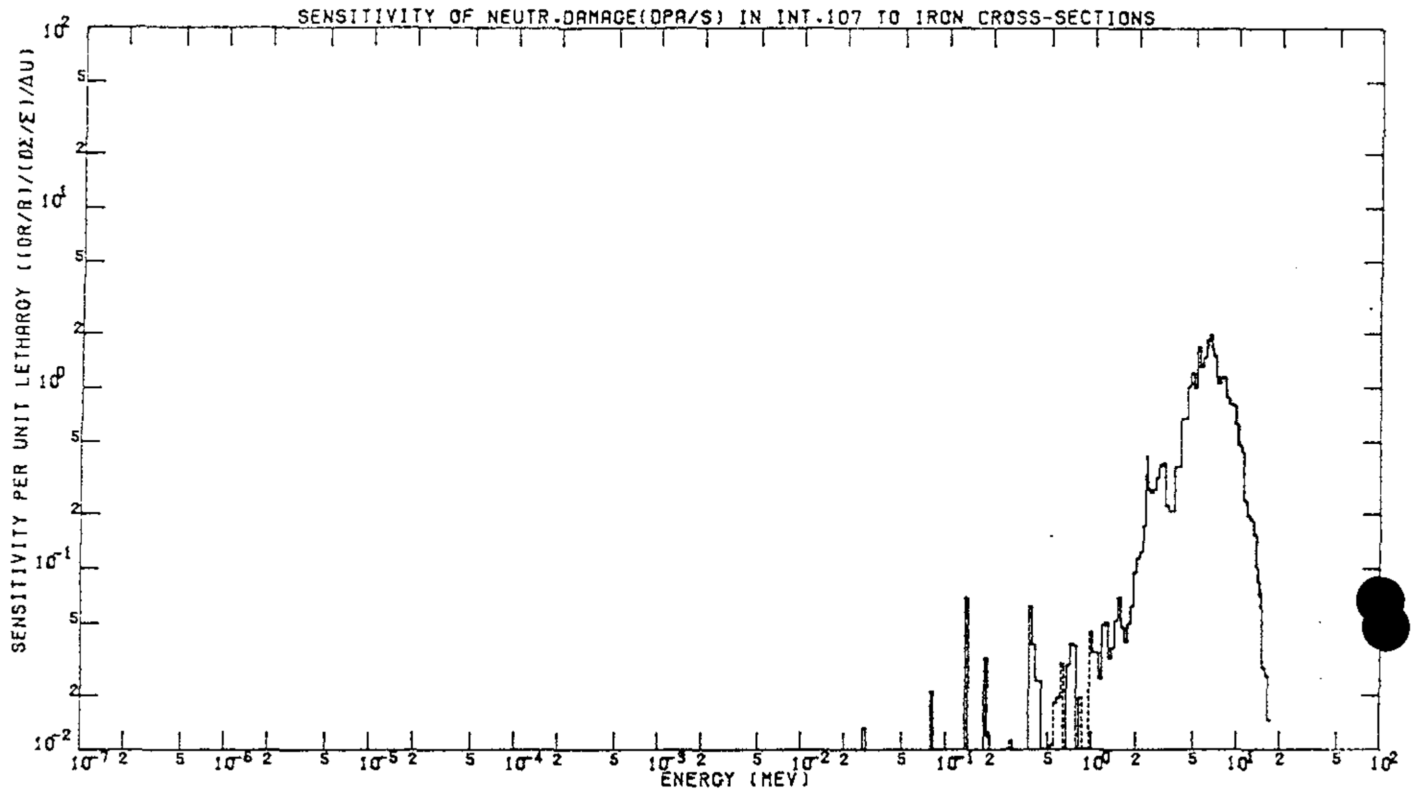

SENSITIVITY OF NEUTR.DAMAGE(DPA/S) IN INT.107 TO IRON CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 16

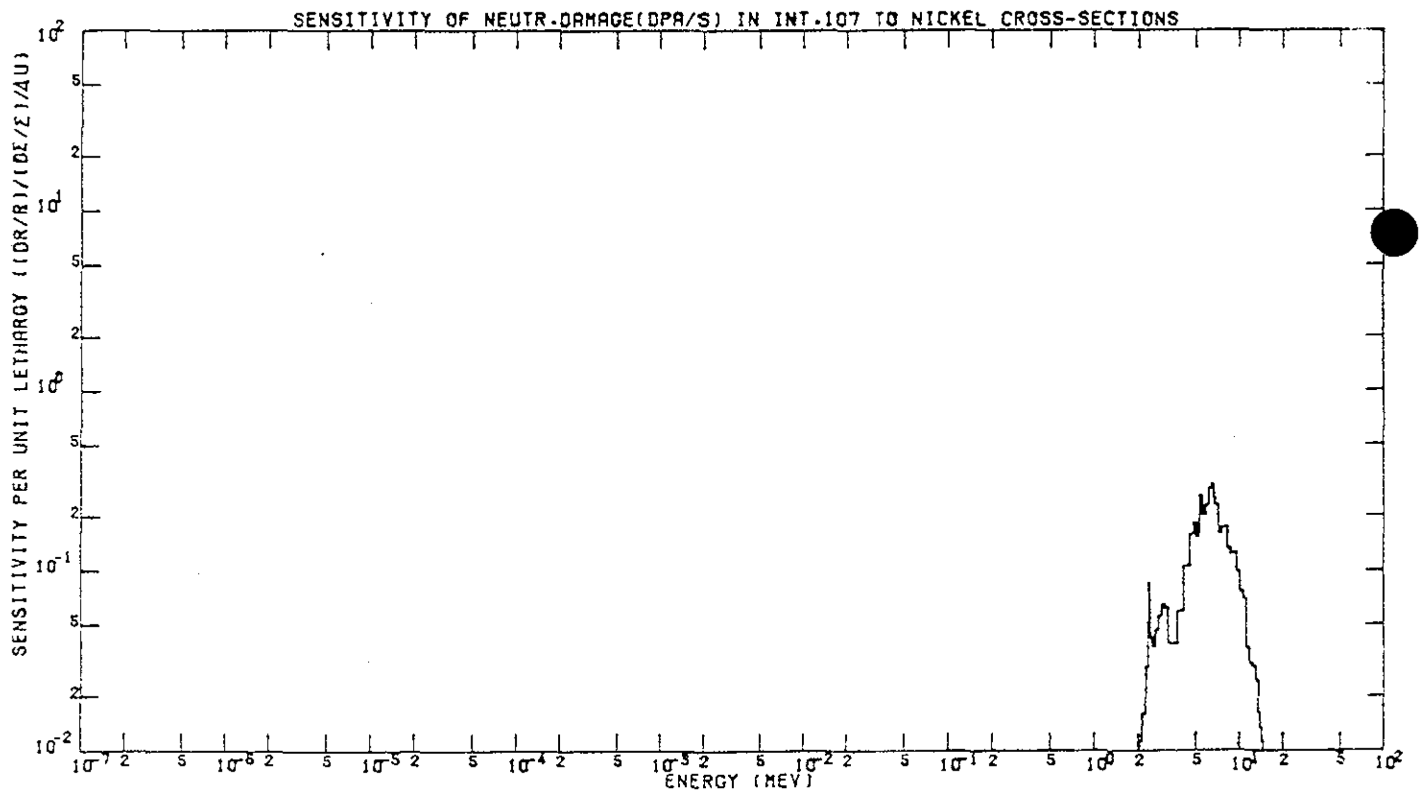

SENSITIVITY OF NEUTR.DAMAGE(DPA/S) IN INT.107 TO NICKEL CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 17

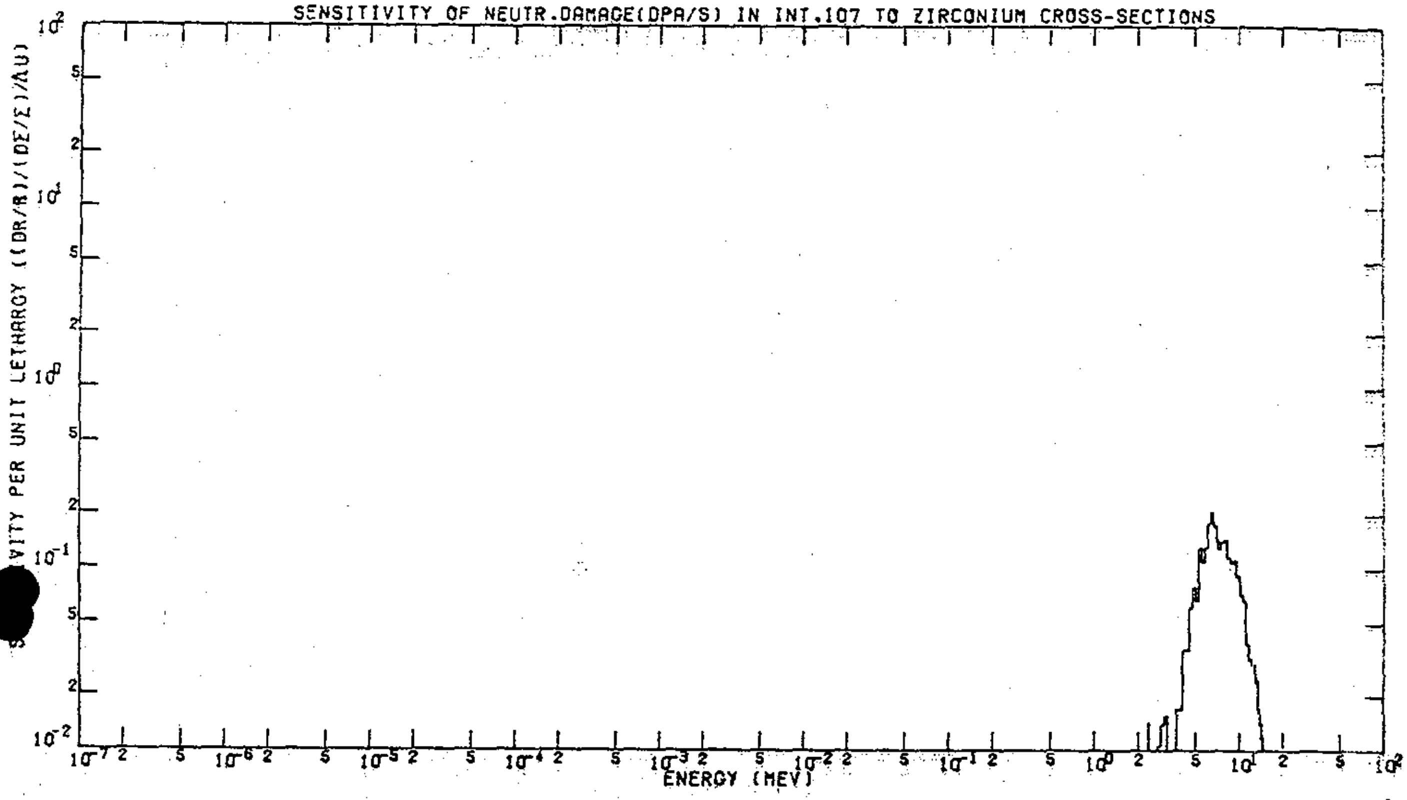

SENSITIVITY OF NEUTR.DAMAGE(DPA/S) IN INT.107 TO ZIRCONIUM CROSS-SECTIONS
ENERGY (MEV)
VITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)

Fig. 18

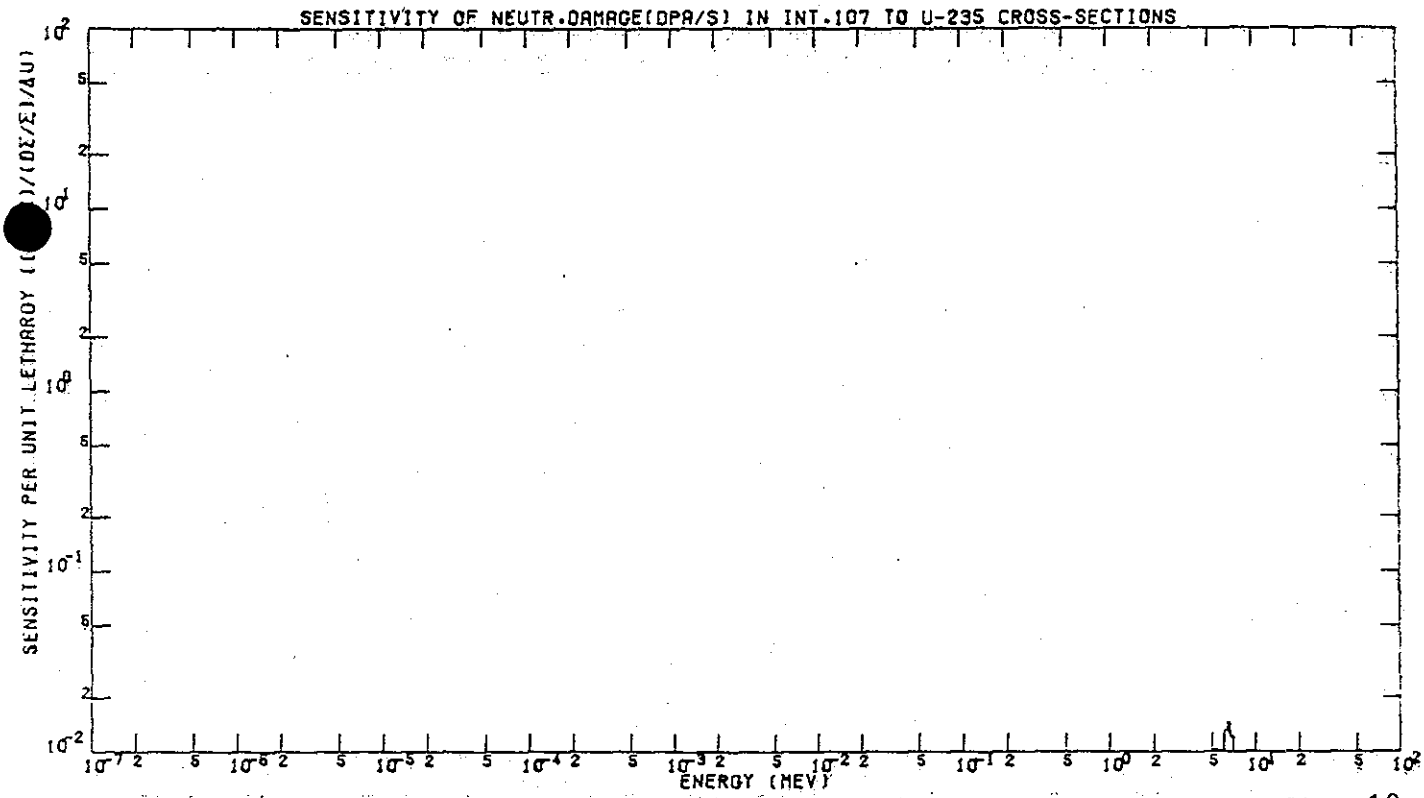

SENSITIVITY OF NEUTR.DAMAGE(DPA/S) IN INT.107 TO U-235 CROSS-SECTIONS
ENERGY (MEV)
SENSITIVITY PER UNIT LETHARGY ((

Fig. 19

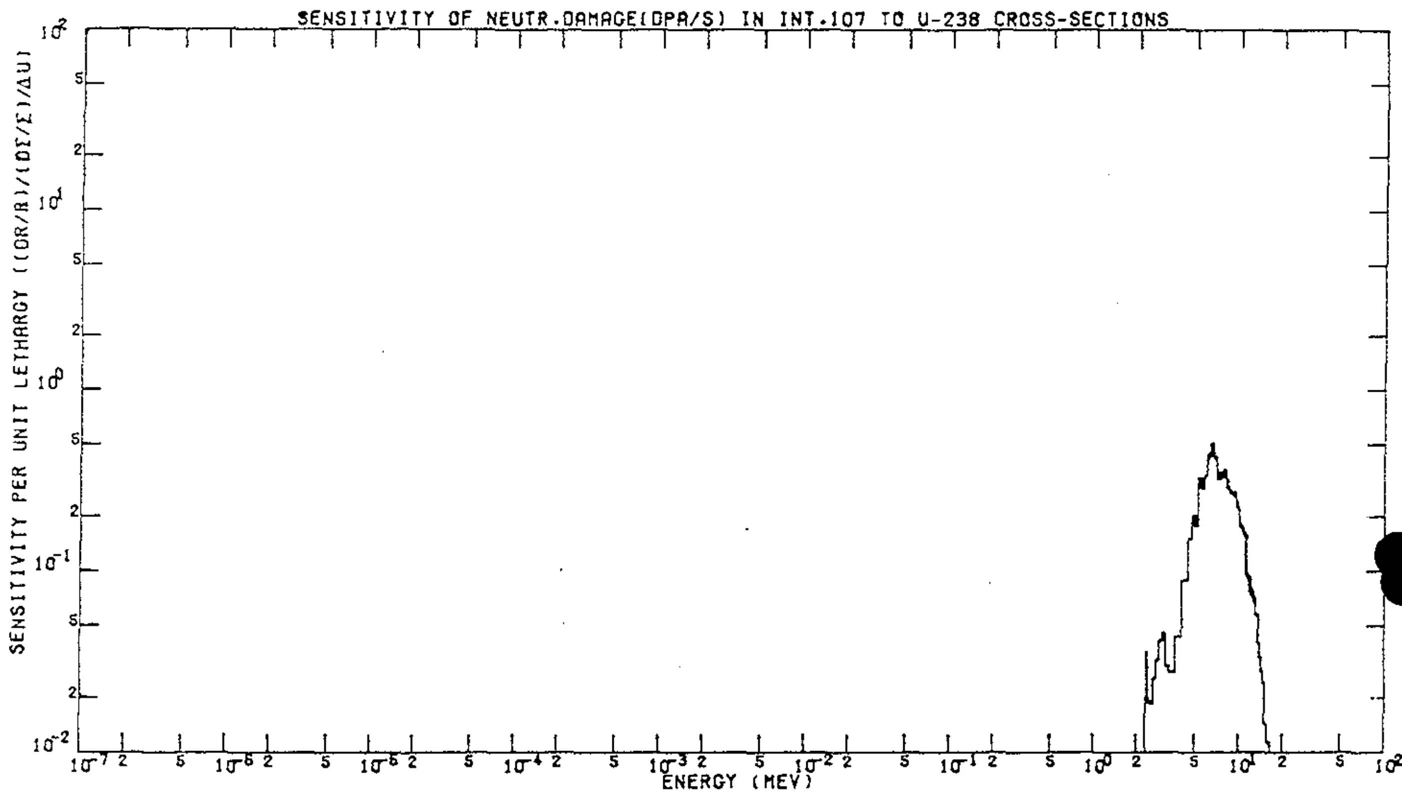


Fig. 20

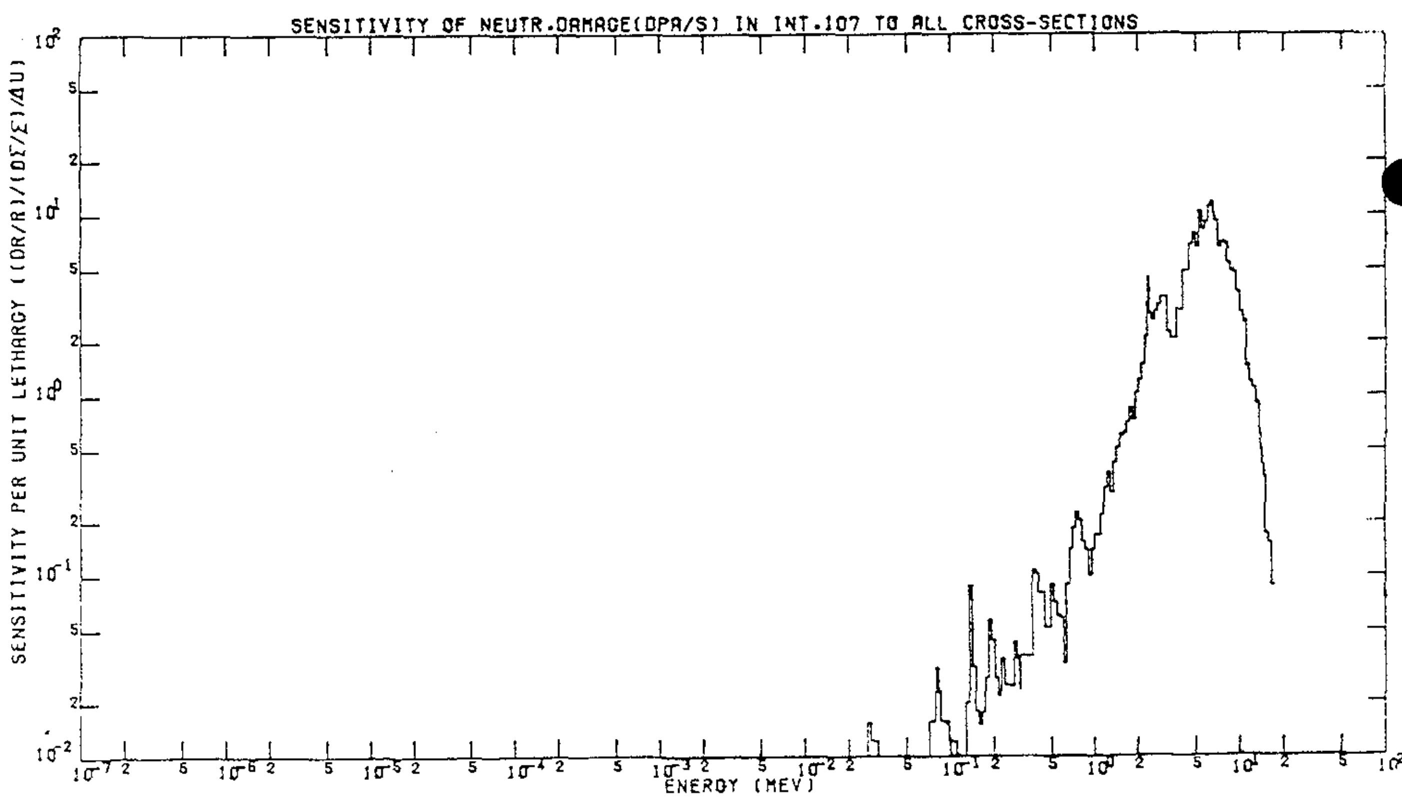


Fig. 21

91090027

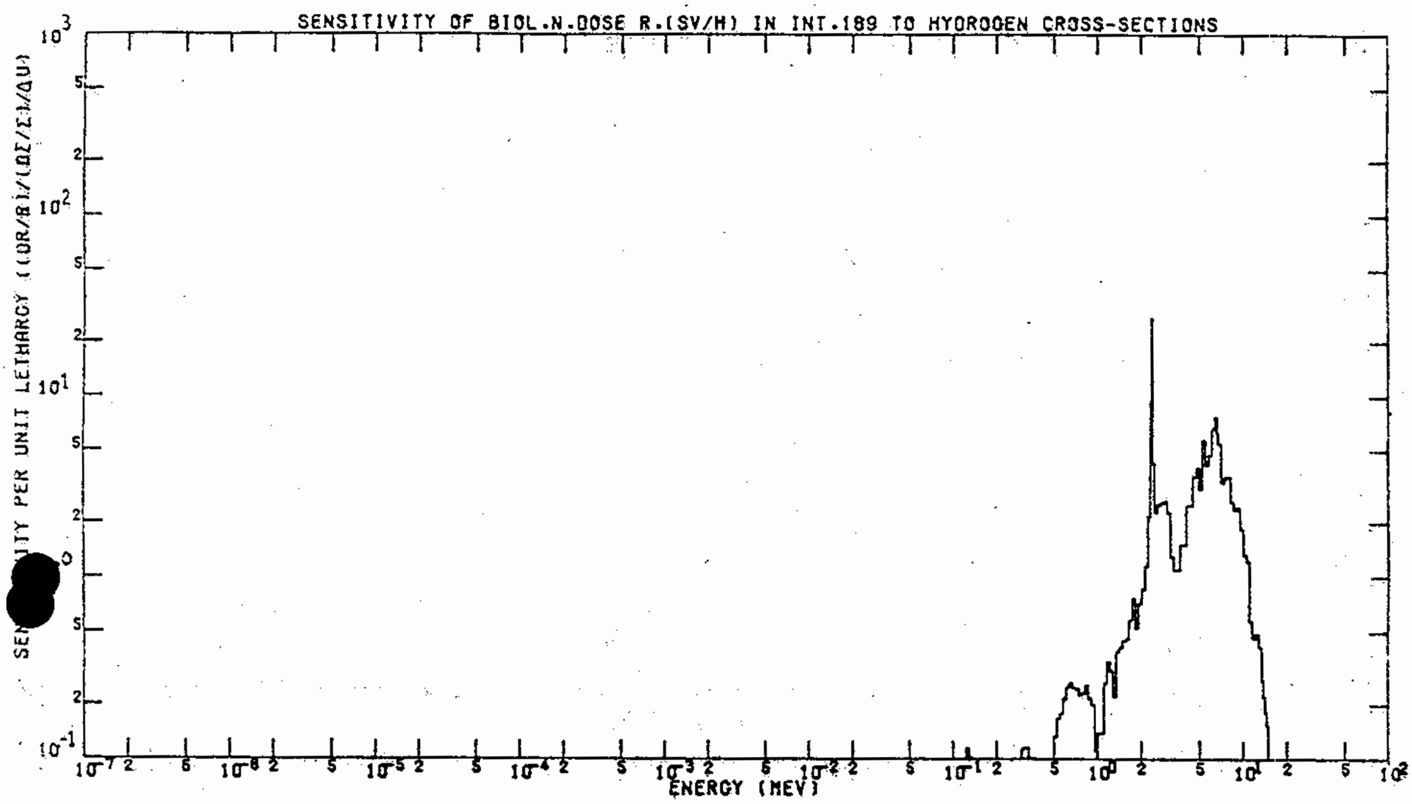

SENSITIVITY OF BIOL.N.DOSE R.(SV/H) IN INT.189 TO HYDROGEN CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 22

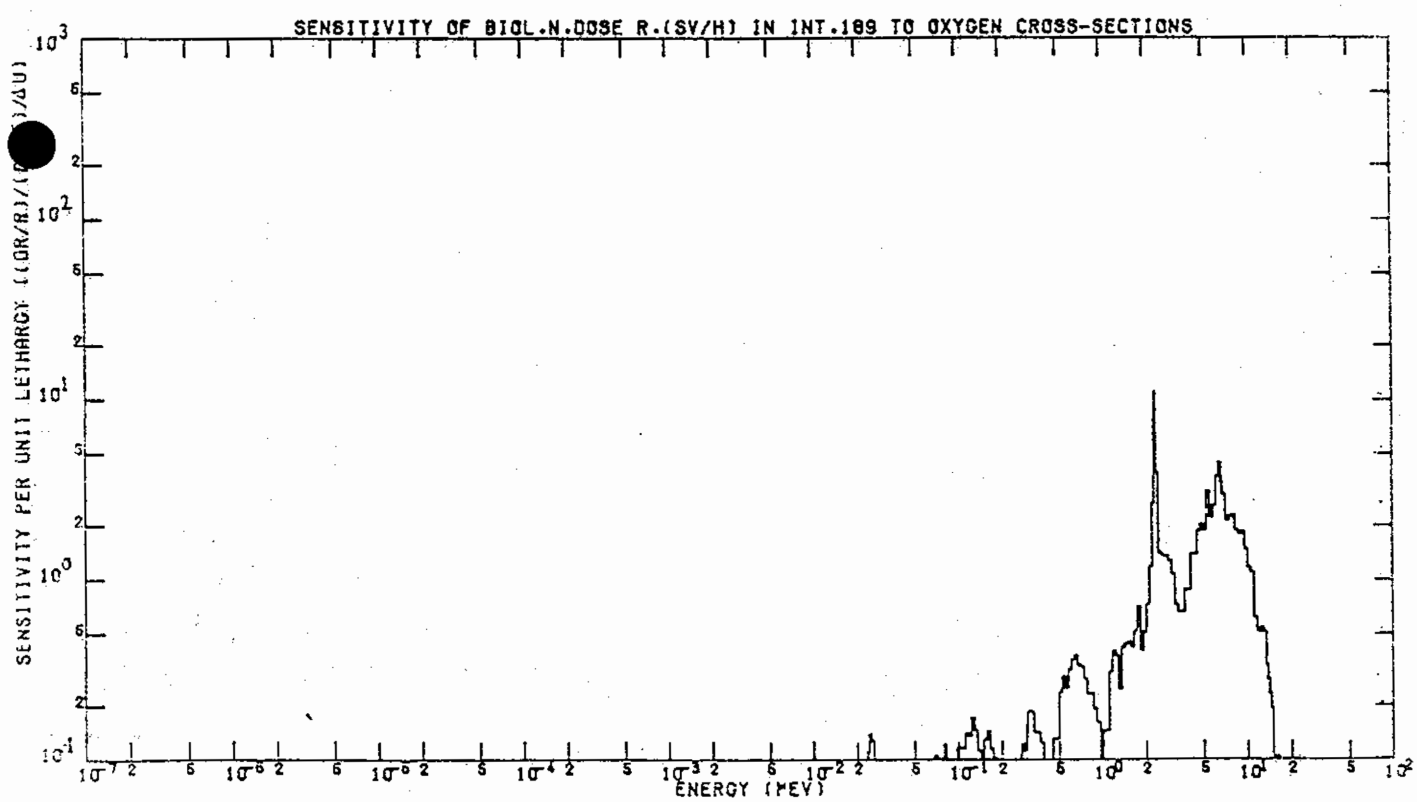

SENSITIVITY OF BIOL.N.DOSE R.(SV/H) IN INT.189 TO OXYGEN CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 23

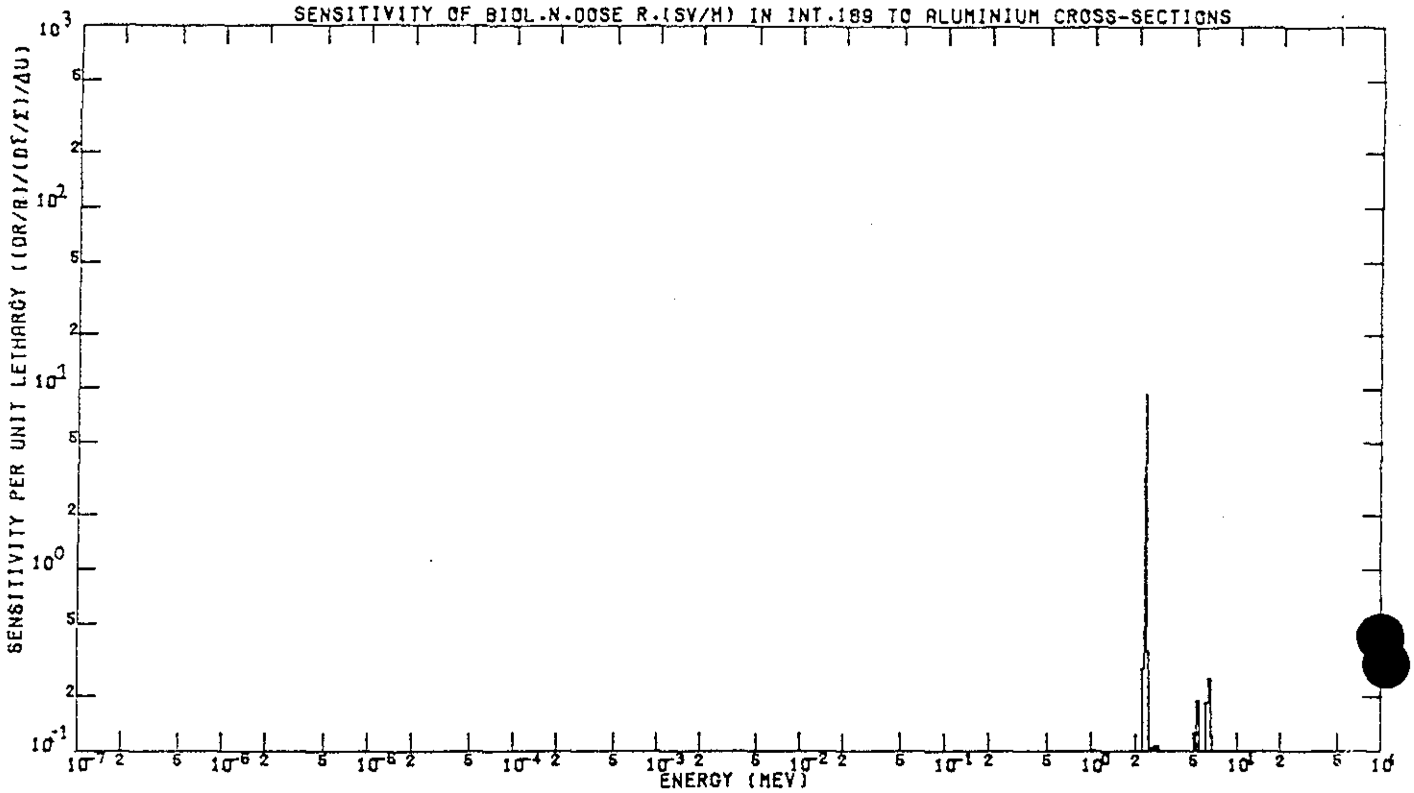

SENSITIVITY OF BIOL.N.DOSE R.(SV/H) IN INT.189 TO ALUMINIUM CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 24

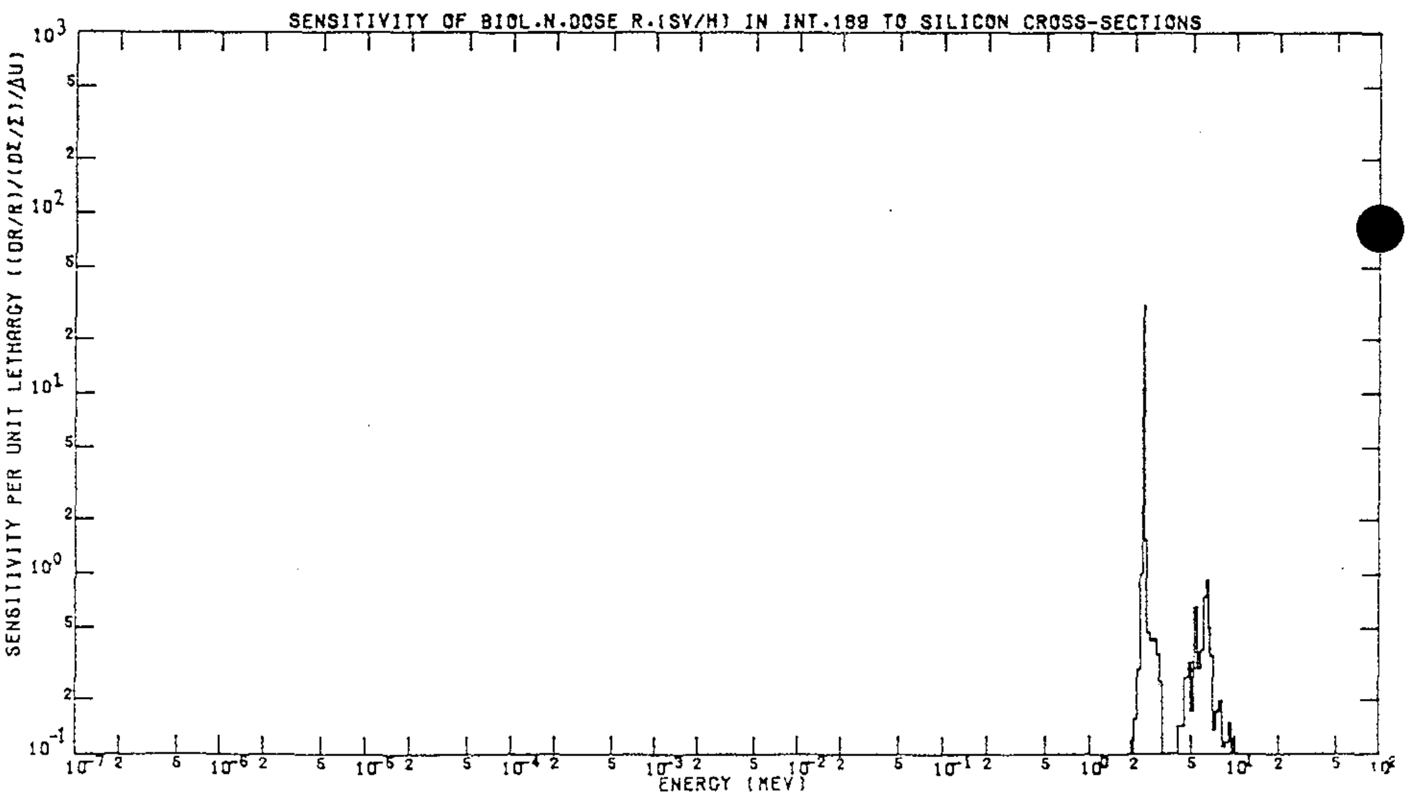

SENSITIVITY OF BIOL.N.DOSE R.(SV/H) IN INT.189 TO SILICON CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 25

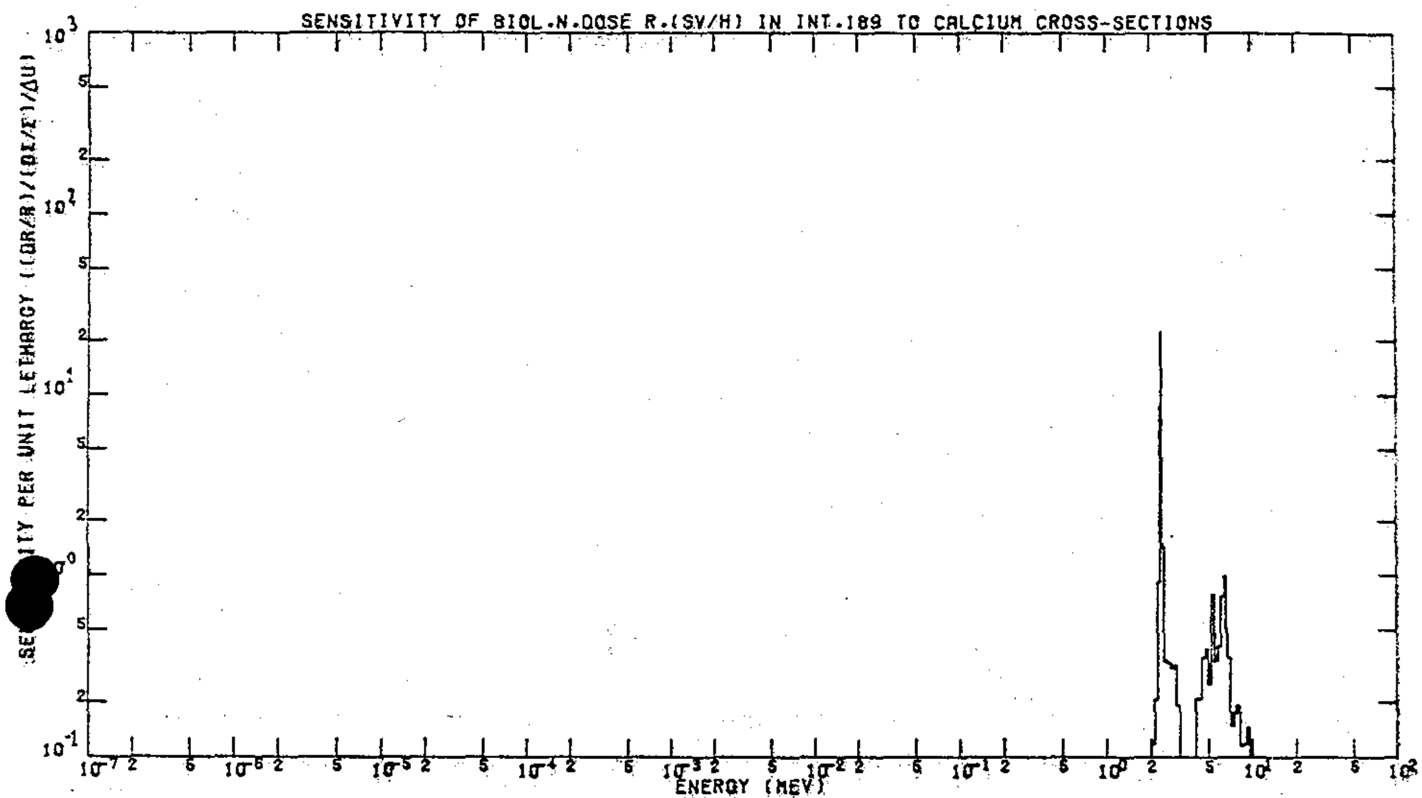

SENSITIVITY OF BIOL.N.DOSE R.(SV/H) IN INT.189 TO CALCIUM CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 26

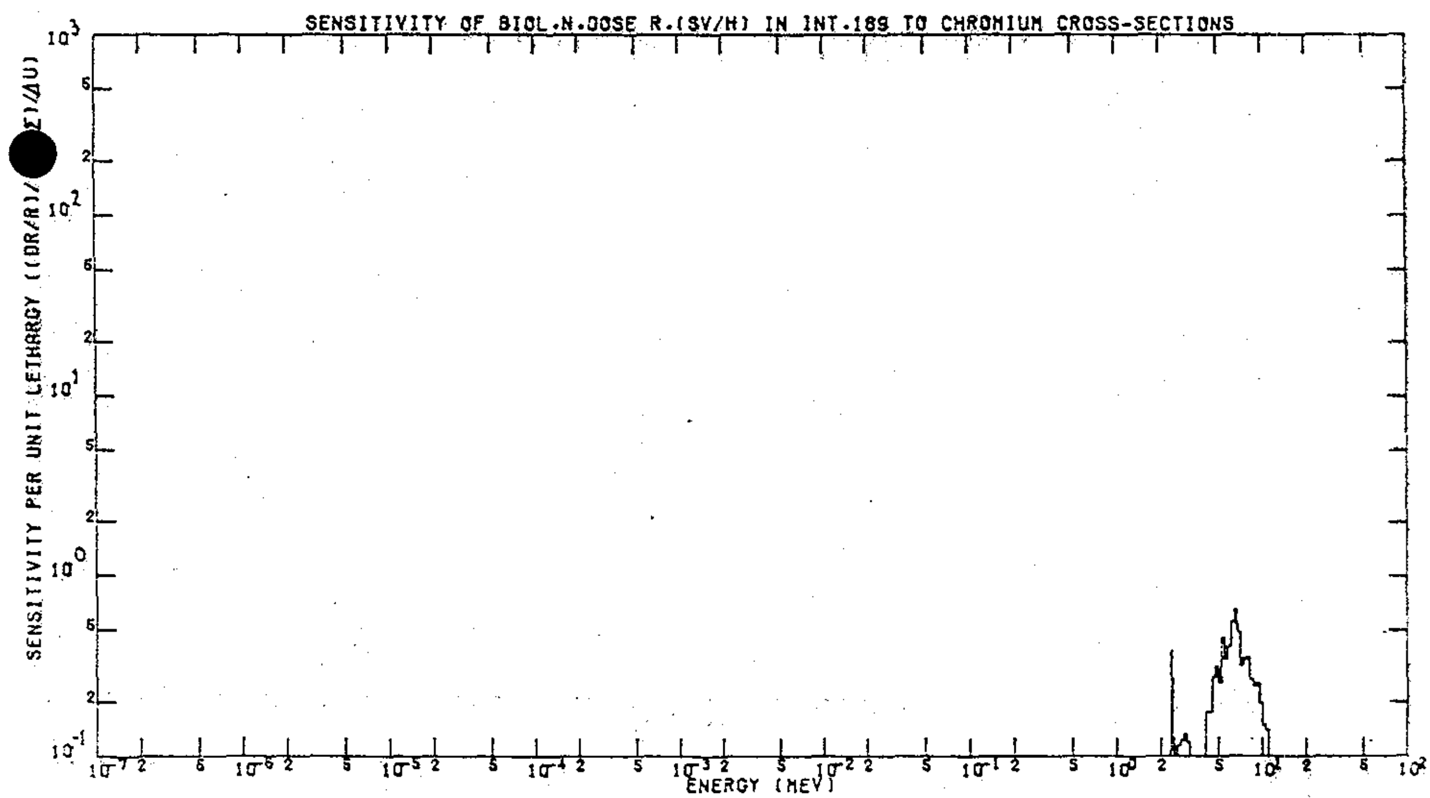

SENSITIVITY OF BIOL.N.DOSE R.(SV/H) IN INT.189 TO CHROMIUM CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 27

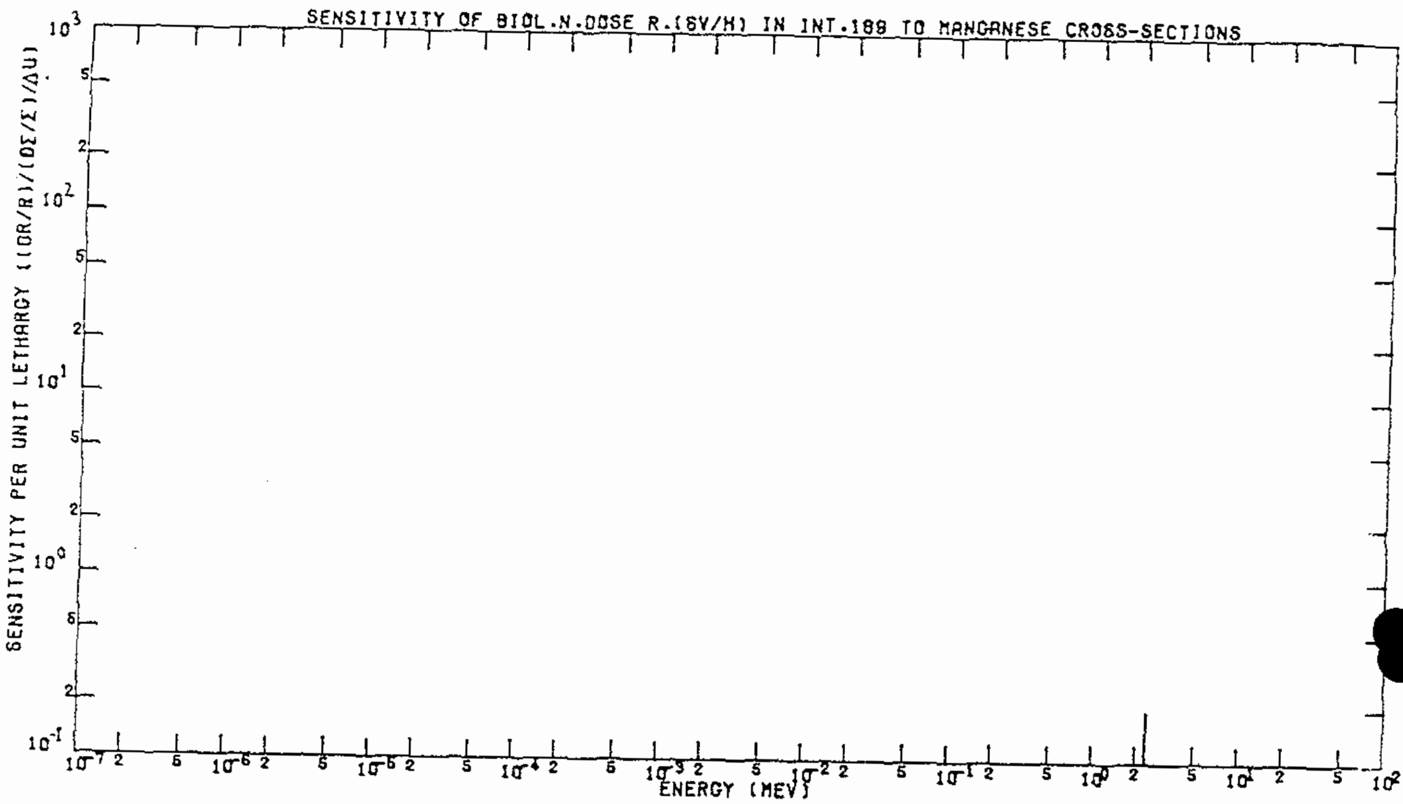

SENSITIVITY OF BIOL.N.DOSE R.(SV/H) IN INT.189 TO MANGANESE CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 28

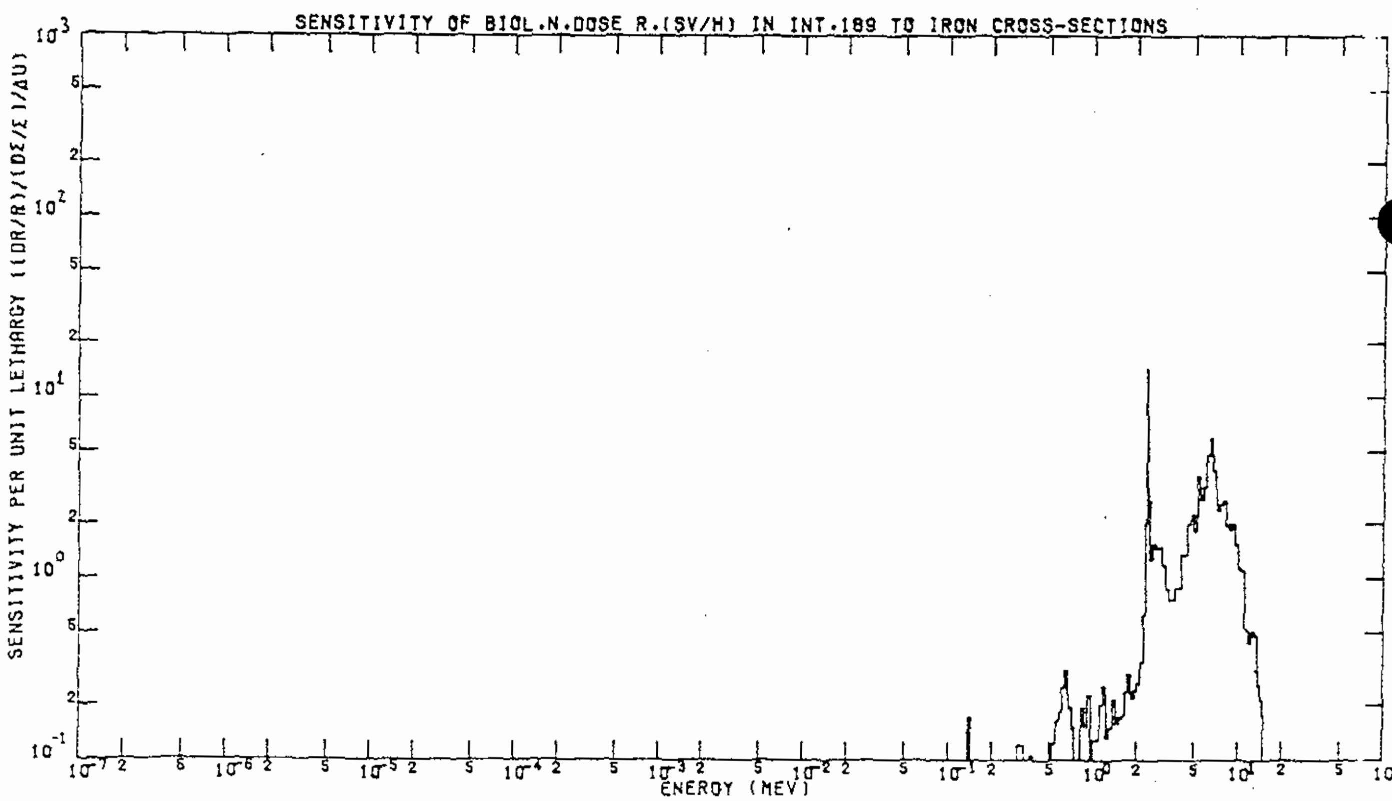

SENSITIVITY OF BIOL.N.DOSE R.(SV/H) IN INT.189 TO IRON CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 29

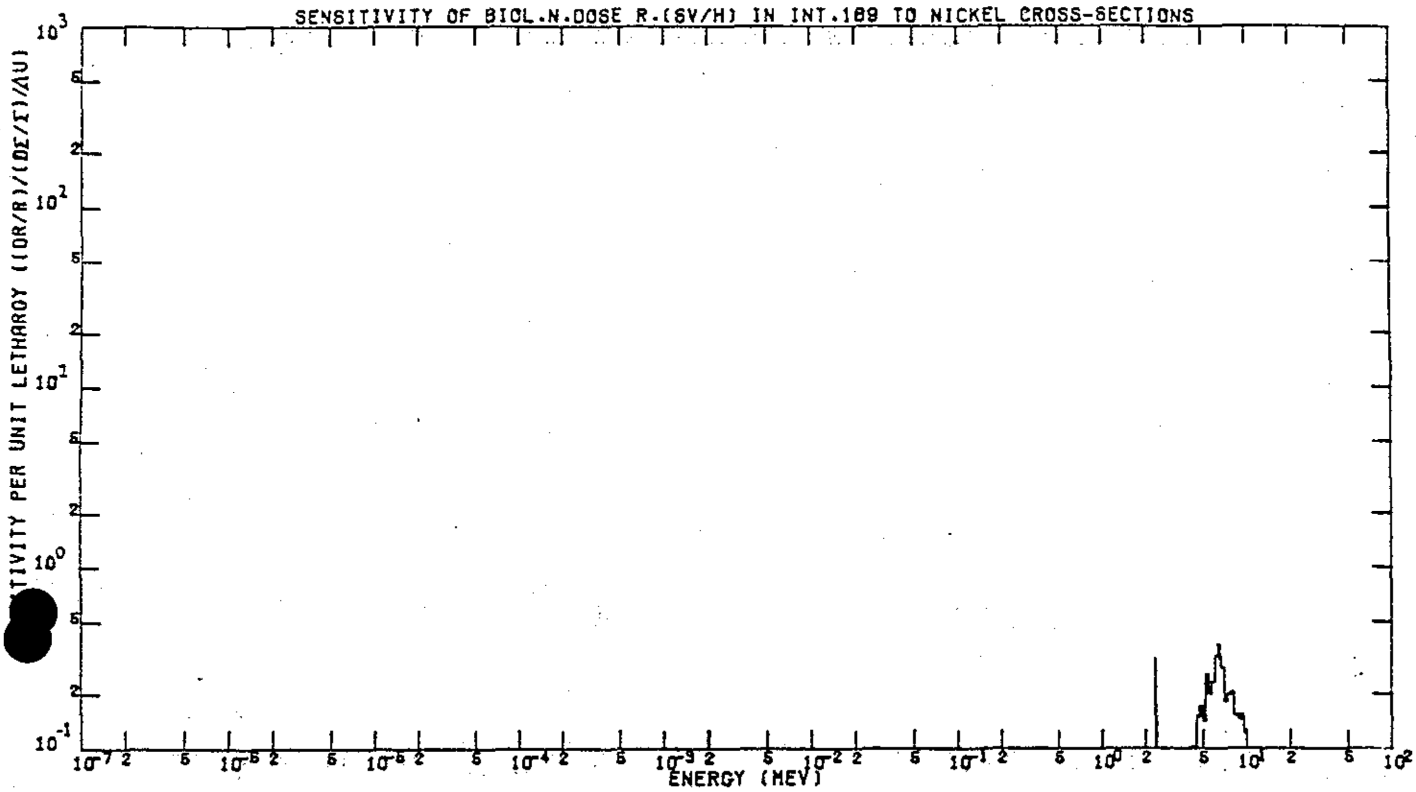

SENSITIVITY OF BIOL.N.DOSE R.(SV/H) IN INT.189 TO NICKEL CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 30

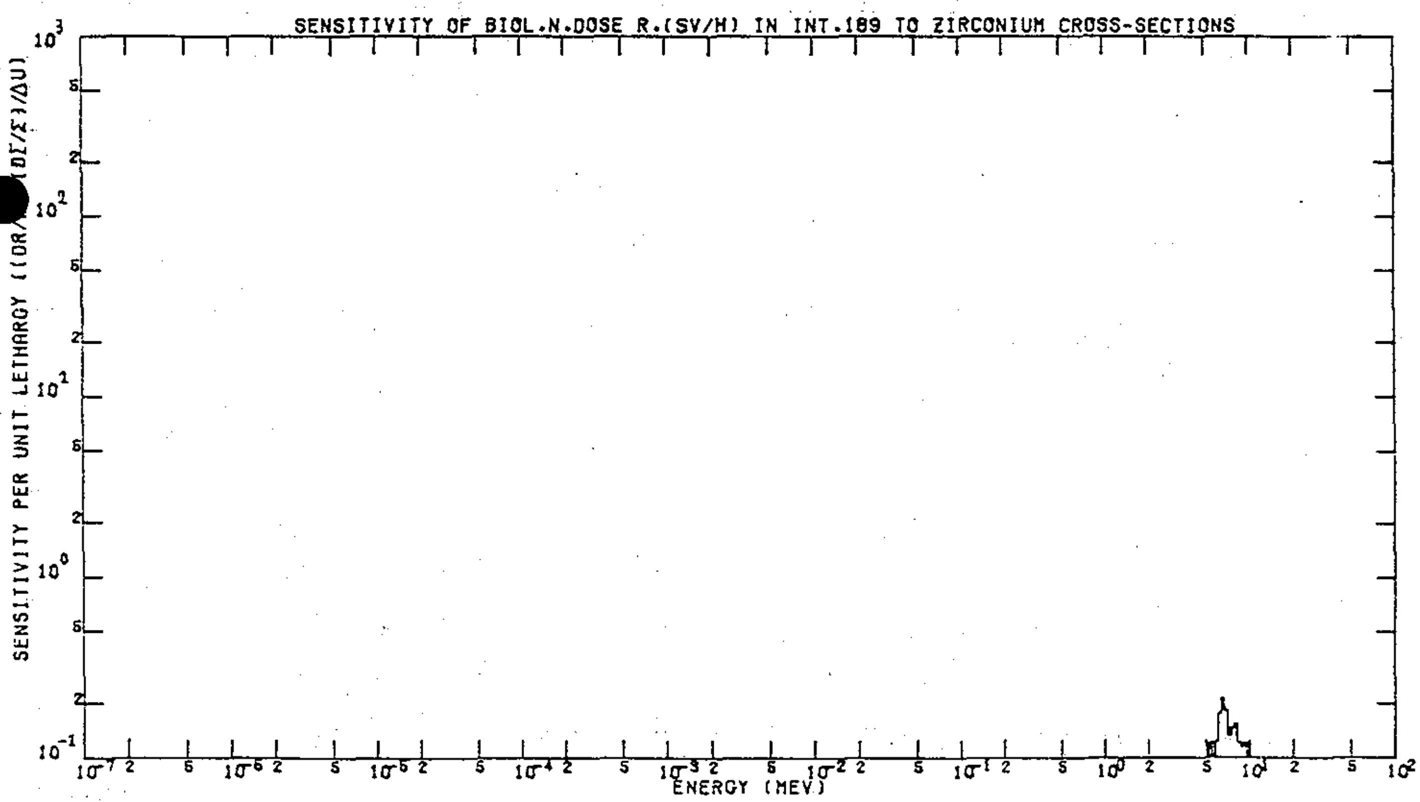

SENSITIVITY OF BIOL.N.DOSE R.(SV/H) IN INT.189 TO ZIRCONIUM CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 31

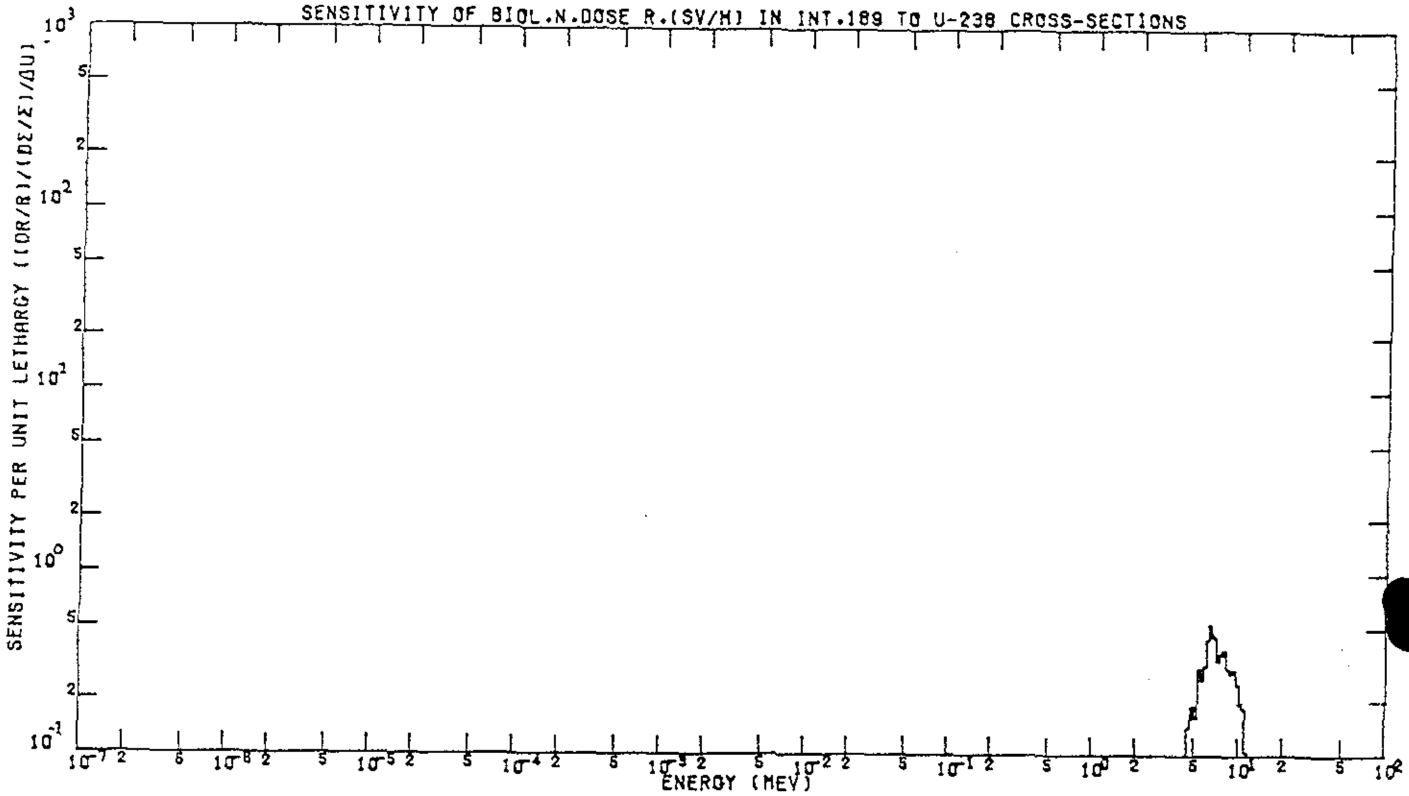

SENSITIVITY OF BIOL.N.DOSE R.(SV/H) IN INT.189 TO U-238 CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 32

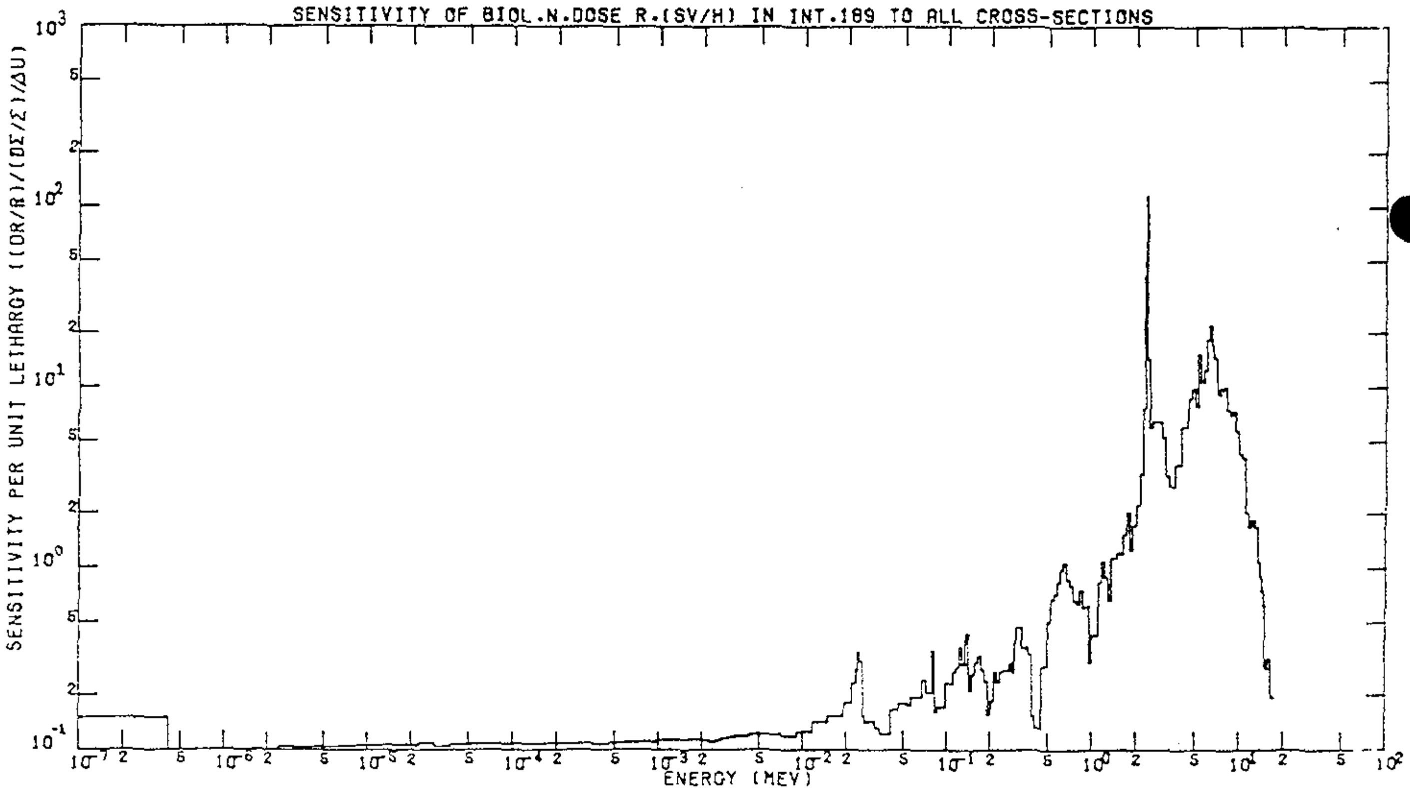

SENSITIVITY OF BIOL.N.DOSE R.(SV/H) IN INT.189 TO ALL CROSS-SECTIONS
SENSITIVITY PER UNIT LETHARGY ((DR/R)/(DΣ/Σ)/ΔU)
ENERGY (MEV)

Fig. 33

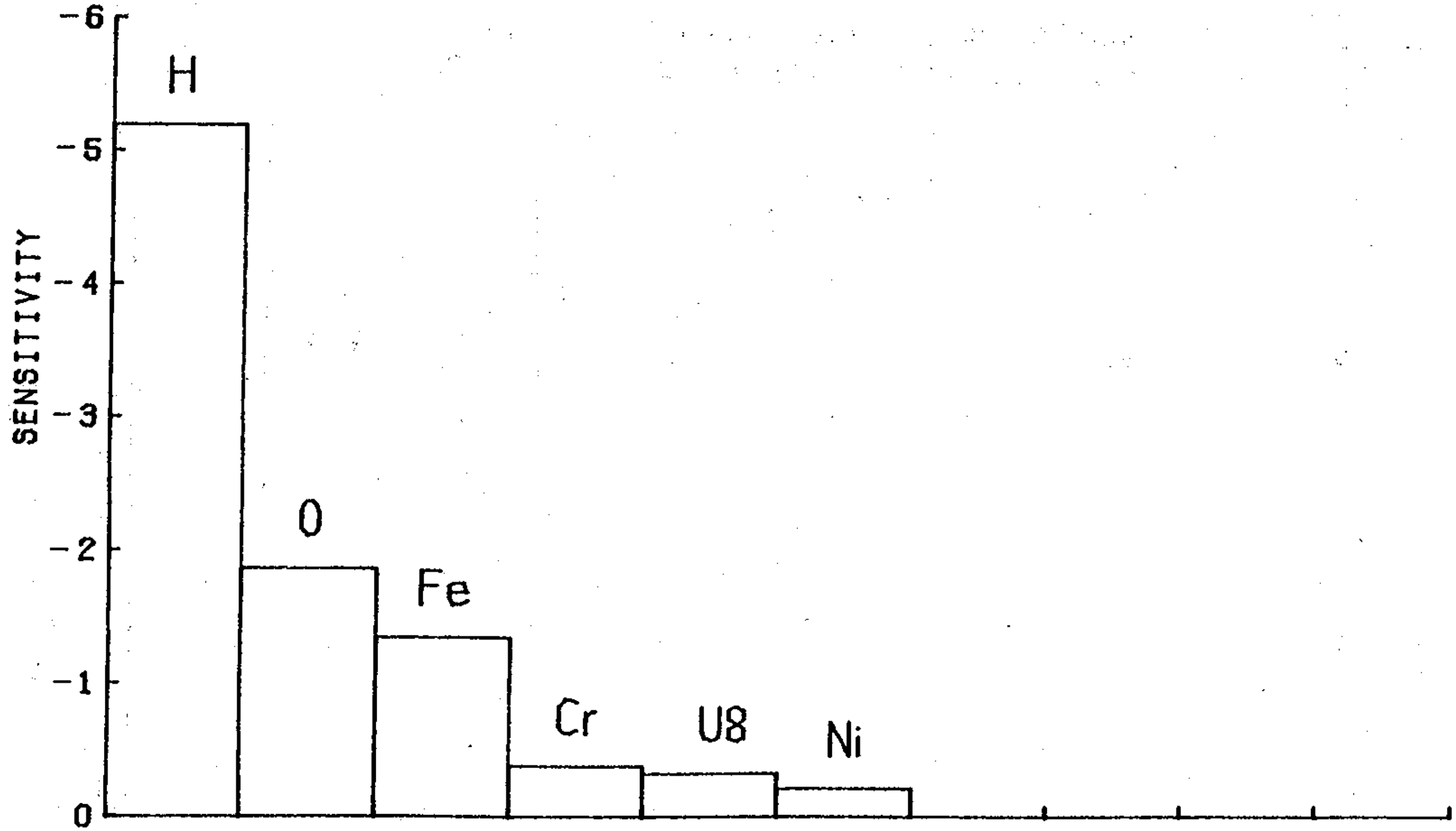


Fig. 34 CROSS SECTION SENSITIVITY OF DAMAGE RATE IN PRESSURE VESSEL SUMMED OVER ALL ENERGY GROUPS AND ZONES

91090034

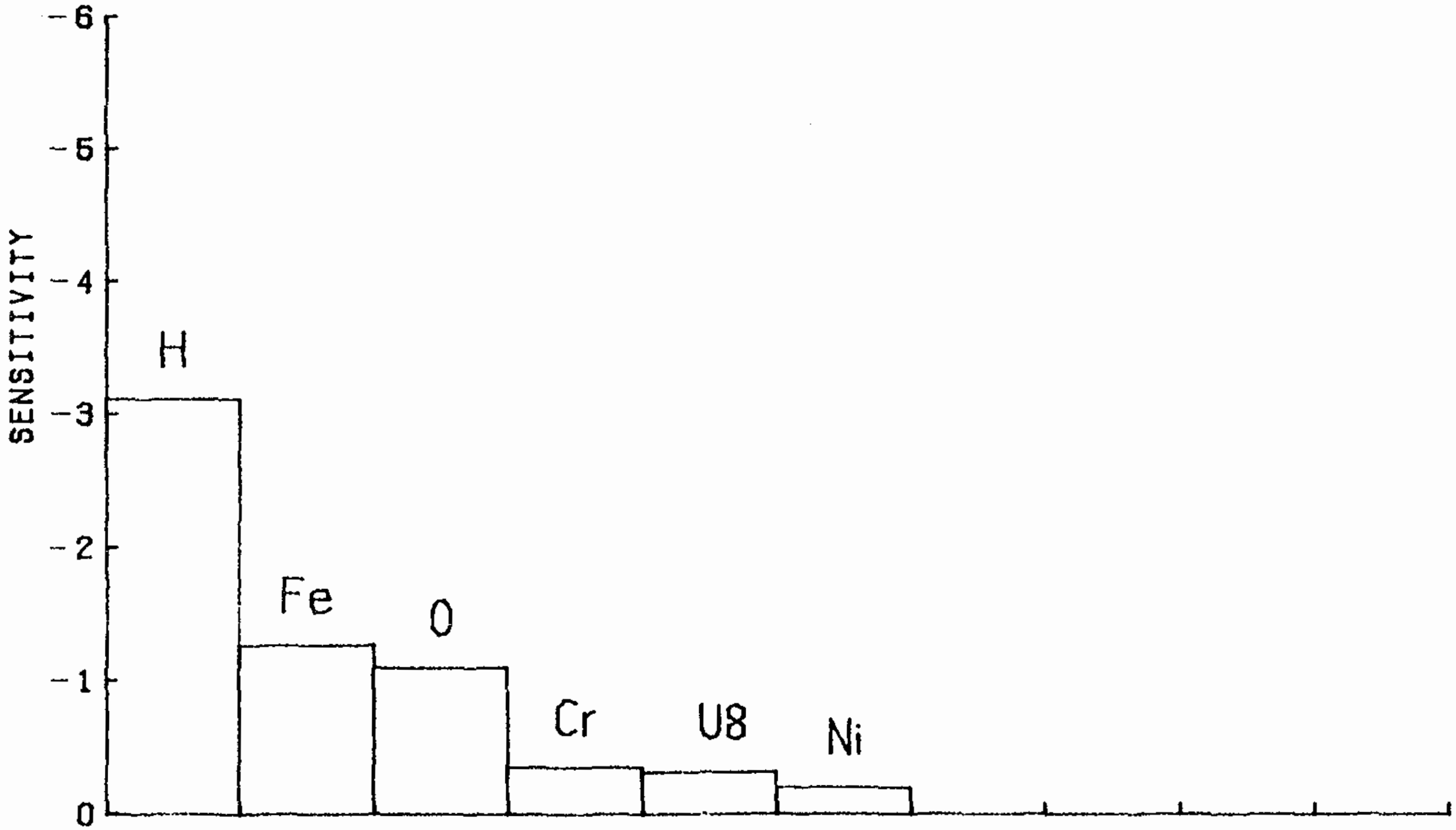


Fig. 35 CROSS SECTION SENSITIVITY OF ACTIVATION RATE IN CORE BARREL SUMMED OVER ALL ENERGY GROUPS AND ZONES

91090035

91090036

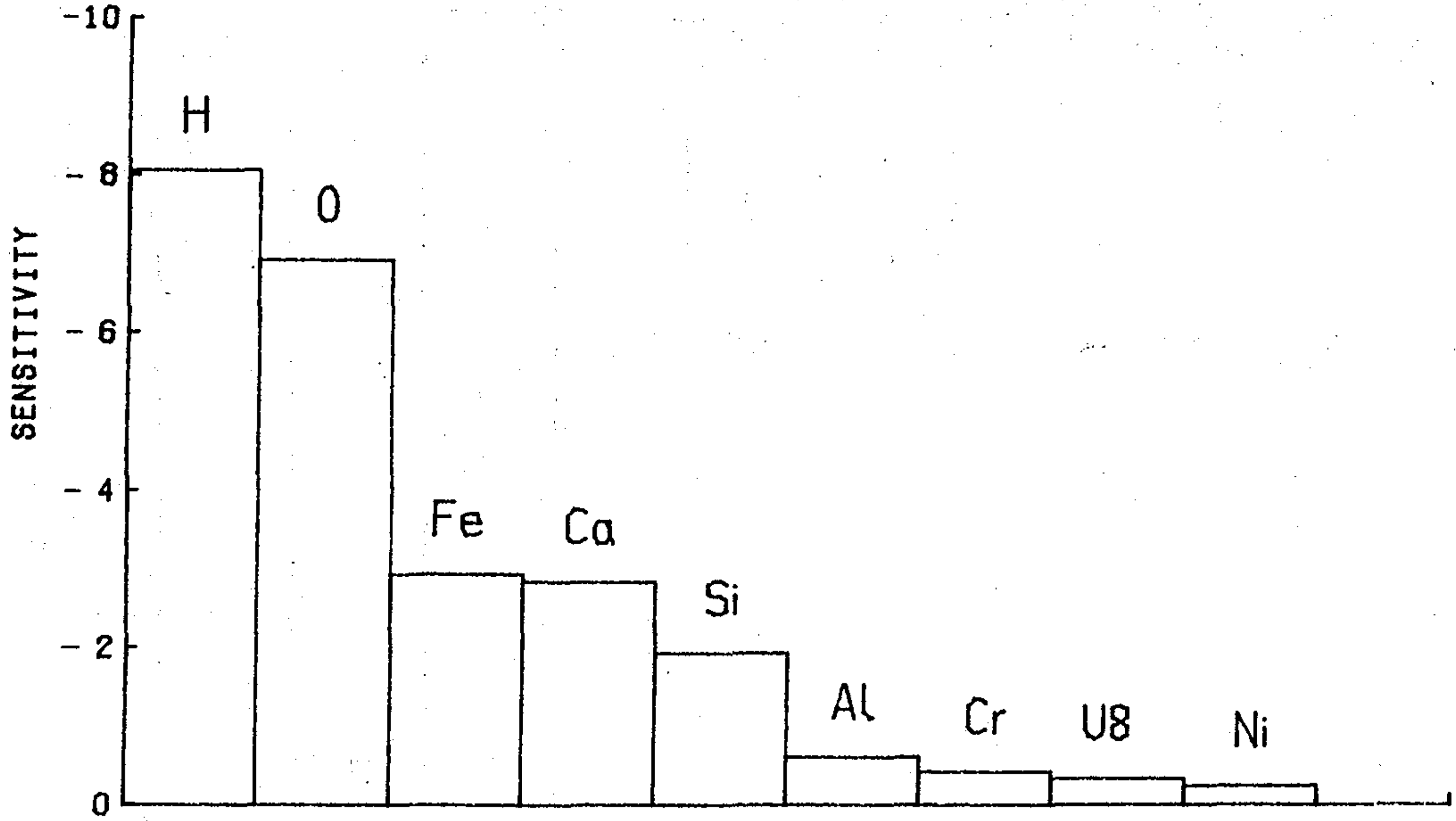


Fig. 36 CROSS SECTION SENSITIVITY OF NEUTRON DOSE RATE AT OUTER SIDE OF CONCRETE SUMMED OVER ALL ENERGY GROUPS AND ZONES

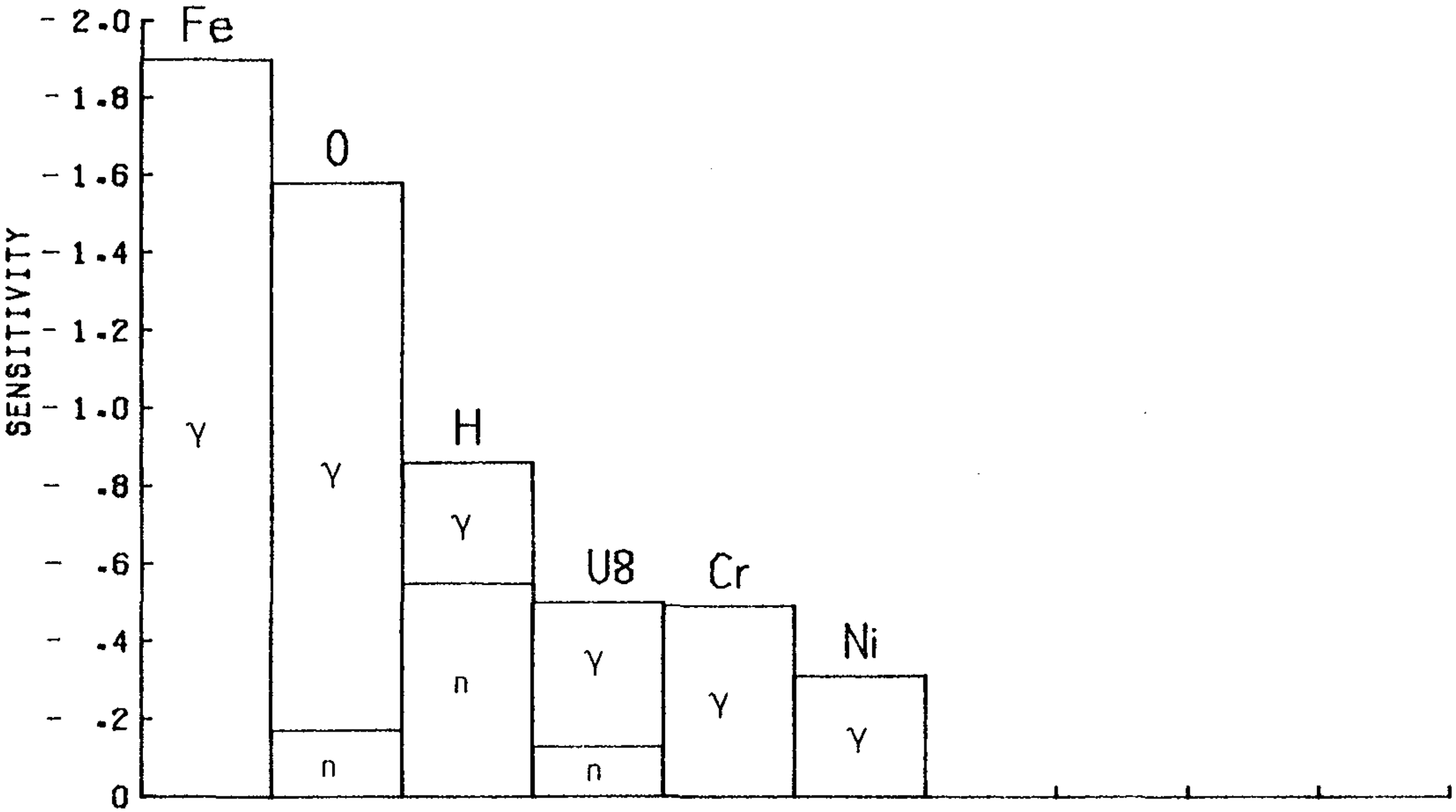


Fig. 37 CROSS SECTION SENSITIVITY OF GAMMA HEATING IN THE AUST. CLADDING SUMMED OVER ALL ENERGY GROUPS AND ZONES

91090037

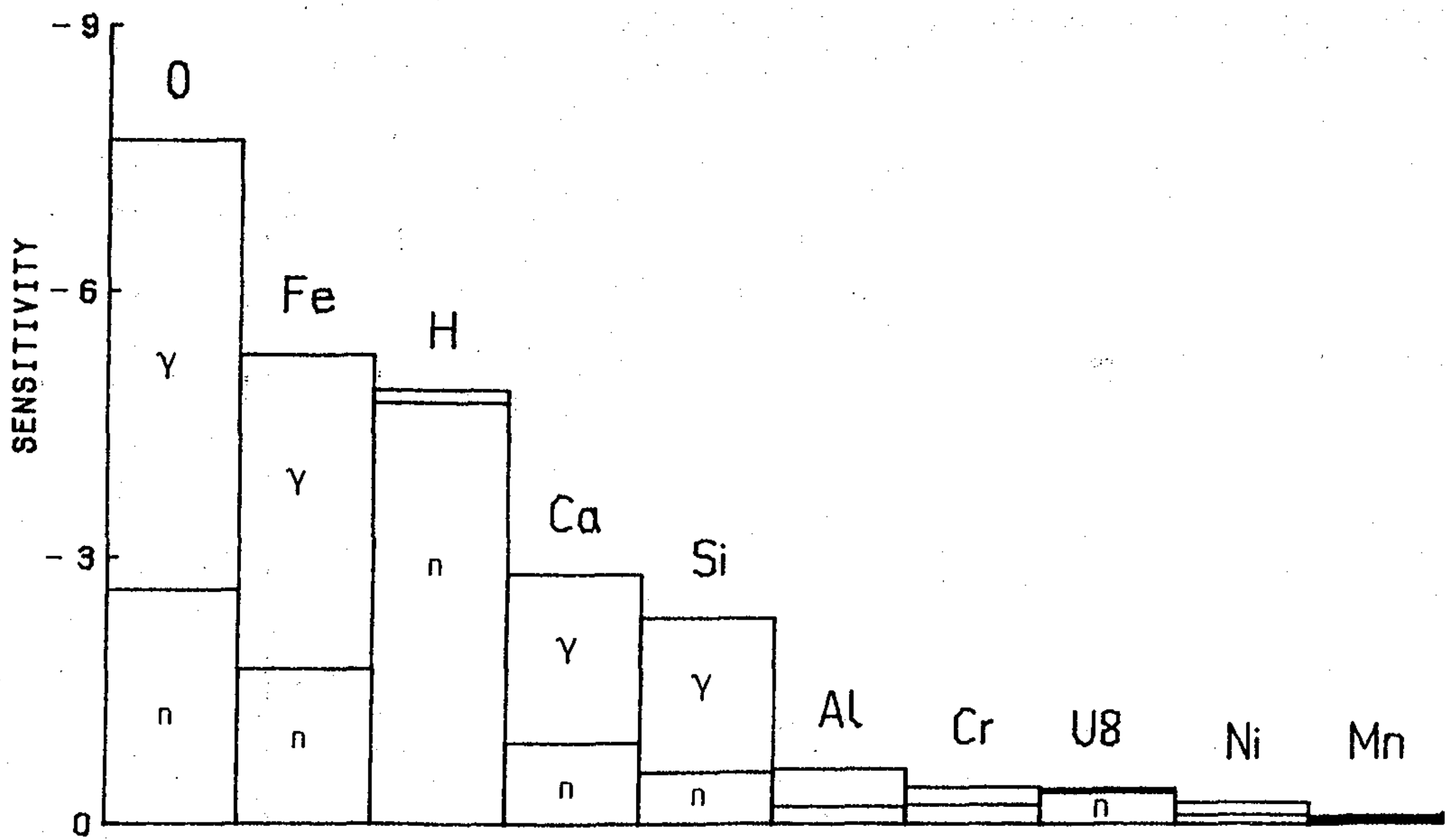


Fig. 38 CROSS SECTION SENSITIVITY OF GAMMA DOSE RATE AT OUTER SIDE OF CONCRETE SUMMED OVER ALL ENERGY GROUPS AND ZONES

91090038

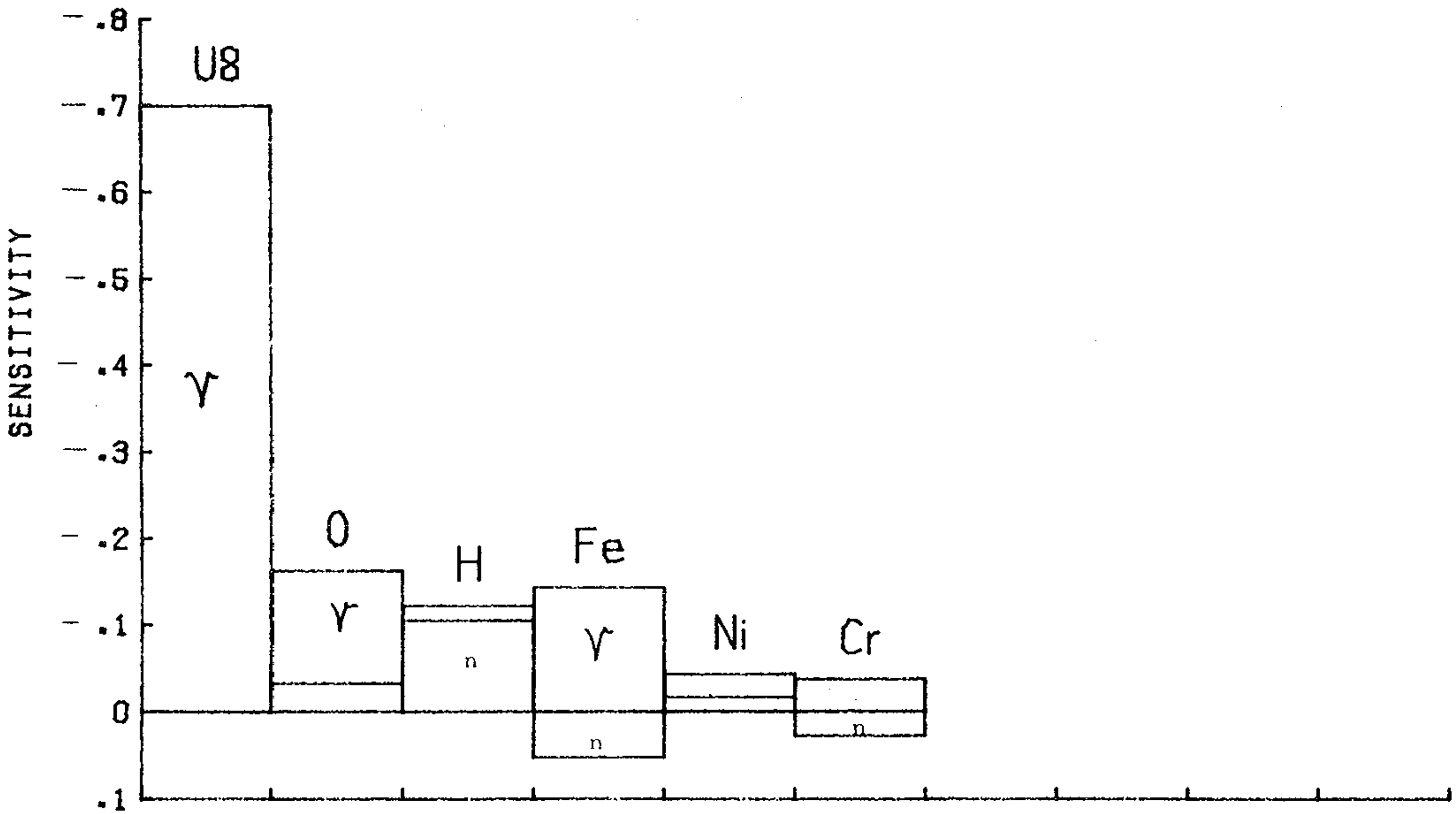


Fig. 39 CROSS SECTION SENSITIVITY OF GAMMA HEATING IN THE CORE BAFFLE SUMMED OVER ALL ENERGY GROUPS AND ZONES

91090039

Annex 1

PWR Benchmark for Studying the Effect of Nuclear Data Uncertainties in Reactor Shield Design

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J. Koban

KWU, Erlangen, FRG

91090040

1 Introduction

The PWR shield benchmark /1/ presented in 1976 has been revised and extended to demonstrate the present status of nuclear data in use for practical shield design. The nuclear data in use have been improved steadily, but for some shielding quantities the target accuracies have been also increased. This second intercomparison of a PWR shield calculation within the NEACRP will show, where we meet the advanced requirements of reactor safety and radiation protection and where further improvements are needed urgently.

2 PWR System Description

The radial shield of a PWR (1300 MW_{el}) has been chosen as design benchmark. 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 ^{o}C inlet temperature and 326 ^{o}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 is normally designed in two separate layers with a total thickness of 1,95 m.

/1/ G. Hehn, J. Koban
Reactor Shield Benchmark No. 2
ESIS Special Issue No. 4, 1976

3. Model Description, Material Composition, and Source Data

The radial dimensions of a one-dimensional cylindrical configuration are given in table 1 and the material compositions in table 2. The composition given for normal concrete corresponds to a mass density $\varrho = 2.27\ g/cm^3$ with 3 % water content in dry concrete according to the standard DIN 25413. In the transport calculations the air interval 189 can be considered approximately as belonging to the concrete zone using a density reduction of 10^{-20} for this interval in ANISN. 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 with the assumption of 194 MeV/fission. For the EURLIB multigroup structure with 100 neutron groups the neutron fission spectrum are given in table 5 containing the proposal in ENDF/B-V as well as an evaluation of NBS/Washington, for which error estimation is supplied /2/.

The neutron yield per fission reaction is then

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

with $\nu = 2.419$ for thermal fission of U-35

The total gamma yield (prompt and delayed) per fission reaction as well as the better known prompt component are given in table 6 for the group structure of 20 gamma groups in EURLIB. The spectral shape was taken from the ORIGEN-2 library /3/ normalized to the newest data of Sher /4/ with integral error information.

/4/ R. Sher
Fission energy release for 16 fissioning nuclides, BNL-Workshop on Evaluation Methods, Sept. 22-25, 1980

4. Aims of the calculations

4.1 Determination of integral target quantities of primary interest in shielding. For this a forward ANISN calculation is needed in routine design S8/P3-approximation with an increased point convergence of 10^{-5} and left boundary condition IBL = 1 and right boundary IBR = 0. For intercomparison the following quantities should be provided:

4.1.1 Activation rate ($cm^{-3}s^{-1}$) of ^{54}Fe (n,p)^{54}Mn in the barrel material near the surveillance capsules (interval 72) and in the pressel vessel cladding (interval 105) for comparison with measurements in the future.

4.1.2 Production rate of neutron damage in the pressure vessel as displacement rate per atom (dpa/s) at interval 107.

4.1.3 Gamma heating rate (W/cm^3) in core baffle (interval 18), core barrel (interval 60), pressure vessel (interval 105) and concrete shield (interval 134).

4.1.4 Biological neutron and gamma dose rate (Sv/h) at outer side of the concrete shield (interval 189).

The fission spectrum of NBS should be taken primarily, because it allows an error consideration of all target quantities. The ENDF/B-5 fission spectrum is within the error given and can be taken optionally for comparison in a second forward calculation but not for sensitivity and error determinations. The effect of a higher calculational effort like S_{16}/P_5 - approximation will be supplied for comparison and mustn't be studied.

/2/ J. Grundl, C. Eisenhauer, E. Mc Garry
Benchmark Neutron Fields for Pressure Vessel Surveillance Dosimetry, NUREG/CR-0551

/3/ A. G. Croff, R. L. Haese, and N. B. Gove, Updated Decay and Photon Libraries for the ORIGEN Code, ORNL/TM-6055 (February 1979).

4.2 Calculation of cross section sensitivities according to the following lists of priority e.g. for neutron target quantities:

a) damage rate at interval 107

b) activation rate at interval 72

c) neutron dose rate at interval 189

and for gamma target quantities:

a) gamma dose rate at interval 189

b) heating rate at interval 18

c) heating rate at interval 1o5

4.2.1 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.

4.2.2 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.

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

4.3 Calculation of the relative standard deviation for neutron target quantities according to the priority list given in 4.2.

4.3.1 Determination of the error contribution from source data uncertainties, e.g. resulting from the NBS error estimate of the neutron spectrum for thermal fission in U-235 as given in table 8.

4.3.2 Determination of the error contributions from neutron cross section uncertainties with available error covariance matrices of the nuclides H and O in EURLIB structure and Fe and U-238 in 26 group COVERX structure.

4.3.3 Estimation of the error contributions from the nonelastic neutron cross section of the nuclides Cr, Ni, and Zr with the help of the nonelastic covariance matrix given for Fe.

4.3.4 Inclusion of the error from detector cross section uncertainties e.g. for the elastic and nonelastic part of the iron displacement cross section as well as for the activation data from ENDF/B-5-Dosimetry File.

4.4 Estimation of the relative standard deviation for gamma target quantities according to the priority list in 4.2.

4.4.1 Determination of the error contribution of the neutron field according to 4.3.1, 4.3.2, and 4.3.3.

4.4.2 Free estimation of the error contribution from the gamma fission yield and the gamma production data stating the assumptions made.

4.4.3 Free estimation of the error contribution from gamma cross sections stating the assumptions made.

5 Presentation of the final results

For easy intercomparison the final integral target values should be given together with their total data uncertainties in tabular form as shown in table 9. Then the partial data uncertainties (in relative standard deviation) can be reported according to table 10 and 11. Finally in similar tables the integral sensitivities according to 4.2.1 and 4.2.2 can be stated and for the most important nuclides energy profiles according to 4.2.3 can be plotted. All assumptions made in the error analysis should be mentioned.

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	174.8	2.3
1. Water layer	210.5	35.7
Core barrel	218.5	8.0
2. Water layer	250.0	31.5
Austenitic cladding	250.6	0.6
Pressure vessel	275.6	25.0
Concrete shield	470.6	195.0
Air	471.0	0.4

Zone / Nuclide	Reactor core	Core baffle Core barrel Aust. cladding	Press. vessel	1. Water layer	2. Water layer	Concrete shield
H	2.77 E-2	-	-	4.92 E-2	4.98 E-2	4.41 E-3
O	2.68 E-2	-	-	2.46 E-2	2.49 E-2	4.78 E-2
B-10	2.73 E-6	-	-	4.86 E-6	4.90 E-6	-
Al	-	-	-	-	-	2.46 E-3
Si	-	-	-	-	-	9.44 E-3
Ca	-	-	-	-	-	6.61 E-3
Cr	2.00 E-4	1.69 E-2	3.00 E-4	-	-	-
Mn	1.27 E-5	1.12 E-3	7.40 E-4	-	-	-
Fe	5.23 E-4	5.88 E-2	8.30 E-2	-	-	-
Ni	2.36 E-4	8.54 E-3	6.44 E-4	-	-	-
Zr	4.29 E-3	-	-	-	-	-
U-235	2.10 E-4	-	-	-	-	-
U-238	6.28 E-3	-	-	-	-	-

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

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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 [W/cm^3]
1	103.5	105.8	89.54
2	108.1	110.4	89.26
3	112.7	115.0	89.35
4	117.3	119.6	89.35
5	121.9	124.2	89.35
6	126.5	128.8	78.63
7	131.1	133.4	75.95
8	135.7	138.0	74.20
9	140.3	142.6	73.64
10	144.9	147.2	74.66
11	149.5	151.8	97.02
12	154.1	156.4	93.32
13	158.7	161.0	84.92
14	163.3	165.6	71.15
15	167.9	169.05	55.44
16	170.2	170.85	45.28
17	171.5	172.0	37.42
	172.5		

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
	14	4.6	167.9
	1	2.3	170.2
	1	1.3	171.5
17	1	1.0	172.5
Core baffle			
	1	0.4	172.9
	3	0.5	174.4
5	1	0.4	174.8
1. Water layer			
	4	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
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	241.4
	5	1.0	246.4
	2	0.8	248.0
	2	0.6	249.2
32	2	0.4	250.0

cont.

Table 4 (continuing)

Zone and total number of intervals	Number of intervals	Thickness of interval [cm]	Radius [cm]
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
	1	0.8	276.8
	1	1.2	278.0
	1	2.0	280.0
	2	3.0	286.0
	44	4.0	462.0
	1	3.0	465.0
	2	2.0	469.0
	1	1.0	470.0
55	1	0.6	470.6
Air			
1	1	0.4	471.0

IM = 189

Table 5 Neutron fission spectrum of U-235 for 100 neutron groups (EURLIB) as given by NBS /2/ and from ENDF/B-5

g	χ_g^{E-5}	χ_g^{NBS}	g	χ_g^{E-5}	χ_g^{NBS}
1	5.34342E-05	5.49591E-05	51	3.24166E-03	3.48304E-03
2	1.42717E-04	1.41883E-04	52	2.81275E-03	2.99242E-03
3	3.39810E-04	3.30155E-04	53	2.44508E-03	2.58043E-03
4	7.28119E-04	6.97937E-04	54	4.77515E-03	4.96971E-03
5	1.42673E-03	1.35998E-03	55	3.32900E-03	3.40807E-03
6	2.56189E-03	2.44474E-03	56	2.31397E-03	2.33821E-03
7	4.27344E-03	4.10523E-03	57	1.60211E-03	1.60209E-03
8	2.99330E-03	2.89751E-03	58	1.10992E-03	1.10101E-03
9	3.66564E-03	3.57130E-03	59	6.34594E-04	6.25743E-04
10	4.45837E-03	4.37658E-03	60	1.31358E-04	1.29161E-04
11	5.32057E-03	5.26593E-03	61	1.22320E-04	1.20168E-04
12	1.35501E-02	1.33758E-02	62	1.66971E-04	1.63831E-04
13	1.79336E-02	1.74133E-02	63	2.39760E-04	2.34837E-04
14	1.07153E-02	1.03099E-02	64	3.64730E-04	3.56279E-04
15	1.19145E-02	1.14097E-02	65	2.51100E-04	2.44523E-04
16	2.75664E-02	2.62599E-02	66	1.72932E-04	1.68049E-04
17	3.22774E-02	3.06275E-02	67	1.19038E-04	1.15426E-04
18	3.66940E-02	3.47808E-02	68	8.19268E-05	7.93249E-05
19	4.04579E-02	3.33956E-02	69	5.63292E-05	5.44777E-05
20	4.36102E-02	4.15167E-02	70	3.87578E-05	3.74505E-05
21	4.58447E-02	4.38426E-02	71	2.66497E-05	2.57329E-05
22	2.34987E-02	2.25647E-02	72	1.83312E-05	1.76910E-05
23	2.37079E-02	2.28333E-02	73	1.25995E-05	1.21543E-05
24	4.77167E-02	4.61816E-02	74	8.65866E-06	8.34999E-06
25	4.75288E-02	4.63250E-02	75	5.95917E-06	5.74528E-06
26	4.65968E-02	4.57517E-02	76	4.09024E-06	3.94263E-06
27	4.49339E-02	4.43581E-02	77	2.81377E-06	2.71181E-06
28	4.31742E-02	4.29352E-02	78	1.93361E-06	1.86332E-06
29	4.07015E-02	4.06027E-02	79	1.32375E-06	1.28032E-06
30	3.81172E-02	3.81562E-02	80	9.13897E-07	8.80526E-07
31	3.54411E-02	3.56090E-02	81	6.27744E-07	6.04785E-07
32	3.26367E-02	3.29667E-02	82	4.31487E-07	4.15689E-07
33	2.99250E-02	3.03057E-02	83	2.96645E-07	2.85774E-07
34	2.72282E-02	2.77968E-02	84	2.03929E-07	1.96450E-07
35	2.46409E-02	2.55454E-02	85	1.40107E-07	1.34966E-07
36	2.21879E-02	2.33707E-02	86	9.62951E-08	9.27601E-08
37	1.98958E-02	2.12516E-02	87	6.62039E-08	6.37727E-08
38	1.77573E-02	1.92240E-02	88	4.54746E-08	4.38042E-08
39	1.58019E-02	1.73074E-02	89	3.12541E-08	3.01059E-08
40	1.39995E-02	1.55057E-02	90	2.15066E-08	2.07163E-08
41	1.23886E-02	1.38503E-02	91	1.47548E-08	1.42222E-08
42	1.09234E-02	1.23221E-02	92	1.71301E-08	1.65005E-08
43	9.61650E-03	1.09344E-02	93	4.79515E-09	4.61889E-09
44	8.43834E-03	9.66680E-03	94	5.58006E-09	5.35568E-09
45	7.38522E-03	8.51203E-03	95	2.62587E-09	2.52934E-09
46	6.47165E-03	7.41552E-03	96	1.24091E-09	1.19530E-09
47	5.64743E-03	6.37253E-03	97	5.85144E-10	5.64595E-10
48	4.92327E-03	5.47865E-03	98	3.07488E-10	2.96184E-10
49	4.28787E-03	4.71154E-03	99	1.00036E-10	9.66061E-11
50	3.73858E-03	4.06279E-03	100	1.17111E-10	1.12883E-10

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Table 6 Total and prompt fission gamma yield - spectral shape from ORIGEN-2 /3/ with normalization of total yield corresponding to 13.3 MeV/fission and of prompt gamma yield to 6.97 ± o.5o MeV/fission /4/.

group g	energy MeV	total yield	prompt yield
1	1.40E+07	0.	0.
2	1.20E+07	1.00635E-04	5.25132E-05
3	1.00E+07	1.74763E-03	9.11951E-04
4	8.00E+06	8.13669E-03	4.24589E-03
5	6.50E+06	4.17365E-02	2.17790E-02
6	5.00E+06	1.02354E-01	5.34104E-02
7	4.00E+06	3.04429E-01	1.58857E-01
8	3.00E+06	3.32314E-01	1.73408E-01
9	2.50E+06	5.73138E-01	2.99075E-01
10	2.00E+06	6.11931E-01	3.19318E-01
11	1.66E+06	8.86953E-01	4.62830E-01
12	1.33E+06	1.55737E+00	8.12668E-01
13	1.00E+06	1.49589E+00	7.80587E-01
14	8.00E+05	2.13483E+00	1.11400E+00
15	6.00E+05	2.39986E+00	1.25230E+00
16	4.00E+05	1.19951E+00	6.25929E-01
17	3.00E+05	1.19951E+00	6.25929E-01
18	2.00E+05	1.14553E+00	5.97762E-01
19	1.00E+05	6.49288E-01	6.49288E-01
20	5.00E+04	3.89573E-01	3.89573E-01
	2.00E+04		

TABLE 7 – Energy limits of energy groups: EURLIB structure

100 Groups		100 Groups	
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 – 1.3534E 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	100	1.0000E-05 – 4.1399E-01

Table 8: Error estimates for neutron fission spectrum of U-235 /2/

Group	1 σ (%)
1 - 6	5.3
7 - 17	4.8
18 - 22	2.0
23 - 27	3.1
28 - 33	3.0
34 - 45	4.1
46 - 74	16.0

Table 9 Integral target quantities with errors

	Interval No. 18	60	72	105	107	134	189
1. activation rate [cm^{-3} s^{-1}]							
1.1 NBS - fission spectrum			X	X			
1.2 rel stand. deviation [%]			X	(X)			
1.3 ENDF/B-5 fission spectrum			(X)	(X)			
1.4 other values eventually							
2. damage rate [dpa/s]							
2.1					X		
2.2					X		
2.3					(X)		
3. gamma heating rate [W/cm^3]							
3.1	X	X		X		X	
3.2	X	(X)		X		(X)	
3.3	(X)	(X)		(X)		(X)	
4. neutron dose rate [Sv/h]							
4.1							X
4.2							X
4.3							(X)
5. gamma dose rate [Sv/h]							
5.1							X
5.2							X
5.3							(X)

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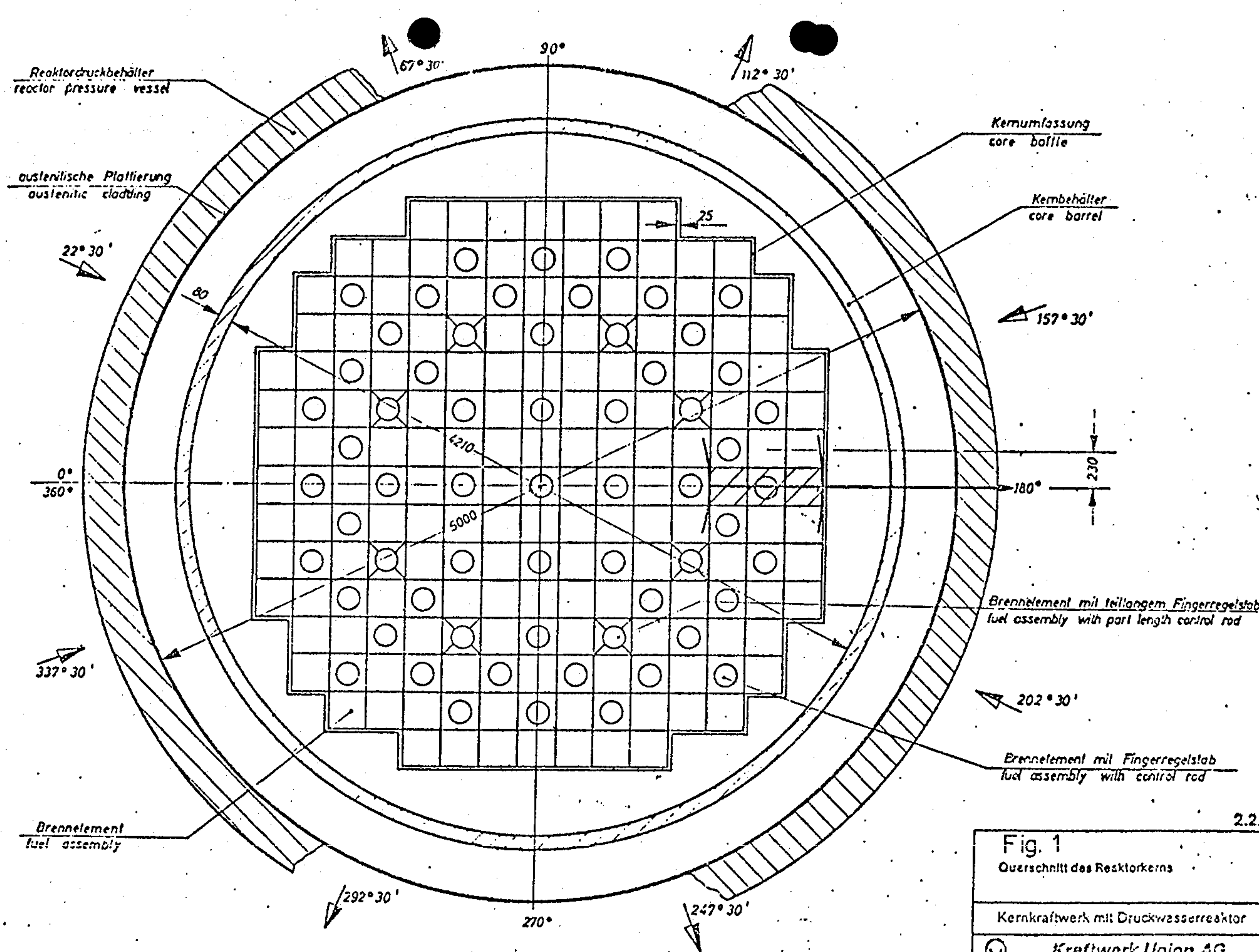
Table 10 Relative standard deviation resulting from source and detector data

	% error contribution	
	fission source neutron/gamma	detector data
1. activation		
1.1 at interval 72		
1.2 at interval 105		
2. damage		
3. gamma heating		———
3.1 at interval 18		
3.2 at interval 105		———
4. neutron dose		———
5. gamma dose		———

<u>Table 11</u> Relative standard deviation resulting from important partial cross sections of the most important nuclides

nuclide	reaction type	relative standard deviation % activation at int. 72	damage	gamma heating at int.18	gamma heating at int.105	neutron dose	gamma dose
Fe	σ_n						
	σ_X						
	$\sigma_{\gamma\text{-prod.}}$						
	$\sigma_{\gamma,\gamma}$						
	tot.err.						
H	σ_n						
	$\sigma_{\gamma\text{-prod.}}$						
	tot.err.						
O	σ_n						
	$\sigma_{n'}$						
	$\sigma_{n,p}$						
	$\sigma_{\gamma\text{-prod.}}$						
	$\sigma_{\gamma,\gamma}$						
	tot.err.						
U-238	⋮						
Zr							
Cr							
Ni							
Ca							
⋮							



2.2/1

Fig. 1
Querschnitt des Reaktorkerns

Kernkraftwerk mit Druckwasserreaktor

Kraftwerk Union AG

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