SEPTEMBER 1990 ECN-RX--90-062

NL91C0223

SENSITIVITY AND UNCERTAINTY ANALYSIS OF NET/ITER SHIELDING BLANKETS

A. HOGENBIRK
H. GRUPPELAAR
K.A. VERSCHUUR

THIS REPORT IS A CONTRIBUTION TO THE 16TH SYMPOSIUM ON FUSION TECHNOLOGY, LONDON, 3-7 SEPTEMBER 1990

The Netherlands Energy Research Foundation ECN is the leading institute in the Netherlands for energy research. ECN carries out pure and applied research in the fields of nuclear energy, fossil fuels, renewable energy sources, environmental aspects of energy supply, computer science and the development and application of new materials. Energy studies are also a part of the research programme.

ECN employs more than 900 staff. Contracts are obtained from the government and from national and foreign organizations and industries.

ECN's research results are published in a number of report series, each series serving a different public from contractors to the international scientific world.

This RX-series is used for publishing pre-prints: the preliminary versions of articles that will appear in a journal, or conference or symposium proceedings. Please *do not* refer to this report but use the reference provided on the title page: 'To appear in ...' or 'Submitted for publication to ...'.

The Netherlands Energy Research Foundation ECN
Scientific and Technical Information Department
P.O. Box 1
NL-1755 ZG Petten
The Netherlands
Telephone: +31 2246 4323
Fax : +31 2246 3483

This report is available on remittance of Dfl. 20 to:
ECN, TWI Department,
Petten, The Netherlands.
Giro account (postal account) No. 3977703.
Please quote the report number.

SENSITIVITY AND UNCERTAINTY ANALYSIS OF NET/ITER SHIELDING BLANKETS

A. HOGENBIRK, H. GRUPPELAAR and K. A. VERSCHUUR

Netherlands Energy Research Foundation ECN, P.O. Box 1, 1755 ZG Petten, The Netherlands

In this paper the results are presented of sensitivity and uncertainty calculations based upon the European Fusion File (EFF-1). The effect of uncertainties in Fe, Cr and Ni cross sections on the nuclear heating in the coils of a NET/ITER shielding blanket has been studied. The analysis has been performed for the total cross section as well as partial cross sections. The correct expression for the sensitivity profile was used, including the gain term. The resulting uncertainty in the nuclear heating lies between 10 and 20%.

1 INTRODUCTION

The blanket of a fusion reactor has in principle several tasks, but in order to simplify the design, one of the blanket options for the NET/ITER reactor only has a shielding function. The design of this shielding blanket plays an important role in the development of the NET/ITER reactor. Especially the design of the inboard blanket is crucial.

The subject of the accuracy with which design parameters are calculated is treated in cross section sensitivity and uncertainty analyses (see e.g. ref.[1]). For an extensive introduction on such calculations the reader is referred to references[2–5].

The subject of our study consists of a neutron source, which emits neutrons that are shielded from the region of interest (the detector region) by a suitable radiation shield. The aim is to calculate the uncertainty of a certain integral parameter R (defined as $R = \langle \Sigma_R, \phi \rangle$, in which Σ_R is the response function that relates the flux ϕ to the response parameter R, e.g. a kerma factor) due to uncertainties in the cross sections that are used in the transport calculation.

In this study the 1-D cross section sensitivity and uncertainty code SENSIT[5] was used, which was substantially modified in order to calculate sensitivity profiles $P^g_{\Sigma_i}$ for *arbitrary* cross sections Σ_i according to

$$P^g_{\Sigma_i} = \frac{1}{R}\{-\Sigma^g_i \chi^g + \sum_{l=0}^{LMAX} \sum_{g'=g}^{GMAX} \Sigma^{g\rightarrow g'}_{i,l} \psi^{gg'}_l\}, \quad (1)$$

which is derived in ref.[5]. Equation (1) is used in all our calculations. The first term in eq. (1) is generally referred to as the *loss* term, the second term as the *gain* term (the *direct* term[6] is omitted from our analysis). Eq. (1) holds for neutrons as well as for photons.

A first-order approximation of the relative variance $(\frac{\Delta R}{R})^2$ is given by

$$(\frac{\Delta R}{R})^2 = \sum_i \sum_j \sum_g \sum_{g'} P^g_{\Sigma_i} P^{g'}_{\Sigma_j} \frac{Cov(\Sigma^g_i, \Sigma^{g'}_j)}{\Sigma^g_i \Sigma^{g'}_j} \quad (2)$$

(see ref.[3]), in which $Cov(\Sigma^g_i, \Sigma^{g'}_j)$ is the covariance matrix element for cross sections Σ^g_i and $\Sigma^{g'}_j$.

Thus, if sensitivity profiles for the reactions of interest can be calculated and the the necessary covariance matrix elements are available, the variance of the parameter R can be calculated.

2 CODE SYSTEM

The code system used is described in ref.[1]. The cross sections were obtained from the 217-group

MATXS library MAT175[7], based upon the European Fusion File (EFF-1). The processing code TRAMIX[8,9] (based on TRANSX-CTR[10]) was used to generate libraries with transport cross sections for use in the codes ANISN[11] and SENSIT[5]. The SENSIT library contains scattering tables for in principle arbitrary cross sections, for which reason TRAMIX was substantially modified. In the ANISN calculations angular fluxes are produced for the forward and the adjoint cases. These fluxes, cross section tables and covariance data are read by the code SENSIT[5], in which the sensitivity and uncertainty analysis is carried out.

3 MODEL DESCRIPTION AND CALCULATIONS

A specific sensitivity and uncertainty analysis has been performed on a NET/ITER iron/water shielding blanket[12,13] (with identification number 541 [ref. [14]]), of which a one-dimensional model has been made (slab geometry). The aim of this study was to obtain the sensitivity of the total nuclear heating in the first interval of the coil case to the cross sections of Fe, Cr and Ni and the uncertainty in the heating due to uncertainties in these cross sections. The main reason for studying only the effect of these cross sections is the large amount of stainless steel present in the shield.

Coupled $n\gamma$ transport calculations were performed in an S_8P_3 approximation. The region of interest was subdivided into 30 zones and 159 intervals; details can be found in ref.[1].

For the forward calculation a homogeneous source was used in group 1 of the GAM-II structure. The neutron source strength was taken to be $5.278 \cdot 10^{11}$ $cm^{-2}\ s^{-1}$, which corresponds to a neutron wall loading of 1.0 MW m^{-2} on the first wall surface area.

The source in the adjoint calculation is in this case equal to the kerma factor of the coil mix. The kerma factors on the MATXS library only contain the *prompt* nuclear heating and have the disadvantage that, because of inconsistencies in the evaluation, they are negative in some cases [see ref.[15]]. We therefore used in our calculations the kerma factors of MACKLIB-IV[16] instead.

For computational convenience in our analysis only cross sections in three innermost stainless steel zones (containing about 60% of the stainless steel present in the shield) were perturbed, assuming that the perturbation of the complete shield could be extrapolated linearly. For Fe sensitivity profiles $P^g_{\Sigma_i}$ were produced[1] for the total cross section and for the cross sections of the following reactions: (n,n), (n,n'), $(n,2n)$, (n,γ), (n,p) and (n,α). For Cr and Ni the analysis was confined to the total, elastic and inelastic cross sections.

For Fe detailed results are presented in figs. 1 and 2, which will be discussed below. For Cr and Ni only the numerical results are given (see table 2). In all figures the sensitivity profiles have been normalized to the lethargy width Δu_g of the g^{th} group.

From fig. 1 it is observed that for the elastic cross section the gain and loss part of the sensitivity profile $P_{\Sigma_{elas}}$ cancel to a large extent. The cancellation is a result of the fact that neutrons which scatter elastically loose only little energy, causing their contribution to the response parameter to remain approximately unchanged.

In an inelastic scattering event a neutron looses at least an energy E_x, equal to the energy of the first excited state of the nucleus from which the scattering takes place. Thus the cancellation of gain and loss part of the sensitivity profile is less pronounced than it is for the elastic cross section (cf. figs. 1a and b). This causes the total sensitivity profile of the elastic and inelastic cross section to be comparable (see fig. 2), despite the difference in magnitude between the cross sections.

For the remaining cross sections no graphs are given; their contribution to the total sensitivity profile is relatively small (see ref.[1]).

The numerical results of the sensitivity study are

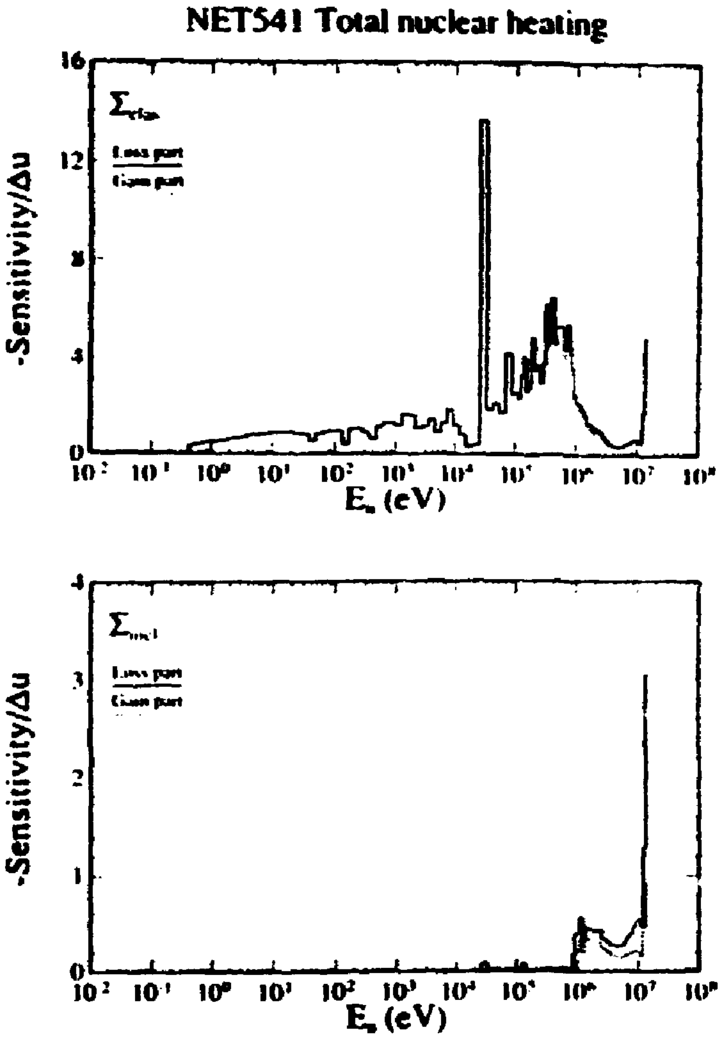

Figure 1: Sensitivity profiles $P^{g}_{\Sigma_{i}}/\Delta u_{g}$ for the gain and loss parts of eq. (1) for the elastic (a) and inelastic (b) cross section of Fe.

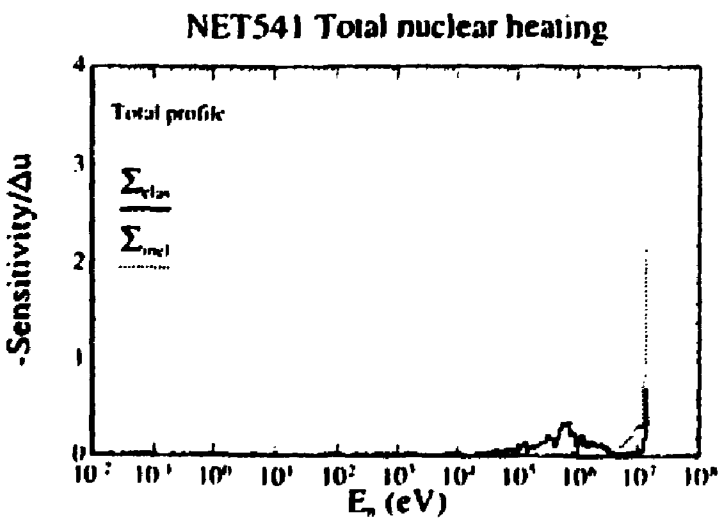

Figure 2: Sensitivity profile $P^{g}_{\Sigma_{i}}/\Delta u_{g}$ for the elastic and inelastic cross section of Fe.

summarized in table 1, in which the integrated sensitivities for all reactions are given. In the row indicated by "Sum" the sum of the individual contributions is given. This sum is about equal to the sensitivity for the total cross section, as it should be.

The final uncertainty in the response parameter due to the uncertainty in a given cross section is determined by folding the sensitivity profiles and relative covariances according to eq. (2), neglecting uncertainties in the matrix elements $\Sigma^{g\to g'}_{s,l}$. The code SENSIT was modified in such a way as to perform this folding correctly for arbitrary cross sections. Covariance data for the total, elastic, inelastic and (n,γ) cross sections were available to us[1] on a covariance library in the BOXER[17] format[18] and were extracted from this library using the processing code ANGELO[18].

The resulting uncertainties in the nuclear heating are given in table 2. Contributions from $(n,2n)$, (n,p) and (n,α) are not given because of lack of uncertainty information. It is observed, that for Fe and Ni the inelastic cross section contributes most strongly to the total uncertainty. The contribution of the (n,γ) cross section is relatively small. The various contributions, if combined by means of the error propagation law, should add up to the column Σ_{total}. As no cross-covariance data are available on the covariance library, this can not be checked, but there is reasonable agreement if cross-correlations are assumed to be small (compare columns Σ_{total}, Σ_{uncorr}).

It is noted, that the assumption that the total uncertainty is proportional to the amount of stainless steel in the shield is only valid if the uncertainty is constant over the neutron energy. However, for several cross sections (e.g. the elastic cross sections of Fe and Cr) the uncertainty strongly increases with increasing neutron energy. As the neutron spectrum in the remaining zones is harder than in the studied zones, the uncertainty will then be underestimated. In order to get an indication of the magnitude of this effect for the elastic cross section of Fe a complete calculation was performed. This resulted in a total

Table 1: Integrated sensitivities for several energy regions for the reactions studied (results shown for Fe only). Region I corresponds to 12.21 MeV < E_n < 14.92 MeV, region II to 1.0 MeV < E_n < 12.21 MeV and region III to E_n < 1.0 MeV. For each reaction also the total integrated sensitivity (integrated over all neutron energies) is given in the last column. In the last row the integrated sensitivity for the total cross section is given (this table contains a correction with respect to ref. [1]).

	region I	region II	region III	total
(n,n)	−0.088	−0.209	−0.630	−0.927
(n,n')	−0.298	−0.408	−0.024	−0.730
$(n,n'p)$	−0.014	−0.001	0	−0.015
$(n,n'\alpha)$	−0.001	0	0	−0.001
$(n,2n)$	−0.141	0	0	−0.141
(n,γ)	$-1.14 \cdot 10^{-4}$	−0.002	−0.011	−0.013
(n,p)	−0.069	−0.023	$-3.32 \cdot 10^{-6}$	−0.092
(n,d)	−0.010	$-1.9 \cdot 10^{-4}$	0	−0.010
(n,α)	−0.022	−0.005	0	−0.027
Sum	−0.643	−0.649	−0.665	−1.957
Σ_{total}	−0.631	−0.628	−0.665	−1.924

uncertainty of 3.54%, which is about 25% higher than the extrapolated value of 2.81%.

Assuming no correlations between the different cross sections, for the nuclear heating in the first interval in the coil a total uncertainty results of 8.6% (see table 1). To be conservative and to take into account uncertainties in (n,p), (n,α) and $(n,2n)$ reaction cross sections, we increase this number to 10%. In order to get an accurate estimate of the uncertainty a complete calculation (in which all zones are perturbed) should be performed. An indication of the error involved in extrapolating from a part of the shield to the total shield is given above; it results in a correction factor of 1.25. A realistic estimate of the total uncertainty is therefore 12.5%. This figure does not include uncertainties in the matrix elements of the scattering cross sections. It is of interest to note that there is no large difference between the values in columns Σ_{total} and Σ_{uncorr}, indicating that there are no large correlations between the various cross section types (assuming that the uncertainty information on the covariance file is correct!).

4 CONCLUSIONS

In this paper the results are summarized of a sensitivity and uncertainty analysis of a realistic NET/ITER shielding blanket. The analysis was confined to only one response parameter (the total nuclear heating in the first interval of the coil case) and to cross sections of Fe, Cr and Ni only, as most of the shield consists of stainless steel.

Due to uncertainties in the reactions mentioned above (neglecting the direct term) a total uncertainty of 12.5% in the nuclear heating in the first interval of the coil case results. This figure must be considered to be a lower limit, assuming no uncertainty in the matrix elements.

It must be stressed that because of the large importance of the gain term uncertainties in the energy distribution and angular distribution of the secondary neutrons may contribute substantially to the total uncertainty. These uncertainties and the uncertainty in the direct term, however, have not been taken into account in the present study. Work on this subject based upon the EFF-2 data file is currently being performed at ECN; the results will be published in a follow-up study. A very rough estimate of the effect of elastic scattering is obtained by assuming all P_n $(n > 0)$ terms in eq. (1) to be equal to zero. This leads to a 6.1% increase in the total uncertainty of the heating. From this observation we would guess that

Table 2: Calculated relative uncertainties ($\frac{\Delta R}{R}$) of the total nuclear heating due to the total, elastic, inelastic and (n,γ) cross section of Fe, Cr and Ni in three zones of the shield (left part of the table) (n.a.= not analysed). In the right part of the table the extrapolated values of $(\frac{\Delta R}{R})_{total}$ for the complete shield are given. In the column indicated by Σ_{uncorr} the summed uncertainty of the reactions (n,n), (n,n') and (n,γ) is given, assuming no correlations. The last row contains the uncertainties due to Fe, Cr and Ni, assuming no correlations between the material cross sections.

		Partial shield (3 zones)				Complete shield (extrapolated)				
		Σ_{total}	(n,n)	(n,n')	(n,γ)	Σ_{total}	(n,n)	(n,n')	(n,γ)	Σ_{uncorr}
Fe	$(\frac{\Delta R}{R})_{total}$	3.79%	1.72%	3.94%	0.13%	6.20%	2.81%	6.44%	0.21%	7.03%
	$(\frac{\Delta R}{R})_{loss}$	33.41%	32.57%	7.31%	0.58%	-	-	-	-	-
Cr	$(\frac{\Delta R}{R})_{total}$	2.75%	2.11%	1.38%	n.a.	4.50%	3.45%	2.26%	n.a.	4.12%
	$(\frac{\Delta R}{R})_{loss}$	81.63%	81.39%	2.48%	n.a.	-	-	-	-	-
Ni	$(\frac{\Delta R}{R})_{total}$	1.05%	1.03%	1.27%	n.a.	1.78%	1.74%	2.15%	n.a.	2.77%
	$(\frac{\Delta R}{R})_{loss}$	31.66%	28.48%	2.07%	n.a.	-	-	-	-	-
Mix	$(\frac{\Delta R}{R})_{total}$	4.8%	2.9%	4.4%	n.a.	7.9%	4.8 %	7.2%	n.a.	8.6%

the uncertainty in the heating due to cross section uncertainties is less than 20%.

Thanks are due to Dr. E. Sartori of the NEA Data Bank for supplying the JEF-/EFF-1 covariance data at an early stage and to Dr. P. Vontobel for sending the TRAMIX code and MATXS data library.

REFERENCES

1. A. Hogenbirk, "Sensitivity and Uncertainty Analysis of the nuclear heating in the coils of a fusion reactor", Report ECN-C-90-034, Energieonderzoek Centrum Nederland, July 1990

2. S. A. W. Gerstl and W. M. Stacey Jr., Nucl. Sci. Eng. 51 (1973) 339

3. D. E. Bartine, E. M. Oblow and F. R. Mynatt, Nucl. Sci. Eng. 55 (1974) 147

4. S. A. W. Gerstl, D. J. Dudziak and D. W. Muir, Nucl. Sci. Eng. 62 (1977) 137

5. S. A. W. Gerstl, "SENSIT: a cross section and design sensitivity and uncertainty analysis code", Report LA-8498-MS, Los Alamos Sci. Lab. University of California, 1980

6. K. Furuta, Y. Oka and S. Kondo, Nucl. Eng. Des./Fus. 3 (1986) 287

7. P. Vontobel and S. Pelloni, "JEF/EFF Based Nuclear Data Libraries", EIR-Bericht Nr. 636, Eidgenössisches Institut für Reaktorforschung, 1987, updated version used (obtained in 1989)

8. J. Stepanek, "TRAMIX: A Code for Interfacing MATXS Cross Section Libraries to Nuclear Transport Code for all Type Fission Reactor, Fusion Blanket as well as Shielding Analysis", Paul Scherrer Institut (draft, 15-4-1988)

9. J. Stepanek and C. E. Higgs, "A General Description of AARE: A Modular System for Advanced Analysis of Reactor Engineering", in: Proceedings of the 1988 International Reactor Physics Conference, Jackson Hole, Wyoming, 18-21 September (1988)

10. R. E. MacFarlane, "TRANSX-CTR: A Code for Interfacing MATXS Cross Section Libraries to Nuclear Transport Codes for Fusion Systems Analysis", Report LA-9863-MS, Los Alamos Sci. Lab. University of California, 1984

11. W. W. Engle Jr., "A users manual for ANISN", Report ORNL/K-1693, Oak Ridge National Laboratory, 1967

12. "Report on ITER Blanket/Shield Work Session, July 17 - August 10 1989, held at the ITER Work Site, Garching, FRG", Report ITER-IL-BL-1-9-9, 1989

13. W. Dänner, "Studies into the new inboard shield configuration for ITER", Report N/R/3320/4/A, 1990

14. W. Dänner, private communication, 1989

15. P. Vontobel, "Assessment of NJOY Generated Neutron Heating Factors Based on JEF/EFF-1", PSI-Bericht Nr. 56, Paul Scherrer Institut, 1990

16. Y. Gohar and M. A. Abdou, "MACKLIB-IV: a library of nuclear response functions generated with the MACK-IV computer program from ENDF/B-IV", Report ANL/FPP/TM-106, Argonne National Laboratory, 1978, updated version used (obtained in 1989)

17. D. W. Muir and R. E. MacFarlane, "The NJOY Nuclear Data Processing System, Vol. IV; The ERRORR and COVR Modules", Report LA-9303-M, Los Alamos Sci. Lab. University of California, 1985

18. I. Kodeli and E. Sartori, "Covariance Data Library ZZ-VITAMIN-J/COVA", Report for NEA Data Bank 1264 package, 1989