MONTE CARLO CALCULATIONS OF THE NEUTRON TRANSMISSION
THROUGH THE ACCESS WAYS OF THE CERN SUPER PROTON SYNCHROTRON

H.G. Vogt
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MONTE CARLO CALCULATIONS OF THE NEUTRON TRANSMISSION
THROUGH THE ACCESS WAYS OF THE CERN SUPER PROTON SYNCHROTRON

H.G. Vogt *)

G E N E V A
1975

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ABSTRACT

Calculations of the neutron attenuation, of interest for radiation-protection purposes, are described for the various access ways (tunnels and shafts for passage of heavy material and personnel as well as cable ducts) of the CERN SPS. The configurations of the duct systems are summarized and the characteristics of the neutron sources (1-line and point geometry) are presented. Source energies from thermal to 15 MeV are taken into account. Different material compositions (concrete with 0-10% water content), the influence of air-scattering, and the effect of traps on the neutron transmission are considered. The Monte Carlo code SAM-CE, for the solution of the energy-dependent and time-dependent neutron and gamma-ray transport equations in complex three-dimensional geometries, and its application are briefly discussed. The results of calculations using this code are presented as graphs showing attenuation curves (dose rates versus distance) for various access ways, energy spectra of neutrons backscattered from concrete slabs having various reflector thickness, and spectral distributions at different depths in the access ways considered.


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1. **INTRODUCTION**

The layout of the access ways to high-energy accelerator installations is determined, apart from operational functions, by the necessities of radiation protection. In general, access ways reduce the shielding mass and provide undesired leakage ways for secondary radiation, especially neutrons. The accelerator operation requires, however, special constructions, which often make the shielding problems rather costly. Accelerators need cable and pipe ducts, and access ways for personnel and equipment such as heavy machine components. The dimensions of the tunnels must at least be sufficient for the installation of such auxiliary equipment as lifts, etc. On the other hand, the attenuation of the particle flux in the ducts must be such that the dose-rates do not exceed the allowable standards for radiation workers in the buildings which house the accelerator's operation equipment. Furthermore, the dose-rate must not exceed acceptable limits near the outlet of the accelerator ducts, which may be located in areas that are accessible to the public. Neutrons are the prevailing radiation component with greater penetration depth as they do not undergo the ionization loss of charged particles from the nuclear cascade. Because of the marked streaming behaviour of neutrons, accelerator ducts are usually designed as labyrinths. Furthermore, obstacles and neutron traps may be necessary, or a neutron absorber is used as coating material for the tunnel walls to strengthen the attenuation effect.

In order to minimize the shielding requirements without loss of safety, much work has been done, in calculations and experiments, to reduce the lengths of the ducts and the number of bends. The duct calculations to be described here use the Monte Carlo method. The configurations of the duct systems and the neutron sources are briefly summarized in the next two sections. Section 4 deals with experiments and methods of calculation for the estimate of the neutron attenuation in ducts. A description of the Monte Carlo code SAM-CE and its application to duct calculations follows in Section 5. The final results are discussed in Section 6, taking into account the dose-rates, energy spectra, and angular distributions of the transmitted neutrons.

2. **GEOMETRY OF THE ACCESS WAYS**

The CERN 300 GeV Proton Synchrotron is located in an underground ring tunnel, of an average diameter of 2.2 km, at a depth between 23 m and 65 m below the surface. There will be six access ways of similar design, each composed of a horizontal connecting tunnel and a vertical shaft running into an auxiliary equipment building at the outlet on the ground surface. Further access pits will be built at the West and North Experimental Areas. The access shafts serve as cable and pipe ducts as well as for air conditioning. They are equipped with a lift, ladders, and working platforms for personnel access (type PP). Some access pits will furthermore be used for the transport of heavy machine components and have consequently bigger dimensions (type PA). The auxiliary equipment in the shafts has light steel structure with metal cladding, just as the construction of the auxiliary buildings has.

As the shapes of tunnels and shafts are mainly determined by requirements of technical engineering, the present calculations were intended to influence not so much the layout of the duct system as the construction of the auxiliary building at the shaft's outlet and potential further attenuation facilities. The dimensions of the access ways considered here are those valid at the time of the calculations. Later modifications have not been included in this study.
Figure 1 shows a typical cross-section of an access pit for magnets and personnel (PA). Starting from the main tunnel with a net internal height of 4.2 m, corresponding to a boring diameter of about 5 m, a horizontal connecting tunnel of rectangular cross-section area ($5 \times 6$ m$^2$) and 30 m length is leading off at an angle of about 50°. It opens directly into a circular cross-section shaft of 9 m diameter. In the forward direction the horizontal tunnel runs into a slightly displaced neutron trap having a length of 12 m. The shaft has a height of 38 m and ends in a concrete building at the ground surface, which is located inside the auxiliary equipment building. The cables and pipes running along the shaft's wall are laid through a special cable duct below the ground floor into the auxiliary building. Thus an opening area of $20 \times 2$ m$^2$ at the front and of $14 \times 2$ m$^2$ at each side of the shaft is to be taken into account for neutron transmission. The geometric configuration in Fig. 1 shows only the simplified duct model, which was taken as a basis for the calculations. Thus all inner equipment and detailed structures have been removed for the sake of simplicity. The rather transparent steel structure of lift and ladders would anyhow give no significant contribution to the neutron attenuation. The walls of the accelerator buildings consist of normal concrete, with the thickness varying between 30 cm and 60 cm. For the duct calculations the walls are assumed to be infinitely thick below the ground whereas the auxiliary buildings on the ground floor have concrete walls up to a thickness of 60 cm.

The six access ways for personnel and machine equipment (PP and PA) are of very similar design. Table 1 gives the variations in the duct dimensions. As can be seen from this survey, the access pit chosen for the present calculations is an example of great transmissibility.

| Table 1 |
| Dimensions of the PA/PP access ways |
| (Centre line distances) |

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<tr>
<td>Length of tunnel</td>
<td>26.86 - 29.60 m</td>
</tr>
<tr>
<td>Height of tunnel</td>
<td>4.65 - 5.50 m</td>
</tr>
<tr>
<td>Width of tunnel</td>
<td>5.30 - 6.10 m</td>
</tr>
<tr>
<td>Radius of tunnel</td>
<td>2.65 - 3.05 m</td>
</tr>
<tr>
<td>Length of trap</td>
<td>10.40 - 12.50 m</td>
</tr>
<tr>
<td>Radius of shaft</td>
<td>2.55 - 4.55 m</td>
</tr>
<tr>
<td>Height of shaft total</td>
<td>28.50 - 67.00 m</td>
</tr>
<tr>
<td>Height of shaft till cable duct</td>
<td>19.90 - 58.40 m</td>
</tr>
<tr>
<td>Height of cable duct</td>
<td>2.0 m</td>
</tr>
<tr>
<td>Length of cable duct</td>
<td>5.50 - 6.00 m</td>
</tr>
<tr>
<td>Width of cable duct</td>
<td>12.00 - 20.90 m</td>
</tr>
</tbody>
</table>
Further duct galleries of interest are the access pits to the North Experimental Area. Figures 2 and 3 give a view of the access type TA 801, which branches off at 90° from the North Target Hall containing target stations and facilities for beam separation and analysis. The hall has a floor area of $18 \times 120 \text{ m}^2$ and a height of 10 m and is located several metres below ground. The connecting tunnel ($4 \times 4 \text{ m}^2$) leads off at the entrance of the beam tunnel into the target hall. After a distance of 30 m it runs into a widened chamber of $7 \times 4 \text{ m}^2$ cross-section, followed by a neutron trap, which has the same tunnel cross-section and a length of 6 m. The rather short vertical shaft of 11 m and a cross-sectional area of $7 \times 9 \text{ m}^2$ is covered by an auxiliary building similar to that already described above.

Another access type is planned to branch off from the target hall some ten metres forward in the beam direction (TA 802). The geometry is illustrated in Fig. 4 as a very simple duct system, consisting of a horizontal tunnel, a vertical shaft, and another horizontal tunnel just below the ground. All ducts have the same cross-sectional area of $3 \times 4 \text{ m}^2$. A neutron trap with a length of 6 m is provided at the end of the first leg. Two variants have been analysed with lengths of 10 m and 30 m for the first leg of the duct system.

The access pit to the Neutrino Target Area is similar to the PA type access. Figure 5 shows the total configuration. The connection tunnel is leading off at 45° from a widened chamber within the main tunnel. Both tunnels have horseshoe cross-sections. The connection tunnel, with a net internal height of 4.65 m, runs into a widened tunnel section followed by a neutron trap. A vertical shaft with a diameter of 9.1 m and a height of 37 m leads to the auxiliary building on the ground floor.

3. DESCRIPTION OF THE NEUTRON SOURCE

3.1 Production

The locations of the neutron sources correspond to beam losses at discrete points as well as broadly smeared parts around the synchrotron ring, where interacting high-energy protons initiate cascades of secondary radiation. The neutrons are one of the most penetrating components of this radiation. Discrete beam losses may occur at bending magnets, beam scrapers, and extraction septa, and at special targets for beam analysis. The discrete beam loss may be expressed in GeV sec$^{-1}$. Other beam losses are assumed to be uniformly distributed around the ring and may consequently be expressed in GeV sec$^{-1}$ cm$^{-1}$.

The beam loss is very uncertain at the different interaction points, especially the uncontrolled loss in the vacuum tube. It may vary by magnitudes, so that the total intensity of the cascade neutrons can only be estimated rather roughly. The determination of the neutron intensity requires information upon intensity and momentum of the protons of the beam as well as knowledge of the geometry and the material composition of the target.

Several studies and measurements of the beam loss have been reported by Goebel and Ranft$^1$ and Charalambus et al.$^2$ using the proton beam of the CERN Proton Synchrotron (PS), and by Routti and Van de Voorde$^3$ at the CERN Intersecting Storage Rings (ISR). Ranft gives an estimate of the beam losses to be expected for wanted and unwanted proton interactions at the components of the 300 GeV machine$^4$. Realistic estimates of the uncontrolled beam losses are of the order of 11-5% of the total proton intensity of $2.5 \times 10^{12}$ protons per
second. In order to get neutron dose-rates, the specified loss values have to be scaled by the neutron production rate per GeV lost. Measurements of the absolute yields of neutrons per interacting proton are reported by Levine et al.\textsuperscript{5)} and Stevenson and Squier\textsuperscript{6)} for point losses of proton beams up to 24 GeV. Corresponding results for distributed beam losses can be derived from calculations for 3 GeV and 200 GeV protons made by Armstrong and Alsmiller\textsuperscript{7}).

The following transmission calculations have been performed for normalized neutron source strengths, so that the attenuation curves may be scaled by the results referred to above. In the present study, one neutron per GeV lost was assumed to be a reasonable approach of the neutron yield\textsuperscript{8)}.

3.2 Geometrical layout

High-energy interacting protons involve a cascade of secondary particles, which is built up in greater depths of the target material. Therefore the neutron source is usually distributed over extended areas. Furthermore, the secondary radiation itself will produce neutrons from interactions in the machine components and the material of the tunnel walls.

An effective line source located on the orbit of the protons should in most cases be a reasonable approach to the real source geometry. The specific source strength is then determined by the total beam loss to be expected, divided by the circumference of the orbit. This assumption may lead, however, to an underestimation of the neutron dose-rates if discrete points of beam loss such as magnets or targets are in a direct line-of-sight with the duct galleries branching off from the main tunnel. The beam loss in such small interaction areas may be much greater than the average loss per unit length of the proton orbit.

For this reason, line sources as well as point sources have been used in the following calculations.

3.3 Angular distribution

The angular distribution of the secondary particles produced by high-energy protons has been measured by Charalambus et al.\textsuperscript{5}) for thin targets. Further investigations by Levine et al.\textsuperscript{5)} confirm a rather flat distribution, at least for large scattering angles and low particle energies. The distribution is, however, forward-peaked even for the thermal neutrons streaming along the machine tunnel. Nevertheless an isotropic distribution of the source neutrons seems an appropriate approach for most low-energy particles emerging from nuclear evaporation.

3.4 Energy spectrum

Several studies at Nimrod and the CERN PS\textsuperscript{9-11)} have shown that neutrons in the energy range below 20 MeV are the dominant radiation component to be taken into account in the calculations of the neutron transmission. Neutrons with higher energies may be neglected if the source cannot be seen from the point of detection in the duct gallery. This effect is due to the ratio of elastic and total cross-section involving a decreasing albedo with increasing energy in this range. Thus high-energy neutrons might give significant contributions to the dose-rate only in the horizontal connection tunnels of the geometries described above. For this reason the following neutron calculations could be performed with methods usually applied to the energy range of reactor neutrons.
High-energy neutrons may, however, contribute to the dose-rate by direct penetration of the bulk shielding weakened by the ducts. This effect has to be studied separately.

The energy spectrum of the source neutrons below 20 MeV used in the duct calculations is roughly known by several experiments and theoretical studies. O'Brien et al.\textsuperscript{12} assumed an energy distribution in the form

$$N(E) = k E^{-x},$$

with $x$ varying between 0.5 and 2. Experimental results for the spectra at shielded and unshielded locations around the CERN PS evaluated from activation detectors, fission counters, and other detectors have been reported by Gilbert et al.\textsuperscript{11}. These measurements with 20 GeV protons confirmed spectra of the shape given above for $x = 1$ within the energy range in question. Further investigations by Goebel and Ramft\textsuperscript{13}, combining Monte Carlo calculations and measurements, provide a good approximation of the $1/E$ spectrum as well.

An analytical solution for the spectrum of the cascade neutrons is given by O'Brien and McLaughlin\textsuperscript{13}, showing good agreement with the $1/E$ distribution. Similar results for the neutron spectra have been received by Armstrong et al.\textsuperscript{7,14} using 3 GeV and 200 GeV protons bombarding iron targets.

On the other hand, several studies have been carried out on neutron attenuation in ducts, based on the existence of a so-called group of 'monitor neutrons' with energies between 2.5 MeV and 4 MeV dominating in the duct transmission\textsuperscript{15,16}. Perret\textsuperscript{16} and Gollon and Awschalom\textsuperscript{17} found out that the attenuation is rather insensitive to the energy of the primary neutrons. For this reason they used in their computations discrete energies for the source neutrons instead of an energy spectrum.

The Monte Carlo calculations to be described here use $1/E$ distributed as well as discrete energy source neutrons and compare the results among each other.

4. NEUTRON ATTENUATION IN DUCTS

4.1 Experiments

Experimental results of the neutron leakage in the ducts of high-energy accelerators are restricted to a few studies. First results have been reported by Shaw\textsuperscript{9} from the 7 GeV proton synchrotron Nimrod. More recent measurements at this accelerator by Stevenson and Squier\textsuperscript{3}, using activation detectors and dosimeters, investigate besides the dose attenuation, the change in the particle spectrum at right-angle bends. The experimental data of Rindi and Tardy-Joubert\textsuperscript{18} as well as those of Schimmerling and Awschalom\textsuperscript{19} have been used to check theoretical concepts of duct transmission. Further investigations at the CERN PS study the dose attenuation in straight tunnels up to a length of 100 m \textsuperscript{11}. Detailed information on the energy spectra of the neutrons is not given by these experimental results.

4.2 Methods of calculation

Many of the methods used in the calculations of the neutron attenuation in ducts and tubes have been developed in reactor shielding\textsuperscript{19}. The application to problems which are connected with high-energy accelerators has turned out to be very useful if certain conditions are taken into account\textsuperscript{20,21}. 
For straight ducts analytic methods give sufficient accuracy. As neutrons can, however, undergo many scatterings without being absorbed or substantially degraded in energy, the introduction of bends is necessary in order to reduce the penetration into the structure. Simple configurations may be analysed by special analytical techniques\textsuperscript{16}). To treat complicated duct geometries, more sophisticated techniques are required in most cases. A rather powerful and flexible technique for the calculation of the neutron transmission is the Monte Carlo method.

There are two types of Monte Carlo techniques which are applied to duct transmission problems. The analog Monte Carlo calculation is a mathematical analog experiment that simulates the most important interactions occurring as the neutrons traverse the duct system. The calculations are based on probability distributions corresponding to the microscopic element cross-sections. In contrast to this, the albedo Monte Carlo method does not follow the neutron's path into the material of the duct walls but substitutes the interactions by reflections at the surface of the walls.

Several analog Monte Carlo codes are available which enable a rather accurate and flexible treatment of the neutron transport\textsuperscript{22,23}). The detailed simulation of the particle flight requires, however, a large amount of computing time in comparison with the albedo concept. On the other hand, the albedo Monte Carlo codes developed by Maerkker and Chain\textsuperscript{24}), and Gervaise and d'Hombres\textsuperscript{25}), suffer from substantial restrictions due to the albedo concept itself or to the layout of the codes.

The albedo concept is based on the assumption that the re-emitting area around the point of incidence on the material surface is small in comparison with the duct dimensions. In this case the re-emission may be interpreted as reflection, described by the material-, energy-, and direction-dependent albedo. Double differential albedos have been measured and are available from Monte Carlo calculations for concrete only\textsuperscript{26-28}), so that only concrete ducts can be investigated. Further limitations concern the reproduction of geometric structures and the treatment of the energy. The multigroup code AMC analyses up to three-legged ducts with rectangular cross-sections. The ZEUS code keeps the neutrons monoenergetic in their scatterings, so that energy spectra are not available.

In spite of these limitations the albedo Monte Carlo codes are rather precious for parametric studies\textsuperscript{15,17}), where analog methods are too slow. The calculations by Alsmliller and Solomito with the AMC\textsuperscript{29}) and by Gallon and Awschalom with the ZEUS code\textsuperscript{17}) are consistent in the main with results of corresponding measurements.

On the other hand, analog Monte Carlo codes can most profitably be applied to complex geometric configurations owing to the more flexible geometry routines of these codes. The monokinetic treatment of the neutrons may be not appropriate in some cases either. Therefore it is useful to subject some realistic structures to an analog analysis, which makes it possible to check albedo techniques and measurement.

The calculations to be described here have been performed with the analog Monte Carlo code S\&M-CE, which has been developed by Cohen et al.\textsuperscript{23}).
5. **MONTE CARLO CODE SAM-CE**

5.1 **Code description**

SAM-CE is a FORTRAN code system for the solution of the energy- and time-dependent neutron and gamma-ray transport equations in complex three-dimensional geometries. It comprises two Monte Carlo codes, SAM-F and SAM-A, and a data processing code SAM-X. The SAM-F code performs the forward neutron and gamma-ray calculation, while SAM-A is designed for adjoint primary or secondary gamma radiation problems. SAM-X provides the element cross-section data from the files of the ENDF/B library.

SAM-F can treat most of the commonly used interaction types, such as isotropic and anisotropic elastic scattering as well as inelastic scattering and absorption. The treatment of the energy during the particle tracking is based on discrete values. All nuclear input data are tabulated in point energy meshes, which are derived by the SAM-X code by means of built-in convergence criteria.

A special feature of the code allows a detailed description of the material composition with many nuclides. For this purpose the whole cross-section range is split into several energy ranges ("bands") to be treated one at a time.

A so-called "Combinatorial Geometry" technique affords extensive geometric capabilities. It subdivides the problem space into regions by the use of seven geometric bodies and their combination in terms of intersections and unions. The provided bodies are as follows: rectangular parallelepiped, box, sphere, right circular cylinder, truncated right-angle cone, right-angle wedge, convex polyhedron.

The source particles may be generated either internally from the input or may be supplied by an external source file. Sources are allowed to be located in several regions, but angular and energy distribution must be the same in all of them. Isotropic as well as monodirectional sources are provided in the original code version. Anisotropic sources must be simulated. The source spectrum can either be read in as an arbitrary step function or taken to be the built-in Cranberg fission spectrum. An external source file may be received by special coding or by use of a neutron interaction file from a previous SAM-F calculation. The creation of an interaction file is a special option of the code, which allows all interactions to be stored in coordinates, direction, energy, weight, etc. Using an interaction file and gamma-ray production data, supplied by SAM-X, sources of secondary gamma radiation can be generated for the following forward or adjoint secondary gamma problem.

The output edit provides fluxes and flux functionals in all those regions of the problem space which are designated in the input as scoring regions. In addition to these volume regions, scoring is possible at particular points in the geometry by use of a special "flux at a point" estimation method. Particles may furthermore be recorded in all their parameters at the surface of arbitrary regions (transmission region), which enables a problem to be run in several steps. This becomes necessary with deep penetration or complex geometric problems.

The code calculates the energy-dependent fluxes in all desired regions. The computation of the average flux value is based on the total distance that all particles travel in the region in question. In addition, flux functionals such as dose, heating, and count rates may
be calculated. Furthermore, the code provides for each region the number of collisions, absorptions, degrades, kills, and births.

In order to get results with reasonable accuracy the code is equipped with an importance sampling technique, which allows one to determine the importance of a particle by arbitrary sets of region, angular, and energy weights.

The SAM-CE system was originally designed for the CDC 6600. Several modifications made it operable on the CDC 7600 at CERN and at the Technical University of Hanover. Apart from alterations arising from the given central core memory capacity, some improvements have been included in the code concerning the flux functionals, the anisotropy of the source, and the geometry checking. An additional routine TTE has been developed for the evaluation of the transmission file information, which is of special interest for the estimation of angular flux distributions.

The calculation time depends upon the geometry being treated, the energy range, and upon the type and the number of detectors. For the duct problems in question the computing times could amount to 30 minutes. The mathematics and coding details for the SAM-CE system are described in the SAM-CE manual[23].

5.2 Application to duct calculations

A complete simulation of the particle tracking through the geometry requires that neutrons are often traced into regions without any significance for the later results. To save the calculation time for these useless particles, the SAM-CE code provides a weighting technique, which allows the problem space to be divided into regions of different importance. A qualified choice of the region weights may emphasize the particle tracking in the important regions, such as the air-filled tunnels and the wall layers near the surface. With greater penetration depth in the tunnel wall the transport neutrons become less important because of the decreasing probability of re-emission into the tunnel, so that they will not contribute at the detection point.

The tunnel walls have therefore been divided radially into regions of decreasing importance with increasing penetration depth. Preliminary studies were made to find out the smallest thickness of the walls, by which the real conditions are simulated without significant particle loss. Furthermore, a favourable subdivision of the tunnel walls and an adequate choice of the region weights was necessary. Calculations have been performed for a simple geometry of a point source in front of a concrete shield. All particles which are re-emitted through the shield surface are analysed in a detection plane which extends parallel to the shield layer. The thickness of the concrete shield was varied between 0.3 m and 3 m. Some results are given in Fig. 6 for monoenergetic 15 MeV neutrons of a half isotropic emitting source. The thick solid line, corresponding to a layer of infinite depth, agrees with the results for a wall thickness of 1 m over the whole energy range. In contrast to this, a thickness of 0.6 m is not sufficient to realize the conditions of infinitely thick walls. The results show particle losses in the lower energy groups, which is not significant for the total dose in this geometry. It may, however, involve a falsification in multi-surface geometries, especially if the neutron spectra are of particular interest.

On the other hand, a wall thickness of more than 1 m involves a useless prolongation of the computing time and greater statistical errors. Figure 7 illustrates the dependence
of the calculation time upon the wall thickness. The ordinate gives the seconds of calculation time necessary to trace one neutron from the source to the detection region. The curve confirms that particles reaching a depth of more than 1 m have very little chance of being re-emitted through the surface and being detected, so that deeper wall regions correspond to lower importance. Figure 7 also shows a subdivision of the tunnel wall with different weight sets, which reduce the calculation time per neutron up to a factor of 1.6. The importance weight set has to be chosen carefully in order to influence the variance but not the results. A subdivision of the tunnel walls into layers of 10 cm : 50 cm : 40 cm with corresponding weights of 1.0 : 0.33 : 0.1 proved to be adequate for the problem in question.

Just as interesting in this context is the radial distribution of the re-emitting neutrons around the point of incidence for an isotropic point source. As can be read from Fig. 8, nearly 90% of the neutrons emerge within an area of 1 m in diameter. This value gives an idea of the lowest duct dimensions to be required by the albedo concept.

Importance sampling can be applied not only to the location but also to the energy of the particles and to their direction of flight. Consequently the 1/E energy distribution of the source neutrons was biased to an almost equal distribution in all groups of the whole energy range. Thus the number of fast neutrons has been enlarged considerably. In addition to this, the emission of source neutrons in the axial direction of the duct has been emphasized by angular biasing with a cos-distribution.

A further variance reduction could be achieved by the energy weighting of the transport neutrons with weights corresponding to the weight standards, attached to the source neutrons. Angular biasing of the transport neutrons was not introduced, owing to the fact that the results were requested at various points along the duct geometry.

The complete calculation could be performed only by a splitting technique, which separates the calculation into several steps, using transmission regions, where the neutrons are recorded in all their parameters. These regions serve as source regions relative to the next calculation step.

The code computes average fluxes per region by the aid of the total distance travelled by all particles in the volume in question. In order to reduce the variance of the results, the detection volumes have been designed as regions of 0.5 m to 1 m in thickness, covering the total duct cross-section. Thus the results are mean values of the flux and flux functionals relative to the duct cross-section.

Dose-rates are obtained from the fluxes in the different energy groups by the flux-to-dose conversion factors of Snyder and Neufeld, which give the dose equivalent at the maximum of the dose-depth curve.

6. RESULTS

The results of the duct calculations are illustrated in Figs. 9 to 22. From the point of view of radiation protection the attenuation curves of the dose-rate, given in Figs. 2: and 22, are most important. The angular distributions as well as the energy and the dose distributions of the neutrons along the duct geometry give, however, profitable details for the qualitative understanding.
Concerning the differential results, the angular distribution of the transport neutrons in the duct system is of special interest. The polar angle distributions, given in Fig. 9 for the PA duct system, have shapes that are characteristic of the neutron extension in ducts. They show increasing forward peaks relative to the axis of the ducts with increasing distance in the legs. The anisotropy is strongest at the end of the legs, while the angular distribution is flatter behind a bend. The ratio of the forward to the backward intensity reaches values up to 25 within small cones.

Figures 10 to 13 show the energy spectra of the neutrons along the duct system for monoenergetic source neutrons, while corresponding results for 1/E distributed source neutrons can be read from Fig. 14. The spectral values give the fluence within each energy bin normalized to one source neutron. Without regard to the statistics the spectra have a rather smooth shape. They show peaks and cuts only in the energy range of the primary and the thermal neutrons.

The thermal neutrons are the dominating flux component. The critical examination must have regard, however, to the special calculation method, which does not allow upscattering in energy but includes the option to track a particle in a thermal energy group, without energy degradation, until absorption. Owing to the lack of a real thermalization calculation the contribution of thermal neutrons to the fluence depends on the choice of the effective energy of the thermal group. For the investigation of this energy, one can start from the assumption that the energy distribution of the thermal neutrons roughly corresponds to a Maxwell distribution, although the true equilibrium is never reached because of neutrons lost by absorption or leakage before their complete thermalization.

The comparison of the mean free paths of the neutrons in concrete (absorption length 1.9 m, collision length 4 cm) and air (absorption length 135 m, collision 25 m) proves that the collision density and the absorption rate is mainly determined by the interactions in the concrete medium. Under conditions of 1/√ν - absorption, Hurwitz et al.31) have shown that a thermal maximum will exist but only in the presence of weak absorption, which means \( \Sigma_a(kT_0)/\xi < 1 \), where \( \xi \) is the average logarithmic energy loss. In the presence of stronger absorption, most neutrons are absorbed before they become thermal. As for concrete, the ratio \( \Sigma_a/\xi \Sigma_s \) is about 0.19, so that the thermal neutrons give a spectrum which is very similar to the Maxwell distribution. The peak corresponds, however, to a higher energy than that of the moderator temperature \( T_0 \). Covey et al.31) obtain for the neutron temperature

\[
T = T_0 \left(1 + 1.82 \frac{\Sigma_a(kT_0)}{\xi \Sigma_s} \right)
\]

\[
T = 1.35 T_0 .
\]

The entire neutron population has therefore to be treated as a one-velocity model at an energy corresponding to a temperature which is about 35% higher than the moderator temperature. Comparable results will be received by replacing the point value cross-sections at the thermal energy by Maxwell-averaged cross-sections. This procedure has been made operable in a later version of the SAM-CE code (Ref. 23, Rev. B).

Using these approaches one should have regard to the fact that the present problem does not concern infinite media as has to be assumed. Furthermore, the shifting of the Maxwell
distribution in energy or the use of average cross-sections are useful for the investigation of reaction rates rather than transport problems. Different intercomparisons with other methods and experiments\textsuperscript{32,33} have shown, however, that the approaches give useful results in neutron transport as well.

The dominance of the thermal neutrons is particularly evident in the energy spectra of Fig. 14. In other energy groups the fluence reaches comparable values only at small distances from the source. The hard components are rather uniformly attenuated with increasing distance from the source and increasing number of bends, thus feeding the lower energy groups. At energies below 100 eV the spectra form an almost horizontal line, which corresponds to the expected $1/E$ energy distribution in the epithermal range. Apart from the thermal neutrons there is no other dominating energy group along the ducts arising from $1/E$ distributed source neutrons. (The unscattered neutrons in the monoenergetic cases can however be distinguished by discrete peaks from the continuous distributions.) In contrast, some preliminary studies\textsuperscript{21} with the Monte Carlo code UNC-SAM\textsuperscript{22} favour the unscattered primary neutrons, such that they dominate in the first leg by an order of magnitude. This follows from the special source geometry used in those calculations, in which a point source is located at the tunnel mouth, emitting particles only into the forward half plane. In the present calculations back-scattering from the walls of the machine tunnel is included. The source geometry considered here is therefore like a superposition of a line source and a plane source. The strong dependence of the energy spectra on the source geometry can be stated, however, only in the first leg, where the high-energy neutrons will undergo the biggest energy losses.

The importance of the different energy groups for the total dose becomes evident if one regards dose values instead of the fluence. Figures 15 to 17 give relative dose distributions of the transport neutrons in the PA duct system arising from different source energy spectra. The distributions are marked by a peak at thermal energies and another one just below the maximum energy. The figures reproduce dose values at different distances from the orbit of the main tunnel. As can be seen very clearly, the high-energy groups give the main contribution to the dose up to some metres within the second leg. The dominating energy groups lie around 1 MeV. With increasing penetration depth, e.g. at the end of the second leg, the maximum of the dose distribution is shifted to lower energies. The contribution of the thermal energy group comes up to nearly 40% in the third leg, as is illustrated in Fig. 17. Monoenergetic source neutrons provide similar results (Figs. 15 and 16).

A comparison of Figs. 10 to 17 suggests that the neutron transmission is rather insensitive to the energy spectra of the source neutrons. Figure 18 compares the neutron dose-rates in the PA duct system for different source energy spectra. The curves are normalized to the dose value at a distance of 7 m from the orbit of the main ring. There is rather good agreement of the curves within the first leg. Only the results corresponding to monoenergetic 3 MeV/15 MeV source neutrons show a smaller slope, because of the relatively great dose portions carried with the extremely forward-peaked high-energy neutrons. On the other hand, the high-energy neutrons lose, at the bend, relatively more in energy than do the low-energy neutrons, which may pass the bend without remarkable energy loss. From this there follows a rather sharp decrease of the dose-rate arising from high-energy source neutrons, while for example the 1 keV curve shows a relative low decrease. The attenuation curve for
1/E distributed source neutrons lies between those for monoenergetic 0.1 MeV and 3 MeV source neutrons over the whole range. As can be verified by interpolation, this curve agrees rather well with the one corresponding to monoenergetic 1 MeV source neutrons. The agreement can be stated in the first and second leg, and should be expected to be more for longer distances, as the energy spectra, though resulting from different source spectra, assimilate with increasing penetration depth.

The small energy dependence of the neutron attenuation proves the equivalence of 1/E distributed source neutrons and monoenergetic source neutrons. The results of the present calculations, providing an effective energy of about 1 MeV, differ slightly from those given by Perret\textsuperscript{16} and de Sereville and Tardy-Joubert\textsuperscript{15}, who use energies of 2.5 MeV and 3-4 MeV, respectively. It has to be considered, however, that the SAM-CE calculations are limited to the energy range from 0.01 eV up to 15 MeV, according to the element cross-section data available from the ENDF/B-III library files.

The influence of variations in the elemental composition of the duct walls on the neutron attenuation is of special interest for the critical examination of the dose-rates at the end of the access ways. Figure 19 illustrates the dependence of the dose attenuation on the water content of the concrete in the PA duct system for 1/E distributed source neutrons. The calculations have been performed for normal concretes with elemental compositions derived from Table 2. The final compositions of the concrete aggregates are derived from

<table>
<thead>
<tr>
<th>Table 2</th>
</tr>
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<tbody>
<tr>
<td>Atomic composition of the concrete used for the calculations (Density 2.3 g cm\textsuperscript{-3})</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Element</th>
<th>Content % by weight</th>
</tr>
</thead>
<tbody>
<tr>
<td>C</td>
<td>4</td>
</tr>
<tr>
<td>O</td>
<td>50</td>
</tr>
<tr>
<td>Al</td>
<td>4.5</td>
</tr>
<tr>
<td>Si</td>
<td>22.5</td>
</tr>
<tr>
<td>Ca</td>
<td>19</td>
</tr>
</tbody>
</table>

this table by addition of different water contents from 1% to 10% by weight, keeping on the total density of 2.3 g cm\textsuperscript{-3}. As can be read from Fig. 19, remarkable differences are to be expected in the absolute values of the dose-rate for water contents in the given range. The dose-rate decreases with the increasing percentage of water, such that the ratio of the dose-rates for 0% and 2.5% water content runs up to a factor of 8 at position A. This dependence becomes weaker, however, towards higher values of water content. The increase from 2.5% to 5% involves a reduction of the dose-rate by a factor of 1.6 only.

Calculations of the neutron transmission through concrete ducts should consequently take into consideration the water content of the wall material rather accurately. Definite
values of the water content of concrete are difficult to obtain. Furthermore, the drying process of the concrete will continue for years, so that the present calculations were standardized on a concrete composition with only 2.5% water content to be on the safe side.

Differences in composition are sometimes large even for normal concretes. Two typical concrete aggregates have been investigated, which differ extremely in their contents of silicon and calcium. There is, however, no remarkable influence on the neutron attenuation, as can be seen in Fig. 20. A variation of the concrete density by about 20% from 2.1 g cm\(^{-3}\) to 2.5 g cm\(^{-3}\) will not cause a significant effect either.

The efficiency of air as absorbing material in ducts is of special interest for comparison purposes because, for example, calculations with the AMC-code do not consider air-scattering. The influence of the air appears in the almost exponential decrease of the dose-rate especially in the second leg of the PA duct system. Figure 20 shows an additional dose curve, which has been calculated for a void duct system. The deviation of the dose curve from that for the corresponding air-filled duct amounts to a factor of 2 at position A.

A further calculation considered the influence of the air humidity on the neutron attenuation. The results, given in Fig. 20, for a relative air humidity of 50% show insignificant differences relative to the comparison values of 0%. The humidity of the air could consequently be neglected in the final calculations.

The final values of the dose attenuation for the geometries in question can be derived from Figs. 21 and 22. The curves give the dose-rate in mrem/h normalized to the total source strength of one source neutron per second as a function of the centre line distance in the ducts. All results refer to 1/E distributed source neutrons isotropically emitted from the source in question into the tunnel system. The air-filled ducts have concrete walls with a density of 2.3 g cm\(^{-3}\) and the elemental composition of Table 2 plus 2.5% water addition.

With respect to the PA duct system (see Fig. 1), the source is represented by a section of the orbit in the main tunnel, which is located in front of the duct mouth for a length of 20 m. The dose-rates in Fig. 21 are therefore related to the specific linear source strength of \(5 \times 10^{-7}\) cm\(^{-3}\) sec\(^{-1}\). Actual values of the dose-rate for a given linear source strength \(s_1\) can be determined by scaling the PA dose values derived from Fig. 19 by \(s_1/5 \times 10^{-4}\).

The PA curve gives dose-rates in three legs for distances of 7 m up to 76 m from the orbit (see Fig. 1). The attenuation within the legs is very similar. Near the bends the calculations did not provide a sufficient number of values, so that the shape could only be indicated by dashed lines. Of special interest is the dose-rate in position A (Fig. 1), where the cable duct runs into the auxiliary building. This value of \(1.1 \times 10^{-1}\) mrem sec/h determines possible further provisions for radiation protection, such that the dose standards for the radiation workers in the buildings are not exceeded.

Also considered was the influence of a neutron trap with a length of 8 m at the end of the first leg. A reaction on the dose-rate in the first leg could not be ascertained. The reduction of the dose-rate in the second leg reaches about 50% at the mouth of the shaft, at a distance of 65 m from the orbit. A prolongation of the trap to a length of 21 m did not bring a further reduction either, going beyond the statistical error of ±25%. These
results differ from calculations reported by Gollon and Awschalom\textsuperscript{17}), who evaluated a factor of 2.5 to 3 for the reduction in the leg following a neutron trap.

A further study concerned the access to the Neutrino Target Area, as shown in Fig. 5. The beam loss was assumed to occur on the beam line about 30 m downstream from the entrance to the duct system. Figure 21 gives the calculated neutron dose-rates at various distances down the centre line of the target hall, connection tunnel, access pit, and cable duct. The curve has a rather smooth and uniform shape, where the effect of bends is hardly obvious. This results from the fact that the connection tunnel branches off at only 45\(^\circ\) and runs into a shaft with a much bigger cross-section.

For the derivation of the actual dose-rate in position A at the entrance to the auxiliary building, the corresponding value of \(1 \times 10^{-16}\) mrem sec/h has to be scaled with the total source strength. Compared to the results for the PA duct system, the attenuation is rather similar but reaches higher orders with the longer distance and the additional bend.

Further calculations have been performed for the access type TA 801, as shown in Figs. 2 and 3. An isotropic point source is displaced by 17 m relative to the entrance of the duct gallery, according to the target geometry. Figure 22 gives the dose-rates at different distances from the tunnel mouth. The attenuation falls off as expected, in a slightly curved slope with a light increase at the end of the neutron trap. In the geometry following the vertical shaft, dose-rates are given only for particular points which do not allow the plotting of a continuous attenuation curve.

The dose-rate at position B in the auxiliary building may be obtained by scaling the value \(1.5 \times 10^{-15}\) mrem sec/h, read from Fig. 22, by the total source strength. An attenuation factor of \(10^{-6}\) is obtained between tunnel mouth and position B, representing the overall attenuation of the duct system.

Another access type investigated is reproduced in Figs. 2 and 4, marked as TA 802. Two variants of this duct system have been examined, distinguished by the length of the horizontal tunnels. A line source of 38 m in length is located in front of the tunnel mouth. The attenuation curves, plotted in Fig. 22, show good agreement until the first bend, where the attenuation for variant 1 falls off deeply. Another break follows from the second bend, which is, however, not so marked. This is due to the small height of the vertical shaft and the lack of a neutron trap. The values of the neutron dose-rate in variant 2 lie, as expected, above those in variant 1, but they approach each other with increasing distance. At the end of the third leg the two attenuation curves fall together again. This effect is to be expected from the small dependence of the neutron attenuation on the particle energy, which involves similar conditions at great distances, if the duct cross-sections and the number of bends correspond in the geometries to be compared. Only near the bends should the differences be obvious. Thus the dose-rates at position A, which in variant 2 is near line-of-sight with the second leg, differ by a factor of 3 in the two variants.

The over-all attenuation of the duct system TA 802-1 from tunnel mouth to position A amounts to \(4 \times 10^{-7}\). For estimation of actual dose-rates, it has to be observed that the TA 802 curves are normalized to the total source strength of 1 sec\(^{-1}\), corresponding to the specific linear source strength of \(2.6 \times 10^{-5}\) cm\(^{-1}\) sec\(^{-1}\). With a given linear source strength \(s_1\), dose-rates follow from that by scaling the values, read from Fig. 22, by the factor \(s_1/2.6 \times 10^{-5}\).
7. **CONCLUSIONS**

In order to get absolute values of the dose-rate at positions near the entrance to the auxiliary buildings, realistic estimates of the beam losses must be available. Table 3 summarizes the final results on the basis of the expected beam losses for the different geometries and a neutron yield of 400 neutrons per interacting proton. In most cases, the dose-rates prove to be within the range of acceptable values for radiation workers. As for the TA 802 ducts, a further reduction may be necessary which can, however, be achieved by simple shielding facilities.

<table>
<thead>
<tr>
<th>Duct system</th>
<th>Source</th>
<th>Beam loss</th>
<th>Neutron generation</th>
<th>Neutron dose-rate at position A/B</th>
</tr>
</thead>
<tbody>
<tr>
<td>PA</td>
<td>line</td>
<td>$2.5 \times 10^7$ p cm$^{-1}$ sec$^{-1}$</td>
<td>$1.0 \times 10^{12}$ n cm$^{-1}$ sec$^{-1}$</td>
<td>0.2 m rem/h</td>
</tr>
<tr>
<td>Neutrino target</td>
<td>point</td>
<td>$2.5 \times 10^{12}$ p sec$^{-1}$</td>
<td>$1.0 \times 10^{15}$ n sec$^{-1}$</td>
<td>0.1 m rem/h</td>
</tr>
<tr>
<td>TA 801</td>
<td>point</td>
<td>$5.0 \times 10^{11}$ p sec$^{-1}$</td>
<td>$2.0 \times 10^{14}$ n sec$^{-1}$</td>
<td>0.3 m rem/h</td>
</tr>
<tr>
<td>TA 802-1</td>
<td>line</td>
<td>$1.65 \times 10^8$ p cm$^{-1}$ sec$^{-1}$</td>
<td>$6.6 \times 10^{10}$ n cm$^{-1}$ sec$^{-1}$</td>
<td>6.5 m rem/h</td>
</tr>
<tr>
<td>TA 802-2</td>
<td>line</td>
<td>$1.65 \times 10^8$ p cm$^{-1}$ sec$^{-1}$</td>
<td>$6.6 \times 10^{10}$ n cm$^{-1}$ sec$^{-1}$</td>
<td>19.0 m rem/h</td>
</tr>
</tbody>
</table>

It is difficult to give reliable information about the accuracy of the results. The statistical errors for the integral dose-rates amount to ±5% at small distances and up to ±25% at the reference points at the end of the duct geometries. The errors for the differential values of the energy spectra reach up to ±35%. Additional inaccuracies follow from the systematic errors of the program and of the element cross-section data, which influence the results in an unpredictable fashion.

The calculations are restricted to the evaluation of the neutron dose attenuation in special duct systems, using reasonable source locations, geometries, and energy spectra. The neutron source strengths, resulting from the different kinds of beam losses, have to be calculated separately. The realistic reproduction of the duct geometry has been made possible by the analog Monte Carlo code SAM-CE. The results confirm, in the main, the estimates of the albedo Monte Carlo codes.

The direct comparison of the two methods is not possible in the present case, as the albedo model codes cannot realize the complex geometries in question. A detailed comparison of the calculation methods and experimental results is, however, planned for simple duct geometries.

Differences between the methods become apparent for the effective energy, which is equivalent to the 1/E spectrum of the source neutrons. Moreover, there seem to be differences in the estimate of the efficiency of neutron traps. One may not always start from the assumption that neutron traps will usually reduce the dose-rate by a factor of about 2 to 3. The present calculations did not verify a factor of more than 1.5. This may depend to a certain degree on the special duct geometry. The importance of neutron traps should anyhow not be overestimated. Their utility may become even questionable considering the
auxiliary equipment in the ducts. If the steel structures for lift and ladders, for example, are located in line-of-sight with the shaft following the neutron trap, the scattering effect of the material may thus prevail against the absorption effect of the trap.

The application of the dose-rates given in Figs. 21 and 22 should take into account that the results represent mean values relative to the duct cross-section, such that higher values of dose-rate might be possible at particular points.

High-energy neutrons of more than 15 MeV could not be taken into consideration within the present calculations. These neutrons will give, however, a substantial contribution only in the first leg of the duct system, being without significant influence in the auxiliary buildings at the end of the ducts.

Further uncertainties follow from the elemental composition of the wall material of the ducts. The neutron attenuation is rather insensitive to variations of the concrete density or changes in the elemental composition of the concrete dry mass. The dose-rate varies strongly, however, with differences in the water (i.e. hydrogen) content. The present calculations have been performed for a concrete of 2.5% water content. Taking account of the crystalline bound water, this value corresponds to a rather low percentage of hydrogen in normal concretes. As can be seen from Fig. 19, the results given are consequently on the safe side.

The dose-rates at the shaft's outlet and at position A give the basis for the layout of the walls of the buildings at the ground level. The energy spectra of the neutrons, which are required for this estimate as well, may be taken from Fig. 14, which shows rather representative results.

Acknowledgements

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Fig. 1 Access pit for magnets and personnel PA
Fig. 2 Access tunnels to North Target Hall

Fig. 3 Access to North Target Hall TA 801
Fig. 4 Access to Neutrino Target Area
Fig. 5 Access to North Target Hall TA 802
**Fig. 6** Energy spectra of neutrons backscattered from concrete slabs

**Fig. 7** Mean value of computation time for the transport of one neutron from the source to the detection region
Fig. 13 Spectral neutron flux density in access pit for magnets and personnel PA, 1 keV source neutrons
Fig. 14 Spectral neutron flux density in access pit for magnets and personnel PA, $E^{-1}$ source spectrum
Fig. 15  Relative dose distribution in PA duct system for 15 MeV source neutrons
Fig. 16 Relative dose distribution in PA duct system for 0.1 MeV source neutrons
Fig. 17 Relative dose distribution in PA duct system for 1/E source spectrum

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Fig. 18 Dose rate in PA duct system for different source energies
Fig. 19 Dose attenuation in PA duct system for different water contents of the concrete walls, $E^{-1}$ source spectrum
Fig. 20 Dependence of the dose attenuation in ducts on composition of wall material and air, PA duct system, E⁻¹ source spectrum
Fig. 21: Dose rate in access ways PA and NTA.
Fig. 22 Dose rate in access ways TA801 and TA802