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D–T Neutron Source Shielding for the Sudbury Neutrino Observatory

by

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Abstract

The Sudbury Neutrino Observatory (SNO) Project requires a D–T neutron source for the activation of ^{16}O gas. The radioactive ^{16}N is used for detector calibration. The D–T neutron generator requires radiation shielding. This report provides details of the shielding analysis performed in support of the shielding design for the SNO neutron source.

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1 Introduction

The Sudbury Neutrino Observatory (SNO) is being built in a hard rock mine near Sudbury, Ontario. The designers of the facility plan to use a deuterium-tritium (D–T) neutron generator to produce radioactive ^{16}N gas via the high energy reaction $^{16}O(n,p)^{16}N$. The ^{16}N will be used in detector calibration. The D–T generator produces 14 MeV neutrons which are emitted isotropically at a maximum rate of 10^8 neutrons per second.

The generator will be placed in a hole blasted out of the floor of an underground chamber and subsequently backfilled with sufficient material to provide shielding against the neutron and secondary photon radiation. A duct penetrating the shielding is required to allow access to the generator. This duct provides a streaming path from the source to the exterior of the shielding which must be dealt with. The purpose of this work was the analysis of the shielding for facility design and licensing support [1].

2 Scoping Calculations

2.1 Discrete Ordinates Methodology

The initial scoping calculations were performed with the one-dimensional discrete ordinates transport code ONEDANT [2] using spherical geometry, S_{16} angular quadrature, P_3 anisotropic scattering, and the BUGLE80 [3] 67-energy-group coupled neutron-photon shielding cross-section library.

The generator was modelled as a point neutron source. Since the discrete energy of the source (14 MeV) lies midway between the mean energies of the top two groups in the BUGLE80 library (15.27 and 13.05 MeV [3]), neither group will accurately model the transport of 14-MeV neutrons by itself. Therefore, the source was split evenly between the top two energy groups which should slightly, but conservatively, under estimate the attenuation. The calculations were performed for water, ordinary concrete, and norite rock, ignoring the access duct. The material compositions are listed in Table 1. ANSI/ANS-6.1.1-1991 anterior/posterior flux-to-dose rate conversion factors were used [4].

2.2 Results

The results of the scoping calculations are illustrated in Figure 1. The thickness of each material studied that is required to achieve a total dose rate of 0.1 mrem/h is: 140 cm of water, 136 cm of ordinary concrete, and 158 cm of norite rock. Ordinary concrete is better than the rock due to the hydrogen content (superior neutron attenuation), and is better

Table 1: Material Compositions

Material	Water	Ordinary Concrete	Norite Rock	Stainless Steel
g/cc	1.0	2.34	2.87	7.9
Element	Atom Density (at/(b·cm))			
H	6.675×10^{-2}	7.767×10^{-3}		
O	3.338×10^{-2}	4.385×10^{-2}	4.861×10^{-2}	
Na		1.048×10^{-3}	2.255×10^{-3}	
Mg		1.487×10^{-4}	2.133×10^{-3}	
Al		2.388×10^{-3}	5.765×10^{-3}	
Si		1.580×10^{-2}	1.600×10^{-2}	
S†		5.635×10^{-5}		
K		6.931×10^{-4}	8.841×10^{-4}	
Ca		2.915×10^{-3}	2.156×10^{-3}	
Cr				1.746×10^{-2}
Mn				1.732×10^{-3}
Fe		3.127×10^{-4}	1.857×10^{-3}	5.828×10^{-2}
Ni				8.511×10^{-3}

† Sulfur is only available in the MCNP library.

than water due to the higher density (superior secondary photon attenuation). Concrete was therefore chosen for further study.

3 Detailed Calculations

3.1 Basic Design

The basic design involved a 4-ft (122-cm) diameter, 7.5-ft (229-cm) high, vertical concrete cylinder containing a co-axial 10-in (25.4-cm) diameter hole. The neutron generator was placed at the bottom of the hole approximately 6.4 ft (194 cm) below the top surface. A 4-in (10-cm) diameter horizontal pipe allows access to the vertical hole. A concrete cap covers both the vertical hole and the horizontal pipe. The distance from the neutron source to (roughly) ground level is 165 cm, and from the source to the base of the cap is 194 cm. The generator tube and gas target chamber were modeled as 0.25-in (0.64-cm) thick stainless steel pipes.

Using $1/r^2$ attenuation from a point source, yields a dose rate of about

$$10^8\frac{\mathrm{n}}{\mathrm{s}} \times \frac{1}{4\pi 194^2}\frac{1}{\mathrm{cm}^2} \times 0.17\frac{\mathrm{mrem}\cdot\mathrm{cm}^2\cdot\mathrm{s}}{\mathrm{h}\cdot\mathrm{n}} \approx 36\ \mathrm{mrem/h}$$

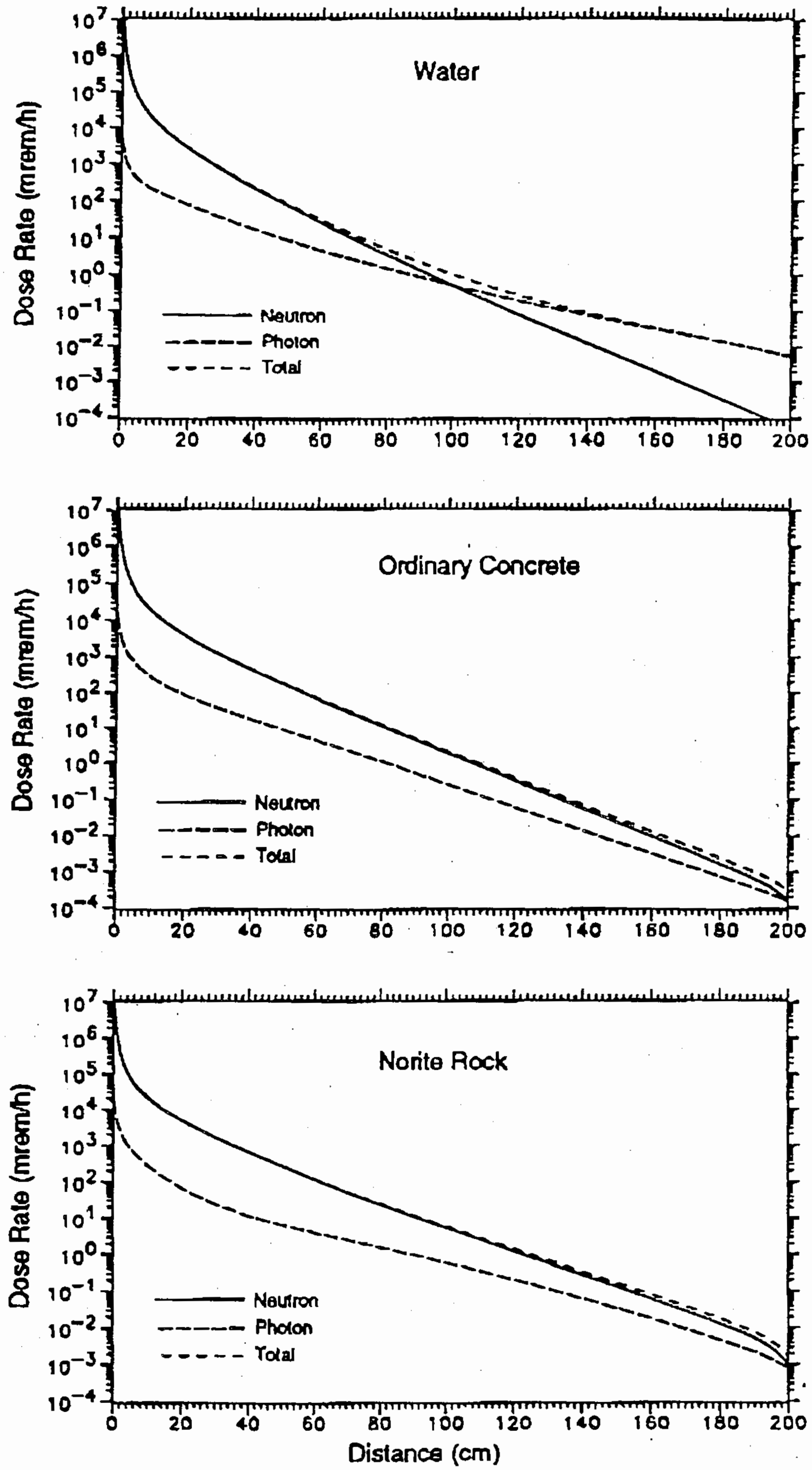

Figure 1: One-dimensional spherical ONEDANT scoping calculations. Dose rates from 14 MeV neutrons in water, ordinary concrete, and norite rock.

striking the underside of the cap. Then applying the slope of the concrete curve in Figure 1 to obtain the desired dose rate of 0.1 mrem/h, yields a cap thickness of roughly 68 cm. The MCNP model of the cap was given a height of 70 cm and a diameter of 100 cm.

3.2 Monte Carlo Methodology

The Monte Carlo calculations were performed with the continuous energy, generalized geometry transport code MCNP [5] using the RMCCS neutron library [6] and the MCPLIB photon library [7]. The upper part of the model used for sizing the cap is shown in Figure 2 and the lower part containing the source is shown in Figure 3. The upper part of the model was modified to determine the streaming through the horizontal access port as shown in Figure 4.

The only variance reduction techniques used were:

1. *Forced Collisions.* Particle collisions are forced to occur, with appropriate weight adjustment, in the D–T generator, gas target chamber walls, and within the first two cm of all inner concrete surfaces. This technique improves the statistics of particles streaming up the vertical hole.

2. *Splitting and Russian Roulette.* This classic technique for improving penetration of thick shields was used through the shield cap.

3. *Point Detector.* This is a deterministic flux estimator at a point of interest from collision events throughout the geometry. When used with forced collisions, point detectors can greatly improve the statistics in *one-bend* streaming situations.

4. *Model Extent Reduction.* Regions at the edge of the geometry, deep in the concrete, far from the point of interest that are extremely unlikely to contribute to the effect of interest were removed from the model. This improves calculational efficiency by reducing the time wasted following unimportant particles.

Calculation of dose rates in the cap was performed separately from the streaming through the horizontal access port, with each case optimized for the particular situation. For example, the horizontal pipe was removed from the cap calculations, and the height of the cap was reduced in the streaming model. Both models used the same lower portion which provided the source of the radiation, and both models had a reduced diameter for the main concrete shield of 60 cm below ground and 100 cm above ground.

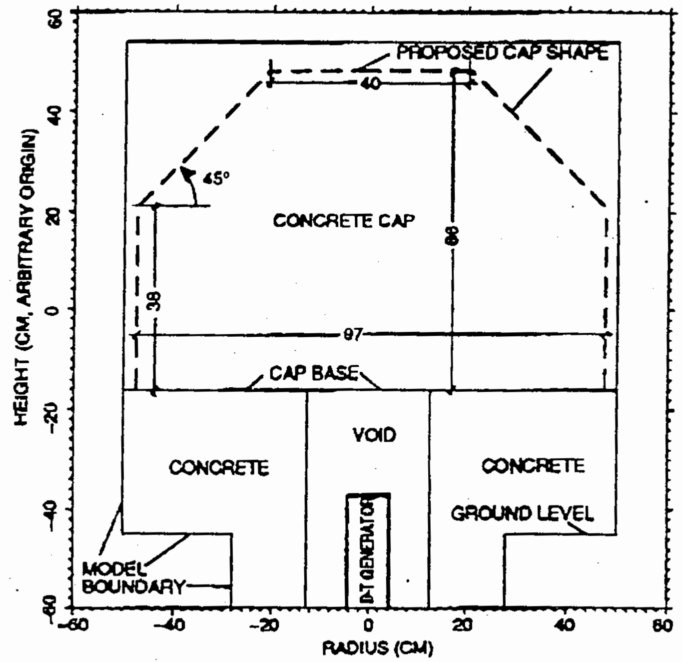

Figure 2: Upper part of the MCNP model used to determine the shape of the concrete shield cap. The proposed cap shape maintains the outer surface dose rate below 0.1 mrem/h.

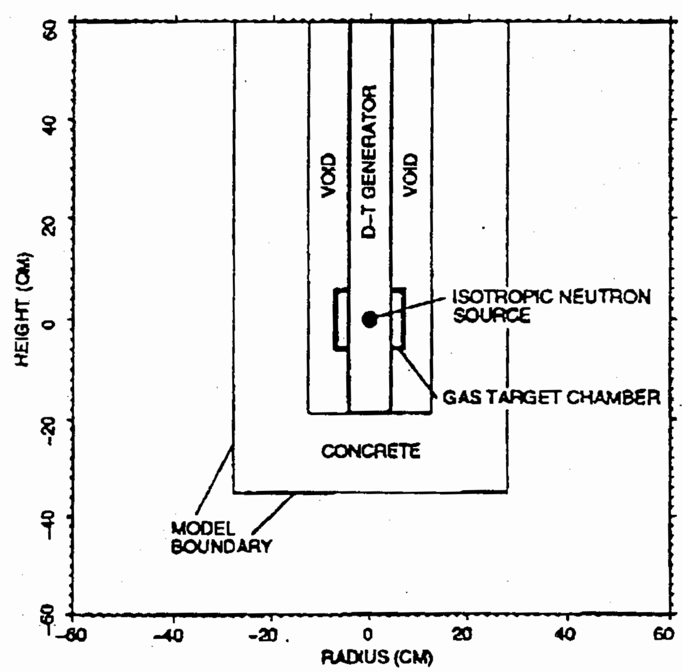

Figure 3: Lower part of the MCNP model used to provide the source of the radiation that streams up the centre hole.

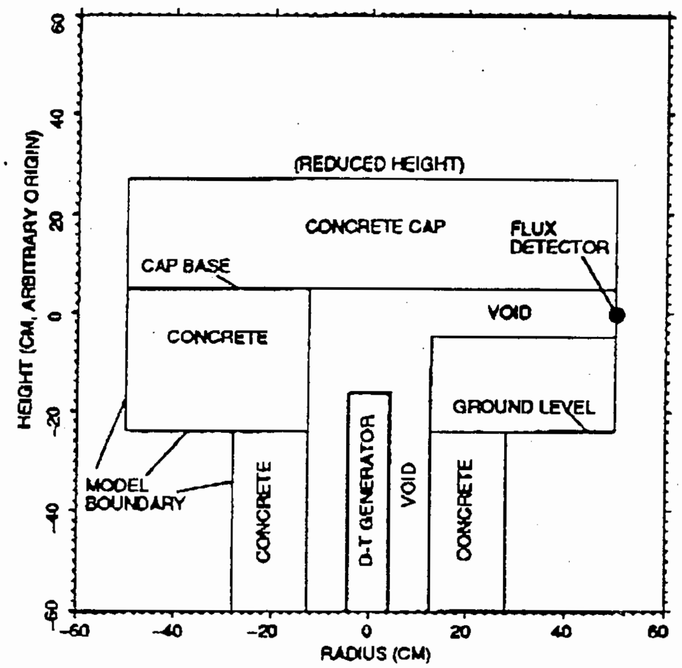

Figure 4: Upper part of the MCNP model used to determine the streaming through the horizontal cable access port.

3.3 Results

3.3.1 Shielding Cap

The total axial dose rate (neutron plus photon) was tallied in the concrete cap through horizontal surfaces at distances from 10 cm to 70 cm above the cap base (abscissa in Figure 5). These surfaces were divided into three concentric radial segments with the dose rates shown as separate curves in the figure.

The total radial dose rate was tallied in the concrete cap through concentric cylindrical surfaces from 20 cm to 50 cm from the cap centerline (abscissa in Figure 6). These surfaces were divided into three height segments with the dose rates shown as separate curves in the figure, where zero is the base of the cap and the segment from −30 to 0 cm is the main shield below the cap.

It is noted that in both sets of results the photon dose rates are roughly 10% of the neutron dose rates. Horizontal lines have been drawn on Figures 5 and 6 at the target dose rate of 0.1 mrem/h. Intercepts between these lines and the dose rate curves provide estimates of the shield thicknesses required to achieve the desired dose rate at various locations. From this information, the proposed cap shape (see Figure 2) was determined as 66 cm high by 97 cm diameter, with the corner removed at 45° to reduce the weight.

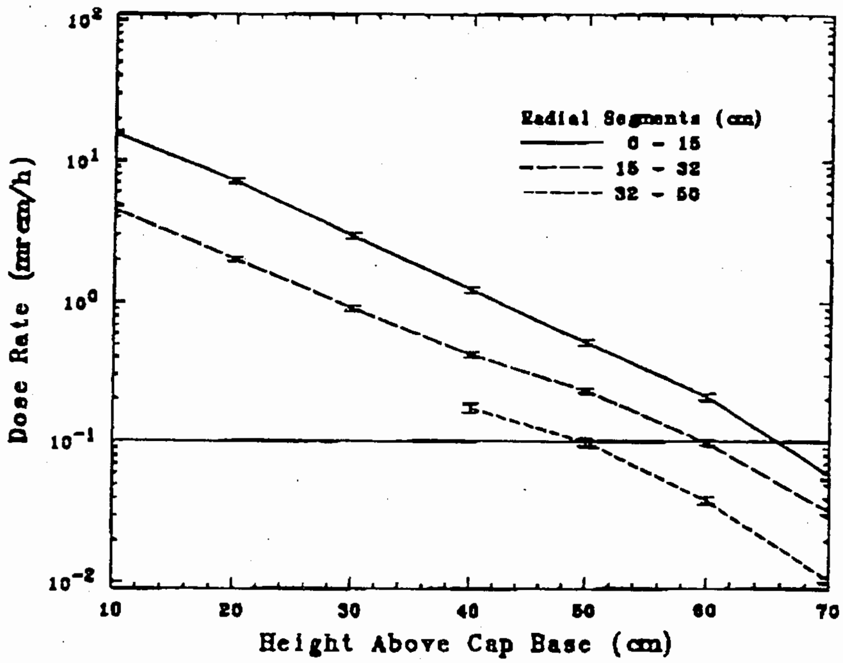

Figure 5: Total dose rates through the shield cap at various heights above the cap base for three concentric radial segments.

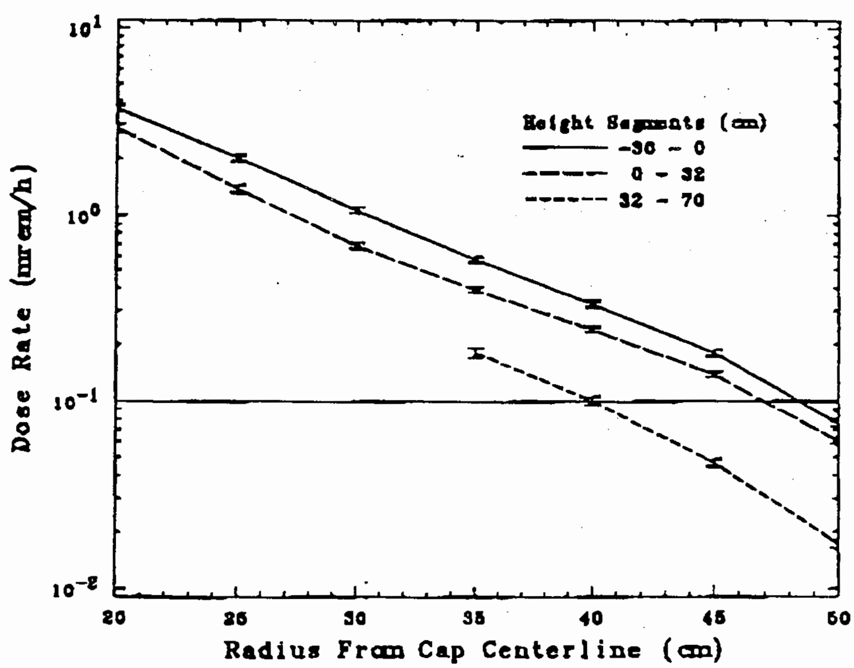

Figure 6: Total dose rates through the shield cap at various radii from the cap centerline for three height segments. Height segment −30 to 0 cm is the main shield above ground level below the cap base.

3.3.2 Cable Access Port

The total dose rate streaming through the horizontal access pipe was calculated to be 0.27 mrem/h. This is a directional beam which exceeds the target dose rate of 0.1 mrem/h, but which may be acceptable through personnel access control. This dose rate can be reduced by lowering the elevation of the horizontal access pipe relative to the base of the cap or by extending the vertical hole a short distance into the cap, such that in both cases, neutrons are prevented from exiting through the horizontal pipe following a single scattering event in the roof of the vertical hole.

3.3.3 ^{16}N Production Rate

The ^{16}N production rate in the gas target chamber was estimated with MCNP using two different dosimetry cross sections and a modified version of the lower model shown in Figure 3 in that the thickness of concrete surrounding the target chamber was increased to 100-cm diameter to more accurately account for neutron backscatter, and the target chamber inner radius was decreased to 2.14 cm [8]. The outer radius and height remained unchanged at 6.67 cm and 10.8 cm, respectively. The calculation involved the modified volume flux tally

$$Y = \int_V \int_E \phi(\vec{r}, E)\sigma(E)\, dE dV$$

where Y is the reaction product yield rate, $\phi(\vec{r}, E)$ is the neutron flux per starting particle, and $\sigma(E)$ is the microscopic cross section taken from one of two point-continuous-energy data sets. Then taking the ^{16}O atom density as 5.377×10^{-5} at/b·cm [8] results in yields per neutron of 9.96×10^{-6} from the LANL data [9], and 8.47×10^{-6} from the LLNL data [10]. The statistical uncertainty in these yields is under 0.5% for one standard deviation. The flux-weighted total cross section was calculated as $\bar{\sigma} = Y/\Phi$, where

$$\Phi = \int_V \int_E \phi(\vec{r}, E)\, dE dV,$$

from the LANL and LLNL data sets to be 44.5 mb and 37.8 mb, respectively. The contribution from neutron backscattering from the concrete is less than one percent of the total.

4 Software Qualification

MCNP has been extensively validated, both internally by AECL and externally by other users, for a variety of coupled neutron-photon transport applications. The MCNP Validation Manual [11] includes a compendium of validation references.

The BUGLE80 cross-section library was developed for application to concrete shielding and the ONEDANT/BUGLE80 combination has been compared to other discrete ordinates codes and libraries [12]. The estimated cap height predicted with MCNP (66 cm) is consistent with the rough estimate from ONEDANT (68 cm).

This analysis was performed under Branch standard software quality assurance procedures [13] and selected computer files have been retained as listed in the Computer Analysis Archive Record (page 11).

5 Conclusions and Recommendations

A shielding assessment of the D–T neutron source for the SNO Project has been completed. Ordinary concrete was chosen over water due both to superior total dose rate attenuation and lack of the requirement for continuing maintenance. Although the native norite rock would have served almost equally well, the hole has already been dug and the use of concrete is most expedient. Finally, concrete has no safety limitations.

The concrete can either be poured in place or pre-formed into *donuts* and lowered into place. If the later is chosen, there is the potential for radiation to stream up the gap between the donuts and wall of the hole. This streaming must be verified as acceptable or provisions taken to shield it.

A concrete cap must be placed over the main concrete shield to stop radiation streaming up the vertical hole. Monte Carlo calculations were used to determine a proposed shape for this cap that minimizes weight while reducing the total dose rate to under 0.1 mrem/h. Radiation streaming out of the horizontal access pipe is estimated to be 0.27 mrem/h. The Monte Carlo statistical standard deviation in these results is typically less than 5%.

If the cap is lifted into place, no further analysis is required. However, if rollers are used to move the cap, there is the potential for radiation streaming through the gap between the cap and the main shield which must be checked. The magnitude of this gap streaming could be reduced by extending the vertical hole a short distance into the cap as discussed in §3.3.2.

References

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Computer Analysis Archive Record (1 Page)

Document Title: D-T Neutron Source Shielding for the Sudbury Neutrino Observatory
Document Reference No: RTN-TN-057
Archive Directory: AECL$RTB_ARCHIVE:[TASKS.RTB-TN-057]
Lead Analyst: G.B. Wilkin
Archive Date: 1995 April 10

No	Code Version	Load Date	Data File	Load Date
1	MCNP 4a.1	93-12-08	MCPLIB2	08-AUG-1985
			RMCCS2	08-AUG-1985
2	ONEDANT 1.1	88-07-15	SNLIB67(BUGLE80)	22-JAN-1992
3	20/20 3.12.13	93-03-05		
4	PLOTMD 2.3A	94-11-04		
5	LaTeX 2.09	91-01-14		
6	M11DRAW 6.2	91-09-18		

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-    0ARCHIVE.              --archive record
1    MC01.DAT               \-cable tray streaming-scoping
1    MC02.DAT               /                     -final
1    MC03.DAT               \-N16 production rate-scoping
1    MC04.DAT               |                     -extended concrete
1    MC05.DAT               |                     -new target chamber size
1    MC06.DAT               /                     -w/out neutron backscatter
1    MC11.DAT               \-cap penetration-scoping
1    MC12.DAT               /                 -final
2    ODNINP4.DAT            \-1 dimensional spherical-water
2    ODNINP5.DAT            |                         -ord concrete
2    ODNINP6.DAT            |                         -air
2    ODNINP7.DAT            /                         -norite rock
4    PLTMC12.DAT            \-data plot files
4    PLTODN.DAT             /
3    RESMC12.M20            \-MCNP result processing
3    RESMC12.TXT            /
5    TN057.TEX              --LaTex document file
5    EPSF.STY;1             \-LaTex resource files
5    RTBSTY.STY;4           /
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