JAERI - M
83-050

SINGLE-FLUID-TYPE ACCELERATOR MOLTEN-SALT BREEDER (AMSB)

March 1983

Kazuo FURUKAWA and Kineo TSUKADA*

日 本 原 子 力 研 究 所
Japan Atomic Energy Research Institute

編集兼発行　日本原子力研究所
印　　刷　㈱原子力資料サービス

Single-fluid-type Accelerator Molten-Salt Breeder (AMSB)

Kazuo FURUKAWA and Kineo TSUKADA*

Division of Nuclear Fuel Research,

Tokai Research Establishment, JAERI

(Received February 17, 1983)

The applications of molten-salt target and blanket to the Accelerator Molten-Salt Breeder (AMSB) are discussed, where the molten fluorides including ThF_4 or UF_4 in high concentration are utilized.

This concept naturally has several significant benefits relating to the target fabrication, design, radiation damage, heat removal, safety, economy, etc. So far, however, a poor spallation neutron yield was mentioned as a severe fault. However, according to the results of neutronics calculation performed on the molten-salt system such as $LiF-BeF_2-ThF_4$, $LiF-NaF-ThF_4$, $LiF-BeF_2-UF_4$ etc., their neutron yields have been found comparable with or superior to the values for heavy metal target such as Bi and Pb, assuring the high performance of AMSB.

The schematic figures of AMSB are shown together with the several tables and figures of their neutronics calculation

---

Originally prepared for a Consultants' Meeting on "Spallation Breeders" of IAEA (November, 1980).

* Atomic Energy Research Institute, Nihon University, Kanda-Surugadai, Chiyoda-ku, Tokyo, 101 JAPAN

models, neutron yields, predicted performances, etc. Neutronics calculations have been performed with the use of NMTC/JAERI modified to include fission processes, TWOTRAN-II and other auxiliary codes. The spatial and energy distributions of neutrons have been also calculated. The chemical aspects of spallation products in these facilities have been examined and estimated that they are practically manageable in processing.

The energy strategical positions of AMSB have been discussed, one of which is the application for producing ^{233}U fuels. This may contribute to the early initiation of self-sustaining Th fuel cycle.

Keywords: Accelerator, Breeding, Accelerator Breeder, Spallation, Molten Salts, Fluorides, Thorium, Uranium, Fissile Materials, Targets, Blankets, Fluids

(additive comment)

The further works on AMSB was presented at Japan-U.S. Seminar on "Th fuel reactors", (Oct. 1982, Nara, Japan) in the title "The combined system of Accelerator Molten-Salt Breeder (AMSB) and Molten-Salt Converter Reactor (MSCR)", by K. Furukawa, Y. Kato, T. Ohmichi, H. Ohno. (Proceeding is printing).

(added to proof)

The related new works are presenting in the "First International Symposium on Molten Salt Chemistry and Technology (April 20-22, 1983, Kyoto, Japan) in the title "Accelerator Molten-Salt Breeding System", Report Series [I] General, [II] Reactor Engineering, [III] Reactor Chemistry and [IV] Proposal of Facility Plan.

単一流体型加速器溶融塩増殖炉（AMSB）

日本原子力研究所東海研究所燃料工学部
古川　和男・塚田甲子男*

（1983年2月17日受理）

溶融塩ターゲットとブランケットを加速器増殖炉に利用することが検討された。この塩の中にはThF_4，またはUF_4が高濃度で含まれる。この方式の利点はターゲット製作，設計，照射損傷，熱除法，安全性，経済性に関連する。$LiF-BeF_2-ThF_4$，$LiF-NaF-ThF_4$，$LiF-BeF_2-UF_4$などをターゲット・ブランケット兼用で使うならば，予期以上の中性子発生効率がえられることが，ニュートロニックスから示された。この加速器溶融塩増殖炉（AMSB）の概要，特性，炉化学問題点が紹介された。また，エネルギー戦略上の位置づけ，特にトリウム燃料サイクル実現化における貢献などが論じられた。

＊　日本大学，原子力研究所（〒100東京都千代田区神田駿河台）

Contents

目　　次

1. INTRODUCTION

The spallation reactions between the heavy nuclei and high energy particles such as protons or deuterons of the energy 0.5~1.5 GeV can effectively generate a lot of neutrons[1]. The prospective utilizations of these neutrons will be as follows:

(1) high intense neutron sources for the studies of material science and nuclear physics,

(2) fissile material productions from the fertile materials,

(3) tritium production by the reaction with lithium,

(4) incineration of the radioactive wastes such as the fission products and/or higher actinoids.

In these applications the target systems will be subjected to the following difficult technical problems:

(a) severe radiation damage of targets,

(b) removal of violently generating heat,

(c) frequent shuffling of the target materials to deal with the localized reaction zone.

To solve these problems it is proposed that the liquid targets will be preferable. These are classified in the following four materials[2]:

(A) aqueous solutions including the slurry or suspension systems,

(B) organic liquids,

(C) liquid metals including the slurry or suspension systems,

(D) molten-salts.

The technological comparison of these liquids is shown in Appendix A, from which the following conclusions will be deduced. The problems in (A) will nearly correspond to those in the aqueous homogeneous (fission) reactors[3], whose disadvantages are corrosiveness, high pressure, relatively low temperature and low performance. The aqueous suspension systems will be unstable under the intense radiations. (B) will be decomposed by the irradiations. In (C) the Pb, Bi or their alloys are exceptionally applicable to the target systems, as already employed in the several proposals[14]. The actinoid metals or alloys will be more effective than the Pb and/or Bi as the target metals. However, the search of the appropriate liquid including actinoids is so difficult as to be equivalent to those in the developments of Liquid Metal Fuel (fission) Reactors[3] or Molten Plutonium Alloy (fission) Reactors[3], which were the unsuccessful projects. On the other hand, (D) the molten-salt system seems to be more useful for the all purposes of (1)~(4). This will lead to realize the utilization of actinoid (especially Th and U) nuclei for the spallation reactions.

In the next chapter a concept of the molten-salt target containing Th or U compounds in the high concentration for the spallation reaction will be presented. Although the atom percents of heavy nuclei in molten-salts are about 10 atom %, the large system will have nearly the same neutron generation efficiency as the pure metal target of Th or U (cf. §3), if any neutron-poisonous element in molten salts does not exist except Th or U.

The molten-salts can accomodate several elements, dissolving them in the simple salts. Therefore the target salts will be able to function as blanket salts containing the fertile materials simultaneously*, which is the most useful aspect for the Accelerator Breeder development (§4).

2. CHARACTERISTICS OF THE MOLTEN-SALT TECHNOLOGY AND SELECTION OF MOLTEN SALTS

The most suitable kind of molten salts for the nuclear-engineering applications is fluoride[(3)(5)], because the element F is one of the most stable nuclei for thermal neutrons, as shown in Table 2.1. Molten fluorides are understood as the typical "ionic liquids" composed of independent positive and negative ions, and have the following characteristics in general:

(1) Molten fluorides are liquids stable at relatively high temperature. They have low vapor pressures, low viscosities and low electric conductivities (10^{-4}~10^{-6} of liquid metals) due to the independence of ions.

(2) Molten fluorides are chemically inert, and don't react violently with air or water.

(3) High-temperature container (structural) alloys such as Hastelloy N (Ni-Mo-Cr alloy) have been developed for the molten fluorides by ORNL. Hastelloy N is serviceable up

---

* Therefore, this reactor concept will be called as a single-fluid-type accelerator breeder. The two-fluid-type Accelerator Molten Salt Breeders are also examined by the other report: K. Furukawa, H. Ohno, J. Mochinaga & K. Igarashi, J. Nucl. Sci. Tech. 17 (1980) 562.

them in the simple salts. Therefore the target salts will be able to function as blanket salts containing the fertile materials simultaneously*, which is the most useful aspect for the Accelerator Breeder development (§4).

2. CHARACTERISTICS OF THE MOLTEN-SALT TECHNOLOGY AND SELECTION OF MOLTEN SALTS

The most suitable kind of molten salts for the nuclear-engineering applications is fluoride[(3)(5)], because the element F is one of the most stable nuclei for thermal neutrons, as shown in Table 2.1. Molten fluorides are understood as the typical "ionic liquids" composed of independent positive and negative ions, and have the following characteristics in general:

(1) Molten fluorides are liquids stable at relatively high temperature. They have low vapor pressures, low viscosities and low electric conductivities (10^{-4}~10^{-6} of liquid metals) due to the independence of ions.

(2) Molten fluorides are chemically inert, and don't react violently with air or water.

(3) High-temperature container (structural) alloys such as Hastelloy N (Ni-Mo-Cr alloy) have been developed for the molten fluorides by ORNL. Hastelloy N is serviceable up

* Therefore, this reactor concept will be called as a single-fluid-type accelerator breeder. The two-fluid-type Accelerator Molten Salt Breeders are also examined by the other report: K. Furukawa, H. Ohno, J. Mochinaga & K. Igarashi, J. Nucl. Sci. Tech. 17 (1980) 562.

to 800 °C.

(4) Molten fluorides are one of the best heat mediums applicable in high temperatures. Adding to the above several superior properties they have high heat capacities (per unit volume) due to the fact that each ion is independent of one another. Ions in molten fluorides have idealistic closed electron shells, which mean the smaller atom-sizes than neutral atoms, so permitting a higher atom-packing number density.

(5) Molten salts are useful as chemical reaction mediums. Ionic liquids are capable of readily dissolving other various ionic species, and they allow a number of chemical reactions to run, dissolving or sustaining reactants or products so as to promote smooth reactions.

(6) Furthermore, molten fluorides are serviceable in the nuclear engineering fields, as they could not be damaged by nuclear radiations. The element F is hardly likely to react with neutrons (cf. Table 2.1).

The molten fluorides are superior to the other materials as the heat mediums, chemical reaction mediums and nuclear engineering mediums and will promise the wide nuclear applications as explained in Appendix B. Therefore it will be able to be recognized that the molten fluorides will be the best liquids suitable for the spallation reaction facilities which have the severe technical problems mentioned as (a)~(c) in the previous chapter.

For the selection of molten fluorides suitable for the target and/or blanket of spallation facilities, the following conditions will be required:

(A) The element Th or U should be chosen as target nucleus preferable to Pb and/or Bi. Their salt compounds are ThF_4 and UF_4, which should be contained in high concentration of 20~35 mol% in the molten-salt mixtures.

(B) Solvent fluorides added to keep the lower melting (liquidus) temperature of 450~550 °C should have the lower thermal neutron absorption cross-section and the good material compatibility. Therefore, they should be chosen from alkali-earth fluorides such as ^{7}LiF, BeF_2, NaF, (KF) and RbF (cf. Table 2.1).

(C) They should have the moderate viscosity coefficient as lower as possible than 20 centipoises.

The molten fluoride mixtures satisfying the above requirements were widely searched, and some promising ones were chosen temporarily which are presented in Table 2.2. All of these phase diagrams have been studied and summarized by ORNL[(7)].

Their viscosities may be estimated to be less than 20 centipoises in working temperature region by making a comparison with the similar molten salts[(8)(5)].

The densities at 100 °C above the melting points are also predicted following the Furukawa's ionic-liquid model[(9)]. They are necessary for the neutronics calculations presented in the next chapter.

3. RESULTS OF NEUTRONICS CALCULATIONS ABOUT TARGET SYSTEMS FOR NUCLEAR FUEL BREEDING*

The neutronics calculations have been performed for several types of molten salt selected in the preceding chapter as the candidate materials for use in the target of the nuclear fuel breeding facility. The detail of the computer code system used in our calculations is described in Appendix C.

The structure of the nuclear fuel breeding system modelled for numeral calculations is shown in Fig.3.1 with geometrical dimensions. The proton beam of 1 GeV enters the target from the top. The beam diameter of 1.5 m is assumed for the beam current of 300 mA.

The calculational results on the neutron yield (the average number of neutrons produced by one incident proton) are summarized in Table 3.1, blank spaces of which are due to the fact that our calculations have not been made. Refering to Table 3.1, we can make a comparison of neutronic characteristics of molten salt and also make observation on the effects of high (>15 MeV) and low (≤ 15 MeV) energy fissions. The cutoff energy for the cascade-model calculations is taken as 15 MeV both for protons and neutrons.

Little difference is seen between the neutron yields from cascade-evaporation processes in ^{7}LiF-ThF_4 (Case 5) and ^{7}LiF-UF_4 (Case 9). From the example for ^{7}LiF-UF_4 (Case 9) it is seen that the inclusion of high energy fissions makes a gain

* Main materials of this section was originally presented in the 4th Meeting of International Collaboration Advanced Neutron Source (Oct. 1980) [cf. Y. Nakahara: Proc. ICANS-IV (1980) p.286 and K. Furukawa, K. Tsukada, Y. Nakahara: ibid. (1980) p.349.]

of 18 % in the neutron yield. The Case 3 shows that the increase due to fissions of Th in the neutron energy range below 15 MeV is only 3 %, while for the molten salt of U it is 23 % as seen in the Case 8. Comparisons between two cases indicate that the values of neutron yield in the nucleon energy range above 15 MeV are determined mainly by the concentrations of Th or U and their dependence on the other compositions of salt is weak. It should be noticed, however, that the contribution from the Be(n,2n) reaction is not taken into account correctly, because the nucleon transport codes available at present are not designed to calculate the Be(n,2n) reaction explicitly. The values of neutron yield for salt including Be can be expected to get a gain further, if the Be(n,2n) contribution is estimated properly. Somewhat larger value of the neutron yield for the Case 3 in comparison with the Case 4 is probably due to the larger statistical error in the Case 3 where relatively large amount of light nuclides are contained. The effect of ^{233}U enrichment on the neutron yield was estimated in the Case 2. The 1 % ^{233}U results in 16.6 % increase of the neutron yield over the case of pure Th.

The spatial distributions of primary and total protons are shown in Fig.3.2 for the $^{7}LiF-UF_4$ case (Case 9), and the mean free path (mfp) of the primary 1-GeV proton is estimated to be 39 cm. In Fig.3.2 also are shown the distributions of total neutrons in the energy range above the cascade cutoff (15 MeV), which are the results of the NMTC[10] and NMTC/JAERI[11] calculations. It is seen that the maximum of neutron distribution appears approximately at 60 cm from the surface of salt. Some irregularities seen in Fig.3.2 are probably due to the insufficient number of samplings in the Monte Carlo calculations.

Two dimensional (r,z) distributions of neutrons in the neutron energy range below 15 MeV are shown in Fig.3.3 for some energy groups. Flux level is normalized to one neutron slowed down or born below 15 MeV obtained as the results of the NMTC/JAERI calculation. It is interesting that the off-central-axis current is seen clearly in the Group 1 distribution. The figures of minus sign on contours in Fig.3.3 indicate the orders in powers of ten.

The group flux spectrum defined as

$$\phi_g = \int_{\Delta E_g} \phi \, dE$$

for the neutrons in the energy range below 15 MeV is shown in Fig.3.4 for a region in ^7LiF-BeF_2-ThF_4 (67-18-15), where the spatial neutron distribution is maximum.

4. DESIGN CONCEPT OF ACCELERATOR MOLTEN SALT BREEDER (AMSB)

At the end of this century one of the most important brob-lems to start up the breeding power reactor systems by fission reaction will be the shortage of fissile materials. Especially if we want to start the Th fuel cycle, the production of ^{233}U will be necessary. To solve these problems the application of the molten-salt target technology to the single-fluid type Accelerator Breeder Concept will be effective. In this Accelerator Molten-Salt Breeder (AMSB) the spallation reaction and fissile production zone in molten salts is chosen to become irregularities seen in Fig.3.2 are probably due to the insufficient number of samplings in the Monte Carlo calculations.

Two dimensional (r,z) distributions of neutrons in the neutron energy range below 15 MeV are shown in Fig.3.3 for some energy groups. Flux level is normalized to one neutron slowed down or born below 15 MeV obtained as the results of the NMTC/JAERI calculation. It is interesting that the off-central-axis current is seen clearly in the Group 1 distribution. The figures of minus sign on contours in Fig.3.3 indicate the orders in powers of ten.

The group flux spectrum defined as

$$\phi_g = \int_{\Delta E_g} \phi \, dE$$

for the neutrons in the energy range below 15 MeV is shown in Fig.3.4 for a region in ^7LiF-BeF_2-ThF_4 (67-18-15), where the spatial neutron distribution is maximum.

4. DESIGN CONCEPT OF ACCELERATOR MOLTEN SALT BREEDER (AMSB)

At the end of this century one of the most important broblems to start up the breeding power reactor systems by fission reaction will be the shortage of fissile materials. Especially if we want to start the Th fuel cycle, the production of ^{233}U will be necessary. To solve these problems the application of the molten-salt target technology to the single-fluid type Accelerator Breeder Concept will be effective. In this Accelerator Molten-Salt Breeder (AMSB) the spallation reaction and fissile production zone in molten salts is chosen to become practically infinite, which corresponds to the size 6 m in depth and 5 m in diameter (cf. Fig.3.1).

The reactor configuration is very simple, as shown in Fig.4.1. The predicted performance of AMSB is shown in Table 4.1.

The vortex is formed on the salt surface in the reactor by using the high-speed molten-salt streams injected from three nozzles 18 in Fig.4.1. This is more or less similar to vane-head design of Bartholomew et al.[12] The free surface in the main molten-salt pumps is about 3 meters below the upper surface in the reactor vessel, balancing with the pressure difference in the static condition.

The reactor vessel is made of Hastelloy N and is 7 m in diameter and 8 m in height, inside of which is shielded by double layers of graphite of about 50 cm each in thickness. The inner graphite layer is replaceable. The total inventory of target salts is about 150 m^3 including residual (about 10 m^3) in the storage tank 12.

The design for the secondary coolant system and electric generating system can be chosen essentially the same as those of 1000 MWe MSBR designed by ORNL[6]. Its electric power generation efficiency is very high and about 43 % (cf. Table 4.1).

Distinguishing points of the present design are as the followings.

(1) Use of the single molten salt for the target and blanket. The merits of them are enumerated in Appendix E.

(2) No window for the incident proton beam into the target by using several sets of the differential pumping systems with many orifice plates and the beam-focussing systems

with the quadrupole magnets.

(3) Proton-beam injection to the center of the vortex formed several-tens cm downward from the upper salt level to improve the heat transfer to the medium and to avoid the neutron generation near the upper boundary of the vessel.

By this design the 1 GeV 300 mA proton beam will be made possible to be injected into the target from the accelerator. The accelerator design will be discussed elsewhere. However, some of the most difficulties in the accelerator design will be:

(1) beam dynamics, that is, beam blow-up, space charge effect and beam spill.

(2) power-to-beam efficiency, which will be increased up to 50 % or more.

(3) activation of the accelerator, main origin of which is the beam spill. If more than 10^{-9} of the beam were spilled, the accelerator would be seriously activated.

Choice of the RF frequency will be important to solve these problems. Halo-scraper will help to decrease the beam spill.

Electric power consumed in this AMSB is estimated to be about 600 MWe, this is comparable with the net electric power generated by the AMSB (see footnote in Table 4.1).

The fissile material production in AMSB of 1 GeV - 300 mA proton is about 800 kg per year in load factor 80 %. The neutronic calculation is also suggesting that the products of spallation and (fast) fission reactions parallelly generated in AMSB are about 40 kg per year in each. If it is assumed that these materials are not separated at all in one year, their total concentrations in molten salts are about 80 ppm in each [cf. with the value 5000 ppm in the MSBR]. Species of the fission products are essentially similar to those of MSBR, although the production rate is about one twentieth in comparison with 1000 MWe MSBR. Therefore, the fission products will be processed following to the MSBR technology[6]. The behavior and processing of spallation products will be discussed in detail at Appendix D. In a word, it will not be difficult that each element of spallation and fission products is kept in low concentration less than 1 ppm.

5. CONCLUDING REMARKS

The concept of accelerator breeding was originally for the military purpose like the other nuclear concepts. However, since the early stage of the development, Canada[13] has suggested its unique position on the energy resource strategy. The AMSB might be one of the first technically feasible accelerator breeder concept except LAFER (Linear Accelerator Fuel Enricher Regenerator) of BNL[14], and will have excellent characteristics by adopting the Molten-Salt Reactor technology and by choosing especially the Th resource utilization in it.

In a summary,

(1) by the application of molten fertile fluorides to the single-fluid-type accelerator breeder, fissile materials for the fuels of the breeder power reactors will be effectively produced not using any fissile materials.

total concentrations in molten salts are about 80 ppm in each [cf. with the value 5000 ppm in the MSBR]. Species of the fission products are essentially similar to those of MSBR, although the production rate is about one twentieth in comparison with 1000 MWe MSBR. Therefore, the fission products will be processed following to the MSBR technology[6]. The behavior and processing of spallation products will be discussed in detail at Appendix D. In a word, it will not be difficult that each element of spallation and fission products is kept in low concentration less than 1 ppm.

5. CONCLUDING REMARKS

The concept of accelerator breeding was originally for the military purpose like the other nuclear concepts. However, since the early stage of the development, Canada[13] has suggested its unique position on the energy resource strategy. The AMSB might be one of the first technically feasible accelerator breeder concept except LAFER (Linear Accelerator Fuel Enricher Regenerator) of BNL[14], and will have excellent characteristics by adopting the Molten-Salt Reactor technology and by choosing especially the Th resource utilization in it.

In a summary,

(1) by the application of molten fertile fluorides to the single-fluid-type accelerator breeder, fissile materials for the fuels of the breeder power reactors will be effectively produced not using any fissile materials.

(2) The technological development of AMSB will be able to proceed in parallel with that of the Molten Salt Power Reactors. The AMSB has a simpler configuration and engineering mainly except for the accelerator development.

(3) The safety of the system will be secured by the subcritical reactor condition and special characteristics inherent in the molten-salt technology (cf. Chap.2).

(4) The economical characteristics may be superior to the other systems by the simple structure, no fabrication, no cladding and no transportation of target, automatic shuffling, easy heat removal, continuous operation and in situ chemical processing. The generated electric power could probably sustain the necessary power to maintain the system.

The comparisons of AMSB with the similar or relating facilities will be considered in Appendix E.

The following development programs should proceed in stepwise:

A. detailed neutronics calculations to find the optimized design conditions by improving the computor programs.

B. physical experiments (neutron and heat productions) on the solid-specimen of target salts.

C. spallation-chemical experiments on the molten salt targets.

D. engineering development of AMSB design.

E. high power Linac development.

F. integral engineering experiment by the 1 GeV - 10 mA class test facilities.

However, in this report, only the preliminary design concepts were proposed, not examined in detail and not optimized concerning the practical design parameters yet. Therefore it has been invited the several suggestions for the engineering improvements and the computer program development to estimates the change of nuclide concentration during the operation and the heat generation in the target and blanket. In parallel it should be performed several partial experiments to confirm the predicted performances.

At present we are planning a preliminary experiment to confirm the neutron yield and heat generation from several fluoride targets.

We have also been improving the computor programs including the effect of $^{9}Be(n,2n)$ reaction.

Acknowledgments

The design concepts have been developed getting the advices of Drs. Y. Kato, H. Ohno, T. Ohmichi and K. Ozawa. Neutronics calculations in this work have been done under the kind leadership of Y. Nakahara on the basis of work for the Light Water Reactor Fuel Regenerator in the group of Dr. H. Takahashi's at BNL. The reactor concept proposed here could not be made unless ORNL U.S.A. carried on the MSBR development spending their twenty-five years' effort and CRNL Canada carried on the accelerator breeder concept development spending their thirty years' effort. Many thanks are expressed to all of them.

References

1) Proc. Information Meeting on Accel. Breeding, CONF-770107 (1977); K. Tsukada: J. Atomic Energy Soc. Japan 20 [8] 533 (1978) (in Jananese).

2) A.R. Ubbelohde: "The Molten State of Matter" (1978, John Wiley and Sons); A. Münster: "Statis. Thermodynam. II (1974, Springerverlag).

3) J.A. Lane, H.G. MacPherson & F. Maslan, ed.: "Fluid Fuel Reactors" (1958, Addison-Wesley); K. Furukawa: "Fluid Fuels" in "Treatise on Reactor Engi." Vol.4 (1972, Baifu-kan in Japanese).

4) K. Furukawa, K. Tsukada & Y. Nakahara: J. Nucl. Sci. & Tech. 18 (1981) 79.

5) W.R. Grimes & D.R. Cuneo: "Reactor Handbook" Vol.I Chap. 17 (1960).

6) M.W. Rosenthal et al.: ORNL-4812 (1972); J.R. Engel et al.: ORNL/TM-7207 (1980).

7) R.E. Thoma: "Adv. Molten Salt Chem." Vol.3, p.275 (1975, Prenum).

8) B.C. Blanke et al.: MLM-1086 (1959); MLM-1079 (1958).

9) K. Furukawa: Discuss. Faraday Soc., No.32, (1962) 53; K. Furukawa: Proc. Symp. Solute-interaction, Nagatsuda (Dec. 1979) (in Japanese).

10) W.A. Coleman and T.W. Armstrong: ORNL-3383 (1963).

11) Y. Nakahara and T. Tsutsui: private communications

12) G.A. Bartholomew and P.R. Tunnicliffe: AECL-2600 (1966) 44, 74.

13) W. B. Lewis: AECL-969 (1953); J. Davis: Nature, 270 (1977) 376 and 375.

14) M. Steinberg et al.: BNL-50592 (1976), BNL-26951 (1980); H. J. Kouts, P. Grand, J. Powell, M. Steinberg and H. Takahashi: BNL-50838 (1978); P. Grand: Nature, 278 (1979) 693; H. Takahashi: J. Atomic Energ. Soc. Japan, 21 (1979) 23.

APPENDIX A Technological Comparison of Various Liquid Targets

Several physical properties of typical liquids chosen from the four types of liquids [(A)~(D) in §1] are shown in Table A.1, where aqueous solutions and organic liquids have been combined as molecular liquids. However, for the selection of the suitable liquid materials for some engineering applications, several technological aspects should be considered from the wide viewpoint. This work is not simple, because each class of liquids has formidable varieties of materials. However, as an aid, in Table A.2, a simple comparison of several type of liquids is tried.

APPENDIX B Several Nuclear Applications of Molten-Salt Technology

As explained in §2, molten salts (especially fluorides) have the superiority as "heat" medium, "chemical reaction" medium and "nuclear engineering" medium simultaneously. Actually molten salts may provide the versatile functions, as shown in Table B.1. In some cases it will be recognized that a single molten-salt phase acts all the above functions.

Nuclear applications of the Molten-Salt Technology have been encouraged by the progress of MSR Program at ORNL. Following the program several other applications have been proposed by the other institutes. The list of typical examples of these nuclear applications is shown in Table B.2.

APPENDIX C Methods of Neutronics Calculations[11]

The neutronics calculations have been performed with the computer code system ACCEL[a] prepared at JAERI for use on the FACOM-M 200 computer. The code system is designed to perform numerical analyses of nuclear characteristics of the accelerator target and blanket systems. Its design concept is based on the BNL code system[b]. Interrelations of computer codes employed in ACCEL are shown in Fig.C.1, where the flow of calculations is also indicated. The function of each code is described briefly in Tables C.1 and C.2.

The first stage of nuclear reactions (cascade evaporation and fission) and nucleon transport processes, initiated by incident protons from a linear accelerator, has been simulated by the Monte Carlo code NMTC/JAERI[a], which is a new version of NMTC[c] modified at JAERI to include the high energy fission in competition with the evaporation. The details of our fission model and the scheme of how to perform the Monte Carlo simulation of the fission process was presented at the ICANS-IV meeting by Y. Nakahara[d]. The importance of the high energy fission and its computational models have been discussed also by F. Atchison[e], H. Takahashi[f] and also V.C. Barashenkov et al.[g]

The cascade cutoff energies for neutrons and protons are taken to be 15 MeV. Why the cutoff is necessary is because the cascade-evaporation model used in NMTC is not applicable in the low energy range. Detailed examinations of the optimum value of the cutoff energy have not been performed yet. The value of 15 MeV, however, is very convenient, because we have compiled cross section data file available in the energy range below it.

The neutron cross sections used in the neutron transport calculations in the energy range below 15 MeV are based on the ENDF/B-4 nuclear data file. The fine energy group cross-section with the GAM-II 100 group structure (h) has been obtained for each nuclide by processing the ENDF/B-4 data with SUPERTOG-JR3. The 100 group cross section sets with up to the P_3 component have been combined by TAPEMAKER into a single file containing microscopic cross section sets of all the nuclides required in the numerical analyses of the molten salt target and blanket systems. TOTAL COUPLE and GROUP INDEPENDENT are used to convert the format of the TAPEMAKER data to that of the one-dimensional neutron transport code ANISN based on the discrete ordinate S_n method. The 100 group data have been collapsed to the 10 group data with ANISN to make neutron cross section input data to be used in the two dimensional neutron transport S_n code TWOTRAN-II.

A collision event file prepared during a NMTC/JAERI calculation is analysed with the HIST3D/A and HIST3D/B codes to make the 100 and 10 energy group neutron source distribution files to be used as the inputs to ANISN and TWOTRAN-II, respectively.

The final neutron transport calculations in the energy range below 15 MeV are performed with TWOTRAN-II. The value of the total neutron yield in the whole energy range (1000 ~ 0 MeV) is obtained from the results of NMTC/JAERI and TWOTRAN-II calculations.

Mass and charge distributions of spallation and high energy spallation products have been obtained by analysing a reaction event file prepared during the NMTC/JAERI calculation with the use of NMTA.

References

(a) Y. Nakahara and T. Tsutui: private communications.

(b) H. Takahashi and Y. Nakahara: Bull. Amer. Phys. Soc., 24, 874 (1979).

(c) W.A. Coleman and T.W. Armstrong: "NMTC Monte Carlo Nucleon Meson Transport Code System", RSIC ORNL CCC-161.

(d) Y. Nakahara: "Studies on high energy spallation and fission reactions", presented at ICANS-IV (The 4-th meeting of International Collaboration and Advanced Neutron Source, Oct.20 ~ 24, 1980, National Laboratory for High Energy Physics, Japan). Proceeding is to be published.

(e) F. Atchison: "Spallation and Fission in Heavy Metal Nuclei under Medium Energy Proton Bombardment", Jül-Conf-34 (1980).

(f) H. Takahashi: "Fission Reaction in High Energy Proton Cascade", BNL-NCS-51245 (1980).

(g) V.C. Barashenkov, V.D. Tonelv and S.E. Chigronov: "On the Calculations of the Electro-nuclear Method of Neutron Production", JINR-R2-7694 (1974).

(h) G.J. Joanou and J.S. Dudek: "GAM-II, A B_3 code for the Calculation of Fast Neutron Spectra and Associated Multigroup Constants", GA-4265 (1963).

APPENDIX D Spallation Chemistry in the Accelerator Molten-Salt Breeder (AMSB)

The spallation reaction products in molten fluoride mixture (cf. Table 2.1) generated by the 1-GeV proton will distribute widely in the periodic table of elements. They include practically all chemical elements as shown in Fig.D.1 schematically. In this lower figure the right-hand peak means the evaporation cascades from heavy nuclei. The middle plateau is high and low-energy fission products, and the left-hand peak presents the products by fragmentation of heavy nuclei spallation of light elements and evaporated particles. They also include their decay products. Since the mechanism of fragmentation has not been clarified yet, the fragmentation process is not incorporated into the computational method. This makes it difficult to predict precisely the light nuclei distribution.

The significant elements among these products are tabulated in Table D.1, qualitatively. These elements and the others can be classified to (A) ~ (F) in Table D.2 depending on their chemical behaviors in the molten fluorides.

The class (A) elements have not any problems. The class (B) elements can easily be separated into the vacuum system and to the cover gas phase. The behavior of hydrogen atoms (C) in these molten fluoride systems was analyzed by the large test loop at ORNL[a]. It has been confirmed that the major parts of T, D and H are permeated into the secondary coolant salt $NaBF_4$-NaF (92-8 mol%) and recovered in the form of water in cover gas of the coolant salt, and the leak-rate reduction more than one thousandth has been guaranteed.[b] The class (D) elements will be contained stably in molten fluorides as halogen salts. Therefore it will be confirmed that the elements of (A) ~ (D) do not induce any severe trouble in the spallation reactor systems. They will compose the major parts of products.

The amount of class (E) & (F) may be less than one percent in total. Some elements [especially the class (E) elements] will be spontaneously isolated by evaporation or by plate-out on the surface of graphite or of piping materials. If necessary, some elements composing of unstable compounds should be separated by the chemical reduction and extraction by the contact with metallic Li, Be or Na in some bypass-cleaning systems, or by the similar simple methods such as filtration, chemical compound formation (in solid of liquid phase), etc..

References

(a) An estimated amount of He produced per annum is about 3.7 m^3 (Standard Temperature Pressure) [cf. Fig.D.1].

(b) G.T. Mays et al.: ORNL/TM-5759 (1977), and the reference (6).

(c) This is one of the most definite and safe techniques developed for the tritium management in the reactor systems, which will provide the industrial tritium-production method (cf. Appendix E).

APPENDIX E Merit of AMSB in Comparison with the Similar or Relating Facilities

The merits of AMSB will be briefly discussed in comparison with the following similar or relating facilities: EP (U enrichment plant), UCR (U converter power reactors), HBR (fusion-fission hybrid reactors), from several viewpoints.

(A) Resources Problems

1. No use of fissile materials as starting materials (cf. EP, UCR, HBR). We can straightly start up the Th-breeder power reactors such as MSBR or MSCR, as shown in Fig.E.1.
2. straight utilization of fertile material Th.
3. no use or less use of net electric-power (cf. EP).

(B) Technology Problems

1. Linac development-confirmed in principle (cf. HBR).
2. MSR technology-R & D completed (cf. HBR). Hardware development is supported by LMFBR sodium components development.
3. no trouble from the radiation damage (cf. UCR, HBR).
4. multiple operation of Linacs & Breeders-load factor improved by setting of multiple AMSB's.

(C) Safety Problems

1. chemically stable and low pressure (cf. UCR).
2. subscritical nuclear reactivity (cf. UCR).
3. lower concentration [10^{-3}] of trans-U elements (cf. UCR, FBR, HBR)

Though, in the case of Th, ^{236}U will be sink, management of it will be easier than that of trans-U elements.

4. (controlled) minor environmental release (cf. reprocess plants)
   However, the levelling down of activation of the accelerator by beam-spill should be developed.

(D) Economic Problems

(a) comparing with UCR and HBR in general

1. low price of Th.
2. no cladding, no fabrication and no transport of target.
3. in-situ reprocessing and continuous operation.
4. high conversion efficiency to electric power [~43 %].

(b) comparing with EP

5. probably selling the excess electric power.
6. byproduct of tritium.

(E) Energy-Strategical Problems

1. Th-U cycle utilization parallel with U-Pu cycle.
   Th resources are 4 times more than U resources.
2. production of ^{239}Pu (or ^{233}U) for UCR and LMFBR.
3. minor environmental impacts.
   Amount of spallation products. (Fission products)
   Thermal polution. Tritium release. Fission gas release.
4. concentrated siting of AMSB in the Regional Center.
5. easy safeguard, non-nuclear-proliferating feature of ^{233}U fuel [high gamma-activity]. (cf. EP, UCR)

Table 2.1 The Elements having tiny Thermal-neutron Absorption Cross-sections.

[Calculated from "Tab. of Isotopes", 7th Ed. (1978)]

element		abundance	[barn]	[natural elem.]
1.	O		0.00018	
2.	D	(0.0148%)	0.00052	[H 0.332]
○ 3.	C		0.0034	
○ 4.	Be		0.0076	
○ 5.	F		0.0096	
6.	Bi		0.032	
○ 7.	^{7}Li	(92.5 %)	0.045	[Li 7.0]
8.	^{11}B	(80.2 %)	$<$0.05 (0.005)	
9.	Mg		0.06	

element		[barn]	element		[barn]
10.	Si	0.16	15.	H	0.33
11.	Pb	0.17	○ 16.	Rb	0.37
			17.	Ca	0.43
12.	Zr	0.18	18.	S	0.52
13.	P	0.185	○ 19.	Na	0.53
14.	Al	0.23	[	K	2.15]
	^{37}Cl (24%)	0.434	[	Cl	32.7]

Table 2.2 Candidate Target Salts and Their Predicted Properties

	mol %	melting point Tm (°C)	density at Tm +100°C (g/cm³)	viscosity coeff (cpoise) 600°C	viscosity coeff (cpoise) 700°C
$LiF-BeF_2-ThF_4$	72−16−12	500	3.35	12	7
〃	71−9−20	540	2.97	16〜18	7〜9
〃	67−18−15	500	2.70	13〜15	6〜7
〃	64−18−18	540	2.7	12〜14	6〜7
$LiF-ThF_4$	71−29	568	3,36	20〜22	12〜14
$LiF-NaF-ThF_4$	54.5-13.5-32	525	3.31	19〜22	11〜13
$NaF-KF-ThF_4$	11−67−22	535	2.54	14〜16	8〜10
$LiF-BeF_2-UF_4$	61−21−18	550	2.87	13〜15	6〜7
$LiF-UF_4$	71−29	525	3.41	20〜22	12〜14
$LiF-NaF-UF_4$	43.5-24.3-32.2	445	3.09	20〜22	12〜14
$LiF-RbF-UF_4$	60−10−30	460	3.52	16〜18	10〜12
〃	57−10−33	470	3.57	17〜19	11〜13
$NaF-RbF-UF_4$	45−27−28	500	2.99	15〜17	9〜11
$NaF-KF-UF_4$	47−20−33	550	2.89	18〜20	11〜13

Table 3.1 Comparison of Neutron Yields per 1 GeV Proton for Some of Target Salts Including Th and U(11)

case	molten salt	cascade + evaporation ($\geq$15 MeV)	cascade + evaporation + fission ($\geq$15 MeV)	whole energy range (1000 ~ 0 MeV)*
1	^7LiF-BeF_2**-ThF_4 (72-16-12)	22.3 ± 2.3	25.1 ± 3.0	
2	^7LiF-BeF_2**-ThF_4 (71-9-20)	26.1 ± 2.1	29.6 ± 2.1 30.9 ± 2.6***	30.7 ± 2.2 35.8 ± 2.7***
3	^7LiF-BeF_2**-ThF_4 (67-18-15)		28.6 ± 3.3	29.5 ± 3.4
4	^7LiF-BeF_2**-ThF_4 (64-18-18)		27.0 ± 2.0	27.8 ± 2.1
5	^7LiF - ThF_4 (71 - 29)	27.3 ± 2.2	33.04 ± 1.9	
6	^7LiF-NaF-ThF_4 (54.5-13.5-32)		32.4 ± 2.1	34.0 ± 2.2
7	NaF-KF-ThF_4 (11-67-22)		25.9 ± 3.0	
8	^7LiF-BeF_2-UF_4 (61-21-18)		31.2 ± 2.4	38.4 ± 3.0
9	^7LiF-UF_4 (71 - 29)	28.1 ± 2.8	33.2 ± 2.6	
10	^7LiF-RbF-UF_4 (60-10-30)	27.7 ± 1.7		
11	^7LiF-RbF-UF_4 (57-10-33)		36.0 ± 4.0	
12	NaF-RbF-UF_4 (45-27-28)	28.0 ± 1.8		

(*) Including fission in the energy range both above and below 15 MeV.

(**) The effect of Be(n,2n) reaction is not enoughly included yet.

(***) 1 % of ^{232}Th is replaced by ^{233}U.

Table 4.1 Example of Predicted Performance of Single-fluid-type Accelerator Molten-Salt Breeder (AMSB)

proton beam	1 GeV, 300 mA
salt example	$^{7}LiF\text{-}BeF_2\text{-}ThF_4$ 64-18-18 m/o
melting point (Tm)	540 °C
density at Tm + 100 °C	2.7 g/cm^3
viscosity coefficient	12 ~ 14 c poise (600 °C)
	6 ~ 7 c poise (700 °C)
salt temperature	inlet 580 °C outlet 680 °C
salt volume	150 m^3
salt weight	405 ton (Th 210 ton)
salt flow	5 m^3/sec
thermal output*	1200 ~ 2000 MWth
elec, power generation	500 ~ 800 MWe
elec, power consumption	~ 600 MWe
fissile material production**	800 ~ 1000 kg/year
	640 ~ 800 kg/year (80 % load)
spallation products	~ 40 kg/year
fission products	~ 40 kg/year

* M. Steinberg: CONF-770107 (1977) p.41.

** Except for the continuous removal, this will be increased more.

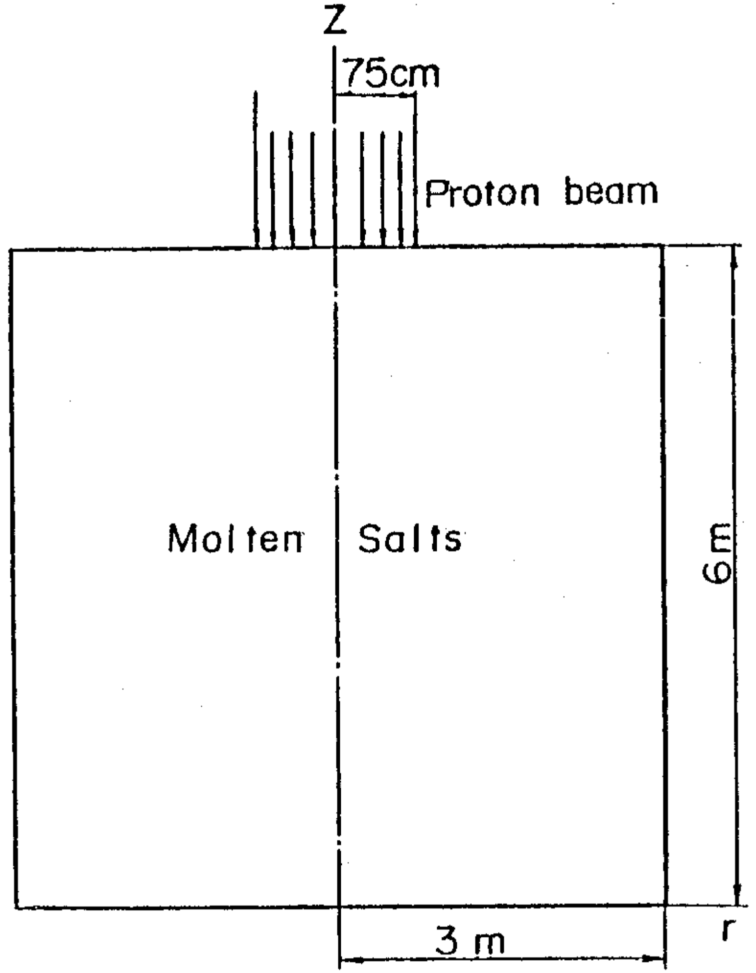


Fig. 3. 1

Cylindrical target model for neutronic calculations of AMSB (Accelerator Molten-Salt Breeder)

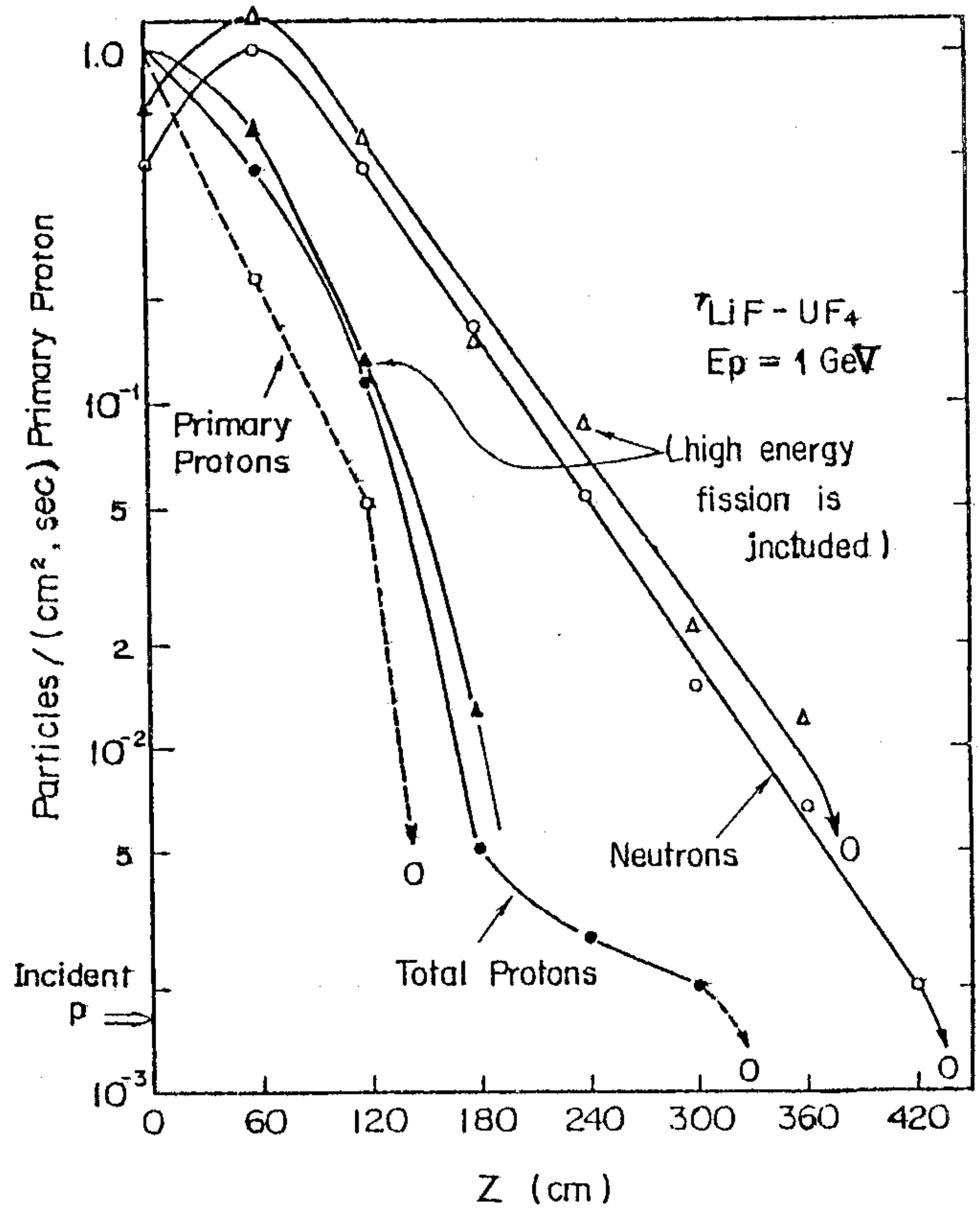


Fig.3. 2 Flux Distributions of Proton and Neutron on the Incident Proton Beam Axis. (Cascade - Evaporation (>15 MeV) Only)[11]

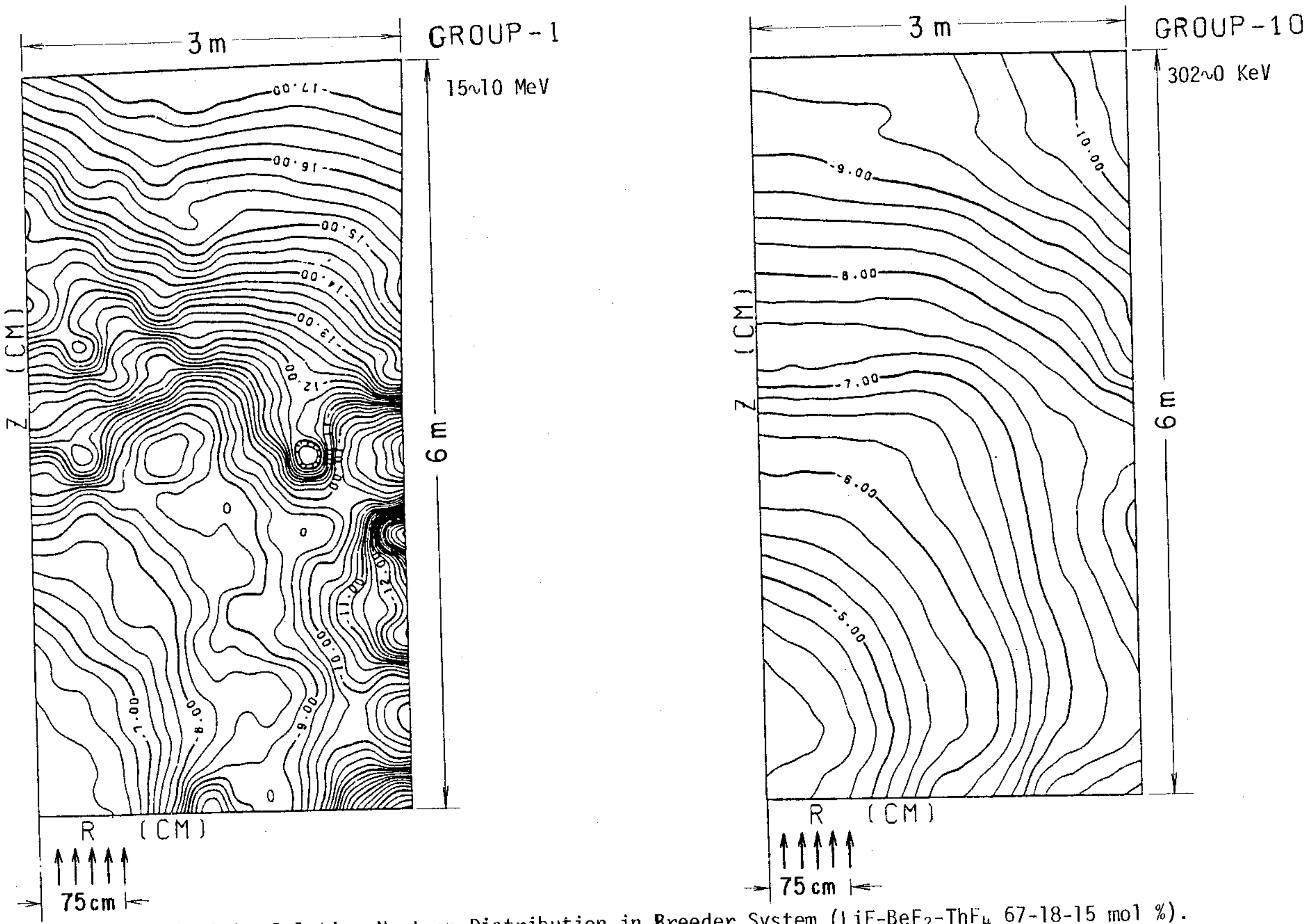


Fig.3.3 Relative Neutron Distribution in Breeder System (LiF-BeF_2-ThF_4 67-18-15 mol %).
[figures with minus sign are the x of 10^x] (11)

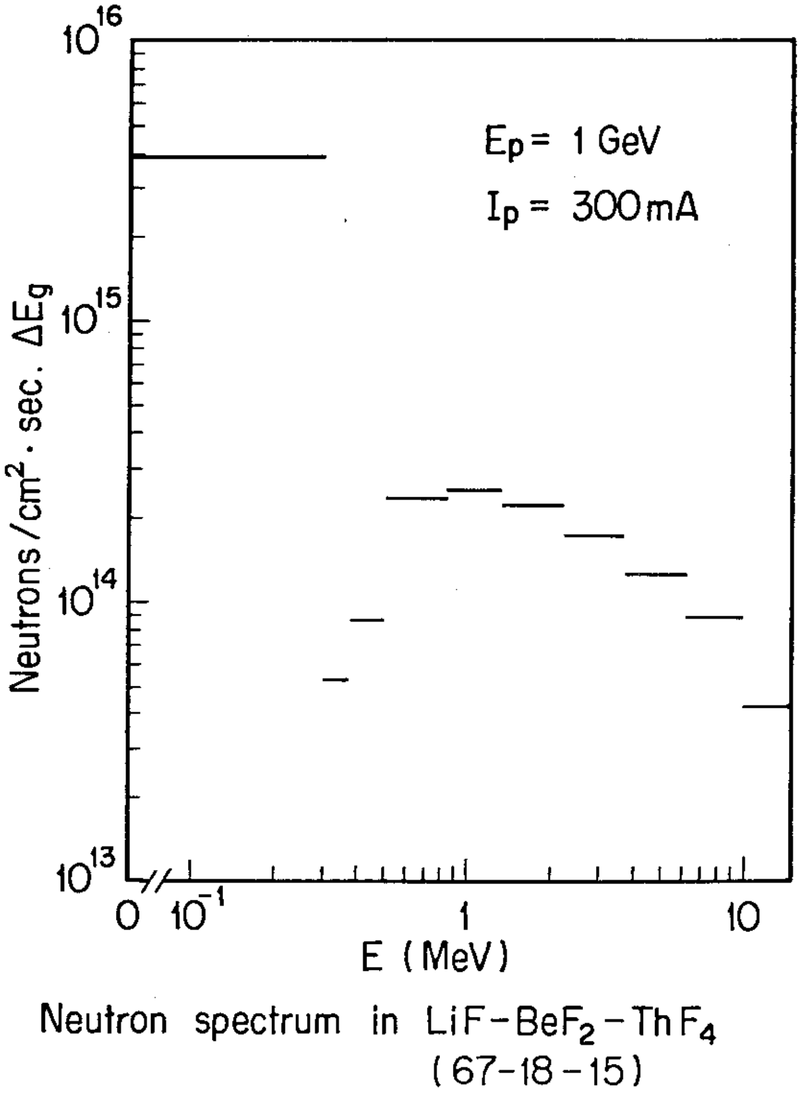


Fig.3.4 Energy Group Flux Distribution of Low Energy Neutrons ($\leq$ 15 MeV) for a Region in LiF - BeF$_2$ - ThF$_4$ (67-18-15), where the Spatial Neutron Distribution is Maximum.(11)

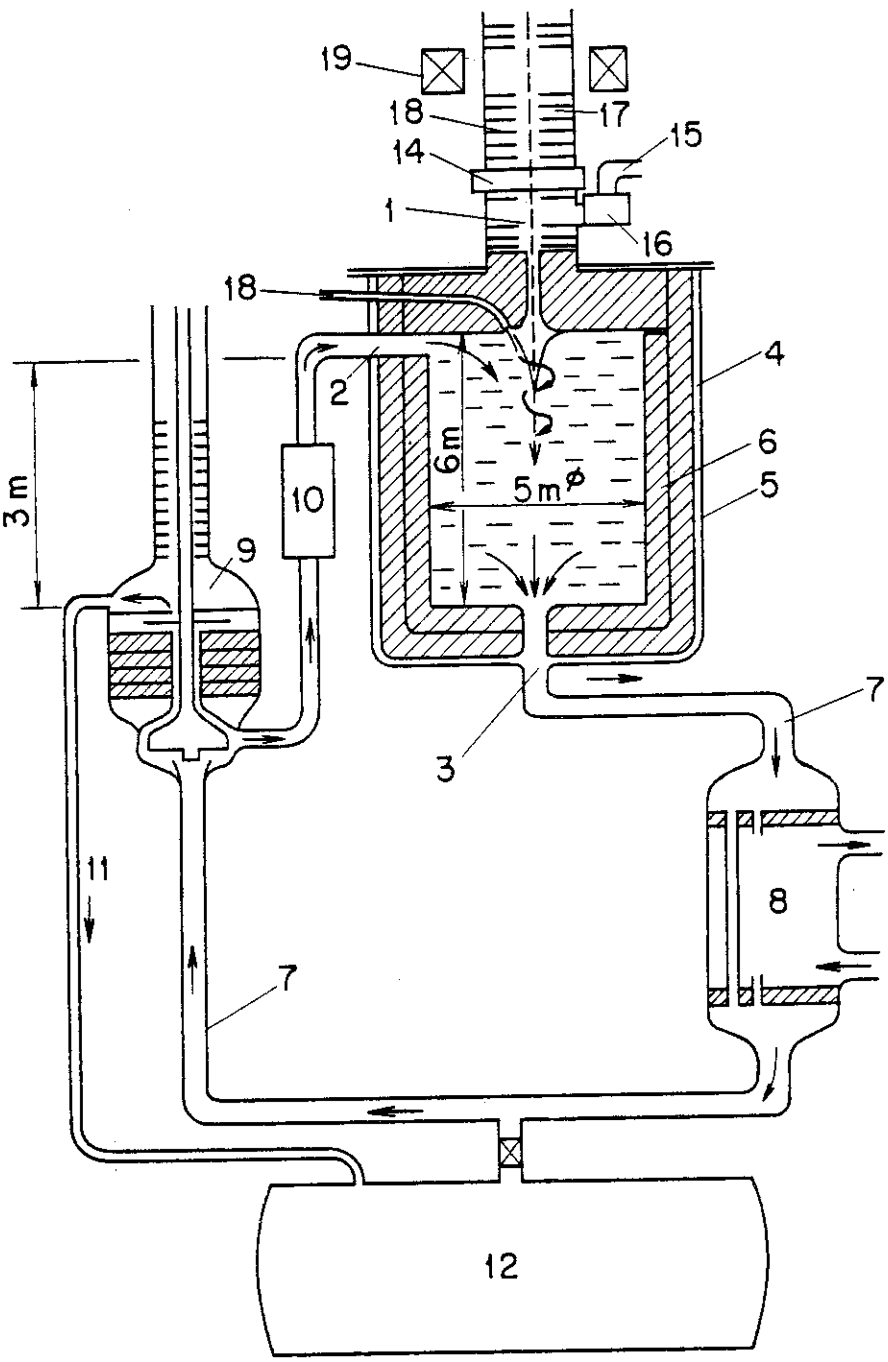


Fig.4.1 Schematic Figure of Single-fluid-type Accelerator Molten-Salt Breeder

1 proton beam, 2 salt inlet, 3 salt outlet, 4 reactor vessel, 5&6 graphite, 7 primary loop, 8 heat exchanger, 9 main salt pump, 10 throttle valve, 11 overflow line, 12 storage tank, 13 high-pressure salt outlet, 14 gate valve, 15 vacuum line, 16 vapor trap, 17 duct, 18 orifice, 19 focussing magnet.

Table A.1 Physical Properties of Typical Molten Materials

	Molten Salts (500°C)			Liquid Metals (500°C)			Molecular Liq.	
	LiF-NaF-KF (Flinak)	LiF-BeF_2 (Flibe)	$NaBF_4$-NaF *	Na	Li	Pb	Dowtherm A [339°C]	H_2O [300°C]
· chemical compo. (mol %)	46.5-11.5-42	66-34	92-8	—	—	—	75 w% $C_{12}H_{10}O$ + $C_{12}H_{10}$	—
· melting point (°C)	454	459	384	98	179	327.4	12	0
· vapor pressure (mm Hg)		<0.1 (458~649°C)	270 (627°C)	18 (548°C)	4.4×10^{-3} (515°C)	1.62×10^{-5}	4.64×10^{3}	6.44×10^{4}
· isobaric speci. heat (J/kg °C)	1.85×10^{3}	2.34×10^{3}	1.51×10^{3}	1.26×10^{3}	4.19×10^{3}	0.137×10^{3}	2.83×10^{3}	5.69×10^{3}
· density (kg/m^3)	2165	2050	1870	833	482	10450	750	712
· thermal conductivity (J/m °C S)	4.51	1.00	0.35	66	37.7 (220°C)	18.2	0.17 (300°C)	0.54
· kinematic visco. (m^2/S)	4.2×10^{-6}	7.44×10^{-6}	0.8×10^{-6}	0.29×10^{-6}	0.91×10^{-6} (285°C)	0.17×10^{-6}	0.44×10^{-6}	0.13×10^{-6}
· speci. heat capacity (J/m^3°C)	4.01×10^{6}	4.79×10^{6}	2.82×10^{6}	1.05×10^{6}	2.02×10^{6}	1.43×10^{6}	2.12×10^{6}	4.11×10^{6}

* Secondary coolant-salt of MSBR proposed by ORNL.

Table A.2 Technological Comparison of Several Kinds of Liquids

characteristics	molecular liq.	molten salts	liq. metals
a. usable temp range (°C)	<400	(100 ~ 3000)	
b. low vapor pressure	×	○	○
c. low viscosity	△	△	○
d. low electro-conduc.	○	△	×
e. high therm.-conduc.	×	△	○
f. low chem. activity	×	○	×
g. optical transparency	○	○	×
h. compatibility with structural metals	○	△	×

Table B.1 The Functions expected by Molten Salts

A. heat transfer, B. heat storage,
C. chemical processing, D. nuclear heat generation,
E. neutron generation, F. tritium production,
G. fission, H. production of fissile materials,
I. radio-waste incineration, J. radiation shielding.

Table B.2 Nuclear Applications of Molten-Salt Technology

	Purposes *	Functions **
(1) Molten-Salt Breeder Reactors[(a)]	fuel	A, C, G, H.
(2) Tritium Producing Molten-Salt Reactors[(b)]	〃	A, C, F, G, H.
(3) Actinoid Incinerating Molten-Salt Reactors[(b)]	〃	A, C, G, H, I.
(4) Molten-Salt Intense Neutron Source[(c)]	target	A, C, E, J
(5) Accelerator Molten-Salt Reactors (single-[(c)] or two-fluid[(d)])	(target (blanket	A, C, D, E, F, G, H, I. A, C, D, F, G, H, I, J.
(6) DT Fusion Reactors	blanket	A, C, D, F, J.
(7) DD Fusion Reactors[(e)]	〃	A, C, D, J.
(8) DT Hybrid Reactor[(f)]	blanket, fuel	A, C, D, E, F, G, H, I, J.
(9) Radioactive-waste Storage[(b)]	coolant	A, B, C, D, J.

*for the reactors (1)~(8) molten-salts will be applied as secondary coolants, too.

** referred to Table B.1

(a) see the reference (5) of ORNL in the main article,

(b) K.Furukawa : "Future Nuclear Appli. M.S. Tech."(in prep.); in the "Fundamentals in M.S. Thermal Tech." ed, T. Ishino (1980)

(c) this report. (d) see the refercnce (4) in the main article.

(e) Y.Nakao, H.Nakashima, M. Ohta & K, Furukawa : J. Nucl. Sci. Tech. 15 (1978) 76.

(f) V.L. Blinkin : Nucl. Fusion, 18 (1978) 893.

Table C.1 Computer codes for the nucleon transport calculations in the nucleon energy range above the cascade cut-off and process codes of history tapes[a]

NMTC/JAERI[a]	performs the intra- and inter-nuclear cascades, evaporation and fission calculations by Monte Carlo method and produces history tapes.
HIST3D/A[A1]	edits a history tape prepared by NMTC to make a source neutron distribution for the ANISN calculation.
HIST3D/B[A1]	edits a history tape prepared by NMTC to make a source neutron distribution for the TWOTRAN-II calculation.
NMTA[A2]	analyzes a history tape prepared by NMTC to compute (a) flux and current from boundary crossing, (b) flux and spectrum from collision density, (c) residual nuclei distribution, etc.

A1) H. Takahashi, J. Beerman, D. Hillman and Y. Nakahara; unpublished.

A2) T.W. Armstrong and K.C. Chandler; "Analysis Subroutines for the Nucleon-Meson Transport Code NMTC", ORNL-4736 (1971).

Table C.2 Computer codes for the neutron transport calculations in the nucleon energy range below the cascade cut-off [a]

SUPERTOG-JR3 [A3]	generates the cross sections for the neutron transport calculation, the energy deposition factor and the atomic displacement factor.
TAPEMAKER [A3]	collect group cross sections for each element or material generated by SUPERTOG-JR3.
TOTAL COUPLE [A3]	makes a coupled neutron and gamma-ray multigroup cross sections and regions-wise macroscopic cross sections.
GROUP INDEPENDENT [A3]	selects the cross section tables for required materials and produce a group independent cross section tape in order to obtain forward or adjoint solutions by ANISN effectively.
ANISN [A4]	solves the one-dimensional neutron and gamma-ray transport problem by the S_n method and produces the collapsed effective cross sections for heterogeneous zones.
TWOTRAN-II [A5]	solves the two-dimensional neutron transport problem by the S_n method.

A3) K. Koyama et al.; "RADHEAT-V3, a Code System for Generating Coupled Neutron and Gamma-ray Group Constants and Analyzing Radiation Transport", JAERI-M 7155 (1977)

A4) W.W. Engle, Jr.; "ANISN, a One Dimensional Discrete Ordinate Transport Code", RSIC ORNL, CCC-82.

A5) K.D. Lathrop and F.W. Brinkley; "TWOTRAN-II; An Interfaced, Exportable Version of the TWOTRAN Code for Two-Dimensional Transport", LA-4848-MS (1973).

Table D.1 Significant Elements in the Spallation Reaction Products

(target)	products from (p,n)(p,2n)(p,3n)(p,α)(n,p)(n,α) etc.	evaporation cascades	fission products
$^{238}_{92}U$, $^{235}_{92}U$	$\underline{\mathrm{Pu}}$, $\underline{\mathrm{Np}}$, $\underline{\mathrm{U}}$, $\underline{\mathrm{Pa}}$,	$\underline{\mathrm{Ac}}$, $\underline{\mathrm{Ra}}$, $\underline{\mathrm{Fr}}$, $\underline{\mathrm{Rn}}$, $\underline{\mathrm{At}}$, Po, $\underline{\mathrm{Bi}}$, $\underline{\mathrm{Pb}}$, $\underline{\mathrm{TL}}$ ·····	mass
$^{232}_{90}Th$	$\underline{\mathrm{U}}$, $\underline{\mathrm{Pa}}$, $\underline{\mathrm{Ac}}$, $\underline{\mathrm{Ra}}$,	$\underline{\mathrm{Fr}}$, $\underline{\mathrm{Rn}}$, $\underline{\mathrm{At}}$, Po, $\underline{\mathrm{Bi}}$, $\underline{\mathrm{Pb}}$, $\underline{\mathrm{TL}}$, Hg, Au, ····	60～140
$^{87\sim}_{37}Rb$	$\underline{\mathrm{(Y)}}$, $\underline{\mathrm{Sr}}$, $\underline{\mathrm{Kr}}$, $\underline{\mathrm{(Br)}}$,	$\underline{\mathrm{Br}}$, Se, As, Ge, Ga, Zn, ····	
$^{23}_{11}Na$	$\underline{\mathrm{Mg}}$, $\underline{\mathrm{Ne}}$, $\underline{\mathrm{(F)}}$,	$\underline{\mathrm{F}}$, O, N, C	
$^{19}_{9}F$	$\underline{\mathrm{Ne}}$, O, (N),	N, C, $\underline{\mathrm{B}}$, $\underline{\mathrm{Be}}$, $\underline{\mathrm{Li}}$, $\underline{\mathrm{He}}$, $\underline{\mathrm{T}}$, $\underline{\mathrm{H}}$	
$^{9}_{4}Be$	$\underline{\mathrm{B}}$, $\underline{\mathrm{Li}}$, $\underline{\mathrm{He}}$, $\underline{\mathrm{T}}$, $\underline{\mathrm{D}}$, $\underline{\mathrm{H}}$	———	
$^{7}_{3}Li$	$\underline{\mathrm{(Be)}}$, $\underline{\mathrm{He}}$, $\underline{\mathrm{T}}$, $\underline{\mathrm{D}}$, $\underline{\mathrm{H}}$	———	

underline : Classes (A) ～ (D) of Table D.2

Table D. 2 Classification of Spallation Reaction Products

(A) actinides : Pu, Np, U, Pa, Th, Ac,

(B) rare-gas elements : Rn, Xe, Kr, Ne, Ar, He,

(C) hydrogen : T, D, H

(D) stable salt elements : Ra, Fr, At, Sr, Bi, Pb, Br, Mg, F, B, Be, Li and the other Ia, IIa, IIIa, IVa, VIIb elements

(E) noble metals : Au, W, Mo etc,

(F) the others including fission products

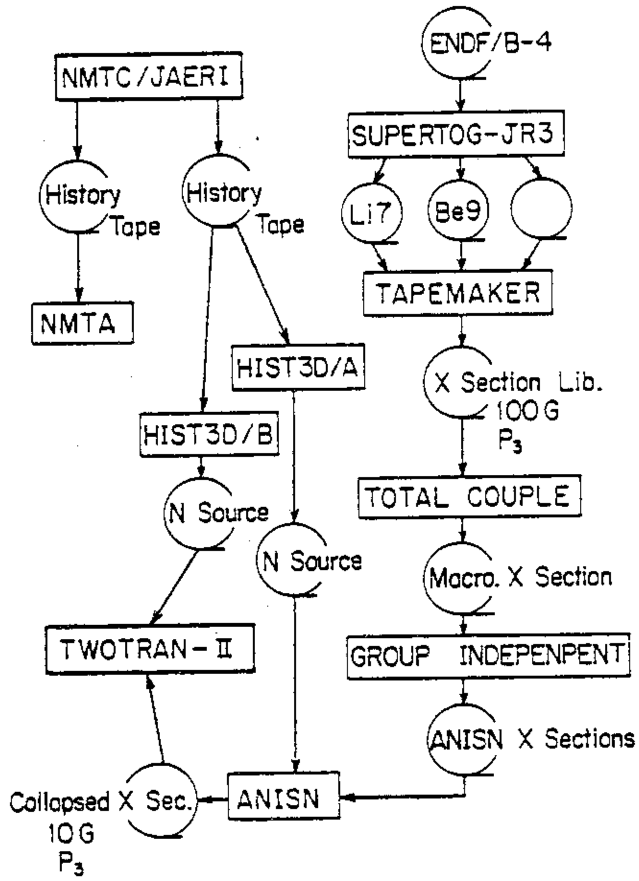


Fig. C.1 Structure of the Computer Code System ACCEL and Flow of Neutronics Calculations.[a]

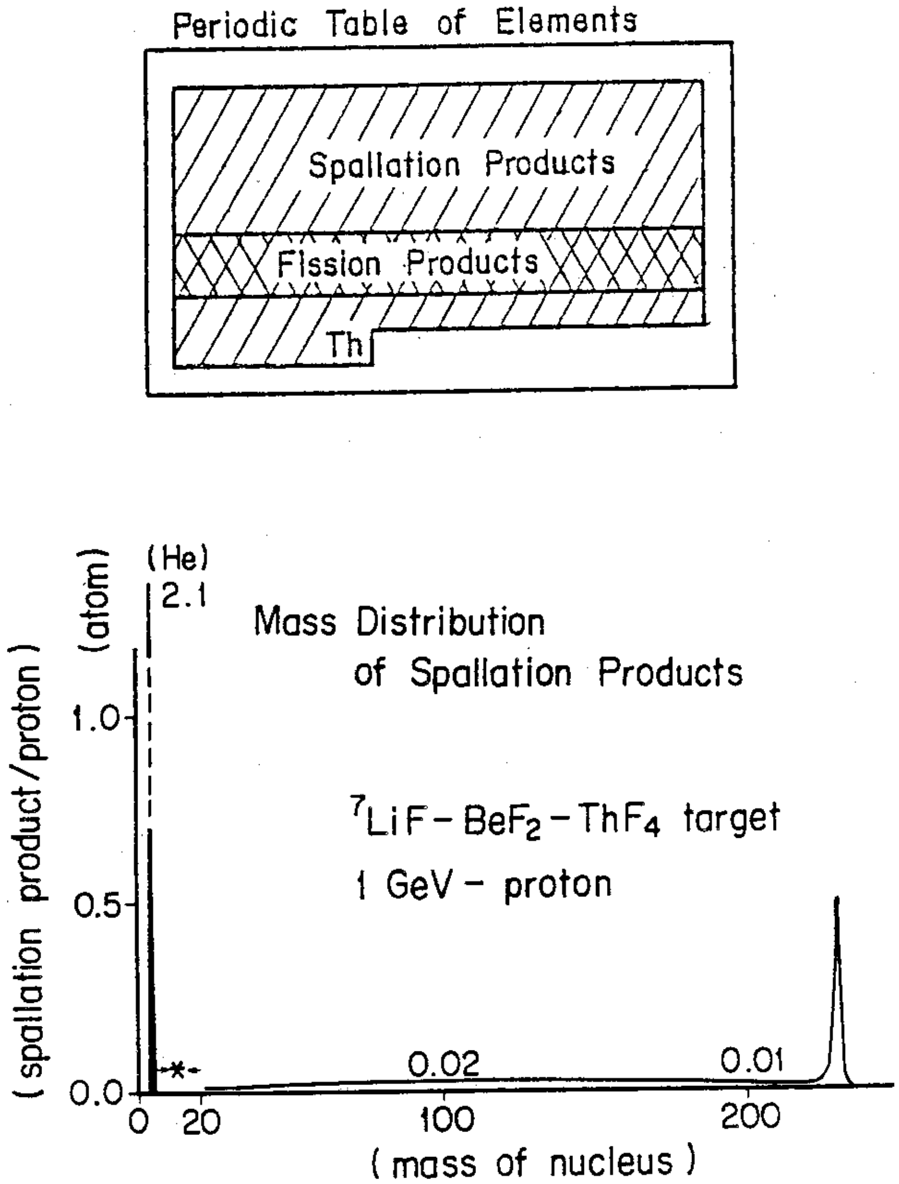
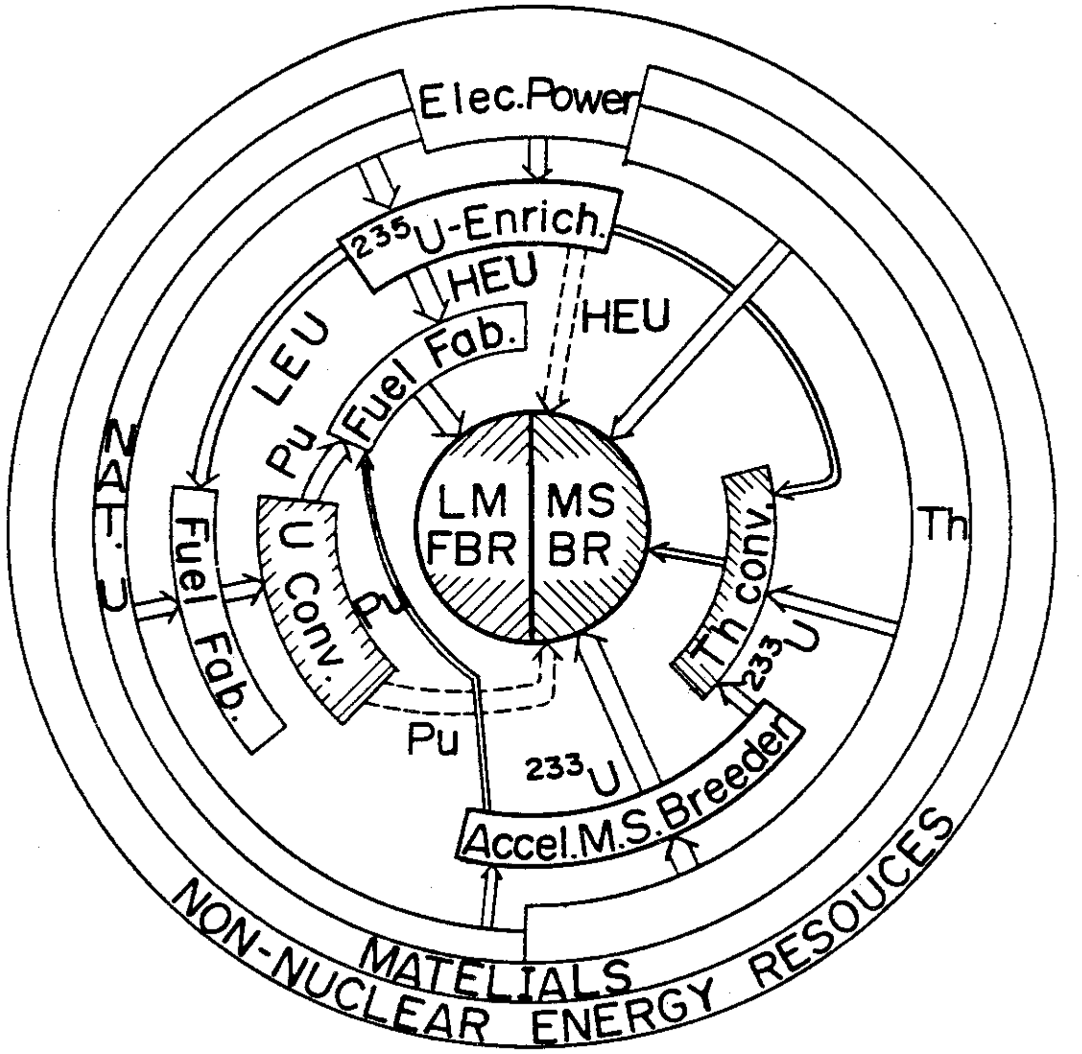


Fig. D.1 Preliminary Presentation of Distributions Spallation Reaction Products. [*The region of low masses is not calculated yet.]



Enrich. : Enrichment Plant

HEU : High enriched Uranium

LEU : Low " "

Fab. : Fabrication Plant

Conv. : Converter Power Reactors

LMFBR : Liquid Metal cooled Fast Breeder Reactor

MSBR : Molten-Salt Breeder Reactor

Fig. E.1 Nuclear Fuel Cycles and the Contributions of Accelerator Molten-Salt Breeder.