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A SYSTEMATIC METHODOLOGY FOR ESTIMATING DIRECT CAPITAL COSTS FOR BLANKET TRITIUM PROCESSING SYSTEMS*

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ABSTRACT

This paper describes the methodology developed for estimating the relative capital costs of blanket processing systems. The capital costs of the nine blanket concepts selected in the Blanket Comparison and Selection Study are presented and compared.

I. INTRODUCTION

The primary goal of the Blanket Comparison and Selection Study[1] was to evaluate different fusion blanket concepts in order to select those which had the most favorable characteristics for a commercial fusion reactor. Detailed selection criteria were generated in four areas: engineering feasibility, economics, safety and R&D requirements.

An area in which major capital costs are incurred is the blanket processing system. Because this system is an important part of each of the nine blanket concepts considered, it was necessary to develop a self-consistent methodology which could be used to estimate the relative capital costs of each blanket processing system. Since there was minimal detailed information for any of the blanket concepts, a relative costing basis was developed. This paper describes the methodology for costing a blanket processing system and compares the results for the nine concepts selected in the Blanket Comparison and Selection Study (BCSS).

II. BASIC ASSUMPTIONS

The blanket system includes not only the blanket itself but also all the necessary systems needed for its operation. This includes the safety systems in the plant for control of tritium release. Because consistency was needed in the analysis of the nine blanket recovery systems it was considered advantageous to assume that each blanket recovery system could be separated into component subsystems which could be individually costed. The total cost for a given blanket concept was the total of all component subsytems. Systems common to each of the blanket concepts would thus be based on a common relative basis. The individual components were these: 1) blanket tritium recovery and associated tritium purification system; 2) processing systems for the blanket coolant to control tritium migration into the reactor hall and/or the heat exchanger systems; 3) coolant purification systems; 4) coolant dump tanks; and 5) tritium extraction systems for tritium recovery from blanket modules removed from the reactor. The cost of tritium control in limiter/first wall or halo scraper/direct convertor systems were also included since for some blanket concepts, a common tritium recovery system could be used which would reduce costs. The summation of each of these costs is the total capital cost of the blanket processing system. To provide a total capital cost of the processing systems needed for a fusion reactor, fuel processing costs and the atmospheric tritium recovery costs were added to the blanket processing costs. The capital cost of water purification systems were assumed to be included in the balance of plant cost.

Information on each of the nine blanket concepts was provided by the responsible design team in the Blanket Comparison and Selection Study. This data is summarized in Table 1 for the tokamak designs and in Table 2 for the tandem mirror designs. The information in these tables is the data used for costing the blanket processing system needed for each blanket concept.

III. METHODOLOGY

The nine blanket concepts are assumed to be operated in either a tokamak (STARFIRE reference)[2] or a mirror (MARS reference)[3] reactor. The fusion power produced by these reactors is 4000 MW and 3200 MW respectively. The capital cost of a fuel processing system[4] for reactors of this size includes both a fixed cost portion (~\$20 M) and a variable cost (\$10-20 M) dependent on the throughput of the reactor. Since this variable cost is dir-

TABLE I. A SUMMARY OF THE TRITIUM PROCESSING PARAMETERS SUPPLIED[a] FOR THE BLANKET CONCEPTS

	Tokamak Concept[b] Description	Mass Flow Rate to the Primary Recovery System (g/d)	Added Protium (g/d)	Tritium Partial Pressure in Breeder (Pa)	Tritium Load into Coolant (g/d)	Tritium Partial Pressure in the Coolant (g/d)	Tritium Load to Steam Generator (Ci/d)	Tritium Inventory[c] Location	Tritium Inventory[c] Amount (g)
A	$LiAlO_2$/Be NS/SG HT-9 Limiter/H_2O[d] Helium Purge	924	9×10^4	1.3×10^{-2}	0.49	Not Available	Not[e] Available	First Wall Purge Coolant Blanket	19 10^{-6} 90 2000
B		---	---	---	---	---	---	---	---
C		---	---	---	---	---	---	---	---
D	Li Li/Na/SG V Limiter/Li Molten Salt/Electrolysis	1121	---	8.6×10^{-9}	0.03	---	3	First Wall Purge Coolant Blanket	6 --- --- 490
E	Li_2O He/SG HT-9 Limiter/H_2O Helium Purge	828	9×10^4	6.9×10^{-1}	1.86	3.6×10^{-4}	32	First Wall Purge Coolant Blanket	19 10.5 --- 136
F	$LiAlO_2$/Be He/SG HT-9 Limiter/H_2O Helium Purge	823	9×10^4	7.5×10^{-1}	12.45	1.3×10^{-3}	48	First Wall Purge Coolant Blanket	19 10 10^{-5} 38
G	Li He/SG HT-9 Limiter/Li Y beds	866	---	10^{-7}	1.22	9.8×10^{-7}	27	First Wall Purge Coolant Blanket	19 --- 10^{-7} 330
H	FLIBE/Be He/SG HT-9 Limiter/H_2O Electrolysis	846	---	2600	172.22	1.9×10^{-1}	260	First Wall Purge Coolant[f] Blanket Blanket Structure	19 --- 0.2 0.5 189[g]
I	$LiAlO_2$/Be H_2O/SG HT-9 Limiter/H_2O Helium Purge	869	5.5×10^3	2.2	1.23	6×10^{-6}	2	First Wall Purge Coolant Blanket	19 0.03 53 2300

[a]The information was supplied by the team responsible for a given design.

[b]The first line is the breeder; the second line is the coolant; the third line is the structure; the fourth line is the coolant for the limiter/divertor; the fifth line is the tritium recovery method used. The abbreviations used are these: Be - beryllium; NS - nitrate salt; SG - steam generator; $LiAlO_2$ - γ-lithium aluminate, HT-9 - a ferritic steel; Li - lithium; Na - sodium; V - a vanadium alloy; Li_2O - lithium oxide; He - helium, and FLIBE - a lithium beryllium fluoride salt.

[c]The tritium inventory was used to size the atmospheric tritium recovery system. The options chosen were a 3-day cleanup for high probability releases (>10^{-2}) and 5-day cleanup for low probability releases (< 10^{-2}).

[d]For all limiters, it was assumed that 1 g/d of tritium permeated into the coolant.

[e]It was assumed that the tritium load was <20 Ci/d. If this is not valid, an additional control system needs to be included and costed.

[f]An assumed value since none was supplied. It was derived by scaling from the other helium concepts.

[g]Only blanket structure with an inventory >10 g.

TABLE 2. A SUMMARY OF THE TRITIUM PROCESSING PARAMETERS SUPPLIED[a] FOR THE BLANKET CONCEPTS

	Mirror Concept[b] Description	Mass Flow Rate to the Primary Recovery System (g/d)	Added Protium (g/d)	Tritium Partial Pressure in Breeder (Pa)	Tritium Load into Coolant (g/d)	Tritium Partial Pressure in the Coolant (g/d)	Tritium Load to Steam Generator (Ci/d)	Tritium Inventory[c] Location	Location (g)
A	$LiAlO_2$/Be NS/SG HT-9 HSDC/H_2O[d] Helium Purge	619	9×10^4	1.3×10^{-2}	0.45	Not Available	Not[e] Available	First Wall Purge Coolant Blanket	3.7 10^{-4} 50 250
B	Li Li/Na/SG HT-9 HSDC/H_2O Molten Salt/Electrolysis	564	---	8.6×10^{-9}	0.02	---	2	First Wall Purge Coolant Blanket	4 --- --- 336
C	LiPb LiPb/SG V HSDC/H_2O Counter Current He	731	4.2×10^6	1.3×10^{-4}	---	---	10	First Wall Purge Coolant Blanket	1 <1 --- <1
D	Li Li/Na/SG V HSDC/H_2O Molten Salt/Electrolysis	735	---	8.6×10^{-9}	0.02	---	2	First Wall Purge Coolant Blanket	1 --- --- 336
E	Li_2O He/SG HT-9 HSDC/H_2O Helium Purge	550	5.6×10^4	7.8×10^{-1}	0.85	6.4×10^{-4}	20	First Wall Purge Coolant Blanket	4 1 10^{-5} 131
F	$LiAlO_2$/Be He/SG HT-9 HSDC/H_2O Helium Purge	554	5.7×10^6	8.4×10^{-1}	8.51	1.9×10^{-3}	43	First Wall Purge Coolant Blanket	4 1 10^{-4} 24
G	Li He/SG HT-9 HSDC/H_2O Yt Beds	558	---	1.1×10^{-7}	0.2	7.8×10^{-7}	17	First Wall Purge Coolant Blanket	4 --- 10^{-7} 233
H	FLIBE/Be He/SG HT-9 HSDC/H_2O Electrolysis	622	---	2600	83.2	1.4×10^{-1}	191	First Wall Purge Coolant[f] Blanket Blanket Structure	4 --- 0.14 0.4 139[g]
I	$LiAlO_2$/Be H_2O/SG HT-9 HSDC/H_2O He Purge	586	3.7×10^3	2.2	0.21	6×10^{-6}	2	First Wall Purge Coolant Blanket	4 0.03 53 1500

[a]The information was supplied by the team responsible for a given design.

[b]The first line is the breeder; the second line is the coolant; the third line is the structure; the fourth line is the coolant for the halo scraper/direct converter the fifth line is the tritium recovery method used. The abbreviations used are these: Be - beryllium; NS - nitrate salt; SG - steam generator; $LiAlO_2$ - γ-lithium aluminate; HT-9 - a ferritic steel; Li - lithium; Na - sodium; V - a vanadium alloy; Li_2O - lithium oxide; He - helium; FLIBE - a lithium beryllium fluoride salt; LiPb - 17Li-83Pb; and HSDC - halo scraper/direct converter.

[c]The tritium inventory was used to size the atmospheric tritium recovery system. The options chosen were a 3-day cleanup for high probability releases ($>10^{-2}$) and 5-day cleanup for low probability releases ($< 10^{-3}$).

[d]For all mirror concepts it was assumed that ~1 g/d of tritium entered the water system cooling the halo scrapers and the direct converter.

[e]It was assumed that the tritium load was <20 Ci/d. If this is not valid, an additional control system needs to be included and costed.

[f]An assumed value since none was supplied. It was derived by scaling from the other helium concepts.

[g]Only blanket structure with an inventory >10 g.

ectly proportional to the fusion power and inversely proportional to the fractional burn, the capital cost of the fuel processing system of the mirror reactor which has a fusion power ~80% of the tokamak's and a fractional burn ~140% of the tokamak's is $31 million and the tokamak's is $40 million.

A. Atmospheric Tritium Recovery

An atmospheric tritium recovery system (ATR) processes the atmosphere to reduce the tritium concentration after a tritium release to levels $\leq$ 50 μCi/m^3. Releases are assumed to occur during maintenance or under accident conditions. The capital cost of the system needed for a given reactor hall is a function of the hall's size, i.e. volume, and also of the processing rate. For BCSS, the baseline volume for both the tokamak and the mirror designs was 2×10^5 m^3. The processing rate considered adequate for a commercial power plant is one which can process releases with a high probability of occurrence ($p > 10^{-2}$) in three days and those with a lower probability of occurrence ($p \leq 10^{-3}$) in five days. The tradeoff between the size of a release and the capital cost of the ATR needed is shown in Table 3. It is assumed that in a commercial power plant, the most likely accident is breakage of a pipe containing gaseous tritium. Amounts released would be of the order of 10 g. This also is the amount of tritium in the first wall of all seven tokamak blanket concepts (Table 1). Accidents with lower probability would include release of all tritium in the blankets of Concepts B, G, or D (<1 kg) or Concepts A or I ($\leq$2.3 kg) (Tables 1 and 2). A system with a capital cost of ~$36 million provides cleanup of events with high probability in ~3 days and events with low probability in ~5 days. Since the ATR in the reactor hall handles releases from systems other than the blanket, this is probably the optimum system that can be specified with the limited information available.

B. Implantation Tritium Control

The size of the tritium control system for the limiter/divertor or the halo scraper/direct convertor depends on the rate of tritium implantation and migration. It is assumed for this study that this is equivalent to 1 g/d for all designs. The limiter coolant is either lithium or water. It is assumed that for a lithium coolant, the cost of removing the tritium is a small increment to the blanket tritium removal system. For water the general equation for the capital cost, C, of removing tritium from a water coolant using CECE[5] (combined electrolysis and chemical exchange) is:

$$C(\$M) = 15\left[\frac{(Ci/d)}{(Ci/L)\ (1000)\ (L/d)}\right]^{0.55}. \qquad (1)$$

If tritium levels of 1 Ci/L are maintained in the water coolant, and water is processed at the rate of 10^2, 10^3, 10^4 or 10^5 L/D, then the capital cost is 6, 15, 50 or 100 million dollars, respectively.

TABLE 3. CAPITAL COST OF ATR SYSTEM AS FUNCTION OF PROCESSING RATE[a]

Release (g)	Cost (72 h) ($M)	Cost (120 h) ($M)
10	36[b]	25
50	40	28
100	42	29
500	46	32
1000	48	33
2300	50	35[b]

[a] Cleanup to 50 μCi/m^3 for a volume 2×10^5 m^3. (Ref. 5).

[b] The same system cleans up a 10 g release in 72 h or a 2.3 kg release in 120 h.

C. Blanket-Tritium Recovery and Purification

There are two types of tritium recovery and purification systems considered: the first is that associated with solid breeders (Li_2O, $LiAlO_2$); the second is that associated with liquid breeders (Li, LiPb, Flibe).

For the solid breeder, an in-situ tritium recovery system consists of a helium purge stream with associated pumps, and a system for removing tritium from the purge stream. In addition, there is a tritium purification system, a system for handling protium for tritium control, and a system for removing tritium from discarded solid breeder modules.

For the liquid breeder, there is a basic tritium recovery system consisting of sets of cycled yttrium getters, a molten salt/electrolysis unit, a counter-current helium flow system, or an electrolysis unit. In addition, there is a tritium purification system and a system for processing protium if this is necessary for control of tritium in the heat exchanger.

The capital cost associated with the tritium recovery system for both liquid and solid breeders was determined as follows. A base design which could process 600 – 800 g/d was assumed for both tokamak and mirror machines. Higher processing rates >800 g/d would be handled by a larger system whose capital cost would be an increment of the base cost. To a first approximation, it was assumed that each blanket tritium recovery system would be of comparable cost. Since there are not pilot plant versions of these systems it was assumed that the capital cost would be approximated by costing similar equipment in other industries. This cost could be increased by an appropriate factor to account for the tritium requirement - i.e., all metal systems, permeation barriers, etc. The unit which served as a baseline for deriving capital costs was a centrifugal contactor. These units are used in the chemical industry and one has been designed[6] for use in fission fuel reprocessing. The cost of this unit was scaled to derive the cost of one needed to process liquid metals. The volume of material in the liquid metal designs was $\sim 10^6$ L. If this volume is processed 100 times a day the processing rate is 4×10^5 L/min. The capital cost of an unheated one-state contactor not designed for use with tritium is \$2.5 M; for a four-state contactor, the cost is \$4.4 M. The estimated capital cost of a four stage unit which could be used at high temperature and with tritium is estimated at \$10 M. The cost of each tritium recovery unit is thus assumed to be ~\$10 M. For processes like molten salt extraction where two different procesing units are needed, a centrifugal contactor and an electrolysis unit, the capital cost of the base system is assumed to be the cost of two base units, i.e. \$20 M. For tokamak Concept D, 1100 g/d is processed; therefore an incremental cost of \$11 M is incurred for the tritium recovery system.

The tritium purification system removes both gamma and other impurities from the recovered tritium. The estimated cost of this unit is \$10 M which includes not only the base units (~\$5 M) but also the support systems needed since these units will be located either in a hot cell area or in a heat exchanger building. The tritium purification system for 17Li-83Pb is expected to cost ~\$15 M because large amounts of gaseous activated impurity is expected.

The addition of protium (5 to 90 kg/d) to the heat exchanger system to reduce tritium permeation in the heat exchanger requires a processing unit. This unit removes tritium from the protium as well as other impurities. To provide perspective, the fuel processing system is expected to handle <10 kg/d of deuterium and tritium. The capital cost incurred for adding a protium unit is determined from the following equation:

$$C(\$M) = 3\times10^6 \; (\text{throughput (g)}/1160)^{0.4} \quad (2)$$

For 5, 50 and 90 kg/d, the capital cost is \$5 M, \$14 M and \$17 M, respectively.

The capital cost for the subsystem blanket-tritium recovery and purification for Tokamak Concept D is \$41 M, i.e. (\$10 + \$10 + \$11 + \$10). This includes 1) the cost of 2 base processing units plus the increment for the larger unit, and 2) the tritium purification system. For Mirror Concept C, the capital cost for the subsystem blanket tritium recovery and purification is \$51 M, i.e. ((10 + 13) + 15 + 13). This includes the cost of 1) a base processing unit plus the increment for a larger unit, 2) a tritium purification system, and 3) a protium cleanup system. See Table 4 for a summary of these costs for all blanket concepts.

D. Tritium Extraction

An extraction system will be needed to remove tritium from discarded solid breeder modules. The capital cost of a unit to be this, including all secondary systems, is estimated at ~\$10 M.

E. Coolant - Tritium Recovery and Purification

The tritium load into the blanket coolant is the sum of the contributions from the first wall ($\leq$0.2 g/d), from the limiter/divertor system ($\leq$1 g/d) from the helium purge ($\leq$0.01 to 113 g/d) and from the beryllium multiplier ($\leq$6 to 9 g/d). The beryllium multiplier contribution applies only to Concepts F and H. In Concept H, the Flibe design, the tritium permeating into the helium coolant from the helium purge stream is extremely large, 113 g/d and 83 g/d respectively for the tokamak and mirror designs.

The coolant tritium recovery system consists of a tritium removal system, a tritium purification system, special heat exchangers and piping, and a tritium removal system for the steam generator. The latter is required if the tritium input to the steam generator exceeds 20 Ci/d. Its cost is determined from Eq. (1).

The capital cost of the tritium recovery systems for coolants other than water is assumed to be \$10 M for tritium recovery and \$10 M for tritium purification. For pressurized helium, the base tritium recovery system is also assumed to cost \$10 M; this base system is assumed to remove 0.2 g/d. When large amounts of tritium are removed from helium, the capital cost increases. An estimate of these costs are \$4, \$14, \$40, and \$60 M as the processing rate increases from 1 to 10 to 80 to 120 g/d.

For Concept I, $LiAlO_2$/water coolant, the major capital cost associated with coolant tritium recovery is included in the cost for tritium control in the limiter/divertor. The incremental capital cost of a larger system with additional capacity is \$10 M. For Concept C, part of the capital cost of coolant-tritium removal is included in blanket-tritium recovery since this is a self-cooled system. The capital costs for special heat exchangers and piping and coolant tritium recovery needed because of the high tritium partial pressure

TABLE 4. COMPONENT AND TOTAL CAPITAL COST ($ M) OF THE BLANKET PROCESSING SYSTEMS FOR NINE DIFFERENT BLANKET CONCEPTS IN BOTH TOKAMAK AND MIRROR REACTORS

	Tokamak Concepts							Mirror Concepts								
Process	A	D	E	F	G	H	I	A	B	C	D	E	F	G	H	I
Blanket subsystems																
Limiter/Divertor (H_2O) or Halo Scraper (H_2O)	50	---	50	50	---	50	50	50	50	50	50	50	50	50	50	50
Blanket-Tritium Recovery and Purification	40	41	37	37	32	21	27	40	30	51	40	34	34	30	23	25
Coolant-Tritium Recovery and Purification	20	20	28	44	28	90	10	20	20	20	20	24	36	24	70	10
Coolant Purification																
NS, He, Na[a]	10	10	10	10	10	10	---	10	10	---	10	10	10	10	10	---
Li, LiPb, FLIBE	---	10	---	---	10	10	---	---	10	20	10	---	---	10	10	---
Coolant Dump Tanks																
NS, Na	5	5	---	---	---	---	---	5	5	---	5	---	---	---	---	---
Li, LiPb, FLIBE	---	5	---	---	5	5	---	---	5	5	5	---	---	5	5	---
Tritium Extraction (Discarded Modules)	10	---	10	10	---	---	10	10	---	---	---	10	10	---	---	10
Blanket Processing	135	91	135	151	85	186	97	135	130	146	140	128	140	129	168	95
Fuel Processing	40	40	40	40	40	40	40	31	31	31	31	31	31	31	31	31
ATR[a]	36	36	36	36	36	36	36	36	36	36	36	36	36	36	36	36
TOTAL (Base)	211	167	211	227	161	262	173	202	197	213	207	195	207	196	235	162

[a] Abbreviations are: NS - nitrate salt; He - helium; Na - sodium; Li - lithium; LiPb - 17Li-83Pb; FLIBE - a lithium beryllium fluoride salt; ATR - atmospheric tritium recovery.

(10^{-4} Pa) in this system is that shown in coolant-tritium recovery - $20 M.

F. Coolant Purification

For each of the coolants, an impurity removal system is necessary; the cost is estimated at $10 M. For concepts with two coolants (B, D, G, H), the cost doubles because there are two impurity removal systems needed. Because the 17Li-83Pb system is regarded as highly corrosive, its impurity removal system is estimated at $20 M.

G. Coolant Dump Tanks

Dump tanks are provided for all coolant and/or liquid metal systems. The estimated cost for the tank heating sytems and tritium control measures is ~$5 M for each dump tank required. Two concepts (B and D) require two dump tanks.

IV. DISCUSSION

The total capital cost of the blanket processing system for each of the nine concepts is shown in Table 4. For a tokamak reactor, the capital costs range from $85 M for Concept G to $186 M for Concept H, the Flibe design. The reason for Concept H's high cost is the high tritium partial pressure in the helium coolant for Flibe. The result is that additional tritium processing systems have to be added to control tritium migration throughout the fusion plant. The individual costs are: $21 M ((10 + 1) + 10) for blanket-tritium recovery and tritium purification; and $90 M ((10 + 60) + 10 + 10) for coolant-tritium recovery, tritium purification and tritium control in the steam generators. The latter is required because of the 260 Ci/d tritium influx (Table 1).

For a mirror reactor, the capital costs of the nine blanket processing systems range from $95 M for Concept I to $168 M for Concept H. Concept H's capital costs are the highest for the reasons cited above. In the mirror designs, the relative capital costs of the concepts would change if the tritium influx to the halo scraper was assumed to be 0.1 g/d rather than 1 g/d. The capital costs associated with implantation (halo scraper) would be reduced by $35 M for all concepts except I; for it, the reduction would only be $15 M since a larger capacity system is needed to handle the blanket-tritium recovery load. The ranking would be: A - $100 M; B - $95 M; C - $111 M; D - $105 M; E - $93 M; F - $105 M; G - $94 M; H - $133 M; I - $80 M. The range is now from $80 M for Concept I to $133 M for Concept H. As can be seen, a change in one of the input parameters can easily change a relative ranking of these concepts by blanket processing capital cost. With 0.1 g/d assumed as the tritium flux to the halo scraper, the capital cost of all Concepts except H fit in the range $95 ± 15 M. Concept H is above this range by ~$25 M.

The final row in Table 4 is a summary of all tritium processing capital costs for the reactors embodying the nine blanket concepts considered. These costs range from $161 M to $262 M if Concept H is included. If it is excluded the range is from \$161 M to \$227 M.

V. RESULTS AND CONCLUSIONS

1) A methodology for determining the capital cost of the blanket processing systems for a fusion reactor has been developed.

2) The methodology is based on the premise that the costs can be broken down into component parts and then summed if interfaces are taken into account.

3) The methodology can be used to estimate the capital costs of blanket concepts which are not highly detailed.

4) When several blanket processing systems are costed, the results indicate which systems have comparable costs and which systems have excessive capital costs.

5) The costs derived from this methodology are a good estimate of the order of magnitude of the capital costs.

6) A detailed study of the total processing system for one blanket recovery system is needed to confirm the usefulness of this model.

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