ORNL/TM-5750
Distribution Category
UC-34c

Contract No. W-7405-eng-26

Neutron Physics Division

PRELIMINARY REPORT ON THE PROMISE OF ACCELERATOR BREEDING AND CONVERTER REACTOR SYMBIOSIS (ABACS) AS AN ALTERNATIVE ENERGY SYSTEM

ABACS Study Group

F. R. Mynatt, Study Group Leader
Target Physics Studies - R. G. Alsmiller, Jr., J. Barish,[1] and T. A. Gabriel
Nuclear Engineering - D. E. Bartine and T. J. Burns
Accelerator Design - J. A. Martin and M. J. Saltmarsh
Heat Transfer and Mechanical Design - E. S. Bettis[3]

Date Published - February 1977

[1] Computer Sciences Division
[2] Physics Division
[3] Consultant

NOTICE This document contains information of a preliminary nature. It is subject to revision or correction and therefore does not represent a final report.

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

DISCLAIMER

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

LIST OF TABLES (Cont'd)

ABSTRACT

A preliminary study has been performed to evaluate the promise of accelerator breeding and converter reactor symbiotic systems (ABACS) as an alternate fission power technology which can make full utilization of the energy content of uranium and thorium ores. ABACS is, therefore, considered as an alternative to fast breeder reactors for extending our energy supply An explanation is given of the fundamentals of accelerator breeding in which U^{233} or Pu^{239} fissile fuel is produced in a target/blanket system as a result of irradiation with an intense high-energy proton beam. Neutronics and heat transfer analyses are performed for three accelerator breeder concepts based on technologies of the liquid metal fast breeder, molten salt, and gas-cooled fast breeder reactors. Several converter reactors are considered, and the mass flows and economics of the complete symbiosis are presented. Particular attention is given to the potential advantages of ABACS relative to the fast breeder reactor in the areas of inherent safety and in the implementation of the U^{233}-U^{238} denatured fuel cycle as a proliferation and diversion deterrent. Advantages and disadvantages of the present accelerator breeder concepts are summarized and development needs are indicated.

I. THE POTENTIAL ROLE FOR ACCELERATOR BREEDING

Fission power technology offers the world a long-term supply of abundant electrical power and process heat at generation costs lower than that for current coal and oil-fired central power plants and with essentially no environmental insult in normal operation. Why, then, are we turning to high-cost, long-term development options such as solar and thermonuclear and planning extensive use of coal with its substantial environmental insult both in mining and in power generation? The reason, of course, is that the public is concerned about fission power technology in abnormal circumstances of accident, sabotage, etc. and in long-term problems associated with nuclear waste. Furthermore, the public has become involved through political, judiciary, and regulatory channels to the extent that the great promise and performance of fission power is lost in concern over potential perils. The major problems of fission power technology are listed in Table I-1. While these may not be all of the problems which have led to doubt, regulatory delays, and capitalization problems which diminish the promise of fission power, the point of reviewing them here is that implementation of the Accelerator Breeder Concept may provide some amelioration of these problems. The importance of accelerator breeding is that (1) it offers an alternative to the fast-breeder in that both allow full energy utilization of uranium and thorium ores, (2) it offers safety advantages over fast breeders, and (3) it provides a source of U^{233} without using U^{235} and facilitates fuel cycles which lessen the threat of fuel diversion and greatly reduce actinide wastes. This paper summarizes the results to date of the ORNL study group, and reviews the potential performance and role for ABACS (accelerator breeder and converter reactor symbiosis).

Table I-1. Fission Power Technology Problems

---

1. The threat of proliferation of nuclear weapons through diversion of reactor materials and low-technology processing and fabrication.

2. The concern that nuclear wastes may find their way into the ecosystem. The elimination or permanent isolation of the actinides are of particular concern.

3. Concerns regarding the production and use of plutonium and the possibly greater accident risk associated with the LMFBR or GCFR.

4. Concern over the shortage of known high-grade uranium ores and the uncertain social, environmental, and economical problems of mining low-grade ores. This becomes particularly important if the high-gain breeder is not developed or is late in implementation.

The production of fissionable materials with an accelerator is not new. The first minute quantities of Pu^{239} and other transuranic materials were produced in accelerators. The use of accelerators for intense neutron sources is also not new and several concepts have been proposed.[1] Why the current interest in accelerator production of fission reactor fuel? The reasons are that the problems listed in Table I-1 are newly emphasized, and the technology to produce high neutron production accelerators is now more promising than in the past. Nevertheless, even with current estimates, the plutonium fast breeder reactor appears more economical than an accelerator system. However, the current fission reactor technology requires the development and use of the plutonium fast breeder for long-term energy production and therefore requires direct solution of the problems on Table I-1. The Accelerator Breeding Concept (ABC) allows alternatives which in principal permit elimination of these problems. We have thus far not conceived of a practical system which alleviates both the safeguards and waste problems, but either one could be greatly alleviated.

Figure I-1 shows schematically the basic processes in accelerator breeding. One accelerates a proton (or a deuteron) to an energy of approximately 1 GeV and directs it onto a target. Interactions with target nuclei produce many secondary particles in a cascade with ultimate production of 20 to 50 neutrons. Experiments have been performed to establish this observation and to determine that the proton energy should be 1 GeV or greater to achieve efficient neutron production.[2]

ORNL-DWG 76-19899

Leakage + Removal = Scattering + Fission + External Source

$$\Omega \cdot \nabla \phi + \Sigma_T \phi = \int \Sigma_S \phi' + X \int \nu \Sigma_f \phi' + S$$

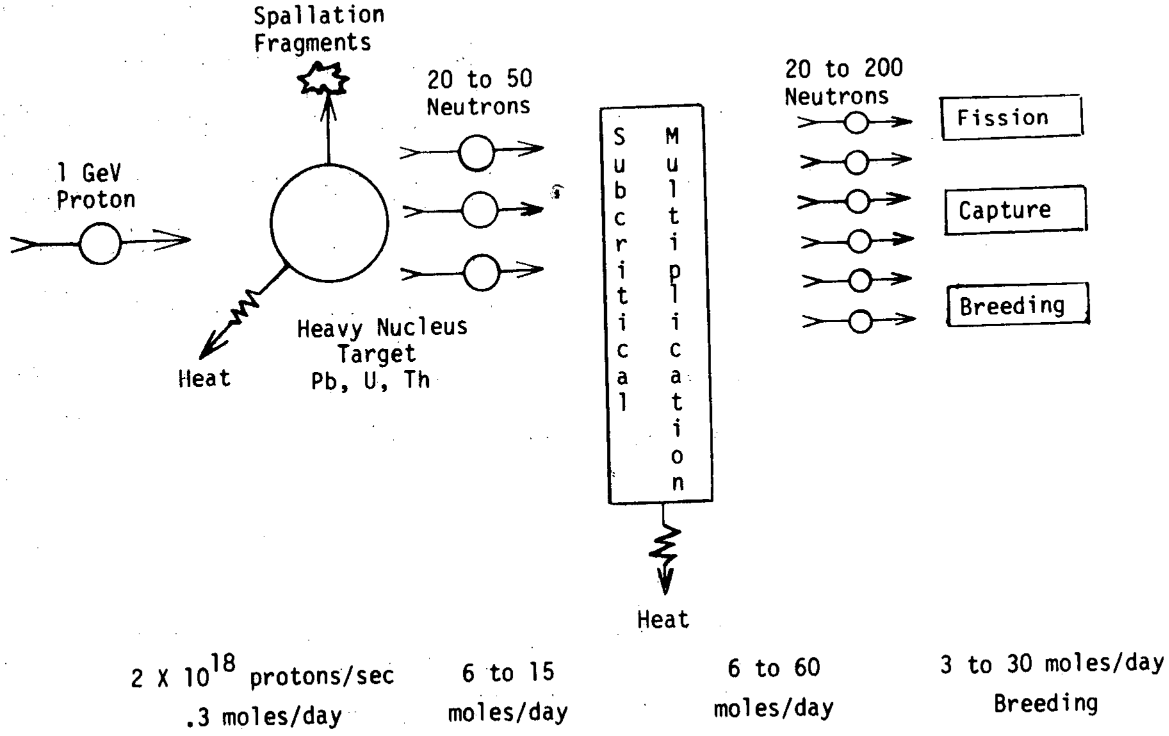


Fig. I.1. Schematic of Accelerator Breeding Process

These data are shown on Figures I-2 and I-3. Analysis of these experiments have demonstrated that the accuracy of calculations using Monte Carlo methods is approximately ±20%.[3]

When one considers a target/blanket of Th^{232} or U^{238}, most of the neutrons are captured producing U^{233} or Pu^{239}. The presence of this fissile material and fast fission of U^{238} lead to multiplication in the blanket which is assumed to be subcritical so that a finite number of daughter neutrons is produced. Since we will later show that one must operate with low multiplication, Figure I-1 indicates that the total number of neutrons is in the range from 20 to 200 per incident proton. These neutrons are absorbed in fission reactions, non-breeding capture, and in the fertile material to produce U^{233} or Pu^{238}. The net fuel production is given by the breeding captures less the fission of bred fuel (not U^{238} fast fission). At the bottom of Figure I-1, we see that a beam of 300 mA of 1 GeV protons (2×10^{18} protons/sec) leads to 3 to 30 moles/day of breeding product. For Pu^{239}, this would be from 0.7 kg/day to 7 kg/day. Heat is given off by the target interactions (the beam power is recovered) and by fission events.

A difference between reactor physics and accelerator breeder physics is that in a critical reactor the power level is essentially arbitrary and is controlled through reactivity control, while in an accelerator breeder, the power level is determined by the accelerator neutron source and the multiplication determined by the target/blanket geometry and enrichment. As fissile material is created the power increases, and to hold the power constant, the fissile material must be continuously removed or control poisons introduced. The accelerator power could be reduced, but fuel production would be proportionally

ORNL-DWG 65-7146A

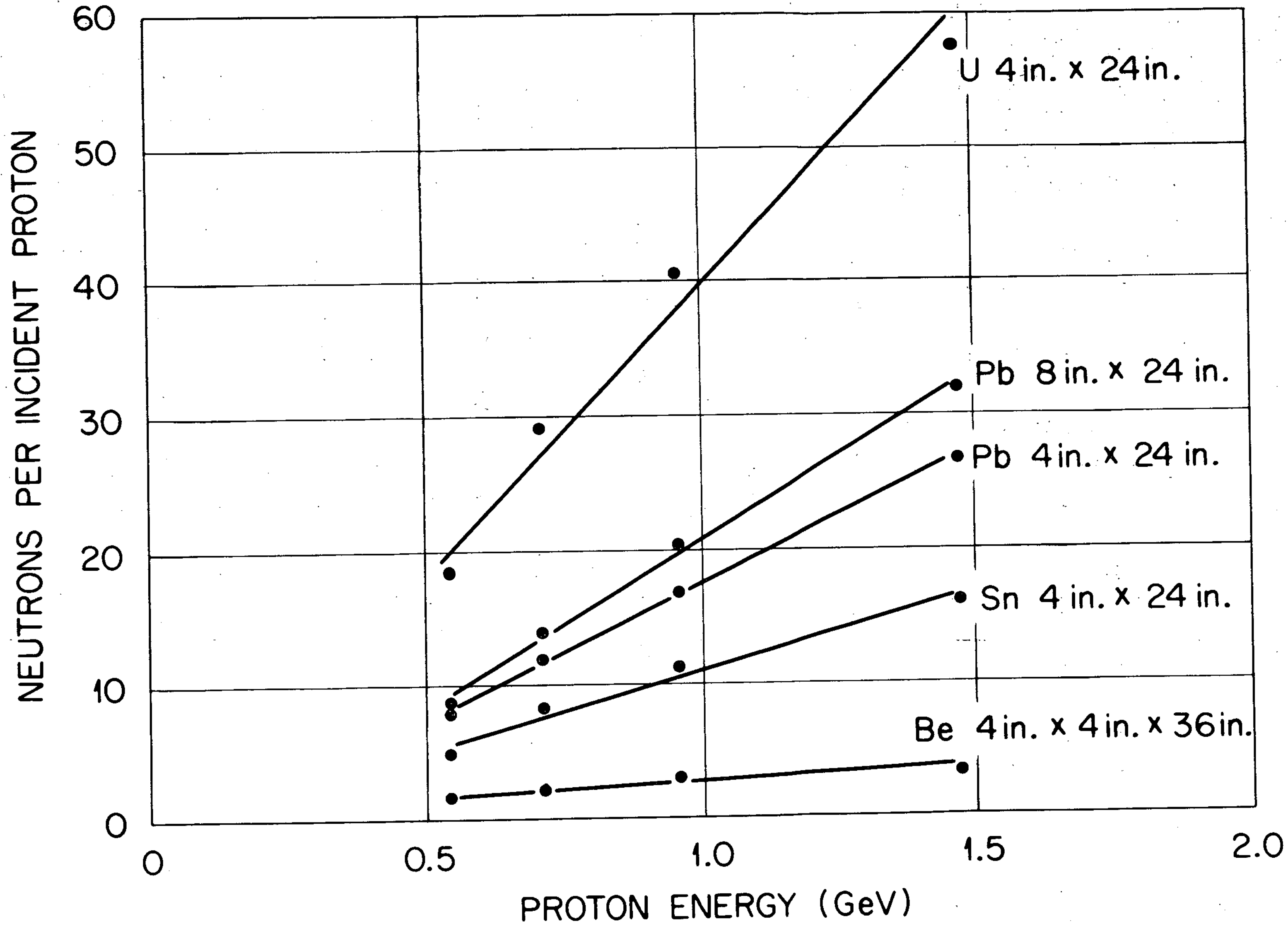


Fig. I.2. Neutron Yield as a Function of Target Material and Proton Energy

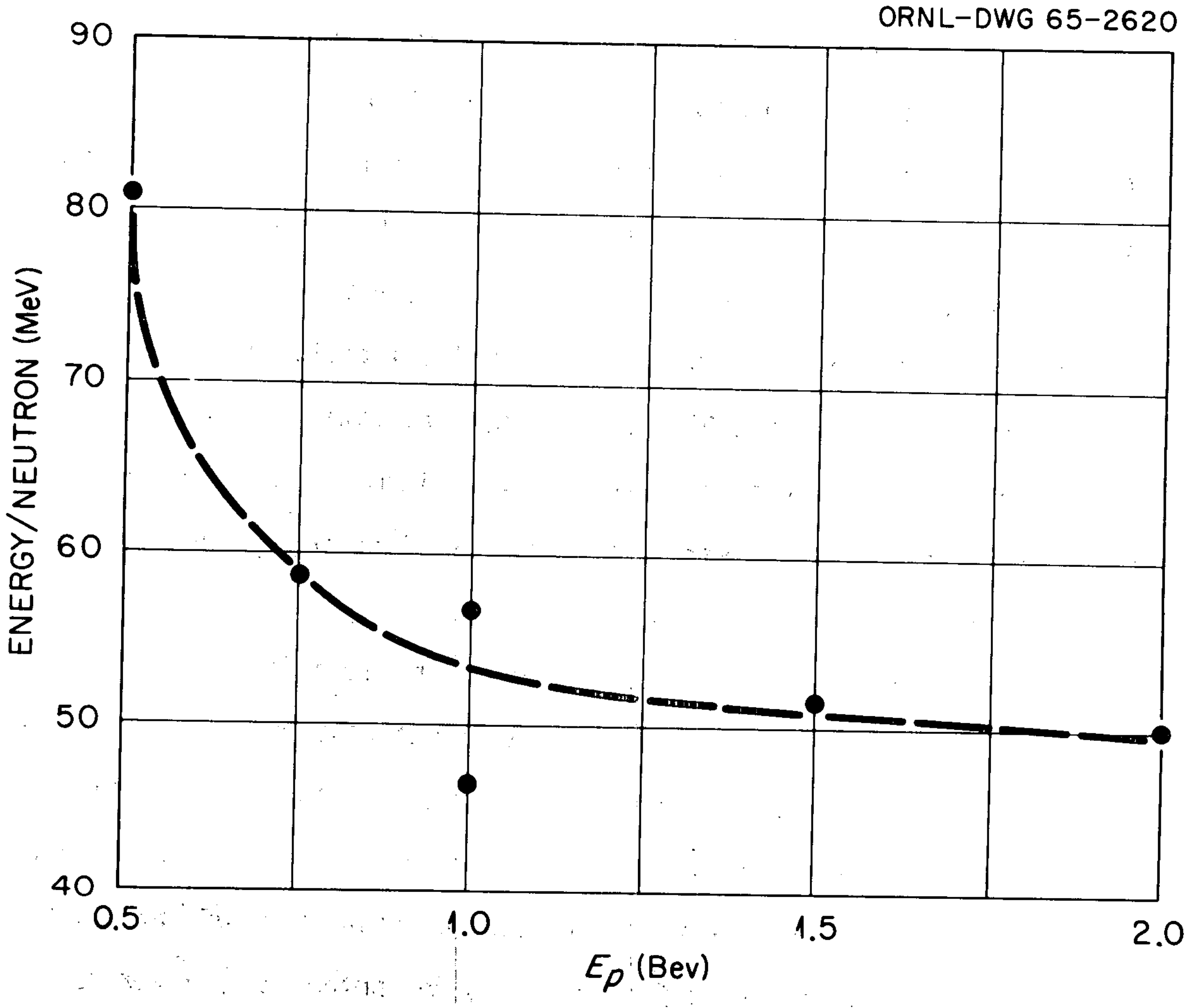


Fig. I.3. Energy Required to Liberate Low-Energy Neutrons by High-Energy Proton Bombardment of Pb Target reduced. A nearly constant power level is needed to allow efficient design of the heat exchangers, turbogenerators, cooling towers, etc. which make up a large part of the plant cost. This difference accelerator breeders and fission reactors is a major concern and to a large part dictates the mode of operation. For example, fuel handling should be continuous or nearly so. One cannot leave the blanket material fixed for six months or a year. Also with an accelerator source of 5 x 10^{19} neutrons/sec, a subcritical multiplication factor of two implies a fission power of about 700 MW thermal, a multiplication of four implies a fission power of about 2000 MW, and a multiplication of ten gives 6400 MW. Therefore, one cannot operate at low enrichment and high power or high enrichment and low power. This will be discussed further in the analysis of specific target/blanket systems.

The primary interest in the accelerator breeder is that one can produce U^{233} or Pu^{239} directly from fertile material without burning U^{235} or Pu^{239} fuel. Particularly with U^{233} production, this gives a fission power alternative remarkably different from present systems, all the way from ore mining to electricity generation yet using much of the existing technology. Also, some U^{233} fuel cycles offer potential safeguards improvements. The questions remain as to whether the accelerator breeder system is of reasonable scale and economy in performance and of whether the advantages warrant development.

In order to address these questions, a reference concept was specified for analysis. The concept is plausible through maximum use of existing technology but is not intended to represent an optimum design. The purpose was to specify a concept and perform sufficient analysis to determine performance levels, sensitivity to design variables, and to uncover potential problems. Because of the complexity of the accelerator breeder system, we turned immediately to analysis by sophisticated computer codes rather than analytic estimates.

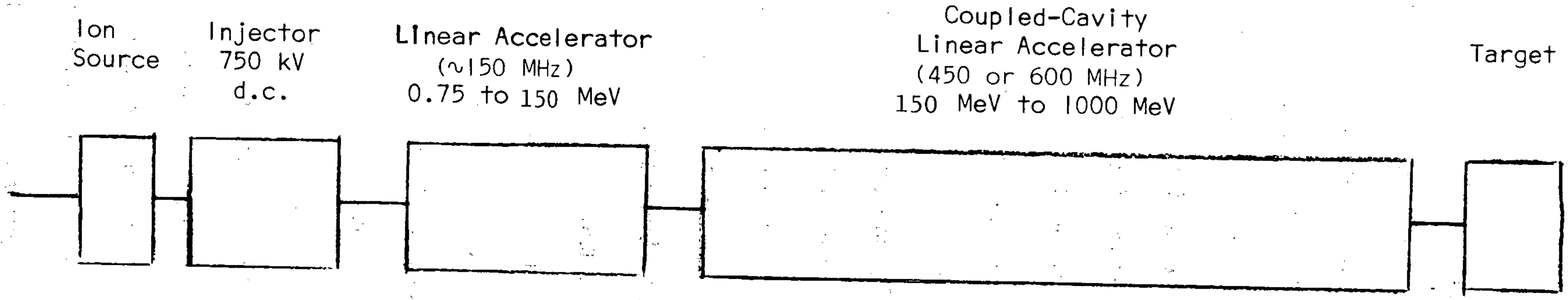
II. ACCELERATOR CONCEPTS

The accelerator for the electronuclear breeder should be able to deliver a continuous beam of 300 milliamperes of 1000 MeV protons to a target/blanket assembly. Linear accelerators are well suited to this application because of high current capability and low beam losses. Operating accelerators already approach the desired specification in output energy; lower energy accelerators approach the desired beam current on a pulsed basis. Design studies for accelerators with similar characteristics suggest that obtaining the beam current on continuous basis will require no more than modest extensions of existing technology. Development of the technology for minimization of beam losses is essential.

The Reference Concept

The reference concept for the accelerator is derived from the Intense Neutron Generator (ING)[4] design developed at the Canadian Chalk River Nuclear Laboratories in the period 1963-1967. That accelerator was designed to deliver a continuous current of 65 milliamperes (mA) to a bismuth-lead eutectic target assembly. The ING accelerator was patterned after the 800 MeV proton linear accelerator of the Los Alamos Meson Physics Facility (LAMPF) that will deliver a 16 mA pulsed beam with 6% duty factor (1 mA average current).

The reference accelerator concept (shown in Figure II-1) includes the following principal sections: (1) the ion source and high voltage dc injector, (2) several sections of Alvarez-type (drift tube) accelerator to accelerate protons in the low velocity range up to 150 MeV ($\beta = v/c = 0.51$), and (3) a high velocity coupled-cavity structure such as the LAMPF developed side-coupled cavity structure to accelerate the



Characteristics

Voltage gain:		1 MeV/m
Length:		1000 m
Beam Current:		300 mA continuous
RF Power		330 MW
Cavities	30 MW	
Beam	300 MW	
RF Efficiency:		50%
Input Power		∿660 MW

Fig. II.1. Outline Characteristics of Breeder Accelerator protons to the final energy of 1000 MeV (β = 0.88).

The ion source and injector system would require additional development of existing concepts. Injection systems for linear accelerators operated in the pulsed mode, as used for synchrotron injection, can routinely deliver peak currents substantially in excess of 250 mA. Development of sources for operation in the continuous mode to provide analyzed currents in the 500-600 mA range would follow the experience gained at Oak Ridge in CTR ion injection programs. High reliability for the ion source system would be an important development goal. Multiple ion sources would probably be provided to eliminate significant lost time from source maintenance.

The low energy stages use the Alvarez-type linear accelerator which is well suited to the acceleration of very large beams of low velocity ions. For this application the characteristics of the Alvarez section such as rf frequency and energy gain rate should be carefully chosen and the design of the ion source, dc injection optics, buncher and early stages of the linear accelerator would be coordinated to achieve the required beam current capability with extremely high reliability.

Linear accelerators now in operation are capable of delivering the required currents. The Fermilab 200 MeV linac operating at 200 MHz delivers 270 mA routinely; plans have been made to go to 350 mA.[5] Theoretical studies by Batchelor [6] have shown that a 200 MHz linac could accelerate 400 mA with somewhat extreme conditions of focusing lens strength. Batchelor's results also suggest that a lower frequency should be used in a conservative high-reliability design. We believe that a frequency of about 100 or 150 MHz may prove to be an acceptable choice. The rf power amplifier tubes now used in pulsed mode at 200 MHz could be used in this application but may require derating for CW operation. A power amplifier development program would probably prove cost effective but may not be essential.

The high energy stages of the linear accelerator would employ a coupled cavity accelerator structure which can be designed to have a much better efficiency for acceleration at high energy. The side-coupled cavity as developed for the LAMPF project[7] or a variant of that structure can be used to provide efficient acceleration in the range from 150 MeV to 1000 MeV.

The frequency of the coupled-cavity stages must be an exact multiple of the frequency of the Alvarez sections, the maximum usable frequency being limited by phase-space considerations. Although both the rf power required and the cost of cavities decrease with increasing frequency, the upper limit for a reliable high current accelerator would probably be no more than 600 MHz,[8] and 450 MHz appears to be a better choice.

The power source for the coupled-cavity stages would probably be a klystron-amplifier especially developed for CW operation and high anode efficiencies. The efficiency of the 800 MHz klystrons used at LAMPF is about 40%.[9] The 353 MHz amplifier being developed for the PEP project[10] will each provide 500 kW at an efficiency in the 60-70% range.

Accelerator Development Requirements

Design of the accelerator system appears to be straightforward but will require attention to a different set of priorities than have been followed in the past. In particular, beam control and minimization of losses will be essential to prevent overheating and large residual induced radioactivities in the accelerator and beam transport structure.

The cavity excitation power, $\sim$ 30 MW, is only about 10% of the beam power. Hence for optimizing the accelerating structure insensitivity to beam loading will be more important than power losses. Reliability will be extremely important in every aspect of the design. Development and testing programs will be required to assure that each component and system will perform as required with known reliability.

The efficiency of conversion of ac power to rf power for the high frequency coupled cavity stages will be the single most important factor influencing overall accelerator efficiency. A power amplifier development program for the coupled-cavity section would aim toward producing highly reliable CW klystrons with efficiency in the 70-80% range. To some extent the frequencies used in the accelerator may be dependent on the constraints imposed by power amplifier design. This suggests that the accelerator physics and power amplifier development efforts be very closely coupled. The principal major development needs are listed in Table II-1.

Accelerator Costs

In the absence of a detailed design, we have taken the cost estimate prepared for the Canadian ING project prepared in 1967 and appropriately modified it to account for escalation of costs and differences in rf power requirements. We believe that this procedure gives a representative cost that will be useful for planning until an outline design study can be completed. Table III-2 gives the cost estimate.

Table II-1. Breeder Accelerator - Development Requirements

1. Develop reliable ion source and injection systems capable of delivering at least 300 mA of analyzed and bunched beam to a linear accelerator.

2. Develop a linear accelerator system using an Alvarez section (nominal frequency 150 MHz) and a coupled-cavity system at a higher multiple (nominally 600 MHz) which will provide "lossless" acceleration of protons to an energy of 1 GeV. System reliability must be a primary goal.

3. Develop appropriate ac to dc and dc to rf conversion systems to provide efficiency and reliable conversion of ac to dc power with a goal of achieving an overall conversion ratio of 75%.

4. Develop a control system that will provide for fast shutdown in case of fault and will provide continuous monitoring of beam loss and systems status.

5. Develop a beam transport system that will safely and reliably deliver the 300 mA beam to the target-blanket assembly with the appropriate beam-shape specifications and with acceptable losses.

Table II-2. Accelerator Cost Estimates

Reference: ING Study (1967 AECL)
INGRID Proposal (1976 ORNL)

Accelerator Structure & Building	$ 70,000,000		
RF Gallery, Mechanical & Electrical Services	74,000,000		
RF Power @ $0.50/watt	165,000,000		
TOTAL		$300,000,000	
10% Contingency		30,000,000	
TOTAL DIRECT COSTS			$330,000,000
40% Indirect (Eng., Man., Etc.)			132,000,000
30% Interest (9%, 5 Year Const.)			138,000,000
TOTAL PROJECT COST			$600,000,000

III. ANALYSIS OF A REFERENCE ABACS DESIGN WITH LMFBR TYPE BLANKET

A reference accelerator-target-blanket design was analyzed in order to identify the potential performance, the most important problems, the sensitivity of performance to system parameters, and a preliminary estimate of fuel costs. The reference design which was analyzed is shown in the schematic drawing on Figure III-1. The accelerator design and performance parameters are those given in Section II of this paper. The target and blanket are of heterogeneous design to optimize their separate functions and to make use of ING design for the target and LMFBR design for the blanket. Lead is the choice target material because it has a high neutron output relative to energy deposited in the target; and, as a circulating liquid, the target heat can be removed by reasonable mass flow rates. It was assumed that heat removal from a small uranium target was impossible and that the resultant Pu^{239} production might be unwanted in a U^{233} breeder. The target is in optimum geometry for neutron output. The blanket is a cylindrical annulus surrounding the target and is composed of 50% UO_2 by volume with coolant and structure omitted for preliminary analysis. Although Th^{232} is the fertile material of greatest interest, U^{238} was calculated because of availability of processed cross sections.

Neutronics

Two types of calculations were performed. Based on previous HETC[11] calculations, a 1/E source spectrum was assumed for the energy range of 15 MeV to 1 keV for use in a large sequence of ANISN[12] calculations which were performed to investigate the performance characteristics of the blanket. Also a HETC-MORSE[13] calculation was performed starting with

ORNL-DWG 76-19896

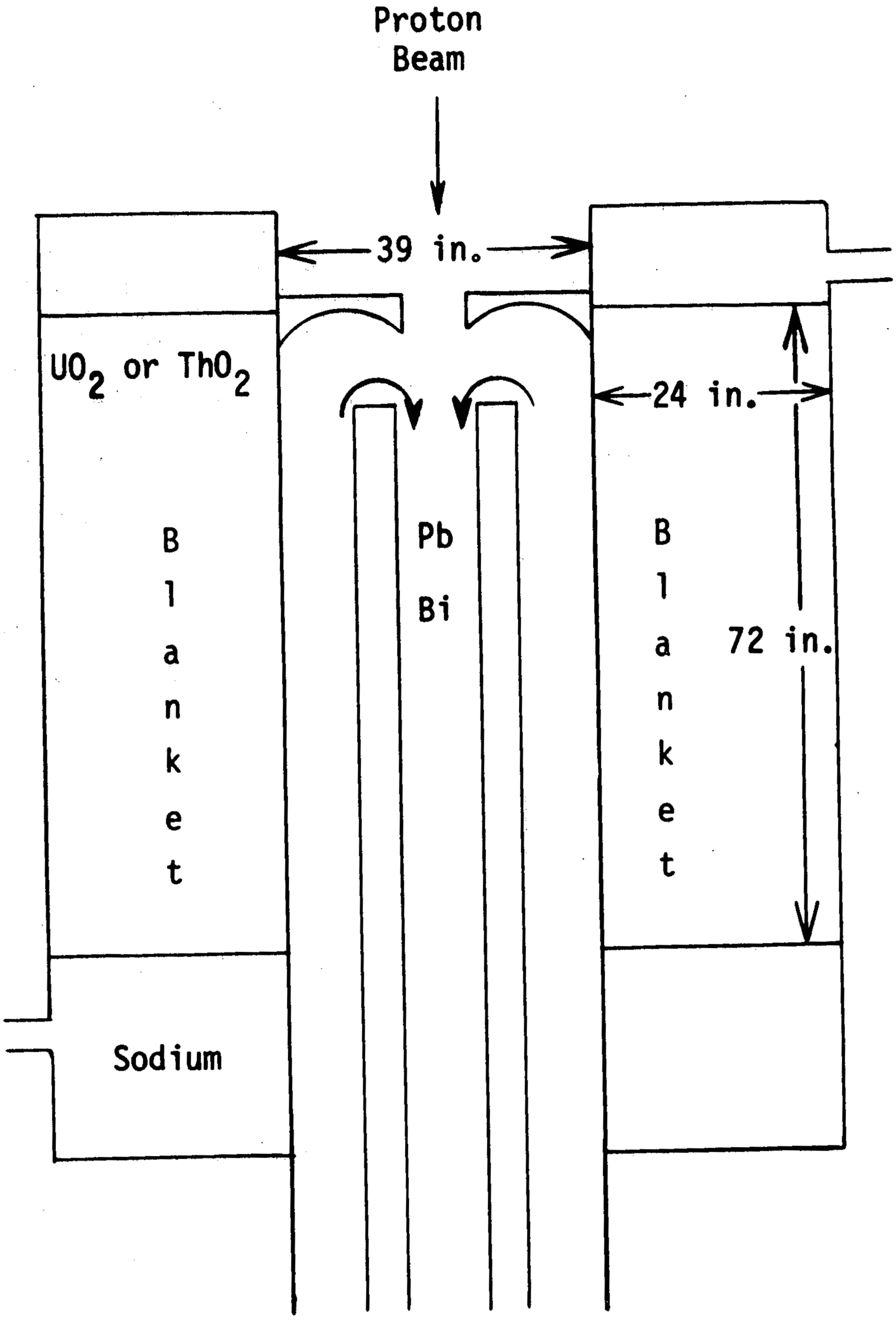


Fig. III.1. Schematic Target and Blanket for the Reference Concept with an LMFBR-Type Blanket

1 GeV protons and following subsequent particles down to low energy neutron capture and leakage. This is the most rigorous calculation which can be performed for such a system and is used to normalize ANISN results when reported in text or summary tables.

ANISN calculations for the idealized UO_2 blanket without structure and coolant were performed for several different enrichments to study the behavior as Pu^{239} breeds in while assuming the fuel is shuffled to keep the enrichment uniform. Figure III-2 shows the power and net Pu^{239} production as functions of enrichment, and Table III-1 gives a breakdown of reaction rates. Tables III-2 and III-3 give key performance parameters. The zero enrichment production rate is 2.8 kg/day of Pu^{239} so that the enrichment increases at the rate of 1% in about 50 days. The initial total power is approximately 650 MW of which 186 MW is in the target. At an enrichment of 4% (obtained in about 200 days from a cold start), the total neutron source is multiplied by a factor of four. Pu^{239} is produced at a rate of 6.1 kg/day but is fissioned at a rate of 2.5 kg/day for a net production of 3.6 kg/day and a blanket power of 2520 MW. Net production of Pu^{239} peaks at about 5% enrichment at a rate of 4.5 kg/day. The peak neutron flux varies from 3.9×10^{16} neutrons/cm^2 sec at 0% Pu to 8.0×10^{16} neutrons/cm^2 sec at 5% Pu indicating substantial radiation damage problems.

To consider the effect of coolant and structural materials, ANISN calculations were performed for 50% UO_2, 25% stainless steel type 316, and 25% sodium which is essentially a CRBR type fuel assembly. Figure III-3 shows net Pu^{239} production and thermal power as functions of enrichment. Table III-4 gives a reaction rate breakdown, and Table III-5 gives key per-

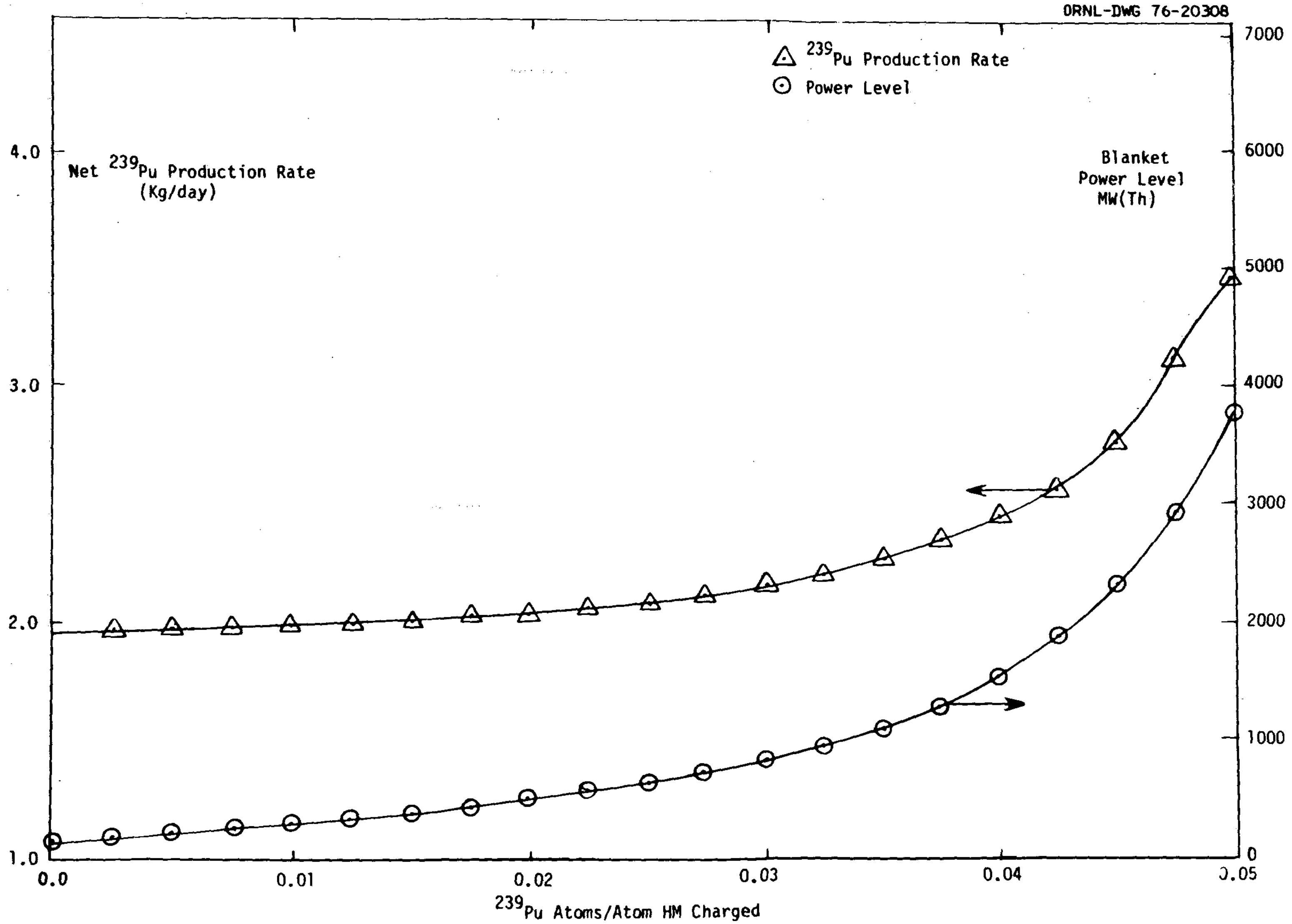


Fig. III.2. Blanket Power and Net Pu^{239} Production as Functions of Enrichment for LMFBR-Type Blanket Without Sodium and Steel

Table III-1. Reaction Rates for UO_2 Blanket Without Sodium or Steel

Reaction	Reactions/Source Neutron				
	0 *	1	2	4	5
U^{238} F	0.066	0.088	0.119	0.250	0.432
U^{238} C	1.153	1.292	1.492	2.359	3.552
Pu^{239} F	0.000	0.075	0.179	0.625	1.249
Pu^{239} C	0.000	0.032	0.071	0.218	0.407
Other	0.060	0.075	0.097	0.218	0.417
Total	1.279	1.561	1.958	3.670	6.057
	% of Total Reaction Rate				
U^{238} F	5.19	5.64	6.06	6.81	7.13
U^{238} C	90.12	82.77	76.18	64.28	58.64
Pu^{239} F	0.00	4.78	9.11	17.04	20.62
Pu^{239} C	0.00	2.03	3.63	5.93	6.72
Other	4.69	4.80	4.95	5.94	6.88
K	0.2019	0.3456	0.4819	0.7245	0.8336
Fission Neutron/S.N.	0.253	0.533	0.929	2.630	5.010
Net Pu/S.N.	1.153	1.186	1.243	1.516	1.896
Pu Rate (kg/day)	1.967	2.024	2.121	2.587	3.236

*% Pu

Table III-2. Target Parameters for LMFBR-Type Blanket Concept*

Type - Heterogeneous target/blanket design

Material - Pb

Size - Diameter 100 cm, Length 60 cm, Cylinder

Neutron Efficiency - 25 neutrons/proton

Neutron Output - 5 x 10^{19} neutrons/sec (target leakage)

Energy Deposition (target only) - 186 MW

Energy Density - 0.40 MW/ℓ

Energy Density Peak/Average - 2.5

Neutron Flux at Liner - 5 x 10^{16} neuts/cm^2/sec

Liner Lifetime for 10^{23} nvt - 35 days

*Based on HETC-MORSE calculation and ANISN calculations normalized to HETC-MORSE calculation

Table III-3. Blanket Parameters for LMFBR-Type Blanket Concept Without Sodium and Steel*

Type - Cylindrical annulus surrounding target

Material - UO_2 at 50% density (no coolant or structure)

Geometry - 100 cm ID, 222 cm OD, 90 cm L, 2.78×10^6 cm^3

Inventory - 14.5×10^3 kg UO_2

Pu^{239} Production Rate - 2.8 kg/day at 0% Pu^{239},
3.6 kg/day at 4% Pu^{239},
4.5 kg/day at 5% Pu^{239}.

Energy Deposition - 460 MW at 0% Pu^{239},
2520 MW at 4% Pu^{239},
3780 MW at 5% Pu^{239}.

Peak Flux - 3.9×10^{16} neuts/cm^2/sec at 0% Pu^{239},
7.0×10^{16} neuts/cm^2/sec at 4% Pu^{239},
8.0×10^{16} neuts/cm^2/sec at 5% Pu^{239}.

---

*Based on ANISN calculations normalized to HETC-MORSE calculation

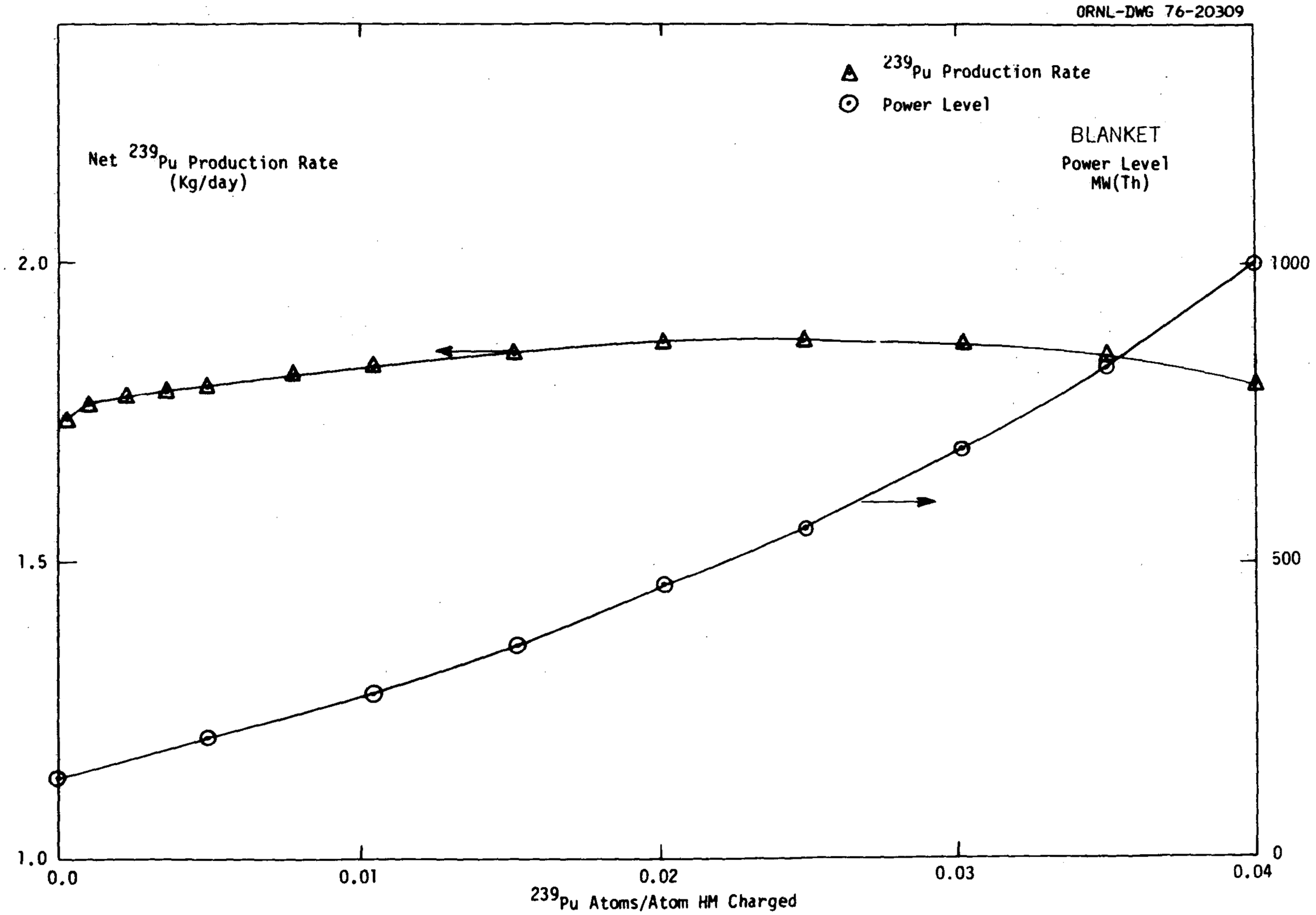


Fig. III.3. Blanket Power and Net Pu^{239} Production as Functions of Enrichment for LMFBR-Type with Sodium and Steel

Table III-4. Reaction Rate Distributions for LMFBR-Type Blanket with Sodium and Steel

Reaction	Reactions/Source Neutron	
	0*	4
U^{238} F	0.042	.108
U^{238} C	1.003	1.570
Pu^{239} F	0.0	.367
Pu^{239} C	0.0	.148
Other	0.205	.330
Total	1.249	2.523
U^{238} F	0.033	0.043
U^{238} C	0.803	0.622
Pu^{239} F	0.000	0.145
Pu^{239} C	0.000	0.059
Other	0.164	0.131
K	0.199	.604
Fission Neutron/S.N.	0.249	1.523
Net Pu/S.N.	1.003	1.055
Pu Rate (kg/day)	1.72	1.81

*% Pu

Table III-5. Blanket Parameters for LMFBR-Type Blanket Concept With Sodium and Steel*

Type, Geometry, and Inventory - Same as in Table III-3

Material - 50% UO_2, 25% Na, 25% Stainless Steel Type 316

Pu Production Rate - 2.4 kg/day at 0% Pu^{239},
2.6 kg/day at 2% Pu^{239},
2.5 kg/day at 4% Pu^{239}.

Energy Deposition - 460 MW at 0% Pu
640 MW at 2% Pu
1400 MW at 4% Pu

Peak Flux - 3.3 x 10^{16} neuts/cm^2/sec at 0° Pu

---

*Based on ANISN calculations normalized to HETC-MORSE calculation formance parameters. For a target source of 5×10^{19} neutrons/sec, the Pu^{239} net production is 2.4 kg/day at zero Pu enrichment in natural UO_2, and the peak production is only 2.6 kg/day at 2.4% enrichment. The power level varies from 460 MW at zero enrichment to 1400 MW at 4% enrichment. Parasitic capture can therefore greatly reduce and change performance characteristics. For the conditions shown on Figure III-4, one might choose to run at low power levels since production is not correlated to power. At a low power level, the structure and coolant volume fraction could be reduced moving performance back closer to Figure III-3 which suggests the effectiveness of a higher power level. The conclusion from this is that much work must be done to determine the desirable enrichments, power levels, and production rates which are highly dependent on mechanical design.

Figure III-5 shows the radial power level distribution in the blanket for the two cases with and without coolant and structure. This shows another problem requiring one of several possible techniques to accommodate the factor of ten peak to average power. The possible accommodations include a compensating enrichment gradient achieved by fuel management or the use of a fluid blanket.

The detailed calculations of proton and neutron transport for the reference lead target and UO_2 blanket with no coolant or structure were performed for the geometry shown in Figure III-6. The proton beam was assumed to have zero width and to be incident along the axis of symmetry. All of the results were obtained by coupling the high-energy transport code HETC with the low-energy transport code MORSE. All charged particles and neutrons with energies > 15 MeV are transported with HETC and neutrons with energies < 15 MeV are transported with MORSE.

ORNL-DWG 76-20311

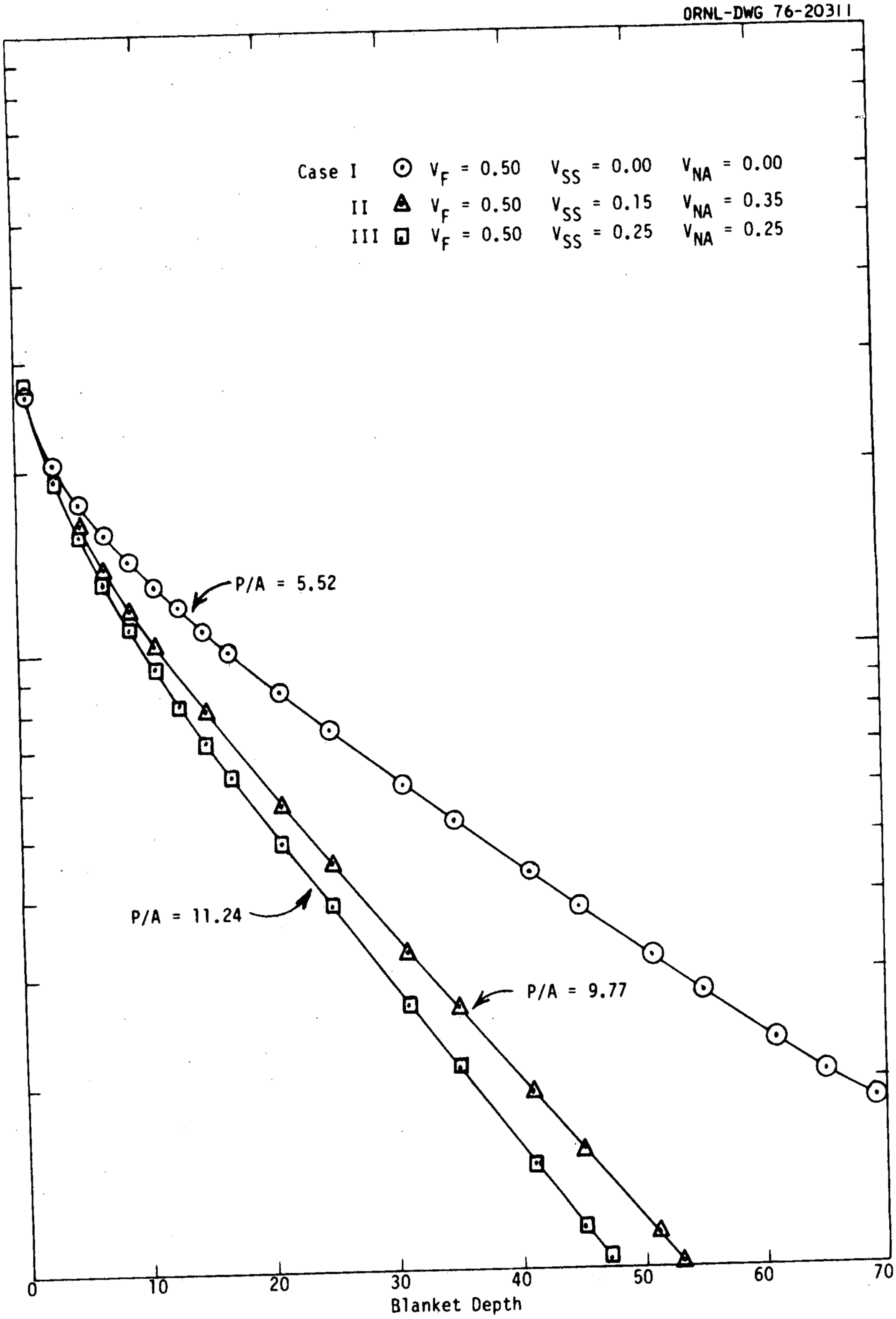


Fig. III.4. Relative Radial Power Distributions for LMFBR-Type Blanket

ORNL-DWG 77-2897

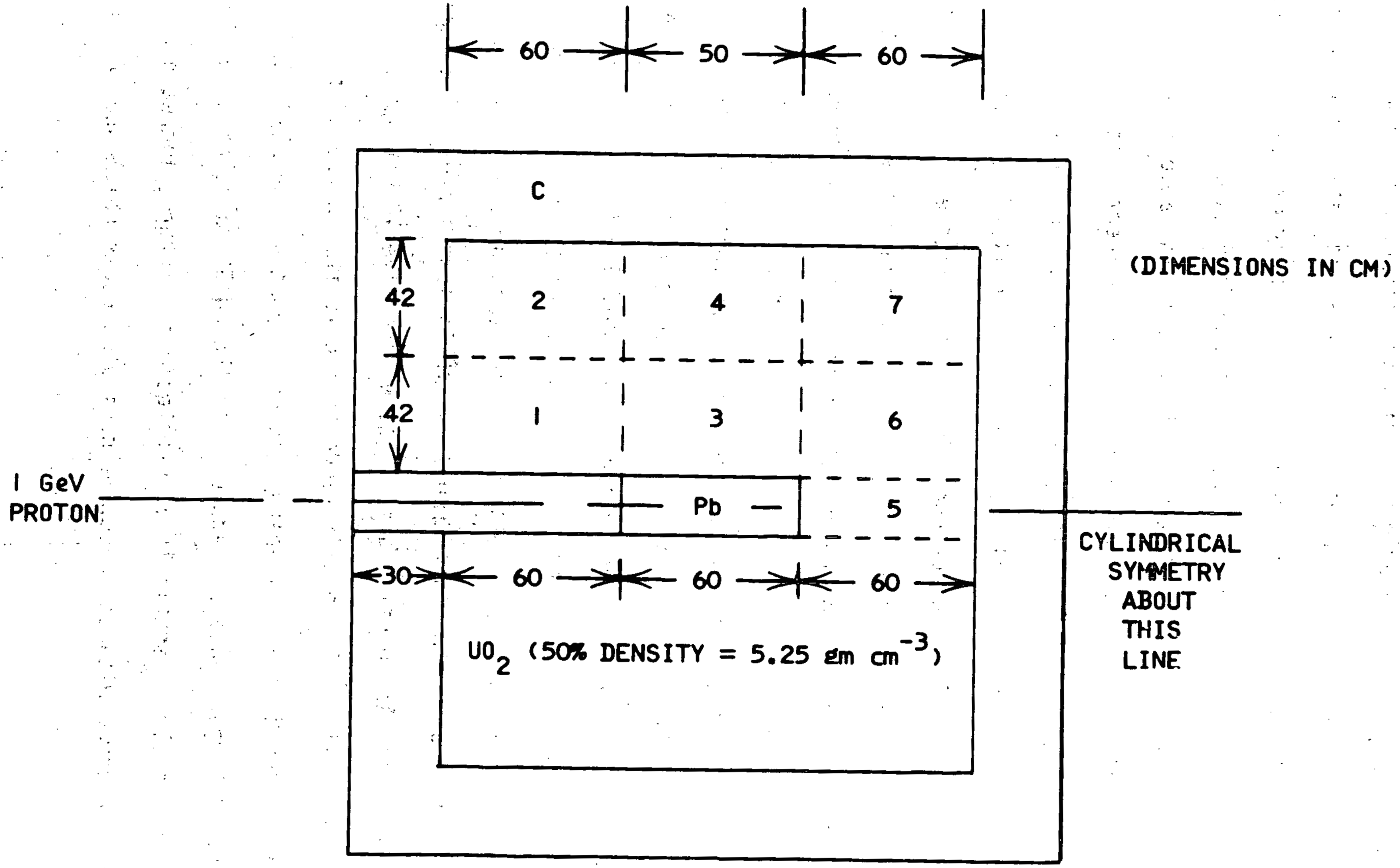


Fig. III.5. Geometry for HETC-MORSE Calculations Showing Definition of Spatial Regions (cylindrical annuli) in the UO_2 used in Presenting the Results

For the purposes of providing some information on the spatial dependence of the results, the UO_2 cylinder has been divided into regions as shown in Figure III-5. In Table III-6 the number of neutron captures in U^{238} is given for the various regions shown in Figure III-5 and for the sum over all regions. Note that the results in Table III-6 and throughout this section are normalized to one incident proton. In Table III-7 the number of U^{238} fissions in the various regions and summed over all regions is given. Also in Table III-7 results are given for the number of fissions produced by neutrons with energies < 15 MeV and by all charged particles (protons, $\pi\pm$) and neutrons with energies > 15 MeV since a substantial number of the fissions are produced by the charged particles and the higher energy neutrons. In Table III-8 energy depositions in various materials and spatial regions are given.

The HETC-MORSE calculations show improved performance relative to the earlier ANISN results as indicated in Table III-9. The production of Pu^{239} is increased from 1.97 to 2.81 kg/day, and the power in the blanket is increased from 130 MW to 463 MW. Comparisons of the calculations showed that most of this difference was the assumption of a 1/E source spectrum in the ANISN calculations rather than the 15 MeV upper energy limit. Virtually all the difference is due to the increase of U^{238} fission in the 2 to 15 MeV energy region. Input of the HETC-MORSE calculated target leakage spectrum into the ANISN calculation would bring the results essentially into agreement. Future blanket analysis can be performed with the discrete ordinates codes ANISN and DOT coupled to the HETC code by a neutron showing down source for E < 15 MeV in the same manner as the present MORSE calculation. Alternatively, the energy

Table III-6. Neutron Captures in U^{238} in Various Spatial Regions (See Figure III-3)

Region	No. of Neutron Captures in U^{238} Per Incident Proton
1	8.9
2	3.4
3	15.7
4	5.1
5	0.2
6	5.2
7	2.9
Sum over all regions	41

Table III-7. U^{238} Fissions in Various Spatial Regions (See Figure III-3)

	U Fissions Per Incident Proton		
Region	From Neutrons With E < 15 MeV	From Charged Particles and Neutrons With E > 15 MeV	From All Particles
1	0.87	0.14	1.01
2	0.11	0.03	0.14
3	2.43	0.83	3.26
4	0.23	0.12	0.35
5	0.03	0.04	0.07
6	0.43	0.32	0.75
7	0.15	0.10	0.25
Sum over all regions	4.2	1.6	5.8

Table III-8. Energy Deposition in Various Materials and Spatial Regions
(See Figure III-3)

Material	Region	Energy Deposition Per Incident Proton (MeV)
Pb		583
C		7
UO_2	1	256
	2	46
	3	768
	4	99
	5	25
	6	185
	7	67
Sum over all UO_2 regions		1446
Sum over all materials and regions		2036

Table III-9. Comparison of HETC-MORSE Results with Preliminary ANISN Calculations for UO_2 Blanket with No Sodium or Steel at Zero Pu Enrichment

	HETC-MORSE	ANISN (1/E)
Pu^{239} Production	2.81 kg/day	1.97 kg/day
Target Power	186 MW	160 MW
Blanket Power	463 MW	130 MW
U^{238} Fission ($E > 15$ MeV)	3.2×10^{18} fission/sec	0
U^{238} Fission ($E < 15$ MeV)	8.4×10^{18} fission/sec	3.3×10^{18} fission/sec

range of ANISN, DOT, and MORSE can be extended up to 400 MeV as has been previously demonstrated.

Heat Transfer

In the heterogenous concept, a stream of molten Pb-Bi is used as a circulating target; the general arrangement is shown schematically in Figure III-6. Three sump-type centrifugal pumps discharge into a common pipe of 40" diameter. There is a concentric pipe 28" in diameter that returns the lead through the tubes of a tube and shell heat exchanger to the suction side of the three pumps.

The open ends of these concentric pipes terminate two-thirds of the way into the annular blanket tank. The molten metal flows in the outer annulus of the concentric pipes and by a diverter plate is turned into the center pipe and down to the heat exchanger. The center pipe is uncovered, and the proton beam impinges directly onto the flowing surface of the target. The portion of the center pipe which is in the high flux fluid can be rather easily replaced if necessary because of radiation damage. Since it carries no mechanical load, it may not have to be replaced until a large fluence has been experienced. The walls of the blanket tank as well as the fuel assemblies will likely be limited in lifetime because of radiation damage, but the limits depend on the requirements placed on the structural materials as well as the damage mechanism and rate.

By proper location of the heat exchanger in relation to the blanket, the force exerted by the lead column on the diverter plate could be minimized. Also the height of the molten column could be chosen so that hydrostatic head would provide driving force to circulate the lead through the tubes of the heat exchanger.

ORNL-DWG 76-19897

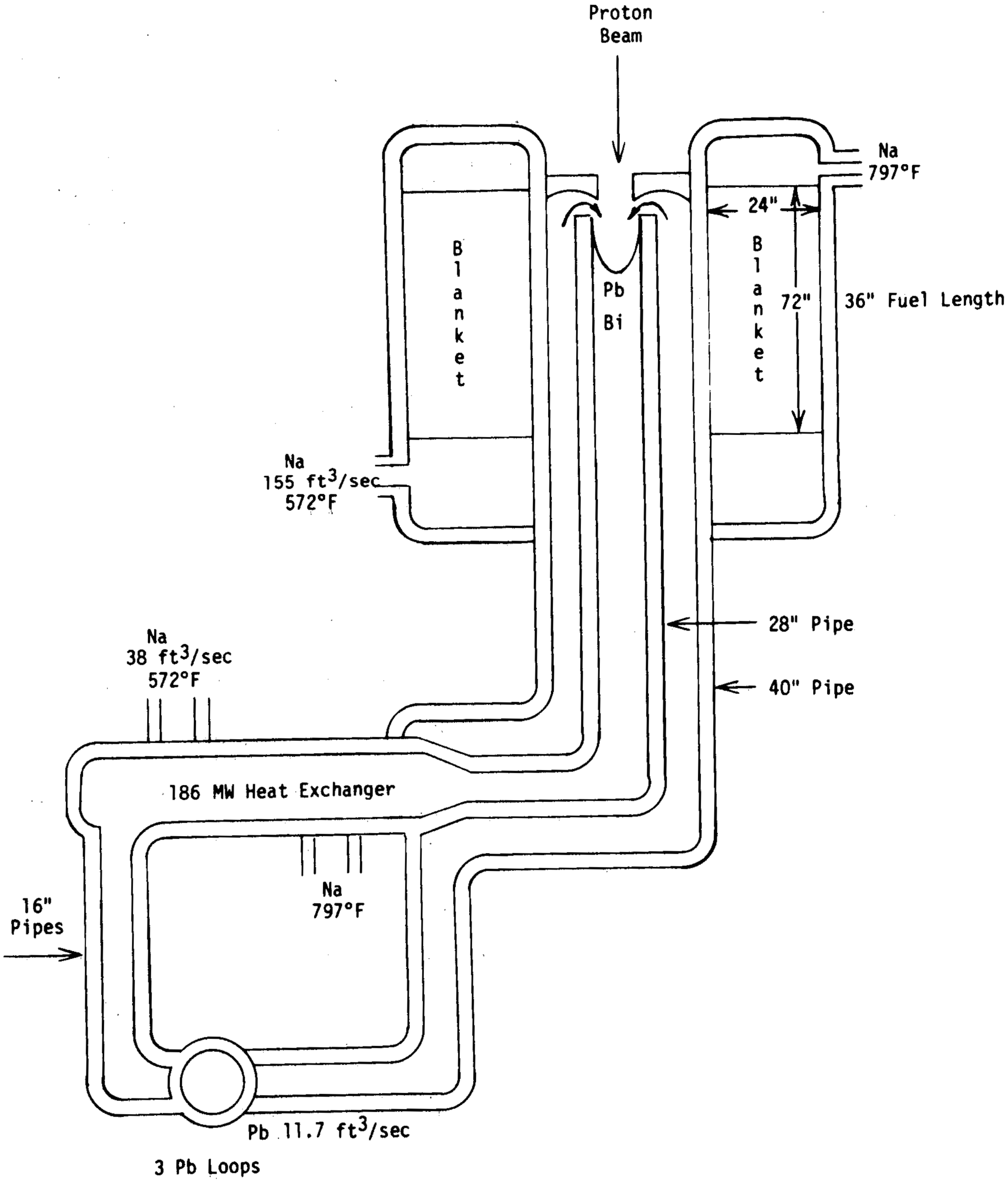


Fig. III.6. Heat and Mass Transfer for LMFBR-Type Blanket

Some characteristic thermal-hydraulic parameters for this concept are given in Table III-10. Calculations for tube shape to accommodate thermal changes have not been done. A likely alloy for the lead bismuth system is tantalum - 10% tungsten.

Fuel Management

Because of the strong variation of thermal power with enrichment, it is desirable to operate at a constant value. While this is possible with a fluid or quasi-fluid continuous processing system, it is interesting to examine the potential for an LMFBR-type blanket. Table III-11 gives the design parameters and calculated parameters for such a blanket. There are 216 assemblies, 4.9 in. across the flats. The pins are assumed to be 6 ft long with a 3 ft active zone. The fuel management scheme is to replace 18 assemblies every 17 full power days and shuffle the blankets to flatten the power distribution. A given assembly resides 200 days. The power swing in a 17-day cycle is from 595 MW to 686 MW as the average enrichment increases from 1.85% to 2.15%. The average power by assembly is 3 MW so that a 91 pin CRBR-type blanket assembly would have an acceptable linear power value of 12 kW/ft. The 18 assemblies removed each cycle contain 2.4 kg Pu^{239} each. In order to accommodate radiation damage criteria, the blanket assemblies could be removed as they reach the specified maxima. The blanket assemblies are at sufficiently low enrichment to be processed in an LWR-type reprocessing plant.

Loss of Coolant Accident

While the Accelerator Breeder Concept is relatively free from concern regarding energetic accidents from overpower transients or sodium voiding, the problem of loss of coolant remains to a reduced degree. ORIGEN

Table III-10. Heat and Mass Transfer Parameters for LMFBR-Type Blanket Concept

Target

Target Power - 186 MW

Blanket Power - 640 MW

Pb-Bi Flowing Target in Concentric Pipes

Target Diameter - 28"

Inlet Pipe Diameter - 40"

Pb-Bi Volumetric Flow Rate - 11.7 ft^3/sec

Inlet Temperature of Pb-Bi = 572°F

Outlet Temperature of Pb-Bi = 797°F

Pb-Bi To Na Heat Exchanger, 2000 3/8" tubes 10' long

Shell of Hx - 30" Diameter

Pb-Bi pumped by 3, 5300 gpm centrifugal pumps

Blanket

Containment in annular tank

- Center hole for target - 42" diameter
- Height of tank - 10 feet
- OD of tank - 10.85 feet

Blanket Elements 19,440 - 0.5 in. OD pins

- UO_2 or ThO_2

Coolant Na - 155 ft^3/sec

- Coolant Inlet Temperature - 572°F
- Coolant Outlet Temperature - 797°F

Table III-11. Fuel Management for an LMFBR-Type Blanket

Number of Subassemblies - 216

Number of Pins per Assembly - 91

Subassembly Width (Flats) - 4.90 In., 12.44 cm

Subassembly Area - 22 In^2, 142 cm^2

Subassembly Fuel Length - 35.43 In., 90 cm

Assembly Inventory - 67.1 kg UO_2/assembly

Number Refueled - 18

Refueling Interval - 17 fpd

Assembly Residence - 200 fpd

Reload Enrichment - 0%

Discharge Inventory - 2.4 kg Pu^{239}

Maximum Discharge Enrichment - 4%

BOC Average Enrichment - 1.85%

EOC Average Enrichment - 2.15%

BOC Total Power - 595 MW

EOC Total Power - 686 MW

Average Assembly Power = 3 MW

Average Linear Power = 12 kw/ft

calculations were performed for the sodium cooled UO_2 blanket operating at 2% enrichment and 648 MW th. The maximum decay heat after a long period of operation and instantaneously after shutdown was 6.8% or 44 MW. This heat could be removed by natural circulation and an auxiliary heat exchanger. If as a result of catastrophe, the blanket melts, calculations with ANISN have shown that K_∞ for the fuel is much less than 1.0 such that recriticality is impossible. The liquid blanket concept is more attractive in this regard because the large mass accommodates the decay heat problem.

Summary of LMFBR-Type Blanket Concept

This concept has several advantages and disadvantages. The separate systems for target and blanket allow more appropriate designs for each. The liquid lead target design makes maximum use of earlier work on the ING design, minimizes energy deposited in the target, and assures good neutron production efficiencies by not including light nuclei in the proton beam. The target heat transfer loop appears feasible.

The blanket makes maximum use of LMFBR technology although one may produce U^{233} rather than Pu^{239} by use of thorium oxide fuel rods. Energy self-sufficiency is not quite achieved with 826 MW th giving an electrical output of 290 MW compared with the accelerator load of about 600 MW. Compared to the LMFBR, the ABC blanket is not concerned with reactivity excursions resulting from sodium voiding or other sources of reactivity. Sodium voiding does give a positive power coefficient, but the system remains subcritical. Decay heat does present a problem which must be attacked by assuring natural convection and auxiliary heat removal. Also a catastrophic sore meltdown cannot result in criticality since k_∞ for the fuel is ~ 0.5. The radial power peaking is extreme and presents serious thermal hydraulics problems.

The most important overall observations of the analysis of this concept are:

(1) The production of fissile fuel is directly proportional to the neutrons produced from protons for zero enrichment. The capital cost of the accelerator/target per neutron produced must be minimized.

(2) The thermal power produced in the blanket is a strong function of the fissile enrichment. To the extent possible, the blanket should be operated at a fixed enrichment while producing maximum net fissile fuel. This requires continuous or near-continuous fuel processing.

(3) For this concept, radiation damage to the wall between the target and the blanket is a serious concern with an indicated need to replace the wall frequently. Also the lifetime of blanket assemblies is likely to be limited by radiation damage.

(4) Compared to the LMFBR, this concept can be designed to eliminate the potential for a core disruptive accident or a re-criticality after core meltdown. A loss of coolant accident does pose a serious problem requiring a guard vessel and auxiliary heat exchanger loops to assure cooling of the decay heat.

(5) For the reference conceptual design, Pu^{239} can be produced at 2.6 kg/day or 760 kg/year at 80% capacity. U^{233} production should be approximately 1.8 kg/day for a thorium oxide blanket.

The desire for continuous processing and the need to mitigate radiation damage led us to look at several fluid or quasi-fluid concepts. One of these was chosen for more consideration and is described here.

IV. ANALYSIS OF A REFERENCE DESIGN WITH A SALT BLANKET

Several concepts were examined in an attempt to accommodate the needs for continuous processing, to eliminate structure susceptible to radiation damage, to eliminate the loss of coolant accident potential, and to better accommodate the more desirable production of U^{233}. A promising concept appears to be a system composed of a large vat of salt (27% ThF_4, 71% LiF, and 2% BeF_2 (mole percents)) with a liquid lead target created by the inertia of a lead column falling into the salt vat. Figure IV-1 shows a schematic drawing of this concept. We have obviously gone to some length to preserve the heterogeneous target/blanket geometry and the need for a pure heavy-metal target must be determined by further calculations and experiments. The salt can contain U^{238} and Pu^{239} if desired and can be processed with a continuous side stream to remove transmutation products.

Neutronics

Compared to the LMFBR-type blanket, much less analysis has been performed for this salt blanket concept. A few ORIGEN and ANISN calculations have been performed, but the non-availability of cross sections for Pa^{233} in a suitable group collapsed form has precluded detailed analysis. Table IV-1 shows reaction rate results from the ANISN calculations. The relatively long 27 day half-life of Pa^{233} and the high flux levels possible with ABC give the potential for Pa^{233} capture to compete strongly with decay. Also the Pa^{233} capture cross section is much higher at low neutron energies giving a spectral dependence to the capture/decay ratio. It is likely that for maximum U^{233} production in a thermal or near-thermal

ORNL-DWG 76-19898

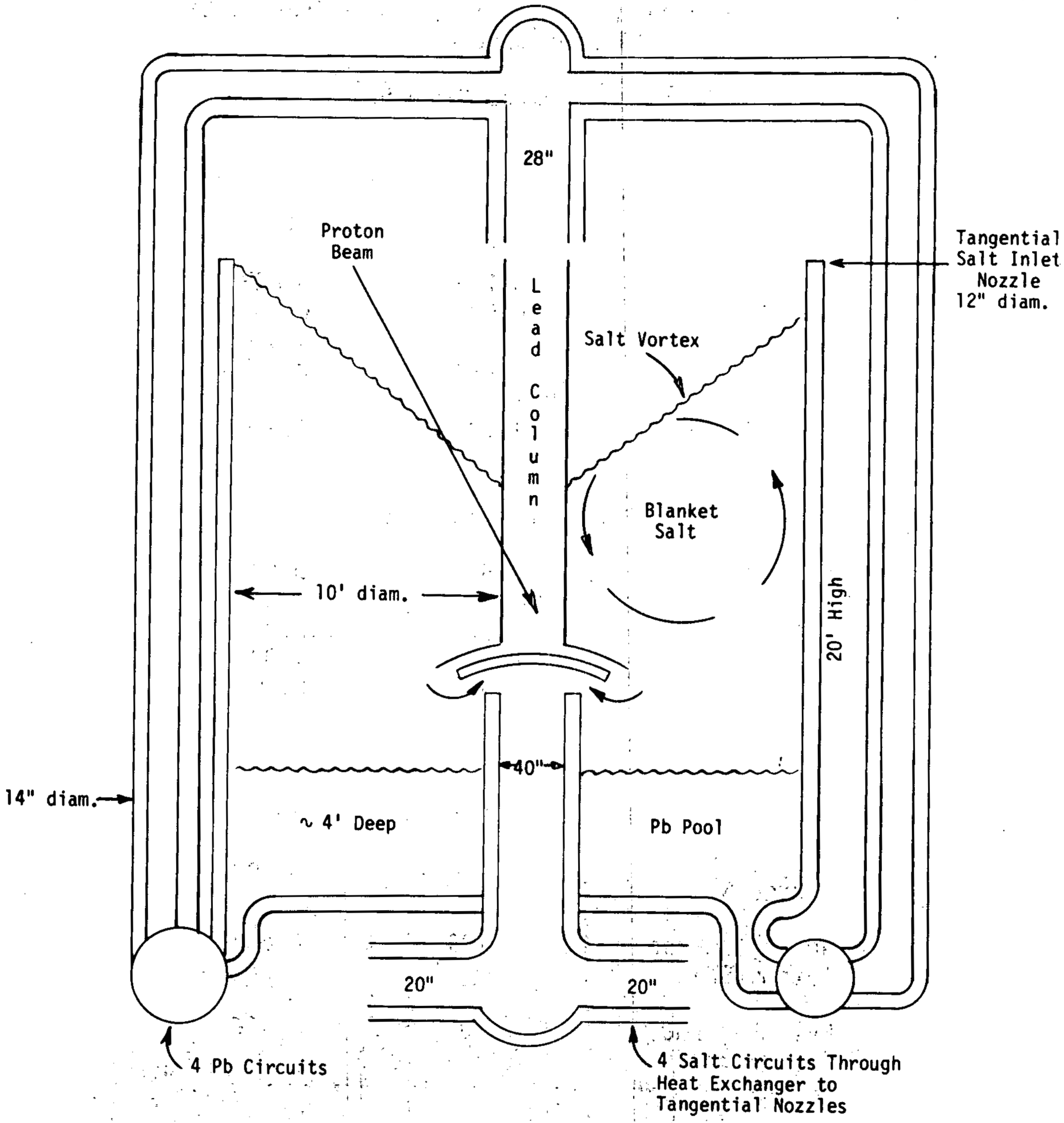


Fig. IV.1. Target and Blanket Arrangement for Salt Blanket Concept

Table IV-1. Reaction Rate Summary for a Thorium Salt Blanket Concept

	Reactions/Source Neutron	
Reaction	0*	4
Th^{232} F	0.006	0.009
Th^{232} C	0.724	0.888
U^{233} F	0.000	0.184
U^{233} C	0.000	0.020
Other	0.297	0.395
TOTAL	1.027	1.496
	% of Total Reaction Rate	
Th^{232} F	0.55	0.60
Th^{232} C	90.51	59.31
U^{233} F	0.00	12.32
U^{233} C	0.00	1.34
Other	28.95	26.40
K	0.016	0.328
$\frac{\text{Fission N}}{\text{Source N}}$	0.016	0.487
Net U^{233} Prod. (atoms/SN)	0.729	0.683
Net U^{233}Prod. (kg/day)	1.205	1.136

* % U^{233}

system the Pa^{233} should be removed by on-line processing and allowed to decay in storage. If production rates are high enough, it might be attractive to produce U^{233} and U^{234} by allowing more of the Pa^{233} to stay in the blanket. A fuel cycle based solely on Th^{232} and U^{233} would produce wastes with actinide hazard reduced by factors as large as 10^6 relative to a U^{235}, U^{238} system. Thus far, the predicted production rates for U^{233} in the salt blanket are too low being on the order of 1.1 kg/day for a target neutron source of 5 x 10^{19} neutrons/sec. Because of the very strong sensitivity to spectrum and flux level for U^{233} production, it is necessary to extensively investigate a wide range of blanket designs.

Heat and Mass Transfer

In the salt blanket concept, mass and heat transfer are combined for the blanket as well as the target fluids. Furthermore, the target heat is transferred to the salt so that only one system of heat exchangers is involved. No wall separates the target from the blanket so that the radiation damage concern is at the wall of the salt vat and is reduced by at least an order of magnitude compared with the LMFBR-type blanket. The containment and removal of gaseous fission products will be a problem but is beneficial from a safety standpoint since one would not have a large inventory of such fission products associated with the fuel. There is also the possibility of free fluorine liberated from the salt, but this has not been observed experimentally. The low vapor pressure of the lead and the salt are compatible with the accelerator vacuum. Some preliminary design parameters are shown in Table IV-2.

Table IV-2. Design Parameters for Salt Blanket Concept

Target Power = 186 MW

Blanket Power = 2000 MW

Lead Flow = 30 ft^3/sec

Blanket Flow = 96 ft^3/sec

Blanket Inlet Temperature - 1100°F

Blanket Outlet Temperature - 1400°F

Lead Target Temperature - 1625°F

Lead Outlet Temperature - 1400°F

Vessel Diameter - 10 feet

Vessel Height - 20 feet

The salt is circulated in four loops through tube and shell heat exchangers where the target heat and the blanket heat are removed. The salt returning from the heat exchangers is introduced tangentially near the top of the tank and creates a vortex in the salt. The proton beam enters the lead at a small angle to the vertical and within the salt vortex just above the surface. The lead, within a very short distance, comes to thermal equilibrium with the salt.

In many years past some considerable experimental work was done with molten lead - molten salt systems. There is always some entrainment of salt in the lead (a few hundredths of percent), but we never experienced even ppm amounts of lead in the salt. The chemical compatibility and physical immiscibility of these two fluids has been well established. The lead also exerts a strong pumping action on the salt giving a rotational flow of the blanket fluid. The salt is taken out near the center where the rotational velocity is lowest. The lead forms a pool four feet deep in the bottom of the tank where it is taken off to the four circulating pumps.

The entire lead circulating system is isothermal at the salt exit temperature of 1400°F (760°C). It heats to 1625°F in the proton beam and is immediately cooled by the salt which enters the tank at 1100°F.

This concept has certain attractive features. All structural members can be kept in a relatively low neutron flux so that radiation damage to structures is minimized. Since the lead circuit is isothermal, there is a minimum of mass transport with no plugging of passages. The chemical processing of the blanket can be done continuously.

While one mechanization of this idea has been discussed, there may be other arrangements that are more attractive. It will require some study to determine the most propitious arrangement of salt inlet and outlet lines. The interaction of the forced vortex in the salt with the strong pumping action of the lead column makes it impossible to determine the internal flow patterns by inspection.

Summary of Salt Blanket Concept

This concept is attractive in reducing problems associated with radiation damage to structural components and allows continuous on-line processing so as to hold enrichment and power level constant. Unfortunately, the calculations currently performed indicate a low U^{233} production rate. Design changes may substantially improve production, so that this concept should not be discarded prematurely. Also HETC-MORSE calculations with a more accurate neutron source spectrum may indicate higher production rates. Direct irradiation of the salt with protons and deuterons should be considered. If production is reasonably good, the elimination of the lead loop would be possible.

V. GAS COOLED TARGET/BLANKET CONCEPT

If production efficiency is to be maximized, it appears that a hard neutron spectrum is desired. For U^{233} production a hard spectrum also reduces capture reactions in Pa^{233}. High efficiency coolants such as sodium, lithium, molten salt, or water all tend to soften the neutron spectrum, and one is left with a gas coolant (helium) as the best candidate for a hard spectrum. Fortunately much information on helium coolant is available from the HTGR, GCFR, and pebble bed reactor designs. GCFR conditions are given in Table V-1.

Figure V-1 shows a gas cooled target/blanket design, and Table V-2 gives performance parameters. The major problems in a gas cooled system are the power density and the severe requirements on the window separating the high pressure helium from the accelerator vacuum. GCFR conditions give the highest power densities, on the order of 240 kW/liter as shown in Table V-1. With a proton beam of 2×10^{18} protons/sec and a large uranium target, we are assured of a power level of around 1500 MW. For a thorium target, the power may be as low as 800 MW because of the substantial reduction in fast fission. Also power peaks on the order of 2.5 occur because of the fact that the protons are incident on one end. For this reason, power density should be limited to 100 kW/liter requiring a target of 1.5×10^4 liters.

As the schematic in Figure V-1 shows, the target and blanket are combined in the gas cooled approach and the proton beam is spread over the incident surface for dilution. The low density (.23 volume fraction) of UO_2 also increases the proton penetration to a range of 190 cm which also dilutes the power density. In the schematic, the core is hung upside down GCFR fashion with all support structure at the top (cantilever

ORNL-DWG 74-9456R2

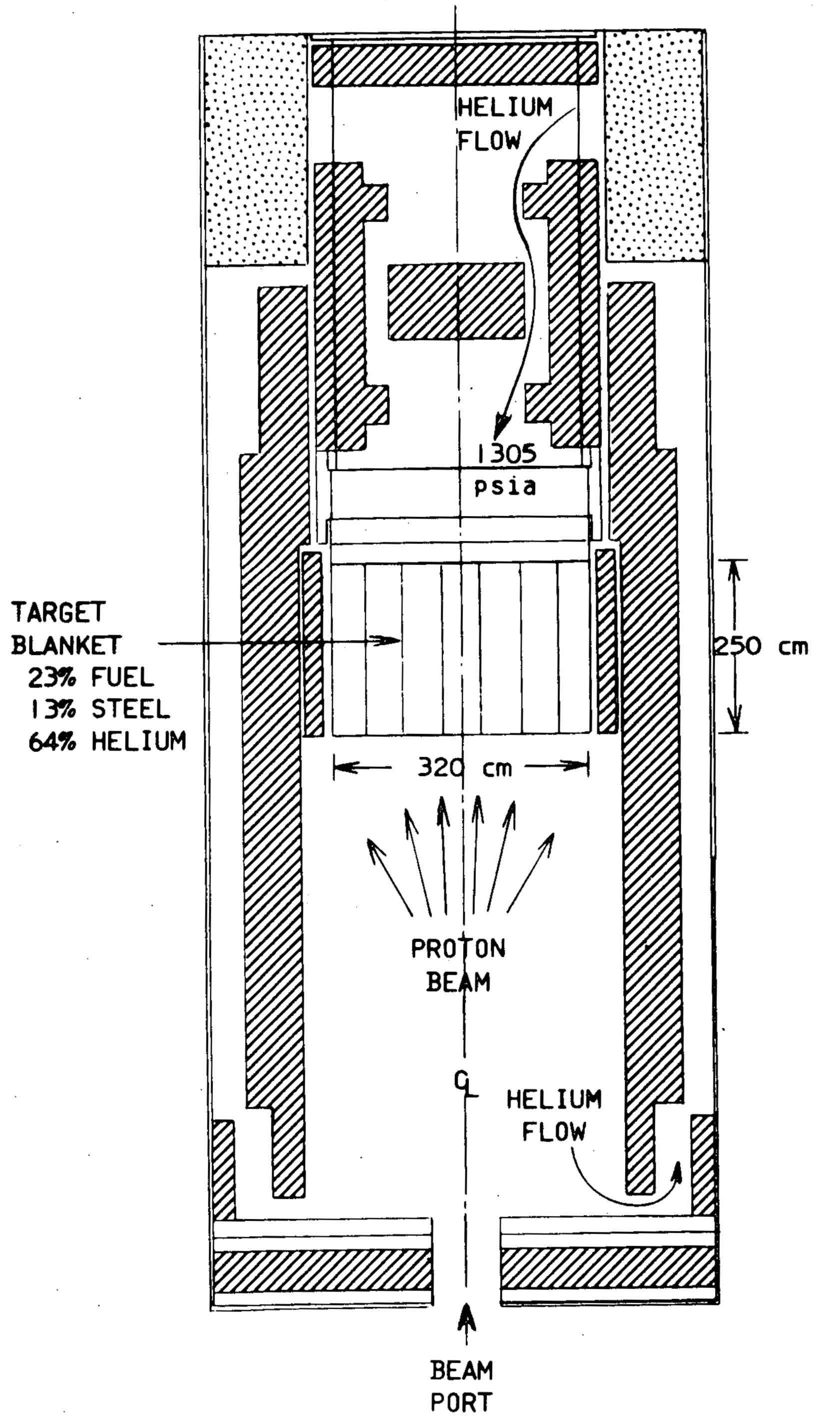


Fig. V.1. Conceptual Design for GCFR Type Design

Table V-1. GCFR Parameters

Core Volume Fractions - Fuel 23%, Stainless Steel 13%,
Helium 64%

Average Power Density - 240 kW/liter

Power Peaking Factor - 1.25

Helium Inlet Pressure - 1305 psia

Helium Pressure Drop - 42 psia

Pin Size - 0.66 cm OD pellet, 270 pins/assembly

Linear Power - 12.5 kW/ft

Table V-2. Parameters for Gas Cooled Blanket

Target Type - Integral with blanket, diffused proton beam

Fuel Form - GCFR type

Average Power Density Limit - 100 kW/liter for GCFR

Power Peak/Average - 2.5

Target Composition - 23% UO_2 or ThO_2, 13% steel, 64% helium

Average Enrichment - 1%

Target Power - 1500 MW at 1% enrichment; 800 MW for U^{233}

Target Volume - 2 x 10^4 liters

Target Dimensions - 8.2 ft length by 10.5 ft diameter

Fuel Inventory - 4.83 x 10^4 kg

Fuel Production - 4.0 kg/day Pu^{239} or 2.8 kg/day U^{233}

Number of Assemblies - 200

Assembly Inventory - 241.5 kg

Assembly Fissile Content at EOC - 4.83 kg

Assembly Residence Time - 242 days for 2% EOC enrichment

Processing and Fabrication Costs - $\sim$\$22/gm fissile

Number of Assemblies Refueled Each Interval - 20

Refueling Interval - 24 Full Power Days

restraint) and the helium flow is downward. In a depressurization accident, any fuel melting will fall down from the core to a suitable core catcher. This feature seems to justify the inconvenience of an upward proton beam which is incident at a slight angle to avoid a parallel orientation with the fuel pin.

This design is sized for Pu production at an average enrichment of 1%. U^{233} production in Th might give substantially lower minimum power levels and substantial design revisions. The heavy metal inventory is 48.3 metric tons which is large but not unreasonable.

Fuel production is given from a value of 70 captures/proton for an infinite uranium slab as calculated by the HETC-MORSE method and an estimate of the deleterious effect of stainless steel and oxygen on the production of neutrons from the protons. If one assumes that the ratio of proton nuclear collisions in the steel and oxygen to collisions in the uranium is given by:

$$R = \frac{VF(SS) \times \rho(SS) \times (56)^{2/3} + VF(0) \times \rho(0) \times (16)^{2/3}}{VF(U) \times \rho(U) \times (238)^{2/3}}$$

where $VF(SS) = .13$, $VF(0) = VF(U) = .23$, $\rho(SS) = 7.9$ gm/cm^3, $\rho(0) = 1.23$ gm/cm^3, and $\rho(U) = 9.28$ gm/cm^3, then

$$R = .21$$

If about 20% of the collisions are in the light elements, the production is assumed to be reduced from an ideal value of 5 kg/day to a value of 4 kg/day. For U^{233} the estimated production is 2.8 kg/day because of the reduced fast fission effect.

For 200 assemblies the inventory for each is 241.5 kg and the Pu content at 2% end-of-cycle conditions is 4.83 kg. The assembly residence time to achieve this enrichment is 242 full-power days. If one assumes refueling of 20 assemblies each for 10 refueling intervals, the interval would be 24.2 full power days. This would given an enrichment swing of 0.2 percent which would hold the power relatively constant.

The radiation damage problem for the target and blanket will likely be severe, but the hanging core concept is such that only removable fuel assemblies see the proton beam and highest neutron fluxes. A lifetime much shorter than 242 days would produce lower enrichments however. The $22K/gm estimate for processing and fabrication is high because of the large amount of heavy metal that must be handled. If discharge enrichments were greatly lowered, the cost would rise to an unacceptable value. Heating and radiation damage in the window separating the accelerator vacuum from the high pressure helium is a major problem in this concept and needs much study.

VI. ACCELERATOR ECONOMICS AND ACCELERATOR - REACTOR SYMBIOTICS

In this section, the potential market for fissile material produced by the accelerator is assessed, and the cost of the fissile product is estimated and compared with ^{235}U cost projections. The fissile cost estimates are then used to further estimate the cost of producing power in LWR's existing when the accelerator-produced fuel becomes available, and in advanced converters built to take advantage of the available ^{233}U.

<u>Identification of Potential Market (Reactor Fissile Requirements)</u>

In order to assess the economic potential of the accelerator-breeder, it is necessary to determine the worth of the fissile material produced. Table VI-1 gives the annual makeup requirements for a variety of thermal reactors and fuel cycles, and indicates the power sustained for each combination by a fissile production rate of 1 Kg per day.

The no-recycle cases indicate the potential for power production on a "throwaway" basis with the fissile material provided by the accelerator. Note that makeup requirements for Pu are slightly higher than those for ^{235}U (the current no-recycle operational mode), while ^{233}U makeup requirements are slightly lower than those for ^{235}U. This is an indication of the relative value of the three primary fissile materials as thermal reactor fuels. At present it is not clear that thermal reactor fuel could be prepared in the accelerator in an acceptable fashion with constraints on reprocessing, but the concept is worthy of further investigation.

The cases considering current reactor design with recycle are assumed to indicate makeup requirements at the time when the accelerator-produced fuel becomes available. The ^{233}U makeup requirements are low for both PWR and HTGR cases, indicating that the higher thermal fuel value of ^{233}U is not a function of reactor type. The denatured case considers 14% ^{233}U in

Table VI-1. Fissile Requirements and Potential Power Sustained For Selected Thermal Fuel Cycles[14,15]

Reactor Type	Fuel Description (All Oxides)	Conversion Ratio	Reprocessing	Annual Makeup (Kg/MW(e)•Yr)	Power Sustained by Accelerator (MW(e)/Kg Fissile/Day)
Current Design, No Recycle, U^{235} Makeup					
PWR	U^{235}/UO_2	0.60	No	0.81	325 (at 0.90 Pu Makeup)
HTGR	U^{235}/ThO_2	0.66	No	0.63	479 (at 0.61 U^{233} Makeup)
Current Design with Recycle, U^{235} Makeup					
PWR	U^{235}/UO_2	0.60	U^{235}, Pu	0.53	
HTGR	U^{235}/ThO_2	0.66	U^{235}, U^{233}	0.32	
Current Design with Recycle, Supplied Pu or U^{233} Makeup					
PWR	Pu/UO_2	0.65	Pu	0.59	495
PWR	U^{233}/ThO_2	0.75	U^{233}	0.31	943
PWR	U^{233}/U^{238} (14% denatured/ThO_2)	0.70	U^{233}	0.38	769
PWR	U^{233}/UO_2	0.67	Pu, U^{233}	0.44	664
HTGR	U^{233}/ThO_2	0.69	U^{233}	0.27	1,082
Advanced Design, Supplied U^{233} Makeup					
PWR	U^{233}/ThO_2	0.90	U^{233}	0.12	2,435
Pebble Bed	U^{233}/ThO_2	0.97	U^{233}	0.04	7,305

^{238}U, which is comparable in critical mass to the often-stated 20% ^{235}U in ^{238}U constraint.[16] Overall, the system is approximately 80% ThO_2 and 20% $^{235}UO_2$ - $^{238}UO_2$. Its makeup requirements are not too high, even though Pu recycle is not allowed. The ^{233}U-^{238}U cycle indicates the desirability of ^{233}U makeup in LWR's even if Pu recycle is allowed.

Advanced converters, including advanced PWR designs, the pebble bed reactor, the high-gain HTGR and even the marginal breeders such as the LWBR and molten salt represent the most advantageous use of ^{233}U in thermal systems. However, the implementation of these concepts requires an initial ^{233}U inventory and hence, in current scenarios, a prebreeder-reactor with an accompanying economic penalty. These advanced converter systems constitute the most logical use for the ^{233}U produced in the accelerator-breeder, which would then be operating in the place of the prebreeder-reactor. Presumably these systems could be developed rapidly if a reliable source of ^{233}U were established. The cases shown here indicate the low makeup requirements characteristic of such systems.

Table VI-2 explains why ^{233}U fuel is required for advanced converter systems. The eta value, which represents the net neutron production per neutron absorbed in a fuel, is consistantly better for ^{233}U in the energy ranges shown. Since one neutron is necessary to maintain criticality, (eta-1) represents the number of neutrons which can undergo fertile capture to produce a fissile atom (breed). It is clear from the values given that for a typical mixed epithermal-thermal spectrum, only ^{233}U provides a realistic opportunity to design a breeder, or even a high-gain converter, since non-productive absorption in structure or coolant and leakage losses are not considered.

Table VI-2. Representative Eta (η) Values[15,17] for U^{233}, U^{235}, Pu^{239}

	U^{233}	U^{235}	Pu^{239}
η (0.025 eV)	2.30	2.07	2.11
η_{th} (Typical PWR Thermal Spectrum)	2.27	2.06	1.84
η_{epi} (Typical PWR Epithermal Spectrum)	2.16	1.67	1.88
$\bar{\eta}$	2.24	1.96	1.85
$(\bar{\eta}-1)$	1.24	0.96	0.85

Estimated Accelerator Product Costs

The capital cost of the accelerator was taken to be \$600 M. The capital cost assumptions used in estimating the capital cost of the balance of plant (everything but the accelerator), and the capital charge rate and formula for computing annual capital cost charges are given in Table VI-3. An 80% capacity factor was assumed for the accelerator, while a 75% capacity factor was assumed for all reactors considered. Annual costs were estimated for two design concepts (the Pb target with LMFBR-type blanket and the GCFR-type target-blanket), and at three blanket power production levels for each design [200 MW(e), 600 MW(e), and 1600 MW(e)]. Since the accelerator requires 600 MW(e) for operation, the net power productions are then -400 MW(e), 0 MW(e), and +1000 MW(e), respectively. A full set of assumptions used is given in Table VI-4. The cost of power assumed, 30 mills/kw.hr, is an estimate of power costs in the 1983-1985 time frame, and is used for both the cost of power purchased for the -400 MW(e) case, and as the price received for power sold in the +1000 MW(e) case. Note that the power and heavy metal densities are assumed to remain constant, so that the blanket size must increase in order to increase power and fissile production. All dollar values quoted are in 1976 dollars.

Table VI-5 gives the estimated annual costs for the accelerator-breeder and the resulting cost of Pu produced for the cases considered. Note that there is no large decrease in fissile product cost with increasing power production unless the system becomes a large energy producer, while an increase in fissile production has a marked effect on product cost.

Estimated ^{235}U Fuel Costs

In assessing the economic viability of accelerator-breeder produced fissile material, it is interesting to compare the estimated prices of Pu given on Table VI-5 with estimated prices for ^{235}U. Table VI-6 presents cost

Table VI-3. CAPITAL COST ASSUMPTIONS[14]

\$800/kW(e) Basic Capital Cost (Balance of Plant)

16% Capital Charge Rate

Consists of: 9% Bond Interest/Equity
1% Sinking Fund (Amortization)
6% Taxes

Capacity Factor: 80%, Accelerator Breeder

Formula for Computing Annual Capital Cost Charges:

$$\text{CAPITAL COST}\left(\frac{\text{Mills}}{\text{kW}\cdot\text{hr}}\right) = \left[\frac{\$800}{\text{kW(e)}} \times \frac{0.16}{\text{Yr}} \times \frac{10^3\ \text{Mills}}{\$} \times \frac{\text{Yr}}{8760\ \text{hr} \times 0.80}\right]$$

Table VI-4. Assumptions Used to Estimate Annual Costs for the Accelerator Breeder[14]

Accelerator Power Production	Comparison of low-power, break-even, and 1 GW(e) power Producer
Capital Cost, Accelerator	\$600M at 16% charge rate
Capital Cost, Balance of Plant	\$800/kW for power generating capacity, 16% charge rate
Initial Inventory	Fabrication at \$150/kg h.m.
Operating Costs, Power	50% accelerator efficiency and power cost of 30 mills/(kW•hr), requires 600 MW(e)
Operating Costs, Personnel and Maintenance	Accelerator requires ∿ 130 employees at \$50k/man-yr and 50% maintenance plus power production cost of 3 mills/(kW•hr)
Fuel Cycle Costs	\$300/kg h.m. processing plus \$150/kg h.m. refabrication (power and heavy metal densities constant)
Fissile Product Cost	2.5 kg/day fissile production at 80% capacity factor; 30 mills/(kW•hr) for power sales price
Fissile Product Cost	4.0 kg/day fissile production at 80% capacity factor; 30 mills/(kW•hr) power sales price, plus \$18.3M additional fuel cycle costs

Table VI-5. Estimated Annual Costs for the Accelerator Breeder

Accelerator Power Production		200 MW(e)	600 MW(e)	1600 MW(e)
Capital Cost, Accelerator		$ 96.0M	$ 96.0M	$ 96.0M
Capital Cost, Balance of Plant		$ 25.6M	$ 76.8M	$204.8M
Initial Inventory		$ 0.3M	$ 0.9M	$ 2.4M
Operating Costs, Power		$ 84.1M	--	--
Operating Costs, Personnel and Maintenance		$ 15.0M	$ 22.6M	$ 43.8M
Fuel Cycle Costs	2.5 kg/day[a]	$ 8.3M	$ 8.3M	$ 8.3M
Total Cost		$229.3M	$204.6M	$355.3M
Fissile Product Cost		$314/g Pu	$281/g Pu	$199/g Pu
Fuel Cycle Costs	4.0 kg/day[b]	$ 26.6M	$ 26.6M	$ 26.6M
Total Cost		$247.6M	$222.9M	$373.6M
Fissile Product Cost		$212/g Pu	$191/g Pu	$140/g Pu

a. Pb target with LMFBR-type blanket.

b. GCFR-type target blanket.

estimates for ^{235}U. as a function of various U_3O_8 ore and separative work unit (SWU) costs, along with explanations of the significance of the selected cost levels. The estimated cost of accelerator-produced Pu given on Table VI-5 are quite comparable to the estimated ^{235}U costs involving additional separative work capacity and low-grade uranium sources. However a comparison strictly on the basis of fissile price per gram does not take into account the relative worths of the fuels for power production. A more meaningful comparison is the cost of power produced in LWR's utilizing accelerator-produced fuel with LWR power costs using ^{235}U obtained under several predicted ore and separative work price conditions.

Estimated LWR Power Costs as a Function of ^{235}U Costs

Estimate LWR power costs as a function of ^{235}U costs are given in Table VI-7. The \$/g values considered are comparable to those presented in Table VI-6 for various ore and separative work costs. Power cost estimates are given for both currently operating LWR's with an assumed non-fissile power cost of 12 mills/kw.hr, and for LWR's now being planned. Presumably the later LWR's, with assumed non-fissile power costs of 25 mills/kw.hr, would dominate the market at the time when the accelerator-produced fuel is available. Total power cost estimates for these later LWR's range from 31 mills/kw.hr using ^{235}U now ordered (escalated to 1983-1985 delivery), to 55 mills/kw.hr for ^{235}U from low-grade uranium sources processed at new separative work facilities.

Initial Estimates of LWR Power Costs Utilizing Accelerator-Breeder Produced Fuel

Table VI-8 presents the LWR power cost and power level sustained for the estimated Pu and ^{233}U production levels in the Pb target - LMFBR blanket and GCFR-type target-blanket designs. The resulting power costs are quite comparable to the 42 mills/kw.hr estimate given in Table VI-7 for \$200/g ^{235}U, except for the lower Pu production level cost. Since the \$200/g ^{235}U price corresponds to \$300/lb U_3O_8 and \$200/SWU, this is in the medium price-

Table VI-6. Estimated Cost of U^{235} (\$/g) for Various Ore and Separative Work Costs[14]

Natural U_3O_8 (\$/lb)		25	40[a]	70[b]	100	300	500[d]
SWU Cost (\$/SWU)		25	32	50	150[c]	200	250
Natural Uranium	U_3O_8	9.2	14.7	25.6	36.6	110	183
(CANDU Fuel)	SWU	--	--	--	--	--	--
	\$/g Total	9.2	14.7	25.6	36.6	110	183
3.2% Enriched	U_3O_8	13.3	21.2	37.2	53.1	159	265
	SWU	3.6	4.5	7.1	21.4	28	35
	\$/g Total	16.9	25.7	44.3	74.5	187	300
93.5 Enriched	U_3O_8	14.2	22.8	39.8	56.9	171	284
	SWU	6.2	8.0	12.5	37.5	50	75
	\$/g Total	20.4	30.8	52.3	94.4	221	359

a. Cost estimates for current delivery.

b. Cost estimates for current order (1983 delivery), assuming 7% inflation rate.

c. Escallation in SWU costs primarily due to necessity for construction of additional separative capacity at ∿ 1980 construction costs.

d. Estimated cost of U_3O_8 recovered from low-grade uranium sources such as Chattanooga shale vary from \$200-\$1000/lb.

Table VI-7. Estimated LWR Power Costs for Existing and Planned Nuclear Plants as a Function of Fissile Cost

Fissile Cost ($/g)	15	20	30	50	100	200	350
Estimated LWR Fissile Costs $\left(\frac{\text{Mills}}{\text{kW}\cdot\text{hr}}\right)$	3	4	5	6	8	17	30
Operating LWR Reactor Power Cost[a] $\left(\frac{\text{Mills}}{\text{kW}\cdot\text{hr}}\right)$	15	16	17	18	20	29	42
Planned LWR Reactor Power Cost[b] $\left(\frac{\text{Mills}}{\text{kW}\cdot\text{hr}}\right)$	28	29	30	31	33	42	55

a. Estimated operating reactor power cost of 12 $\frac{\text{mills}}{\text{kW}\cdot\text{hr}}$ without fissile cost.

b. Estimated planned reactor power cost of 25 $\frac{\text{mills}}{\text{kW}\cdot\text{hr}}$ without fissile cost,[14] based on:

20 $\frac{\text{mills}}{\text{kW}\cdot\text{hr}}$ Capital Cost ($800/kW at 16% charge rate)

2 $\frac{\text{mills}}{\text{kW}\cdot\text{hr}}$ Operation and Maintenance

3 $\frac{\text{mills}}{\text{kW}\cdot\text{hr}}$ Fuel Processing and Fabrication

Table VI-8. Initial Estimates of Symbiotic Accelerator Breeder-LWR Power Costs

Fuel (Fissile Isotope)	Fissile Production (kg/day)[a]	LWR Power Sustained (MW(e))	Power Cost Mills/(kW·hr)[b]
Pu	4.0	1980	42
	2.5	1238	50
U^{233}	2.8	2640	38
	1.8	1697	43

a. Production rates correspond to the Pb target and LMFBR blanket design for lower values (2.5 kg/day Pu and 1.8 kg/day U^{233}, and to the GCFR-type target-blanket for the higher production rates.

b. Assumed ABC break-even power case with total cost of $204.6M/yr for Pb-LMFBR and $222.9M/yr for the GCFR-type design, with non-fissile LWR production costs of 25 $\frac{\text{mills}}{\text{kW·hr}}$.

estimate range for ^{235}U recovered from low-grade uranium ore utilizing separative work capacity not currently available. The neutronic advantage of ^{233}U over Pu as a thermal reactor fuel is translated into an economic advantage in power costs in spite of the lower ^{233}U production rates.

Estimated Power Cost and Power Level Sustained for A Symbiotic Relationship Between The Accelerator-Breeder and Pebble Bed Near-Breeder Reactors

Table VI-9 presents a lifetime history of a symbiotic accelerator-breeder-pebble bed near breeder reactor network system. Time-dependent power production rates and total power production are given, along with average power cost. The accelerator-breeder considered is the GCFR-type target-blanket design, and the reactor inventory, makeup, and power output correspond to the Julich design,[18] derated to 0.90 conversion ratio, including processing losses. Table VI-9 estimates that the accelerator-breeder can provide the inventory and makeup requirements for up to four 1200 MW(e) pebble bed near-breeder reactors. This allows the startup of a 1200 MW(e) plant roughly every six years. In a 30-year lifetime, the fuel produced enables the generation of 130.4 GW(e) over a 66-year power production period. The average fissile cost for the power produced in this system is about 6 mills/kw.hr, which gives a total estimated average power cost of 31 mills/kw.hr, assuming non-fissile power costs of 25 mills/kw.hr. This power cost [31 mills/kw.hr] is quite comparable to that estimated for 1980-vintage LWR-produced power at \$30-\$50/g ^{235}U. Although an 80% capacity factor is admittedly optimistic and may even constitute an upper bound, the economics of accelerator breeder, advanced converter symbiosis appears promising even at lower availability factors. For example, if the above accelerator breeder operates at 50% capacity, the fissile cost for a symbiotic system with the pebble-bed reactor described in Table VI-9 is still only 10 mills/kw.hr, which is comparable to LWR produced-power at \$125/g ^{235}U. Even for the accelerator breeder model with the lower production rate (1.8 kg ^{233}U/day) operating at 50% capacity, the fissile cost for a symbiotic system with the pebble-bed advanced converter from Table VI-9 appeared comparable to LWR produced-power at ∿ \$200/g ^{235}U.

Economic Observations

1. Fissile material produced by the accelerator-breeder appears competitive with low-grade uranium sources.
2. Symbiotic relationships between the accelerator-breeder on a Th-^{233}U cycle and advanced converters appear promising for the development of large-capacity power grids.
3. Production of ^{233}U shows a definite neutronic advantage over Pu, even at lower ^{233}U production rates.
4. Fissile production costs show little variation from low-power to break-even operation.
5. Estimated power production costs appear promising enough to warrant further investigation.

Table VI-9. Speculative Symbiotic Relationship Between Accelerator-Breeder and Pebble Bed Near-Breeders Over 30 Years

1. Accelerator-Breeder produces 2.8 kg U^{233}/day (818.2 kg/yr at 80% capacity) and loses 2% in reprocessing.
2. Pebble-Bed Near-Breeder has inventory of 2900 kg U^{233}, requires 144 kg/yr makeup for 1200 MW(e) plant,[18] derated from 0.97 conversion ratio to 0.90, including processing losses.
3. History of accelerator lifetime:

(Consecutive)	Plant Power (MW(e))	Power Produced (MW(e)·Yr at 75% Capacity)
3.62	0	0
4.41	1200	3,968
5.64	2400	10,159
7.84	3600	21,171
8.49	4800	30,564
TOTAL 30 Years		65,862

4. Period after accelerator shutdown (assuming reactors have 30-year lifetime, and utilizing remaining 1917 kg U^{233} at end accelerator lifetime)

(Consecutive)	Plant Power (MW(e))	Power Produced (MW(e)·Yr at 75% Capacity)
3.33	4800	11,981
4.70	3600	12,690
12.49	2400	22,477
19.35	1200	17,397
TOTAL 40 Years		64,545
GRAND TOTAL 70 Years		130,407

5. Fissile Economics - Assuming accelerator operates at power break-even and ($222.9 M/yr), can produce fuel for 130.4 GW(e) over 70-year period at a fissile cost of about $6\ \frac{\text{mills}}{\text{kW}\cdot\text{hr}}$. This gives a total estimated power cost of $31\ \frac{\text{mills}}{\text{kW}\cdot\text{hr}}$ with non-fissile power charges of $25\ \frac{\text{mills}}{\text{kW}\cdot\text{hr}}$. A full inventory (2900 kg ^{233}U) remains at the conclusion of the above scenario.

VII. CONCLUSIONS

If economic feasibility for ABACS is reasonably close to our current estimates, this is an attractive alternative to the fast breeder. With advanced converter reactors, ABACS could supply a substantial fraction of our electricity demand for a thousand or more years at a cost of about 30 mills/kWh in current dollars. The ABACS alternative without development of advanced converters gives electricity costs of about 45 mills/kWh which may be attractive considering non-nuclear alternatives.

Another important factor of ABACS, however, is that accelerator breeders appears inherently safer than the fast breeders and allow greatly increased flexibility in the fuel cycle which permits mitigation of several of the perceived problems of fission technology. Over the next 30 to 100 years, our concerns may vary regarding safety, waste disposal, and diversion, but the ABACS flexibility allows adjustment of the fuel cycle to meet changing concerns.

Figure VII-1 show two of the many possible symbiotic fuel cycles available with ABACS. Note that the Pu cycle supports about 2000 MWe with 18.8 kg/day in Pu^{239} flow. The U^{233} cycle with an advanced converter with CR = .90 including processing losses supports nearly 7000 MW(e), and there is a cycle flow of 53 kg/day which is proportionally less than that for the Pu cycle. The U^{233} cycle supports the maximum electrical output per accelerator at the lowest power cost. Also the actinide hazard is reduced by as much as 10^6 with the isotope chain essentially terminating with Np^{237}.

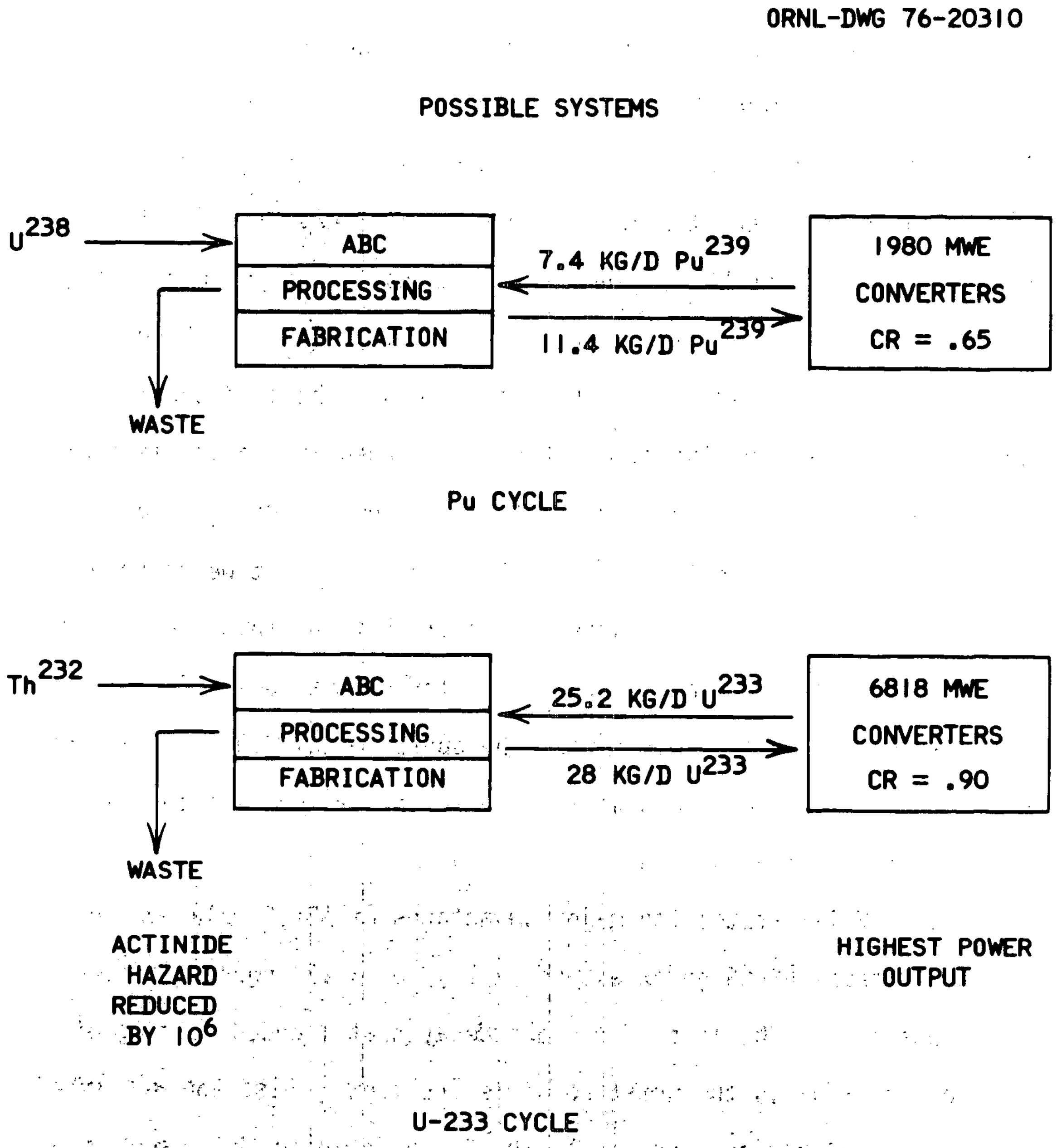


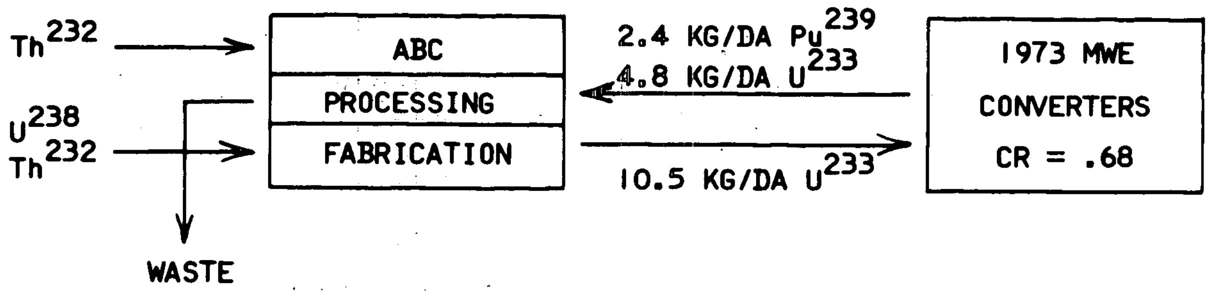
Fig. VII.1. ABACS Flow Sheets for Pu and U^{233} Cycles

Figure VII-2 shows two additional options. In the denatured fuel cycle the U^{233} output from the accelerator is diluted to 7% in U^{238} to make the material unattractive for diversion, and the uranium mixture is further mixed with thorium. The reactor produces Pu^{239} and U^{233} at 1 to 2 ratios with a CR = .68. The accelerator burns Pu^{238} which boosts U^{233} production to about 6 kg/day. Only 2.4 kg/day of Pu^{239} is in circulation for 2000 MW(e) generation and that is only found in highly radioactive spent fuel along with U^{233}, U^{238}, and Th^{232} making it also unattractive for diversion. Actinide waste will be produced in this cycle, but, like the plutonium, it can be burned in the accelerator blanket which effectively converts the actinides to U^{233}. The U^{233}, U^{234}, U^{235} fuel production is not well understood, but we do know that such fuel can be produced in the accelerator blanket and that the conversion ratio for the reactors may exceed unity for such a fuel. The material circulating in the fuel cycle would be approximately 50% fissile which would make it somewhat less attractive for diversion, and the cycle would produce essentially no actinide waste.

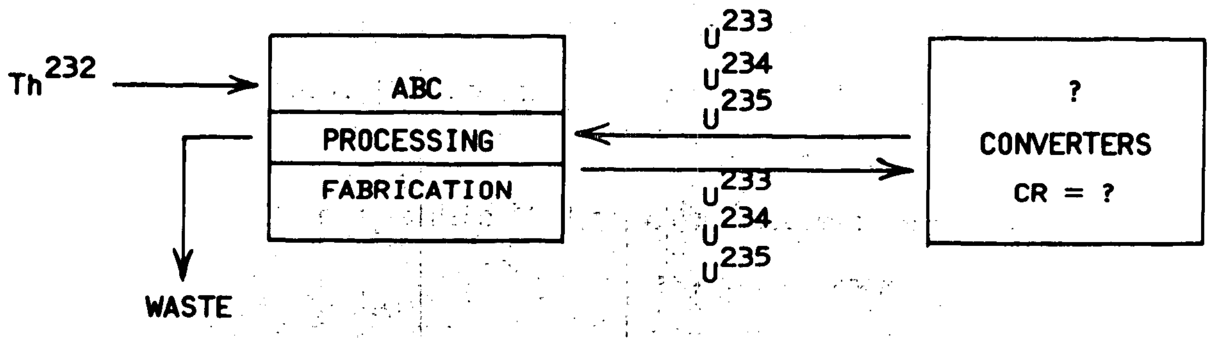
Table VII-1 states the major advantages of ABACS relative to the LMFBR. Because ABACS never exceeds a k_{eff} of $\sim .7$, energetic accidents are impossible. The loss of coolant decay heat induced melting of some blanket material is the worst possible accident. Also the accelerator blanket contains less than 1/4 of the Pu found in an LMFBR and produces about twice the breeding product. Table VII-2 lists the further advantages of the U^{233} cycle ABACS compared with the Pu cycle. Up to four times as much electrical capacity is supported per accelerator. Some possible fuel cycles mitigate diversion threat. A pure U^{233} fuel cycle essentially

ORNL-DWG 76-19900

POSSIBLE SYSTEMS



DENATURED 7% U-233 IN U-238 CYCLE



DENATURED U-233 U-234 U-235 CYCLE

Fig. VII.2. ABACS Flow Sheets for Denatured Fuel Cycles

Table VII-1. Advantages of Pu Cycle ABACS vs LMFBR

1. ABACS eliminates potential for energetic accidents.
2. ABACS Pu inventory in breeder is much less.

Table VII-2. Advantages of U^{233} Cycle vs Pu Cycle ABACS

1. More electrical capacity supported.
2. Denatured U^{233} fuel cycles may mitigate diversion and proliferation threats.
3. A U^{233} fuel cycle would substantially reduce actinide hazard.
4. ABACS breeder supplying U^{233} enables use of advanced converters.

eliminates actinide waste. Lastly, the accelerator breeder by producing U^{233} directly from Th^{232} enables the use of advanced converters which require an initial load of U^{233} rather than U^{235} in order to function well.

Overall conclusions are listed in Table VII-3. The potential application of several of the possible fuel cycles looks promising. Much needs to be done, however, to build up a body of design experience needed to assess the engineering feasibility of accelerator breeding.

It should be noted that these preliminary cost estimates for the accelerator breeder are subject to many uncertainties in accelerator and target performance, availability, breeding, and in the reliability of the combined accelerator breeder power production system. In particular, the U_3O_8 ore and SWU cost estimates given in Table VI-6 assume a fairly aggressive, expanding nuclear economy, and a nuclear slowdown would result in a corresponding reduction in ore price and SWU cost escalation.

Table VII-3. Overall Conclusions

1. Accelerator breeding is technically feasible based on moderate extensions of present technology, but much work must be done before engineering feasibility is known.
2. Although U^{238}-Pu^{239} production may be 25% higher than Th^{232}-U^{233} production, the U^{233} production is more important because of lower makeup requirements in the converters, improved safety and safeguards concerns, and the elimination of actinide waste.
3. In order to keep the power level nearly constant, the accelerator breeder target and blanket fuel must be processed nearly continuously.
4. High efficiency coolants degrade the neutron spectrum and substantially lower fuel production.
5. Radiation damage may severely limit the lifetimes of target/blanket structures.
6. The potential for accelerator breeding justifies an initial study to determine conceptual feasibility within sound engineering and economic judgement.

REFERENCES

1. Livermore Research Laboratory, Status of the MTA Process, Livermore Research Laboratory Report LRL-102 (1954).

2. W. A. Gibson et al., "Low-Energy Neutron Production by High-Energy Proton Bombardment of Thick Targets," Electronuclear Division Annual Progress Report for Period Ending December 31, 1965, ORNL-3940 (May 1966).

3. W. A. Coleman and R. G. Alsmiller, Jr., "Thermal-Neutron Flux Generation by High-Energy Protons," Nuclear Science and Engineering 34, 104-113 (1968).

4. An AECL Study for an Intense Neutron Generator, Edited by G. A. Bartholomew and P. R. Tunnicliffe, Atomic Energy of Canada, Ltd., Report AECL-2600 (1966).

5. P. V. Livdahl, Private communication, (October 1976).

6. "Choice of Initial Operating Parameters for High Current Linear Accelerators," in Proceedings of the 1976 Proton Linear Accelerator Conference, Chalk River, Canada, September 1976 (In Press).

7. B. C. Knapp et al., "Resonantly Coupled Accelerating Structure for High Current Proton Linacs," IEEE Trans. Nucl. Sci., NS-12 No. 3, 159 (1965).

8. P. R. Tunnicliffe, et al., "High Current Proton Linear Accelerators and Nuclear Power," in Proceedings of the 1976 Proton Linear Accelerator Conference (See ref. 6).

9. D. C. Hagerman, "High Duty Factor RF Sources at 800 MHz," IEEE Trans. Nucl. Sci. NS-14, No. 3, 197 (1967).

10. PEP Conceptual Design Report, LLB-4288/SLAC 189, February 1976.

11. K. C. Chandler and T. W. Armstrong, "Operating Instructions for the High Energy Nucleon-Meson Transport Code HETC," ORNL-4744 (1972).

12. W. W. Engle, Jr., "User's Manual for ANISN, A One Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," USAEC Report K-1693 (1967).

13. E. A. Straker, P. N. Stevens, D. C. Irving, and V. R. Cain, "The MORSE Code - A Multigroup Neutron and Gamma Ray Monte Carlo Transport Code," ORNL-4585 (1970).

14. P. R. Kasten, et al., "Assessment of Thorium Fuel Cycle in Power Reactors," ORNL/TM-5565 (December 1976).

15. R. A. Matzie and J. R. Rec, "Assessment of Thorium Fuel Cycles in Pressurized Water Reactors," TIS-5114, Combustion Engineering, Inc. (November 1976).

16. H. A. Feiveson and T. B. Taylor, "Keeping the Plutonium Genie in a Box, A Strategy for the Control of Nuclear Power," Draft, Center for Environmental Studies, Princeton University, Princeton, N. J., Revised as "The Security Implications of Alternative Fission Futures," to be published in December 1976 issue of the Bulletin of the Atomic Scientists.

17. M. J. Bell, "ORIGEN: The ORNL Isotope Generation and Depletion Code," ORNL-4628 (May 1973).

18. E. Teuchert, H. J. Rutten, H. Werner, "Performance of Thorium Fuel Cycles in the Pebble Bed Reactor," Trans. Am. Nucl. Soc., <u>24</u>, 222 (1976).

ORNL/TM-5750
Distribution Category
UC-34c

INTERNAL DISTRIBUTION

1-3. L. S. Abbott
4. N. J. Ackermann, Jr.
5. R. G. Alsmiller, Jr.
6. S. I. Auerbach
7. J. Barish
8. D. E. Bartine
9. H. W. Bertini
10. E. S. Bettis
11. R. S. Booth
12. C. J. Borkowski
13. W. D. Burch
14. T. J. Burns
15. G. T. Chapman
16. R. L. Childs
17. J. F. Clarke
18. J. C. Cleveland
19. C. E. Clifford
20. S. N. Cramer
21. F. L. Culler
22. J. W. T. Dabbs
23. M. B. Emmett
24. J. R. Engel
25. W. W. Engle, Jr.
26. G. G. Fee
27. G. F. Flanagan
28. W. Fulkerson
29. T. A. Gabriel
30. H. Goldstein, Consultant
31. N. M. Greene
32. R. Gwin
33. W. O. Harms
34. E. Herst
35. T. J. Hoffman
36. J. Horak
37. D. T. Ingersoll
38. D. Jenkins
39. P. R. Kasten
40. O. L. Keller
41. C. C. Koche
42. D. C. Larson
43. R. S. Livingston
44. R. E. Maerker
45. F. C. Maienschein
46. J. H. Marable
47. J. A. Martin
48. B. F. Maskewitz
49. F. J. Muckenthaler
50-189. F. R. Mynatt
190. J. P. Nichols
191. E. M. Oblow
192. J. V. Pace
193. R. W. Peelle
194. F. G. Perey
195. L. M. Petrie
196. H. Postma
197. W. A. Rhoades
198. C. R. Richmond
199. J. C. Robinson
200. M. W. Rosenthal
201. J. E. Rushton
202. M. J. Saltmarsh
203. R. T. Santoro
204. D. L. Selby
205. W. D. Shults
206. E. G. Silver
207. D. B. Simpson
208. C. O. Slater
209. I. Spiewak
210. P. H. Stelson
211. P. N. Stevens
212. E. T. Tomlinson
213. D. B. Trauger
214. D. R. Vondy
215. C. R. Weisbin
216. J. E. White
217. G. E. Whitesides
218. M. L. Williams
219-220. A. Zucker
221. P. Greebler, Consultant
222. W. W. Havens, Consultant
223. A. F. Henry, Consultant
224. R. E. Uhrig, Consultant
225-226. Central Research Library
227. ORNL Y-12 Technical Library,
Document Reference Section
228-229. Laboratory Records Department
230. Laboratory Records - RC

EXTERNAL DISTRIBUTION

231. P. F. Fox, Manager, Reactor Engineering, FFTF Project, Westinghouse Advanced Reactor Division, Madison, PA 15663.

232-233. Energy Research and Development Administration, Division of Reactor Research and Development, Washington, DC 20545: Director.

234. Energy Research and Development Administration, Reactor Division, P. O. Box E, Oak Ridge, TN 37830: Director

235. Energy Research and Development Administration, Oak Ridge Operations Office, Research and Technical Support Division, P. O. Box E, Oak Ridge, TN 37830: Director.

236-266. Given High-Energy Accelerator Shielding Distribution.

267-389. For distribution as shown in TID-4500 Distribution Category UC-34c, Physics - Nuclear.