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PRELIMINARY CROSS-SECTION SENSITIVITY ANALYSIS FOR AN AIR-OVER-GROUND ENVIRONMENT*

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Two-dimensional sensitivity calculations[1] have been made for an air-over-ground geometry to determine the effect of air and ground cross-section perturbations on the total neutron and gamma ray dose near the air/ground interface and 415 meters above the ground. ENDF/B 22 neutron and 18 gamma group Version IV cross-sections were used in all computations.

The two-dimensional air-over-ground calculations were made with the DØT code[2] using cylindrical geometry. The cylinder had an upper 1300M high region of air and a lower 50 cm deep region of ground. The maximum horizontal range was 1500M and the source height was 50M. A weapon fission source spectrum was input as a point source in the forward DØT run. For the two adjoint runs, the adjoint source consisting of the Auxier-Snyder neutron dose factors[3] and the Claiborne-Trubey gamma dose factors,[4] were input at heights of 0.5M and 415M and ground ranges of 605M and 485M, respectively; thus keeping the slant range approximately constant. The VIP code[5] provided the integral of the adjoint and forward flux over angle to the SWANLAKE code[6] which calculated the sensitivity of the total neutron and gamma ray dose to changes in the cross-sections as functions of position, energy, and element.

For the two dose detector positions, the calculations showed that a P_1 expansion of the cross-section was adequate to within 1% of that calculated

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with a P_3 expansion. The neutron doses were 1.8×10^{-20} and 4.0×10^{-20} rads tissue per source neutron and the gamma doses were 1.0×10^{-21} and 2.5×10^{-21} rads tissue per source neutron for the detector heights 0.5M and 415M, respectively. Therefore, for this set of problems, the ground acted as an overall absorber. Of the total sensitivity (percent change in the response per percent change in the cross section) of -2.62 and -2.24 for the detector heights 0.5M and 415M, respectively, the nitrogen sensitivities are -2.12 and -1.86. Nitrogen cross-sections in the energy range 2.35 to .00335 MeV accounted for 60% of the total sensitivity for the two cases.

Table 1 shows the percent of the gamma dose produced by thermal (n,γ) capture. For both detector positions, the nitrogen and silicon thermal (n,γ) captures combined to contribute more than 50% of the total gamma dose. Altogether, thermal (n,γ) capture contributes between 73 and 80% of the gamma dose.

The constituent elements used for the ground in this calculation used a water content of 8.6% by weight. Linear perturbation theory predicted that a 1% change in the percent by weight of water would decrease the total dose on the ground by 2.3% and 415 meters above the ground by .8%. The dominant element is hydrogen which accounts for 99% of the reduction.

These calculations show that the total dose is very sensitive to nitrogen neutron cross-sections for detectors both near and far above the ground. For the γ dose, the nitrogen and silicon thermal (n,γ) reactions contribute over 50% to the total gamma dose. Hydrogen accounts for 99% of the reduction in the dose when water content is increased in the ground. Finally, a P_1 expansion of the cross-sections is sufficient to calculate the dose to within 1% of the dose calculated with a P_3 expansion.

Table 1. Percent of Gamma Dose Produced
by Thermal (n,γ) Capture
as a Function of Element

Element	Dose Height = 0.5 Meters Ground Range = 605 Meters	Dose Height = 415 Meters Ground Range = 485 Meters
N(Air)	25.8	29.1
O(Air)	~ 0	~ 0
H(Gr)	14.5	8.6
O(Gr)	~ 0	~ 0
Si(Gr)	26.6	22.5
Al(Gr)	13.9	12.6
Total	80.8	72.8

REFERENCES

1. D. E. Bartine, et al., "ORNL Cross Section Sensitivity Analysis Applications for Radiation Shielding," American Nuclear Society Topical Meeting for Advanced Reactors: Physics, Design, and Economics, Atlanta, Georgia (1974).
2. W. A. Rhoades, F. R. Mynatt, "The DOT III Two-Dimensional Discrete Ordinates Transport Code," ORNL-TM-4280, Oak Ridge National Laboratory (1973).
3. "Protection Against Neutron Radiation," NCRP-38, National Council on Radiation Protection and Measurements, Washington, D.C. (1971).
4. H. C. Claiborne and D. K. Trubey, "Dose Rates in a Slab Phantom from Monoenergetic Gamma Rays," Nucl. Appl. Tech., 8, 450 (1970).
5. R. L. Childs, "VIP, A Computer Program Using Two-Dimensional Discrete Ordinates Transport Calculations for Cross-Section Sensitivity Analysis," UCCND-CSD-1 (to be published).
6. D. E. Bartine, F. R. Mynatt, and E. M. Oblow, "SWANLAKE, A Computer Code Utilizing ANISN Transport Calculations for Sensitivity Analysis," ORNL-TM-3809, Oak Ridge National Laboratory (1973).