SNS Inner Plug Shipping Cask Analysis

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Abstract- ***Calculations were performed to evaluate the dose rates outside the shipping cask containing the Spallation Neutron Source (SNS) inner plug assembly. The analysis consisted of simulating the proton beam interaction with the SNS target, activation calculations with the determined neutron flux levels and assumed SNS operation schedule, and calculation of the decay gamma-rays propagation through the inner plug and shipping*** cask. Several ***materials were consideredfor the inner plug. The results provide guidance for thefinalization of the plug design.***

I. INTRODUCTION

The Spallation Neutron Source (SNS), currently under construction at Oak Ridge National Laboratory, will use a 1-GeV proton beam of 2 MW nominal power to produce neutrons in a mercury target. On the average, approximately 33 neutrons per incident proton are released, resulting in a total source of -4 x 10^{17} n/s. Only a small fraction of these neutrons is successfully directed to the beam-lines and used for experiments. The majority of neutrons slow down and eventually get absorbed in the target, reflector, and shielding, causing intense radiation damage and activation of certain structures. Radiation damage limits the lifetime of some components, such as the steel container of the target and the target inner plug, which will be replaced at regular maintenance intervals during the SNS lifetime. These components will be shipped off-site and it is therefore necessary to assure that the dose rates outside the shipping cask will not exceed the regulatory limits. This paper presents an analysis for the SNS inner plug assembly.

The SNS target model is shown in Figs. 1 and 2. The SNS inner plug assembly consists of a cylindrical structure with a diameter of -90 cm and a height of -120 cm, which houses the mercury target and the moderators. The main materials of the plug are beryllium in the inner zones close to the moderators, and lead in the outer zones. The container and the liners of the numerous cavities in the plug are made of steel. It is planned that the plug will be replaced approximately every three years of SNS operation. Before removal, the mercury will be drained, the body of the target will be removed, and the liquids in the moderators will be drained. It is planned that the rest of the structure, including the moderator vessels, will **be** shipped offsite as an integral unit.

The commercial shipping cask **"CNS–120B"**, which is intended to be used for the inner plug transport, was designed by Chem-Nuclear Systems, L.L.C., (CoC USA/9168/B(U)). The cask has an outer diameter of -188 cm, is -224 cm tall, weighs -33500 kg, and has a payload capacity of -6600 kg. The cylindrical wall consists of a 1.91-cm-thick layer of stainless steel, followed by 8.51 cm of lead, and 3.81 cm of stainless steel. The cask lid and bottom are comprised of 16.51-cm-thick stainless steel. The model of the plug inside the shipping cask is shown in Fig. 3.

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* Managed by UT-Battelle, LLC, for the U. S. Department of Energy under contract DE-AC05-00OR22725.

II. THE ANALYSIS

The analysis consisted of simulating the proton beam interaction with the SNS target, activation calculations with the determined neutron flux levels, spallation production rates, and assumed SNS operation schedule, and calculation of the decay gamma-rays propagation through the inner plug and shipping cask.

II.A. SNS Target Calculations

The proton beam interaction with the target was modeled with the MCNPX code.' The target model was developed for detailed neutronics studies of the target.[2] The central part of the model, which is of interest here, is shown in Figs. 1 and 2. The model also included a proton beam window, not shown in the figures, and a detailed description of the proton beam profile in front of the window.[3] A separate calculation was performed to obtain the cell volumes not otherwise determined by the code. In a target calculation, neutron fluxes averaged over cells were tallied for each of the cells of interest. These fluxes were tallied in 112 energy bins with an upper energy of 20 MeV. The default physics module options were used with pre-equilibrium model after intranuclear cascade. Information about proton interactions and neutron interactions above 20 MeV was written to a HISTP tape.

II.B. Activation Calculations

The objective of this step was to determine the intensity and spectrum of decay gamma rays emitted by the radionuclides generated during the SNS operation. The Activation Analysis System (AAS) was used for all activation calculations.[4] AAS is a modular system of UNIX C-shell scripts, Fortran codes, and data libraries that automates the activation calculations for the systems analyzed with the MCNPX code. AAS minimizes the user's overall effort and input preparation, improves reliability, and greatly reduces the time necessary to complete the calculations. The flow chart of the AAS is shown in Fig. 4. Starting with the list of MCNPX cells, the ASS extracts the material information from the MCNPX input, determines isotopic compositions and calculates necessary number densities. Next, it extracts neutron cross-sections from the FENDL library and processes them into the energy group structure employed in the MCNPX flux tallies. The neutron fluxes are extracted from the MCNPX output. The HTAPE3X code is run to obtain the residual nuclide and gas production from proton and high-energy neutron reactions. The neutron fluxes are folded with cross-sections to obtain the low-energy (below 20 MeV) nuclide production rates, which are then combined with the high-energy production rates and normalized to the specified proton beam current. Next, for the user-specified sequence of beam-up and beam-down times, the buildup and decay of the radionuclides is calculated with the OriHet95 code,' and gamma-ray decay spectra are determined for selected times. The final (optional) step is to merge the activation results from all the cells considered to obtain the total radionuclide inventories, radioactivity levels, decay heat releases, etc.

For the SNS inner plug, 75 non-void cells were processed. The "realistic" SNS operation was modeled assuming -52 days at full power followed by 7 days shutdown. This sequence was repeated four times and was followed by the shutdown until the end of the year (-129 days). Three years of such operation were followed, by a final decay time of up to 3 years. The total gamma-ray source as a function of time is shown in Fig. 5. The calculations to obtain the dose rates outside the shipping cask were performed using the gamma-ray sources at 90 days after the last beam shutdown.

II.C. Gamma-Ray Transport Calculations

For the calculation of dose rates outside the cask, the MCNPX model shown in Fig. 3 was used. All the details of the inner plug were preserved in the model; however, the target body was removed and the liquids were drained. The gamma-ray sources for each cell were obtained from the activation calculations. It was assumed that within each cell the sources were uniformly distributed in space; however, the correct energy distribution for each cell was preserved. The first MCNPX calculation produced a surface source file at the inside surface of the cask. The penetration of the gamma rays through the cask walls was then treated with two MCNPX calculations: one for the cylindrical shell and the other 'for the top and bottom lid. Geometry splitting with Russian roulette was used as a variance reduction technique. The cask shell and lids were divided into several cells and importances were gradually increased from the inside towards the outside.

Cylindrical mesh tallies were used to tally the dose rate distribution on the cylindrical shell of the cask. The mesh grid covering the cylindrical wall of the cask consisted of 15 axial segments covering the cask height (which is -2 meters), 37 azimuthal segments (35 x 10 degrees and 2x5 degrees), and one radial segment 1-cm-thick. On the top and bottom a mesh grid of ten concentric rings with equal areas was used to tally the dose rates.

II.D. Deterministic Transport Calculations

In the preparation phase a simplified analysis — a two-dimensional R-Z discrete-ordinates calculation — that would use spatially homogenized materials and sources in the inner plug, was considered. This approach was abandoned *in* favor of the more detailed analysis described above. However, some deterministic calculations were nevertheless performed. These calculations started at the inner capsule. wall using source information from the surface-crossing file created by MCNPX. This procedure thus avoided the need to treat the geometric complexities of the inner plug in the deterministic calculations.

The MTD[6] coupling code processed the surface-crossing file into a boundary source file for the discrete ordinates transport code DORT.[7] Since a DORT R-Z model requires azimuthal symmetry, the DORT boundary source can not have azimuthal variations. The maximum rather than average gamma flux values were of interest; therefore, the DORT source was prepared from the azimuthal segment(s) around the location where the MCNPX calculations indicated the dose rate maximum. In the Z-direction the source distribution corresponding to the selected azimuthal segment was retained.

III. RESULTS

The calculations showed a significantly uneven distribution of gamma fluxes and dose rates on the outside of the capsule cylindrical shell. In the vertical direction, the maximum fluxes and doses are at the elevation of the plug mid-plane (i.e. at Z=0 cm), and they drop sharply above and below mid-plane, as illustrated in Fig. 6. The azimuthal distributions shown in Fig. 7, also display pronounced structure, which is most significant for the two axial segments closest to the plug mid-plane. The highest dose rates (at an azimuth of -90 degrees) are at the locations facing the back-end of the target cavity in the plug, which corresponds to the right side of the cask as shown in Fig.3. The lower peak of the dose distribution (at an azimuth of -270 degrees) corresponds to the location facing the proton-beam channel shown at the left in Fig. 3. The proton beam and target cavities provide streaming paths for the gamma rays from the inside of the plug, which are otherwise significantly attenuated in the outer lead regions of the plug. Correct modeling of the inner plug is therefore critical for prediction of maximum dose rates.

Dose rate distributions just outside the shipping cask lid and bottom as well as at 1m outside these surfaces are shown in Fig. 8. As expected, the dose rates decrease from the center of the capsule lid or bottom towards the outer edge, and the distributions are flatter further away from the cask. Note that the dose rates are averages for the concentric ring cells (described in section II.C. above); therefore, refining the tally mesh grid would likely lead to higher dose rates near the center.

As mentioned above, some calculations were also done with the hybrid method which used a MCNPX calculation to create a boundary source file, the MTD code to process this file into a DORT boundary source file, and finally the DORT calculation for the penetration through the cask. A comparison of axial distributions of the total gamma flux at the cask outer wall obtained with MCNPX and DORT is shown in Fig. 9. The MCNPX results are for the azimuthal segment, which spans 10 degrees around the location of the second peak, corresponding to the proton beam channel. Results from two DORT calculations are shown in Fig. 9. For one calculation the source was prepared using a lo-degree azimuthal segment from the boundary-crossing file, while for the other calculation a 20-degree segment was used. At the maximum, the gamma flux from DORT agrees with the MCNPX result to within -20%. Given the large attenuation through the cask wall — a dose rate attenuation factor of -2800 — this agreement can be considered reasonable.

So far, all the discussion was limited to the inner plug design which used beryllium, lead (with -12.5 vol% of heavy water for cooling), and stainless-steel liners. Several modified designs were alsq considered in which the geometry of the plug and the beryllium zones was unchanged, but other materials were used instead of lead or steel. In the first modification, steel was replaced with aluminum with the intent to reduce dose rates outside the cask and improve the neutronic performance of the system For the other modifications, in which nickel and stainless steel were substituted for lead, the motivation was to replace the lead with material having a higher melting point, thus avoiding the potential danger that high heating rates in combination with poor thermal contact would cause melting in the lead reflector. The dose rates outside the cask lid and bottom for these different plug configurations are compared in Fig. 10. Starting from the "basic" inner plug configuration, labeled "Lead + SS" in Fig. 10, replacing steel liners with Al (curves labeled "Lead + Al ") reduces the maximum dose rate by almost a factor of 2, and gets the dose rates below the regulatory limit of 1 rem/hour[8]. Replacing lead with stainless steel (steel SS-316 was used, with no heavy water) increases dose rates by about. a factor of 2 (curves labeled "SS316"). Finally, substituting Ni for the lead results in more than a factor of five higher doses (curves labeled "Ni"), which clearly makes Ni the least favorable of the materials considered. Since replacing lead in the reflector with stainless steel also has only a marginally negative effect on the neutronic performance of the target,' this appears to be the most probable configuration for the initial SNS inner plug assembly. The activation and shipping cask analysis will be repeated when the inner plug configuration is finalized. However, it appears probable that some additional in-cask shielding will be necessary to meet the regulatory limits.

IV.CONCLUSION

Detailed calculations of the dose rates outside the shipping cask containing the SNS inner plug assembly were performed. The analysis required an intricate sequence of calculations which included simulating the proton beam interaction with the SNS target, activation calculations with the determined neutron flux levels and assumed SNS operation schedule, and calculation of the decay gamma-rays propagation through the inner plug and shipping cask. To a great extent such a detailed analysis was possible due to the use of the Activation Analysis System for the activation calculations, which would be otherwise very time consuming and potentially prone to user' error.

The inner plug structure was found to have important effects on the distribution and maximum dose rate values outside the cask. Dose rates were determined for the radionuclide inventory after three years of SNS operation and 90 days of cool-down after the last beam shutdown. The lowest dose rates were obtained for the inner plug with lead outer zones and aluminum structure. The use of nickel for the plug outer zones should be avoided. For the plug with lead and a steel structure, as well as for the plug with steel outer zones and structure, additional shielding will likely be needed inside the CNS8-12B cask.

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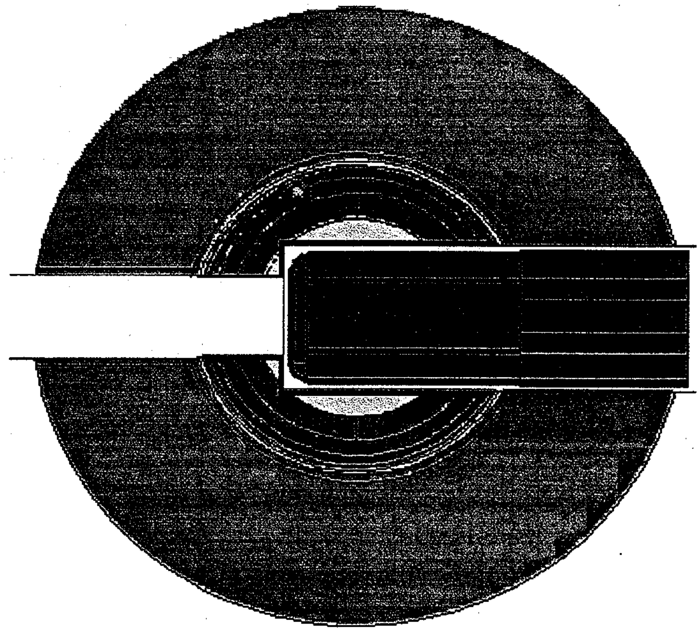

Fig. 1. Horizontal cross-section of the SNS target model. The inner plug consists of beryllium (light blue) and lead (purple) cylindrical regions surrounding the rectangular area of the target body.

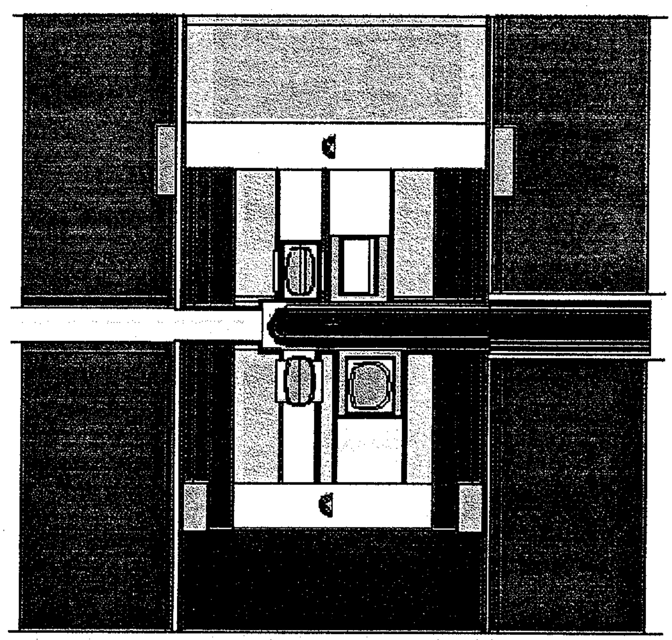

Fig. 2. Vertical cross-section through the SNS target model. The proton beam channel enters from the left (white region in the mid-plane). Also shown are the moderators, the mercury target (purple body between the moderators),, and the outer plug (blue regions).

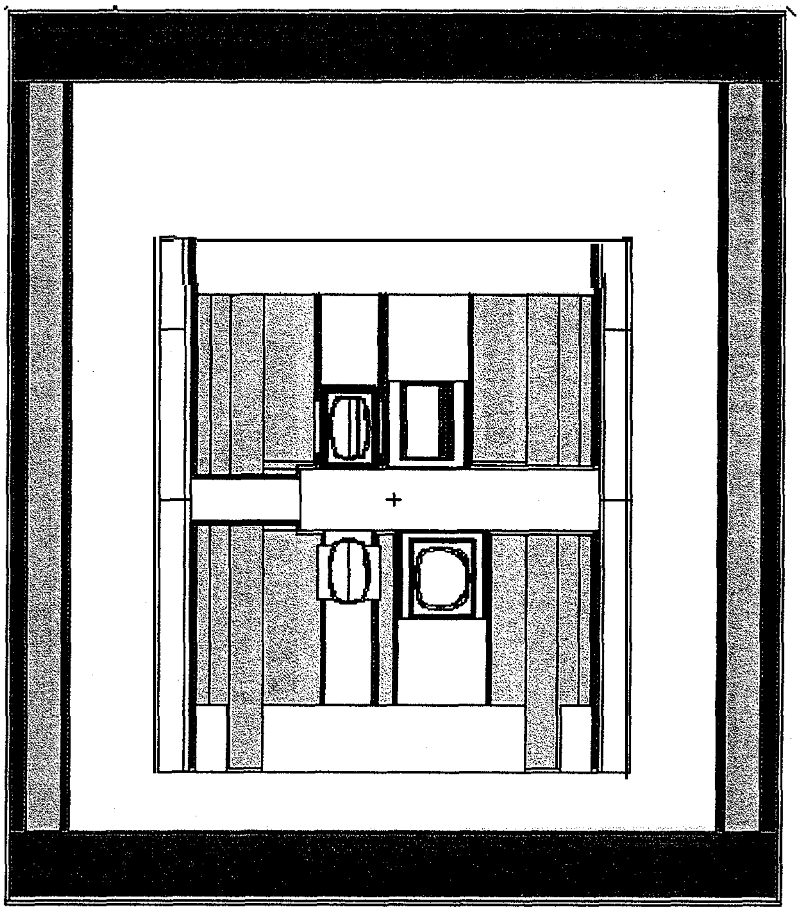

Fig. 3. Vertical cross-section of the model of the SNS inner plug assembly inside the shipping cask.

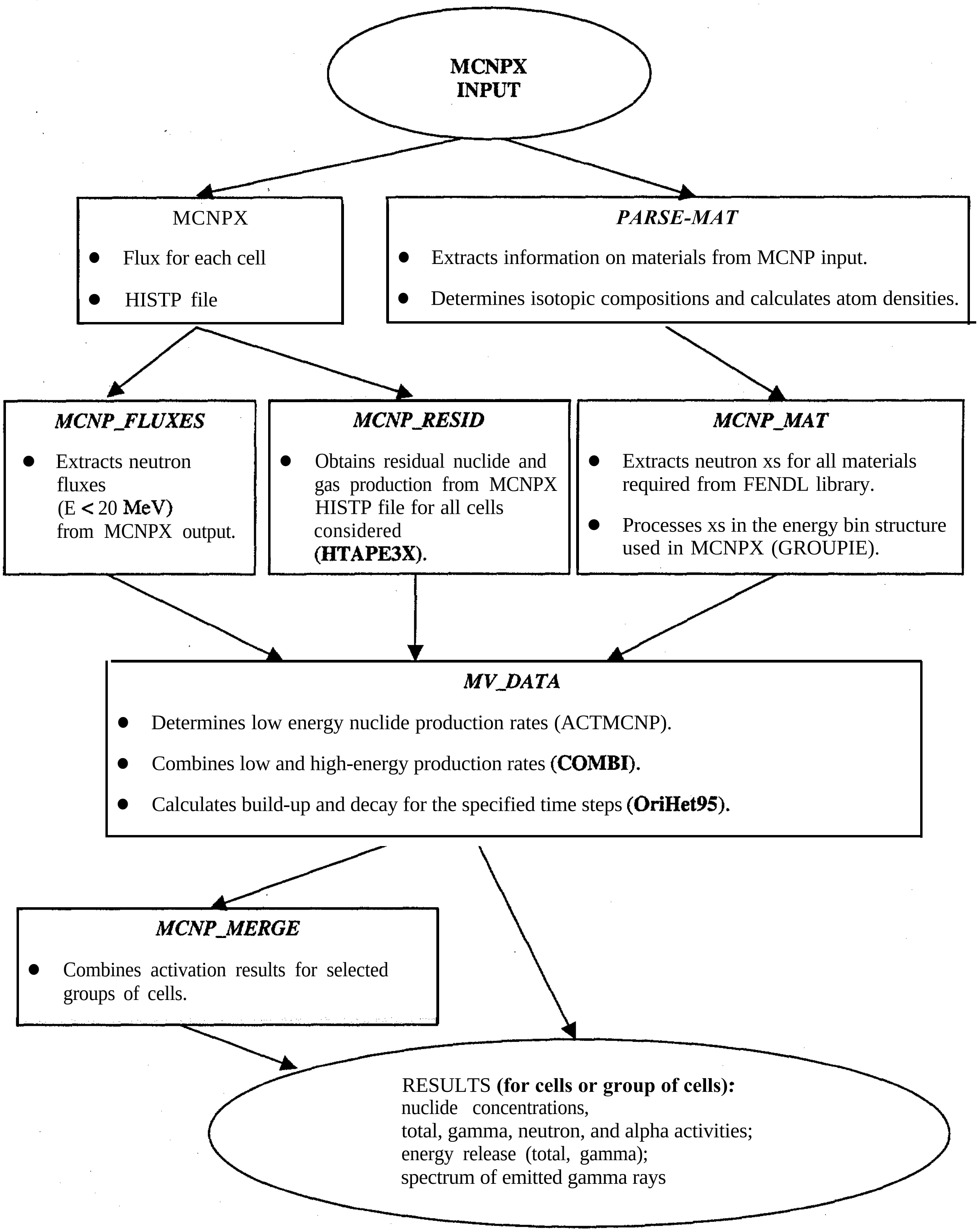

Fig. 4. The Activation Analysis System.

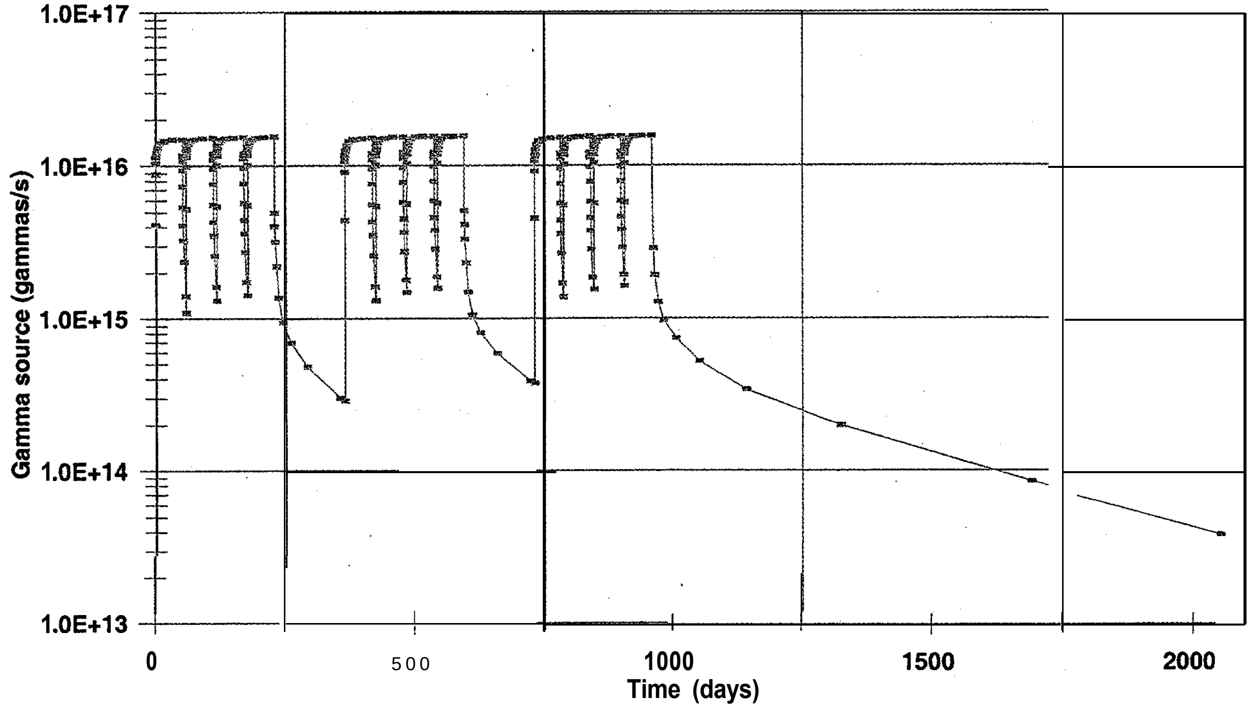

Fig. 5. Gamma source as a function of time for the lead-beryllium SNS inner plug.

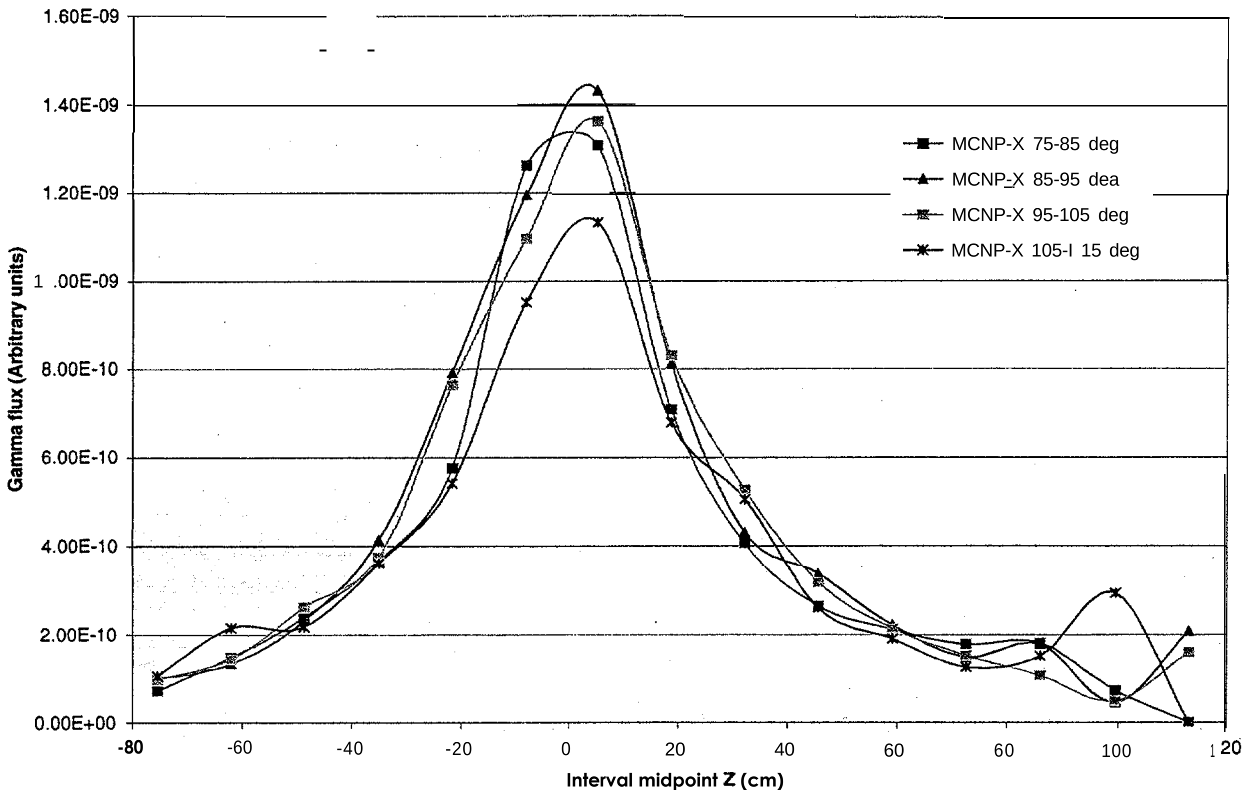

Fig. 6 Variation of gamma flux in the vertical direction at the outer surface of the shipping cask. Total gamma flux is shown for four azimuthal segments around the location of the maximum (at ~ 90 degrees).

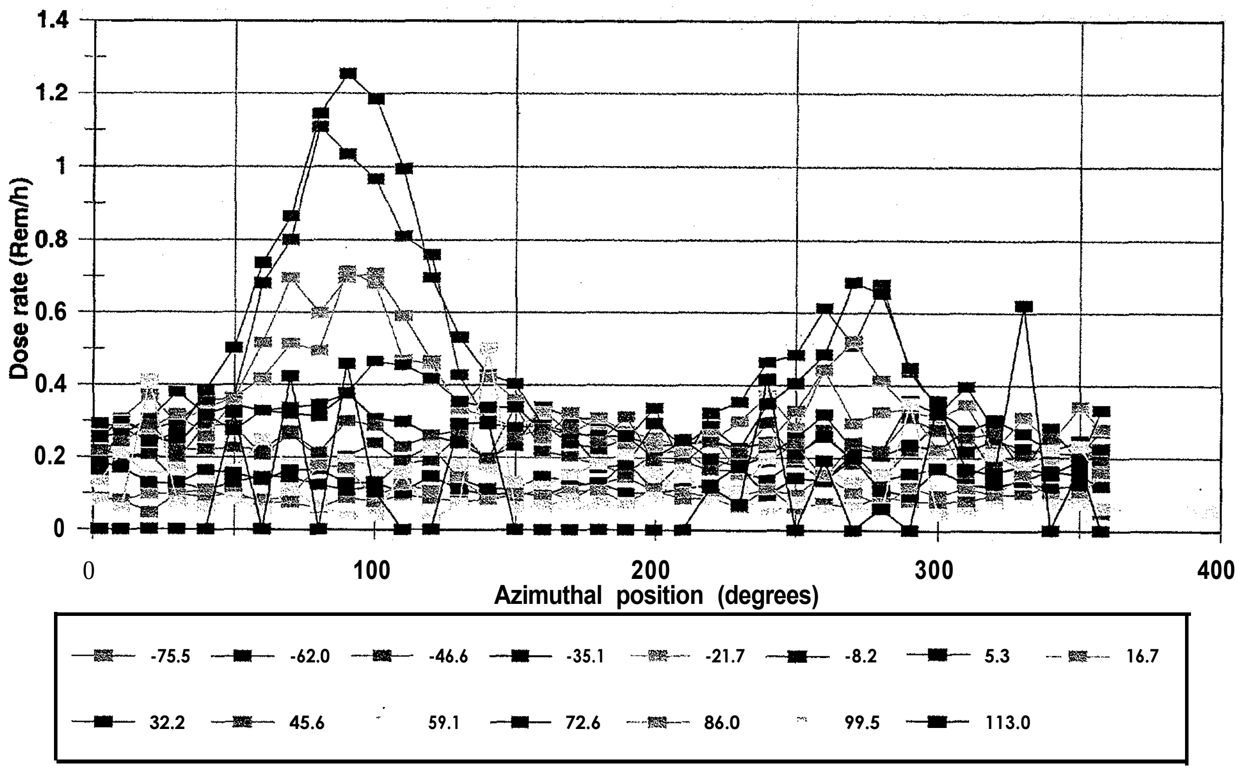

Fig. 7. Distribution of the dose rate on the otside surface of the shipping cask. Each curve corresponds to one axial segment, with the Z coordinate of the center of the segment given in the legend (in centimeters). The plug mid-plane is in the Z=O plane.

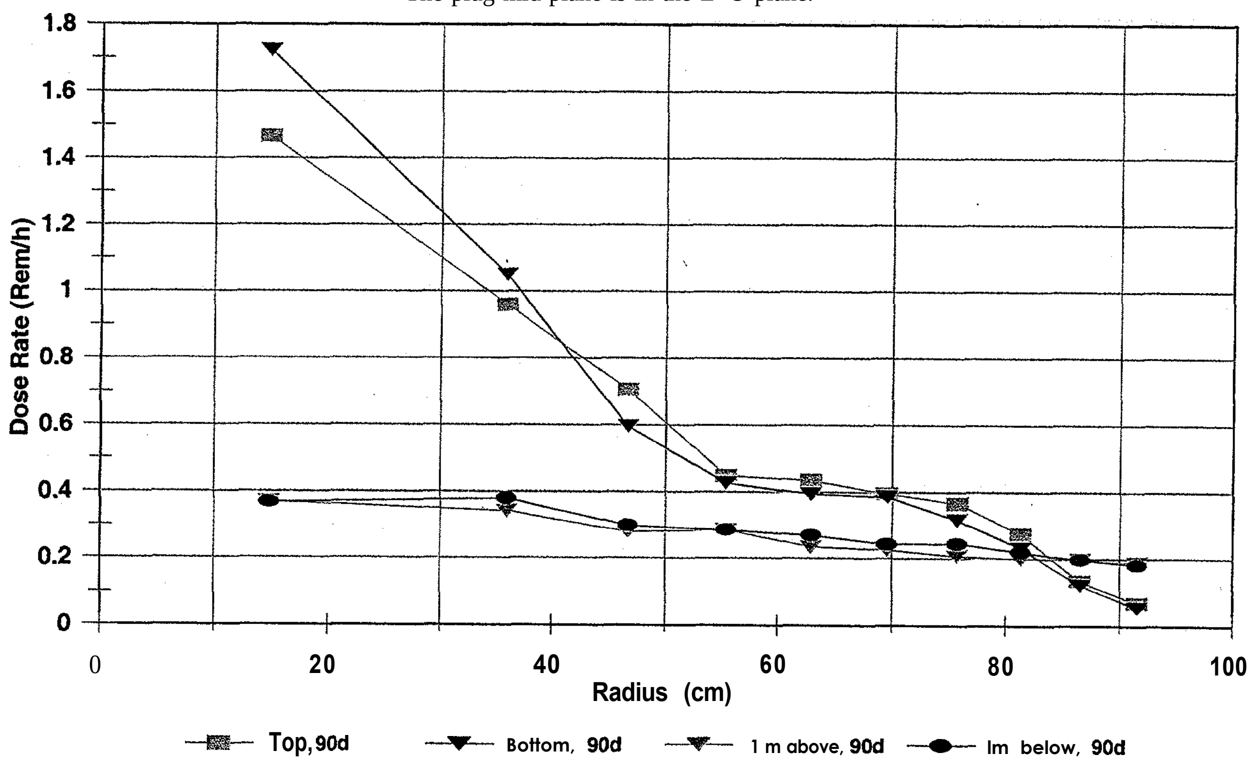

Fig. 8. Dose rate distributions at the shipping cask top lid, bottom lid, and at the distance of 1m above the top and below the bottom.

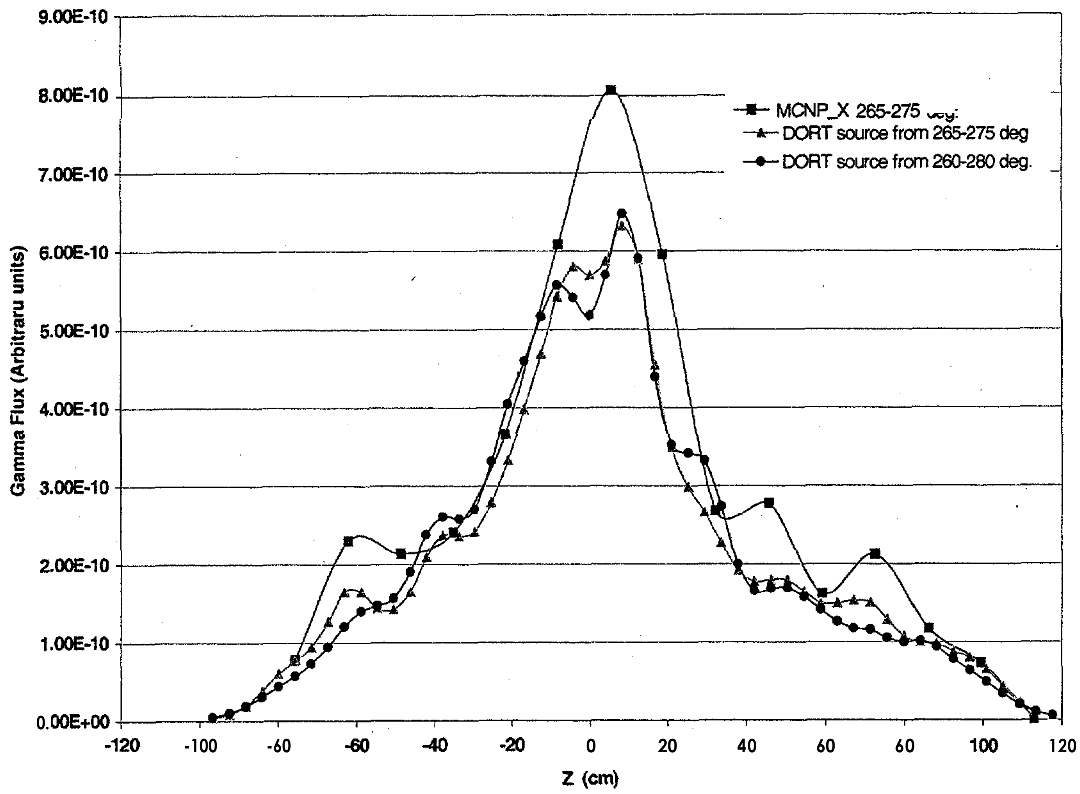

Fig. 9. Axial distribution of gamma fluxes obtained with MCNPX and DORT code.

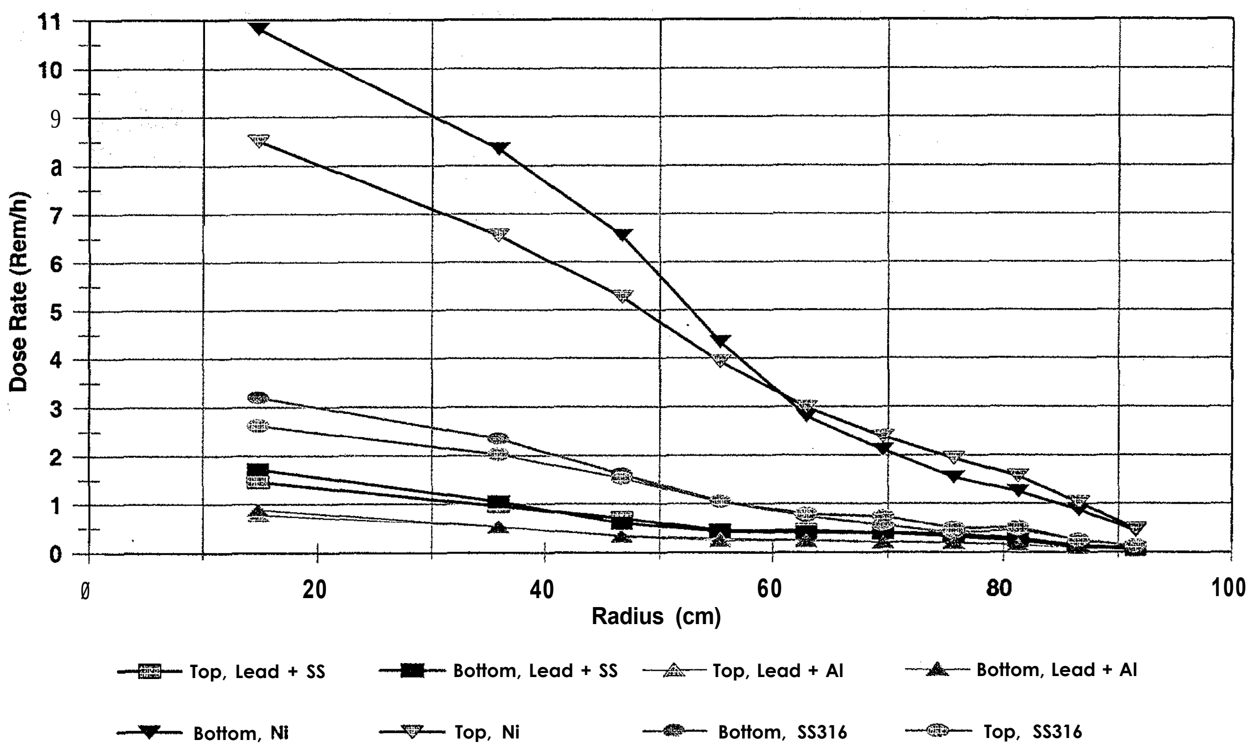

Fig. 10. Comparison of dose rates outside the cask lid and bottom for different inner plug material compositions.