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ENHANCEMENT OF THE SNS SPALLATION TARGET BY U-235 ENRICHMENT

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Introduction

A collaboration between Rutherford Appleton Laboratory (RAL) and Kernforschungsanlage Jülich GmbH (KFA) was initiated in the early 1983 to study a multiplying subcritical booster target for the SNS (1). One proposal to realize a booster target is to enrich the present U-238 target in U-235 or Pu-239 and to enhance the neutron output by additional fissions of these isotopes without changing the overall design of the target. Another concept is to design a totally new booster target regarding from the beginning all requirements of such a type of neutron source. Since there was no preference for one of these proposals at the beginning it was decided to consider first the development potential of the present SNS target assembly and to use this as a baseline for the design of a new subcritical booster target.

For estimating the potential of the enhanced SNS target two-dimensional neutron transport calculations were made for different enriched targets and the original SNS target which is used as reference. In the following, the potential performance of the present SNS target is discussed and first results of the neutronic calculations for the enriched SNS target are given.

Potential Performance of the SNS Target

The power density distribution in the present SNS target is not uniform along the target axis (2, 3) and emphasis is being placed on varying the U-235 enrichment along the target to increase the neutron production at the downstream part. A further measure is to operate the SNS target up to its design limit.

The average power density in the uranium plate with the highest thermal load is 270 W/cm^3 for the present SNS target. Regarding the safety factor 3 in the target cooling system the average power density in the enhanced target could be raised to 810 W/cm^3. This leads to a total power of 1.6 MW in the enhanced target or a total energy deposition of 8100 MeV/p for a proton current of 200 μA.

The energy deposition due to high energy reaction is 646 MeV/p in the present SNS target. Since the high energy deposition will be changed only insignificantly by the enrichment the number of low energy fissions, n_{fiss}, will be increased from 2 fissions /p in the present target to about 38 fissions /p in the enriched target. The average number of fission neutrons per fission is about 2.5 in the enriched target so that 96 fission neutrons /p are created in the enhanced assembly.

In the present SNS target, 28 neutrons are produced per incident proton of which 23 neutrons are due to high energy reactions. Denoting the total number of neutrons per proton by n_{SNS} and the number of spallation neutrons by n_{spall} the neutron gain of the enriched target is given by

$$\text{neutron gain} = \frac{n_{fiss} + n_{spall}}{n_{SNS}} = 4$$

The resulting effective multiplication constant defined by

$$k_{eff} = \frac{\text{fission source}}{\text{absorptions + leakage}} = \frac{n_{fiss}}{n_{fiss} + n_{spall}}$$

will be 0.8 for the enhanced target.

Calculational Model

The model of the target assembly used in the neutron transport calculations is shown in Fig. 1. The target is simulated by a 9 cm diameter by 33.9 cm long cylinder containing a uniform mixture of uranium, coolant, and cladding in their appropriate portions. The coolant is heavy water and the cladding material is Zr. The target is surrounded by pressure vessel which is approximated by a 1 cm thick region made of stainless steel. The reflector, which consists of beryllium rods cooled by heavy water, is represented by a homogeneous mixture of 80 % (vol.) Be and 20 % (vol.) D_2O.

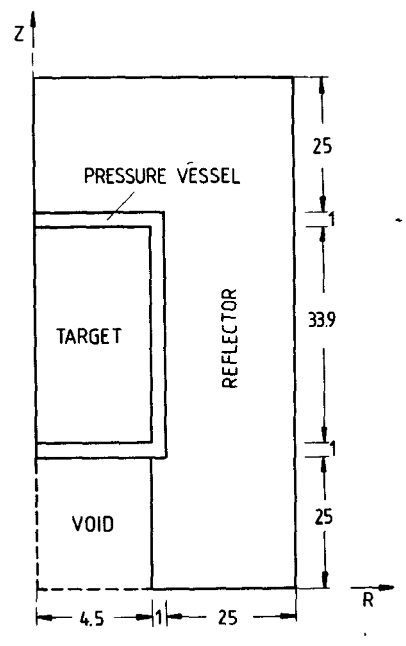

Fig. 1 Model for Transport Calculations (Dimensions in cm).

In the neutronic calculations, the target was divided into eight regions of nearly the same volume in which the enrichment in U-235 was changed (see Fig. 2). The U-235 enrichment increase in the direction of the proton beam to account for the decrease of the primary neutron source. To regard the influence of the Be-reflector the enrichment in the outer radial regions is lower than in the inner regions.

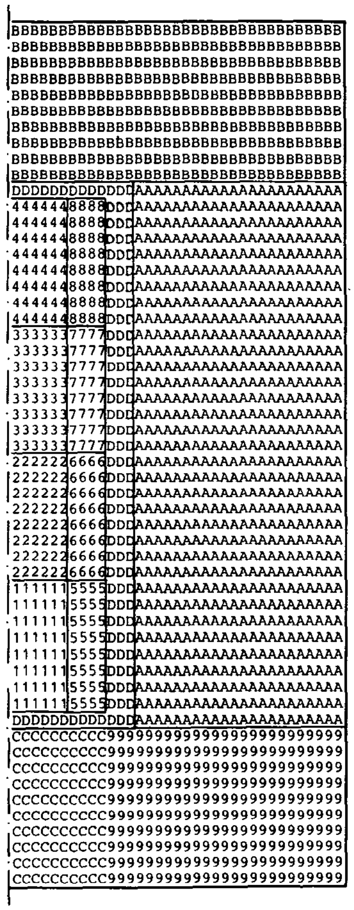

Fig. 2 Zone Number by Interval in Transport Calculations.

The neutronic calculations were made with the diffusion code CITATION (4) and the discrete ordinates transport code DOT-IV (5), using the P1S4 approximation. A 15 broad group library was used in the transport calculations which was collapsed from a 123 fine group cross section library (94 fast plus 30 thermal groups). The weighting functions were calculated by the 1-D discrete ordinates transport code XSDRNPM (6) using infinite cylinder geometry. For this purpose, two spectrum calculations with U-235 enrichments of 30 % and 50 % were made regarding the different enrichments in the lower and upper target zones.

The primary spallation neutron source was approximated by a two-dimensional step function shown in Fig. 3. The source strenght data were taken from Ref. 2.

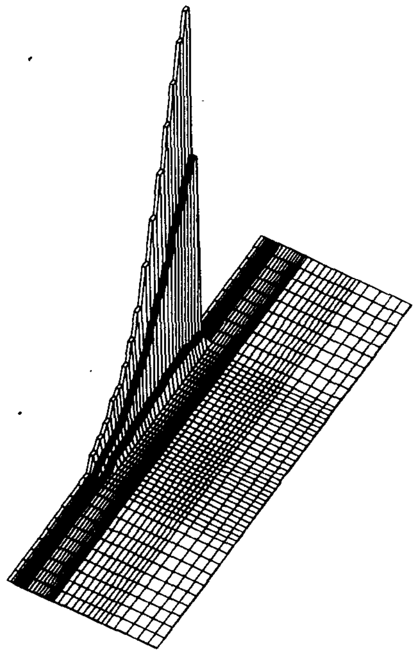

Fig. 3 Neutron Source Distribution (E < 15 MeV) in Target.

Discussion of Results

Two series of calculations were made using the diffusion code CITATION and the discrete ordinates transport code DOT. In the first series, the original SNS target which is used as reference was calculated. The results were compared with earlier results of RAL to make sure that the calculational model is adequate. In the second series of calculations, the U-235 enrichment in the 8 target regions was changed in such a way that a nearly uniform power density distribution with the desired power level of 810 W/cm^3 was achieved regarding also the contribution of the incident proton beam in the front regions of the target. Some quantities of interest obtained from these calculations are given in Table I. To calculate the total power a fixed amount of 125 kW due to the high energy reactions was added to the fission power. It was furthermore assumed that an energy of 6 MeV is released in a neutron capture event.

Further results of the CITATION AND DOT calculations for the enriched target are given in Figs. 4 and 5. They present the required U-235 enrichment and the fission power densities in each of the eight target regions. In order to quantify the benefits of the enriched target at the present stage of calculations, the neutron gain was calculated as the ratio of the radial outflow from the enriched target to the radial outflow from the present SNS target.

The gain in the total radial outflow for the enriched target in comparison to the present SNS design is about 3. The enhancement varies from 2 in the region near the proton beam entrance to 7.7 at the downstream region. The increase in the neutron currents feeding the moderator boxes is 2.3 for the first half of the target and 5.7 for the second half.

TABLE 1

Summary of CITATION and DOT Results.

		Fission Power (kW)	Total Power (kW)	Multiplication Factor	Radial Outflow from Target
Reference	CITATION	57	194	0.1542	$5.06 \cdot 10^{16}$
	DOT	68	207	0.1878	$5.60 \cdot 10^{16}$
Enriched	CITATION	1415	1560	0.7972	$1.49 \cdot 10^{17}$
	DOT	1480	1640	0.8062	$1.71 \cdot 10^{17}$

U-235 ENRICHMENT

FISSION POWER DENSITY (W/CM^3)

		NEUTRON GAIN
REGION 4 70 % 734	REGION 8 50 % 746	7.7
REGION 3 45 % 670	REGION 7 35 % 749	4.3
REGION 2 35 % 662	REGION 6 30 % 749	2.6
REGION 1 25 % 514	REGION 5 25 % 617	2.0

Fig. 4 U-235 Enrichment, Fission Power Density, and Neutron Gain in the 8 Target Regions obtained from the DOT Calculation.

U-235 ENRICHMENT

FISSION POWER DENSITY (W/CM^3)

		NEUTRON GAIN
REGION 4 90 % 659	REGION 8 70 % 733	7.1
REGION 3 60 % 664	REGION 7 45 % 754	4.1
REGION 2 40 % 572	REGION 6 35 % 736	2.5
REGION 1 30 % 440	REGION 5 30 % 628	1.9

Fig. 5 U-235 Enrichment, Fission Power Density, and Neutron Gain in the 8 Target Regions obtained from the CITATION Calculation.

Conclusion

One way to design a multiplying subcritical booster target for the SNS is to enrich the present target in U-235 or Pu-239. First calculations show that an operation of the target up to the design limit by enriching in U-235 leads to gains in the radial neutron outflow from the target that further investigations of this proposal seem to be worthwhile. The next task will be the study of the time structure of the neutron pulse from the target and the moderator. A further important problem to be investigated will be how the coupling between target and moderator will be influenced by the large increase of the fission power in the target.

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