UNCLASSIFIED

MASTER

CLASSIFICATION LEVEL
(S, C, OR U)

ATOMICS INTERNATIONAL
A Division of North American Aviation, Inc.

NAA-SR-MEMO 8369

Do not remove
this sheet

This document contains 36 pages

This is copy 15 of series

UNCLASSIFIED

CLASSIFICATION TYPE
(RD OR DI)

GROUP 1
EXCLUDED FROM AUTOMATIC
DOWNGRADING & DECLASSIFICATION

*NAA-SR-MEMOs are working papers and may be expanded,
modified, or withdrawn at any time,
and are intended for internal use only.*

This report may not be published without the approval of the Patent Branch, AEC.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission to the extent that such employee or contractor prepares, handles or distributes, or provides access to, any information pursuant to his employment or contract with the Commission.

CLASSIFICATION LEVEL
(S, C, OR U)

UNCLASSIFIED

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

UNCLASSIFIED

ATOMICS INTERNATIONAL A Division of North American Aviation, Inc.		NAA-SR-MEMO DOC NO 8969	APPROVALS
TECHNICAL DATA RECORD		PAGE 1 OF 36	
AUTHOR C. Steichen	DEPT & GROUP NO. 727 - 75	DATE 8/2/63	
TITLE Fast Neutron Attenuation Analysis for a SNAP Reactor Shadow Shield Using the Flexible Monte Carlo Code, FMC-N		GO NO 7622	<i>H.E. Buttery</i>
		S A NO 1420	TWR 52973
		SECURITY CLASSIFICATION	
PROGRAM SNAP 2	SUBACCOUNT TITLE Shield Analysis	(CHECK ONE BOX ONLY) UNCL. ☐ AEC ☐ DOD ☐ CONF. ☒ SECRET ☐	(CHECK ONE BOX ONLY) RESTRICTED DATA ☒ DEFENSE INFO. ☐
36pg DISTRIBUTION R. Ashley J. Asquith J. Belcher R. Blaine W. Brown K. Buttery D. Cockeram R. Detterman H. Dieckamp R. Elliott S. Fields W. Flynn R. Gimera C. Goetz J. Gylfe R. Hubner C. Johnson R. Johnson L. Maki L. Mims L. Moss W. Roberts K. Rooney D. Schweiger J. Stewart J. Susnir R. Thomson J. Wetch R. Williams R. Wilson Nuclear & Dynamics Unit(15)	STATEMENT OF PROBLEM Perform calculations to demonstrate the feasibility of using the Flexible Monte Carlo Code FMC-N for fast neutron attenuation analysis for SNAP reactor shadow shields.		
	ABSTRACT Calculations were performed to demonstrate the feasibility of using the G.E.A.N.P. Flexible Monte Carlo Code, FMC-N, for SNAP shielding analysis. The SNAP 10A reactor-shield configuration, with the original design shield, was used as the calculational model. The results of the calculation were somewhat inconclusive due to the lack of any actual comparison between FMC calculations and experimental data for LiH systems. However, the ability of the code to calculate such things as flux, reflector scattering contribution, leakage effects, spectra, and attenuation between shield bottom and dose plane, was demonstrated. Comparisons between the FMC results and previous calculations using ray tracing techniques pointed out areas where the discrepancies in the methods of calculation were sufficiently large to warrant further study.		
<div style="text-align: center;"> UNCLASSIFIED Classification Cancelled Or Changed To By Authority Of <i>DOC</i> By <i>J. E. Buttery</i> Date <i>3 Nov 71</i> Exempt from CCRP Re-review Requirements (per 7/22/82 Duff/Caudle memorandum) <i>HA 9/28/84</i> </div>			

UNCLASSIFIED

UNCLASSIFIED

~~CONFIDENTIAL~~

01155 UNCLASSIFIED

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

NO. NAA-SR-TDR-8969
DATE 8/2/63
PAGE 2 OF 36

INTRODUCTION

The application of Monte Carlo techniques to the analysis of SNAP shielding has been prompted by the failure of existing calculational techniques, and experimental data, to provide results with the confidence required for high performance shields.

This study of the G.E.A.N.P. Flexible Monte Carlo Code, FMC-N, was initiated in an attempt to establish a Monte Carlo calculational technique, suitable for use in SNAP shield analysis, using an existing code.

The SNAP 10A reactor-shield configuration, using the original design shield, was chosen as the calculational model in this study.

The objectives of this study were as follows:

- 1) Determine the feasibility of using the FMC-N code for SNAP shielding analysis.
- 2) To obtain the fast neutron flux distribution through the LiH shield; across a plane within the shield shadow just below the shield; and across a plane within the shield shadow at the mating plane.
- 3) Determine the fast neutron flux attenuation from the bottom of the shield to the mating plane.
- 4) Determine fast neutron spectra at various locations in the shield and below the shield.
- 5) Determine the contribution of neutrons scattered in the reflector to the total number of fast neutrons penetrating the shield within the shadow.

UNCLASSIFIED

~~CONFIDENTIAL~~

UNCLASSIFIED

~~CONFIDENTIAL~~

NO. NAA-SR- TDR- 8969

DATE 8/2/63

PAGE 3 OF 36

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

DISCUSSION

A general description of the FMC-N code is given in Appendix A and a discussion of the methods used to generate the material constants required as input to the code is given in Appendix B. A description of the SNAP LOA reactor shield calculational model is shown in Figure 1. The material compositions of the various regions are given in Table 1. Because of the limited number of elements available in the cross section library, the exact compositions of the various regions could not be used. The materials such as fuel cladding, core can, and grid plates, were assumed to be pure iron, and the NaK coolant was assumed to be pure sodium. The steel honeycomb structure in the shield was not included in the shield composition.

Because of the large attenuation in the shield, it was not possible to obtain accurate results, for reasonable calculational times, using a straightforward unbiased Monte Carlo technique. Therefore, it was necessary to use some technique which increased the probability of an initial source neutron contributing to the final answer. A study of the various biasing, or variance reduction techniques, available in the FMC code, indicated that a combination of biased source sampling, statistical estimation, splitting, and Russian roulette techniques would produce results with the least variance for a given amount of computational effort.

The abundance of neutrons of various energies produced by fission decreases rapidly as the energies of the emitted neutrons increases. However, for shields with large attenuations, the source neutrons contributing most to the transmitted flux are those with relatively high energies. Therefore, the computational effort can be reduced by biasing the source sampling such that most of the computational effort is spent on source neutrons with the greatest probability of penetrating the shield. The importance of the biased source neutrons are weighted to reflect the true abundance in the fission spectrum.

The statistical estimation technique is often called the technique of expected values, and therefore, quantities calculated using this technique are called expectation quantities (i.e., expectation entrance and expectation flux). The expectation entrance is calculated by multiplying the weight of a neutron after each collision, or after leaving the source,

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

NO. NAA-SR-TDR 8969
DATE 8/2/63
PAGE 4 OF 36

by the probability of the particle continuing in the direction chosen and entering the selected region without further collision. The total expectation entrance for a given region is then the sum of all individual entrance tallies. Therefore, the position and nature of a collision is determined by Monte Carlo but the probability of entrance is determined analytically. It can also be seen that the total number of expectation entrances for a given region will be greater than the number of Monte Carlo entrances, and the resulting statistical error should be less. However, the presence of a large number of expectation entrances does not guarantee a correct answer. The overall distribution of Monte Carlo collisions must also be reasonably statistically correct. FMC-N allows both Monte Carlo and expectation entrance tallies to be formed for a given problem. FMC-N also allows either Monte Carlo or expectation flux tallies to be formed, but both cannot be formed in a single problem.

In splitting, the shield is divided into regions and it is decided that, upon entering a given region in a direction away from the source, the neutron will be split into n identical neutrons. Each neutron is given a weight of $1/n$ th of its former value and is treated as a distinct neutron. If the neutron enters the region in a direction heading back toward the source, Russian roulette is used to determine if the neutron history should be continued or terminated. The probability of continuing is set at $1/n$. If the neutron history is continued, its weight is multiplied by n . Therefore, when using Russian roulette, and splitting, a lot of computational time is spent on neutrons penetrating deeply into the shield and very little time is spent on neutrons heading back toward the source. The difficulty encountered when using splitting and Russian roulette is the determination of the separation distance between splitting boundaries and the degree of splitting to be used to obtain minimum variance and computational time. This difficulty becomes more acute when the source contains a distributed spectrum of source energies. Also, the absolute number, in addition to the weighted number, of neutrons entering each region should be known in order to estimate the statistical error.

Initial calculations in this study, using the splitting technique, were made for source energies covering the entire fission spectrum. Because of the difference in the attenuation for the various source energies, it soon became apparent that it was impractical to use the same

~~CONFIDENTIAL~~ 1507030

NO. NAA-SR-TDR-8969

DATE 8/2/63

PAGE 5 OF 36

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

spacing of the splitting boundaries for all energies. Therefore, the fission source was divided into 4 energy groups and separate problems were run for each of the groups. Spacing of the splitting boundaries and the degree of splitting were determined by trial and error. Problems using 2000 source neutrons were run for various splitting boundary locations, until a satisfactory distribution of neutrons crossing each of the splitting boundaries was obtained. The desired distribution of neutrons was such that equal numbers of neutrons crossed each boundary. However, this ideal was difficult to achieve with a reasonable number of trials, and consequently, the final distribution was somewhat different than desired. The degree of splitting used at the splitting boundaries was the same for all of the source energy groups. The neutrons were split by a factor of 2 at the top boundary of the shield and by a factor of 3 at each of the splitting boundaries in the shield. Using a factor of 2 at each splitting boundary would have required a proportionately greater number of boundaries and a corresponding increase in computational time. The locations of the splitting boundaries for each of the source energy groups are shown in Figures 2, 3, and 4. In order to obtain the distribution of the absolute number of Monte Carlo neutrons crossing the splitting boundaries, a change had to be made in the FMC-N code. Previously only the weighted number of neutrons could be obtained. The change was made in the code by normalizing each Monte Carlo entrance tally to 1. When this option is used, the weighted Monte Carlo entrance cannot be obtained. However, the expectation entrance tally is still valid. The actual numbers of Monte Carlo entrances in the various shield regions are given in Table 2.

The FMC code does not obtain flux values at a point but, rather it calculates the total path length of the neutrons in a given region. The average flux in the region is then obtained by dividing by the region volume. Therefore, in order to obtain the desired fluxes, care must be taken in arranging proper regions for the flux calculation. In this study, expectation flux tallies were obtained for 0.1 cm thick regions at each of the splitting boundaries in the shield, just below the reactor within the shield shadow, and at the mating plane. These plane regions in turn were divided into radial regions. The locations of the regions are shown in Figures 2, 3 and 4.

1507030

03 15 07 03 00

~~CONFIDENTIAL~~

NO. NAA- SR-TDR 8969

DATE 8/2/63

PAGE 6 OF 36

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

The first few runs for each of the source energy groups were made using only 2000 source neutrons per run. Although it was realized that considerably more than 2000 source neutrons, in each source energy group, would be required to obtain reasonable accuracy, a number of runs were made for each source energy group in order to obtain some idea of the variation in results between runs. The initial pseudo random number, used in each of the runs for each source energy group was taken as the last pseudo random number used in the previous run.

The calculations were performed for energies between 0.0919 and 18.0 Mev. A total of 35 energy groups, equally spaced in lethargy between the upper and lower energy limits, were used to describe the input material constants. Output tallies were obtained for each of the 35 energy groups.

The source energy bounds for each of the 4 separate problems, used to calculate the complete fission spectrum, were 0.1 to 3.099 Mev, 3.099 to 5.715 Mev, 5.715 to 10.54 Mev, and 10.54 to 18 Mev. The fraction of the fission spectrum assigned to each of the source energy groups, and the cumulative probability distribution of the source energies within each of the groups is given in Table 3. The probability distributions include the true fission spectrum distribution and the biased distribution as used in the calculations.

The total number of neutron histories calculated for each of the four source energy groups is given in Table 2. A total of 68,000 histories were calculated. The computations were performed at the Rocketdyne computing facilities and at Oak Ridge National Laboratory. The total running time for 36,000 histories on Rocketdyne's IBM 7094 computer was about 2 hours. The total running time for 32,000 histories on ORNL's IBM 7090 computer was 3 hours.

ANALYSIS OF RESULTS

The average expectation fluxes obtained in the various regions of the shield are plotted in Figure 5. The total fluxes for the entire fission source energy spectrum, along with the fluxes for each of the 4 separate problems used to calculate the complete spectrum, are shown as a function of the distance from the center of the core. The fluxes are the average fluxes over 0.1 cm high cylindrical regions with diameters equal to the diameter of the core, and located at the distances shown.

~~CONFIDENTIAL~~

~~CONFIDENTIAL~~

NO. NAA-SR-TDR 8969

DATE 8/2/63

PAGE 7 OF 36

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

As can be seen from the plots, the largest contribution to the total flux at the top of the shield was from source energies from 0.0919 to 5.715 Mev. At the bottom of the shield, the largest contribution was from source energies from 5.715 to 10.54 Mev.

Fluxes calculated using the G.E.A.N.P. 14-0 ray tracing code are also shown in Figure 5¹. These fluxes were calculated at points along the center line of the shield using removal theory with a fast neutron removal cross section of 0.128 cm^{-1} . For comparison purposes, the fluxes are plotted for cases with and without the customary addition for neutron scattering from the reflectors. As can be seen from the plots, the FMC flux near the top of the shield (5.82 cm of LiH) is about a factor of 2 greater than the 14-0 flux without the reflector contribution; at the bottom of the shield, the FMC flux is about a factor of 1.8 less than the 14-0 flux. Comparison with the 14-0 cases which include the reflector contribution indicates that the FMC flux is about a factor of 1.3 greater at the top of the shield and about a factor of 3.6 less at the bottom of the shield.

Because of the spatial distribution of the source neutrons in the core, it is difficult to obtain an attenuation coefficient, λ , for a point kernel of the form

$$\frac{e^{-\lambda t}}{4\pi r^2}$$

from the plot of the fast neutron flux. Therefore, an attenuation coefficient was determined for the LiH shield such that the 14-0 code, using this attenuation coefficient, would produce the same flux as the FMC calculations. To do this, the fluxes calculated using 14-0, and various LiH attenuation coefficients, were plotted (in Figure 6) as a function of the coefficients. The required coefficient was then read from this plot at the point corresponding to the required FMC flux. The attenuation coefficients required to obtain the FMC fluxes at two locations were determined. The first location, 71.5 cm from the center of the core, corresponded to the position at which the parabolic portion of the shield joined the conical portion. The second location, 91.5 cm from the center of the core, was just below the bottom of the shield. The calculated attenuation coefficients for these two locations were 0.131 cm^{-1} and 0.136 cm^{-1} , respectively, when the reflector scattering contribution was not considered

~~CONFIDENTIAL~~

~~CONFIDENTIAL~~

03 15047030

ATOMICS INTERNATIONAL
A Division of North American Aviation, Inc.

NO. NAA-SR-TDR 8969
DATE 8/2/63
PAGE 8 OF 36

and 0.144 cm^{-1} and 0.145 cm^{-1} , respectively, when this contribution was considered.

The radial variation of the flux within the shadow of the shield just below the reactor shield, and at the mating plane, is given in Table 4. A comparison of the flux at the bottom of the shield to the flux at the mating plane can also be obtained from Table 4. The ratio of the total average flux in the shadow of the shield at the mating plane (284.3 cm from core center) to the total average flux in the shadow of the shield just below the shield (91.5 cm from core center) was 0.0359. The corresponding ratio of the average fluxes in the central regions (10 cm radius) at these two locations was 0.0405. The statistical accuracy of the ratio of the fluxes over the entire region within the shield shadow is much better than that for the small 10 cm radius regions only and, therefore, is probably a better criterion for comparison of fluxes. When the 14-0 ray tracing code was used (no reflector contribution), the ratio of the flux at the bottom of the shield to the flux at the mating plane, along the reactor-shield center line was approximately 0.1. Therefore, the FMC calculation predicted about a factor of 2.8 greater flux attenuation from the bottom of the shield to the mating plane than the 14-0 calculation.

Flux spectra at various locations in the shield were determined from the FMC calculations. These spectra are plotted in Figures 8, 9 and 10 as histograms of flux contribution per energy group. Flux spectra at various locations in the LiH shield were also obtained from NIOBE² transport theory calculations for a simulated SNAP 10A reactor-shield configuration in spherical geometry. These spectra are plotted as histograms of the total average flux contribution per energy group, and are presented in Figures 11, 12 and 13. The number of neutron histories used in the FMC calculation were not sufficient to obtain statistically accurate and smooth histograms. However, the shapes of the histograms are sufficiently defined so that some idea of the variation in spectra with penetration in the LiH can be obtained. It appears that there is reasonably good agreement between the NIOBE and FMC spectra where comparison can be made. The FMC spectra falls off much more sharply at the high energies ($> 8 \text{ Mev}$). However, the contribution to the total flux is quite small at the high energies where the discrepancies appear to occur. No NIOBE spectrum was available for comparison with the FMC spectrum below the shield. The FMC

~~CONFIDENTIAL~~

~~CONFIDENTIAL~~

NO. NAA- SR-TDR 8969

DATE 8/2/63

PAGE 9 OF 36

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

spectrum below the shield (Figure 10) exhibits an unexpectedly large peak between 5 and 8 Mev. The region in which this spectrum was obtained was a 0.1 cm high cylindrical region with a radius of 10 cm and located just below the shield (91.5 cm from core center). This region was located outside of the shield and could see a large part of the parabolic bottom surface of the shield. Physically, it was then possible for this region to see a somewhat different spectrum than a region located within the shield itself and some distance from a physical boundary. Because of the different locations of the flux tally regions in the shield for each of the separate source energy group calculations, it was difficult to obtain a spectrum at a location near the bottom of the shield but still within the shield material. Because of the small number of tallies at the mating plane, no attempt was made to obtain a flux spectrum at that location.

In order to compare calculated flux levels with experimental data, it is necessary to know how the detector response varies with the spectrum of the flux being measured. The difference in the attenuation of the calculated FMC flux and the instrument response to the flux, was obtained by converting the calculated flux spectra, at the top and bottom of the shield, to relative detector responses. The conversion was made for a polyethylene Hurst type fast neutron dosimeter. A plot of the detector response curve used in the conversion is presented in Figure 7. The ratio of the flux to detector response at the top and bottom of the shield were then obtained. These ratios indicated that the relative detector responses would have a factor of 1.45 less attenuation through the shield than the flux.

Following the adoption of the present configurations of the SNAP reactor-shield system, one of the basic shielding problems has been to determine the relative importance of neutron scattering in the reflector to the total flux intensity at the mating plane. The removal cross section concept has always been used in the ray tracing codes, such as FARSE and 14-0, to calculate the attenuation from a distributed source in the core. However, it has been customary to add to the flux, calculated in this manner, some additional contribution due to fast neutron scattering from the beryllium reflectors. Various methods have been used to calculate this contribution.³

~~CONFIDENTIAL~~

0315087030
NO. NAA-SR-TDR 8969
DATE 8/2/63
PAGE 10 OF 36

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

Going back to the original theory of the removal cross section concept and examining the manner in which the cross sections were determined, some question arises regarding the validity of the manner in which the reflector scattering contribution is added to the flux calculated using removal theory. Removal theory considers attenuation from point isotropic sources through homogenous slabs of material of infinite extent in the transverse dimension. Therefore, all of the flux due to multiple scattering in the semi-infinite slab is included in the transmitted flux. In the case of the SNAP reactors, the attenuation calculated using removal theory is really the attenuation through slabs of core and shield material of infinite extent in the transverse direction. The reflector is actually considered to be an infinitely thick radial region of core material. Therefore, the only change which should be made in the transmitted fluxes, calculated using removal theory, is the difference in the reflector contribution from the actual thickness of Be reflector and an infinitely thick reflector consisting of core material.

The history tallies obtainable with the FMC code for neutrons entering selected regions give the regions in which the neutrons were born, the regions in which the neutrons made their first three collisions, and the regions in which the neutrons made their last four collisions. It is, therefore, possible to follow portions of the path of each neutron history. This feature of the FMC code was used to determine the contribution of neutrons which underwent at least one scattering event in the interface or reflector regions to the total number of neutrons transmitted through the shield. Thus, some idea of the importance of reflector scattered neutrons was obtained. By tabulating the weights of each neutron which underwent a scattering event in the reflector, only the weighted number of neutrons entering a given region was obtained. Therefore, the comparison had to be made on an entrance basis and not on a flux basis. However, at the mating plane, the number of neutrons entering a region is very nearly proportional to the flux, because the neutron must enter the region in a direction close to the normal of the surface. The entrance tallies, calculated using the history tallies, were actually Monte Carlo entrance tallies; whereas, the entrance tallies calculated by the code were expectation entrance tallies. At the mating plane, the total entrance tally was obtained by tabulating all of the history tallies, so a comparison could be made on a Monte Carlo

~~CONFIDENTIAL~~

~~CONFIDENTIAL~~

03 15587030

ATOMICS INTERNATIONAL
A Division of North American Aviation, Inc.

NO. NAA-SR-TDR 8969
DATE 8/2/63
PAGE 11 OF 36

entrance basis. In any case, the expectation entrance tallies and the Monte Carlo entrance tallies were so similar that any difference was relatively unimportant in the comparison.

The ratio of reflector scattered neutrons to total neutrons which crossed the entire region below the shield (91.50 cm from core center), within the shield shadow, was 0.125. The corresponding ratio for the entire region at the mating plane (284.3 cm from core center), within the shield shadow, was 0.134. In other words, only one out of eight neutrons crossing the region below the shield, within the shield shadow, had undergone a scattering event in the interface or reflector regions. The ratio of 1 out of 8 is considerably different than the approximate ratio of 1 out of 2 calculated¹ using the ray tracing techniques with codes such as FARSE, 14-0, and SCAR. In fact, the ratio between reflector scattered and total flux is probably not much different than that to be expected for a reflector consisting entirely of core material, and thus, already included in the removal theory calculation for the distributed source in the core.

There are some considerations which may tend to slightly underestimate the number of reflector scattered neutrons obtained by FMC. The $n, 2n$ cross section in the beryllium reflector was included in the absorption cross section; thus the total number of scattered neutrons was underestimated. However, when the magnitude of the $n, 2n$ cross section and spectra effects are considered, the contribution of the $n, 2n$ event to the total number of scattered neutrons should be small. The other consideration which may tend to underestimate the number of scattered neutrons is the manner in which the neutrons scattering in the reflector were determined. The FMC code only gives the regions in which the first three and the last four collisions occur. If the first three collisions occurred in the source region, and there were more than a total of seven collisions, it was not possible to tell where the intervening collisions occurred. However, the number of neutrons which made the first three collisions in the source region were quite small, and as only a fraction of these scattered in the interface or reflector region, the overall scatter contribution was quite small.

CONCLUSIONS

Several inconsistencies appear when the FMC calculations are com-

~~CONFIDENTIAL~~

03 10587030
~~CONFIDENTIAL~~

NO. NAA-SR-TDR 8969

DATE 8/2/63

PAGE 12 OF 36

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

pared to the results of calculations using ray tracing codes and removal theory. The FMC calculations indicated that the contribution of reflector scattered neutrons was quite small below the shield. The ray tracing codes indicated that the contribution of reflector scattered neutrons was comparable to the non-reflector scatter neutrons below the shield. At the top of the shield, however, FMC predicted a much higher fast neutron flux than removal theory, but the difference was much less when the reflector contribution was added. It is possible, that due to spectral and leakage considerations, the relative importance of reflector scattered neutrons is larger at the top of the shield than at the bottom; however, when the concept of removal theory is considered, it is difficult to see how this could account for the entire discrepancy in the calculated fluxes. In fact, it seems that when the concept of removal theory is considered, the removal theory flux should be greater than the FMC flux, because of the finite radial extent of the reflectors.

This difference in calculated flux at the top of the shield has a serious effect on the LiH attenuation coefficient which must be used in a ray tracing code, in order to obtain the same results as FMC at the bottom of the shield. Although the slope of the FMC flux curve is much greater than the slope of the ray tracing flux curves, the LiH attenuation coefficient which must be used in the ray tracing codes, in order to obtain the FMC flux at the bottom of the shield, must be much less than that indicated by the FMC flux curve.

The quantities proportional to the detector responses of a fast neutron detector, such as the Hurst dosimeter, which were calculated from the FMC flux spectra, were attenuated less than the FMC flux in the shield. In other words, a curve of the Hurst dosimeter response to the flux in the shield, versus distances in the shield, would have a lesser slope than a similar curve of the actual calculated flux itself for neutron energies above 0.1 Mev. Therefore, the slope of the curve for detector responses more nearly follows the slope of the curve for ray tracing calculations. However, since the absolute relationship between actual flux and flux calculated using removal theory, versus the flux spectra, is still not known, the discrepancy between the removal theory flux and FMC flux at the top of the shield cannot be explained.

03 10587030
~~CONFIDENTIAL~~

~~CONFIDENTIAL~~

0315587030

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

NO. NAA-SR-TDR 8969
DATE 8/2/63
PAGE 13 OF 36

As was discussed in the analysis, the FMC flux spectra in the shield agreed quite well with the NIOBE flux spectra except at high energies (>8 Mev) where the NIOBE spectra fell off faster than the FMC spectra. There has been some doubt as to the accuracy of NIOBE flux calculations for some types of calculations, such as small source regions, but there was no evidence that there was anything unusual about the calculation from which the flux spectra was obtained. However, at the same time, there has been no general confirmation of the accuracy of NIOBE for this type of calculation. The FMC flux spectrum below the shield appeared to be considerably harder than had been expected. The fact that this spectrum was obtained in a region below the shield, and not within the shield material itself, undoubtedly had some effect on the spectrum, but this effect should have been relatively small.

The large difference in the ratio of the flux at the mating plane to the flux below the shield, for the FMC and ray tracing calculations, points out the importance of the angular distribution of the flux below the shield. The statistical accuracy of the calculated flux at the mating plane is not as good as that just below the shield but it is not expected that additional calculations would make an appreciable difference in the ratio of the fluxes.

No detailed analysis of the statistical accuracy of the calculations was made; however, comparison of results between individual cases indicated that the variation in results was well within acceptable limits. Also, the computations performed at the Rocketdyne computing facilities, and at ORNL, were analyzed separately, and no appreciable difference was observed in the total flux curve for the two separate determinations.

The lack of any actual comparison between FMC calculations and experimental data for LiH systems, makes it difficult to draw conclusions regarding the accuracy of the FMC calculations. The cross section data, and the validity of the biasing techniques used in the calculations are the two main areas of uncertainty. Although no definite conclusions can be reached regarding the accuracy of the FMC calculations, comparisons between the FMC results and previous calculations using ray tracing techniques have pointed out areas where the discrepancies in the methods of calculation appear to be sufficiently large to warrant further study.

~~CONFIDENTIAL~~

0314587030

~~CONFIDENTIAL~~**ATOMICS INTERNATIONAL**

A Division of North American Aviation, Inc.

NO. NAA-SR-TDR 8969
DATE 8/2/63
PAGE 14 OF 36

The ability of the code to calculate such things as the reflector scattering contribution, leakage, spectra, and attenuation between shield bottom and dose plane, has been demonstrated. The remaining problem is to demonstrate agreement with experimental data. However, before this is done, a complete review of the cross section data should be made. Since the present cross sections were prepared, new data on Li has become available, and a new edition of BNL 400 (differential scattering cross sections) has been published.

~~CONFIDENTIAL~~ 0314587030

0315587030

NO. NAA-SR-TDR 8969
DATE 8/2/63
PAGE 15 OF 36

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

REFERENCES:

1. J. A. Belcher, SNAP 10A/2 Reactor Engineering, personal communication
2. D. Yetman, B. Eiseman, G. Rabinowitz, "Description of Input Preparation and Operating Procedures for 9-NIOBE, an IBM-7090 Code," UNCOR-631
3. R. J. Thomson, SNAP 10A/2 Reactor Engineering, personal communication

=====

0315587030

~~CONFIDENTIAL~~

NO. NAA-SR-TDR 8969

DATE 2/2/63

PAGE 16 OF 36

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

TABLE I

Material Compositions Used in the Calculations

<u>Element</u>	Atom Densities, Atoms/Cm ³ x 10 ⁻²⁴					
	<u>Core</u>	<u>Interface</u>	<u>Reflector</u>	<u>Upper Plenum</u>	<u>Lower Plenum</u>	<u>Shield</u>
H	.05454					.0540
Li						.0540
Be		.09557	.123			
Nb	.00167	.000883		.01154	.0155	
Fe	.004708	.006121		.02336	.0056	
Zr	.02917					
U-235	.001251					

~~CONFIDENTIAL~~

031507030

NO. NAA-SR-TDR 8969

DATE 8/2/63

PAGE 17 OF 36

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

TABLE 2

Absolute (Not Weighted) Number of Monte Carlo
Neutrons Entering Given Regions, and the Number of Source
Neutrons Started, for Each of the Source Energy Groups

Source Energy Groups, Mev							
.0919 to 3.099		3.099 to 5.715		5.715 to 10.54		10.54 to 18.0	
No. of Source Neutrons Started							
16,000		16,000		20,000		16,000	
Region No.	Monte Carlo Entrance	Region No.	Monte Carlo Entrance	Region No.	Monte Carlo Entrance	Region No.	Monte Carlo Entrance
10	545	10	1202	10	1887	10	1438
11	151	11	418	11	704	11	553
12	257	12	818	12	1374	12	1095
14	98	14	337	14	506	14	209
15	152	15	746	15	1259	15	639
17	53	17	283	17	469	17	158
18	115	18	750	18	1130	18	536
20	41	20	272	20	420	20	140
21	88	21	845	21	1162	21	494
23	29	23	274	23	406	23	97
24	56	24	862	24	1322	24	474
26	22	26	264	26	402	26	88
27	52	27	850	27	1442	27	224
29	15	29	244	29	416	30	64
30	34	30	831	30	1462	31	141
32	8	32	231	32	546	32	166
33	29	33	839	33	1138	34	1
35	4	35	247	35	616	35	2
36	24	36	849	36	369	36	11
38	3	38	221	39	604	37	9
39	16	39	700	40	1102	38	18
51	5	51	247	41	738		
52	4	52	407	43	5		
42	1	42	121	44	36		
43	7	43	422	45	61		
44	5	44	338	46	82		
46	0	46	4	47	84		
47	1	47	6				
48	0	48	15				
49	0	49	36				
50	0	50	27				

031507030

CONFIDENTIAL
NO. 1
AA-SR-TDR 8969

DATE 8/2/63
PAGE 18 OF 36

TABLE 3

Fraction of Fission Spectrum Assigned to Each Source Energy Group and the Cumulative Probability Distribution of Source Energies Within Each of the Groups

Source Energy Group, Mev									
0.919 to 3.099		3.099 to 5.715		5.715 to 10.54		10.54 to 18.0			
Fraction of Fission Neutrons Born in Each Source Energy Group									
.8051		.1647		.0295		.000685			
Cumulative Probability Distribution for Each Source Energy Group									
E	True Spectrum	E	True Spectrum	E	True Spectrum	Biased Spectrum	E	True Spectrum	Biased Spectrum
0.919	0	3.099	0	5.715	0	0	10.54	0	0
1.071	.01925	3.612	.3479	6.660	.5288	.25	12.28	.777	.40
1.1975	.04608	4.209	.6333	7.761	.8203	.50	14.31	.962	.70
1.3124	.08769	4.904	.8494	9.043	.9424	.75	16.67	.996	.90
1.4243	.1329	5.715	1	10.54	1	1	18.00	1.0	1
1.4944	.1626								
1.5761	.1981								
1.6714	.2402								
1.7823	.2894								
1.9116	.3460								
1.1062	.4090								
1.1442	.5648								
1.1959	.7406								
1.2660	.9201								
1.3099	1.000								

0315087030

NO. NAA-SR-TDR 8936

DATE 8/2/63

PAGE 19 OF 36

ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

TABLE 4

Average flux per source neutron in the shadow of the shield, and the average flux as a function of radius in the shadow of the shield, just below the shield and at the mating plane.

Below Shield (91.50 Cm from Center of Core)

Radius, Cm	Average Flux, Neutrons/Cm ² -sec per Source Neutron
0 - 10	9.98 x 10 ⁻¹¹
10 - 20	1.402 x 10 ⁻¹⁰
20 - 29.2	1.728 x 10 ⁻¹⁰
0 - 29.2	1.553 x 10 ⁻¹⁰

Mating Plane (284.3 Cm from Center of Core)

Radius, Cm	Average Flux, Neutrons/Cm ² -sec per Source Neutron
0 - 10	4.04 x 10 ⁻¹²
10 - 20	4.58 x 10 ⁻¹²
20 - 30	4.99 x 10 ⁻¹²
30 - 40	5.99 x 10 ⁻¹²
40 - 50.4	6.06 x 10 ⁻¹²
0 - 50.4	5.58 x 10 ⁻¹²

0315087030

FIGURE 1
 CALCULATIONAL MODEL FOR THE SNAP 10A
 REACTOR SHIELD SYSTEM

ALL DIMENSIONS
 MEASURED FROM
 CENTER OF CORE

WVR CASTLE
 8-5-63

ALL DIMENSIONS IN CM

10X10 TO THE CM. 359-14
 KEUFFEL & ESSER CO. MADE IN U.S.A.

FIGURE 2
 LOCATION OF SPLITTING BOUNDARIES
 AND FLUX TALLY REGIONS FOR THE
 .0919 TO 3.099 Mev AND THE 3.099 TO
 5.715 Mev SOURCE ENERGY GROUPS

FIGURE 4
 LOCATION OF SPLITTING BOUNDARIES
 AND FLUX TALLY REGIONS FOR THE
 10.54 TO 18.0 SOURCE ENERGY GROUP

W.R. CASTLE
 8-5-63

10 X 10 TO THE CM. 359-14
 KEUFFEL & ESSER CO. MADE IN U.S.A.

CONFIDENTIAL

FIGURE 5

~~FIGURE 6~~

FAST NEUTRON FLUX
 CALCULATED USING THE 14-0 RAY
 TRACING CODE AS A FUNCTION OF
 THE LH ATTENUATION COEFFICIENT
 AND THE DISTANCE FROM THE
 CORE CENTER, WITH AND WITHOUT
 THE REFLECTOR CONTRIBUTION

NEUTRONS / CM²-SEC-SOURCE NEUTRON

10⁻¹¹
0.12

0.13

0.14

0.15

0.16

LH ATTENUATION COEFFICIENT, CM

91.50 CM WITHOUT REFLECTOR CONTRIBUTION

91.50 CM WITH REFLECTOR CONTRIBUTION

71.40 CM WITHOUT REFLECTOR CONTRIBUTION

71.40 CM WITH REFLECTOR CONTRIBUTION

RESPONCE CURVE FOR AN ETHYLENE FAST NEUTRON DOSIMETER

W. R. CASTLE
2-4-63

JJA-SJ-MDR 2026
 Date: 8/2/63
 Page 26 of 26

FIGURE 8
FMC FLUX SPECTRUM IN
SHIELD AT 26.50 CM FROM
CENTER OF CORE
(5.82 CM LH)

W.R. CASTLE
8-4-63

MA-46-D-809
Date 8/2/63
Page 27 of 36

W.R. CASTLE
9-6-63

NA-5R-11B-0169
Page 8 of 36
Date 8/2/63
Page 20 of 36

NA-SS-RDB-8969
Date: 8/2/83
Page 29 of 36

W.R. CASTLE
8-7-63

FIGURE 12
NI0BE FLUX SPECTRUM IN SHIELD
(46.02 CM LIH)

W.R. CASTLE
8-7-63

NAAS R-TR 8969
Date 8/2/63
Page 31 of 36

FIGURE 13
 NIOBE FLUX SPECTRUM IN SHIELD
 (60.18 CM LIH)

RELATIVE FAST NEUTRON FLUX / ENERGY GROUP

W.R. CASTLE
 8-8-63

Date: 8/2/63
 Page 32 of 36

03105087030

~~CONFIDENTIAL~~**ATOMICS INTERNATIONAL**

A Division of North American Aviation, Inc.

NO. NAA-SR-TDR 8969DATE 8/2/63PAGE 33 OF 36

APPENDIX A

A General Description of the Flexible Monte Carlo Code FMC-N

FMC-N* is a program which applies Monte Carlo methods to simulate neutron histories in a source shield configuration. The program is designed for flexibility in the geometrical, material, nuclear and source descriptions of source-shield configurations. Several variance reduction techniques are available as options in the code.

Any number of regions, of a variety of geometrical shapes, can be represented by any number of surfaces described by the general equation:

$$AX^2 + X_0X + BY^2 + Y_0Y + CZ^2 + Z_0Z - K = 0$$

Each region is composed of a homogenous mixture of any number of materials.

The direction, spatial and energy parameters of source neutrons may be selected by a source particle generator option, or they may be entered as input from cards or magnetic tape.

The neutron collision routine uses random sampling from appropriate discrete distributions to select either an elastic scattering event, an inelastic scattering event, or a fission event. The angular distribution of the scattered neutrons may be isotropic or anisotropic in the center-of-mass system at the discretion of the user. An absorption event is not allowed. In order to eliminate the bias introduced by the event sampling scheme of not allowing an absorption to occur, a neutron emerging from a collision is given a weight equal to the probability that it escaped absorption at that collision multiplied by the probability that it escaped absorption at all prior collisions. The fission event routine can be used to simulate n, 2n events if desired.

Several variance reduction techniques are optionally allowed in addition to the mandatory technique of weighting for absorption escape at each collision. An optional statistical estimation technique

*J. J. Loechler and J. E. MacDonald, Flexible Monte Carlo Programs FMC-N and FMC-G, APEX-706

~~CONFIDENTIAL~~

03105087030

0315387030

~~CONFIDENTIAL~~
ATOMICS INTERNATIONAL

A Division of North American Aviation, Inc.

NO. NAA-SR-TDR 8969
DATE 8/2/63
PAGE 34 OF 36

(expectation technique) may be used for obtaining flux tallies and for scoring entrance tallies and mandatory leakage tallies. Flexibility in sampling from source spectra is achieved by using energy group averaged constants called acceleration factors. This option in the code has been altered somewhat from the original version in order to provide a more useful method of biased sampling from the source spectrum. The other options, which are forms of importance sampling, are splitting and Russian roulette depending on energy, region, and location within a region, Russian roulette on a particle whose weight falls below the cut off, and exponential transformation.

Output available from this program includes independently optional absorption or energy disposition tallies, Monte Carlo entrance and leakage tallies, expectation entrance and leakage tallies, Monte Carlo or expectation flux tallies, and history tallies of particles reaching selected regions. The history tallies include the spatial coordinates, direction cosines, energy, the region where the particle was born, the regions in which the particle made its first three collisions, the region in which the particle made its last four collisions, and the weighted number of particles represented by the particle for each particle entering selected regions.

~~CONFIDENTIAL~~
0315387030

031508 UNCLASSIFIED

ATOMICS INTERNATIONAL
A Division of North American Aviation, Inc.NO. NAA-SR-TDR 8969
DATE 8/2/63
PAGE 35 OF 36

APPENDIX B

Material Cross Section Library for FMC-N

Since there was no cross section library available for FMC-N, all of the material constants had to be assembled before the code could be used. The material constants required were the microscopic elastic scattering cross sections, inelastic scattering cross sections, fission cross sections (for n , $2n$ events), and total cross sections, the cumulative probability angular distribution tables for anisotropic elastic scattering, and the cumulative probability energy distribution tables for inelastic scattering and fission (n , $2n$).

The material constants for 8 elements were included in the library. These elements were H, Li, Be, Na, Ti, Fe, Zr, and U-235. The constants were prepared for 50 energy groups ranging from 0.01 Mev to 18.0 Mev, and were equally spaced in lethargy intervals. The library was later changed to 35 energy groups with energies ranging from 0.0919 Mev to 18.0 Mev with the same lethargy spacing as in the 50 group library.

All of the microscopic cross section data, with the exception of Ti, was obtained by using the Genda cross section assembly program which is included as an integral part of the NIOBE* transport code. The Genda program calculates point-wise data for any equally spaced lethargy group- ing from point-wise input data of any other equally spaced lethargy group- ing using linear interpolation. The Genda program was used to calculate the point-wise data at the mid-points of the lethargy groups chosen for the FMC-N library. This data was then assembled for the FMC-N library such that it was constant between the end points of the energy groups. The microscopic data for Ti was obtained from BNL-325.

The Genda program also calculates the point-wise Legendre Polynomial coefficients used to describe the anisotropic elastic scattering. The cumulative probability angular distribution tables for anisotropic elastic scattering, with the exception of Ti, were obtained by using a code which generates the cumulative distribution tables using Legendre Polynomials with the Legendre coefficients as input. The code was a modified version

*D. Yetman, B. Eiseman, G. Rabinowitz, "Description of Input Preparation and Operating Procedures for 9-NIOBE, an IBM-7090 Code," UNNCCOR-631

UNCLASSIFIED

011000 UNCLASSIFIED

~~CONFIDENTIAL~~**ATOMICS INTERNATIONAL**

A Division of North American Aviation, Inc.

NO. NAA-SR-TDR 8969
DATE 8/2/63
PAGE 36 OF 36

of a code described in a GEANP document, GEMP-115. The cumulative probability distribution table for Ti was calculated from differential scattering cross sections tabulated in UCRL-5573.

The neutron spectra cumulative probability distribution tables for inelastic scattering were obtained by using a code which determines the tables from the probability density distribution given by the evaporation model of the nucleus. This code was a modified version of a code described in the GEANP document GEMP-116.

The non-elastic portion of the Be cross section, usually associated with inelastic scattering, was assumed to consist entirely of the $n, 2n$ event. This cross section is treated as a fission cross section in FMC-N. The non-elastic portion of the Be cross section is treated as absorption if the fission option is not used.

UNCLASSIFIED

~~CONFIDENTIAL~~
011000