UCRL- 17083

Lawrence Livermore Laboratory

MIRROR REACTOR BLANKETS

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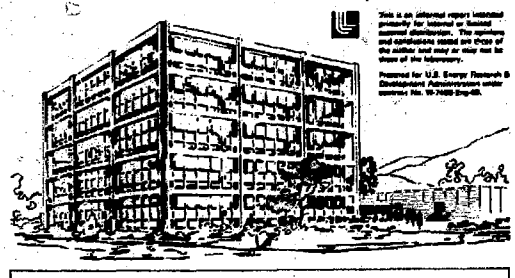
March 25, 1976



This presentation is being prepared for the "Blanket/Power Systems Fusion Reactors: Technology Workshop" to be held March 29 - April 2, 1976 at BNL

UCID- 17081

MIRROR REACTOR BLANKETS*

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*Work performed under the auspices of the United States Energy Research & Development Administration, Contract No. W-7405-ENG-48.

SECTION 1

INTRODUCTION AND GENERAL REQUIREMENTS, T. D. Lee

The process of developing conceptual blanket designs for mirror reactors as well as for reactors based on other methods of plasma containment requires satisfying in a consistent fashion a variety of requirements. Major requirements are:

- Performance requirements imposed by plasma characteristics
- Geometry
- Neutron performance
- Thermal conversion
- Materials (structural and chemical)
- Tritium handling
- Thermal hydraulics
- Maintenance
- Economics
- Safety and Environmental
- Material resources

We can only address some of these points at this time.

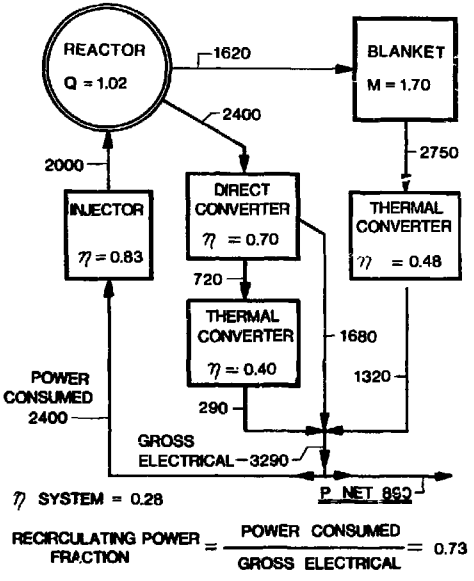
Conceptual design work is underway on two blankets. The first is for a commercial hybrid (fusion-fission) reactor whose principle product is ^{233}U. The second is for a commercial power reactor. The fundamental difference between the two blankets is the choice of their principle moderators, uranium for the first and Be for the second. We are on our second iteration of the hybrid blanket and our first for the power reactor blanket. While they share the same geometrical design, their structural materials, coolants, and thermal conversion systems are presently different. A summary table of parameters for the two blankets is appended. (Appendix I). These are example parameters, as they are

subject to change.

A dominate feature of presently conceived mirror fusion reactors is the fact that its plasma is sustained constant in time by the continuous injection of energetic fuel atoms. The ratio of fusion power to injection power is ∿ 1.0. We call this ratio "Q". To get an interesting power balance from a reactor system using a plasma with a Q-1 requires efficient circulation of large amounts of power and a blanket system that can multiply the kinetic energy of the fusion neutrons and efficiently convert blanket heat to electricity. These points are demonstrated by Figures 1-1 through 1-3. Figure 1-1 is the power flow diagram of a representative mirror reactor system. Figure 1-2 displays the sensitivity of this reactor system to changes in blanket energy multiplication (M). Figure 1-3 shows the effects on system economics at variations in blanket thermal conversion efficiency. The circled points in Figures 1-2 and 1-3 refer to the design points used in the power flow diagram (Figure 1-1).

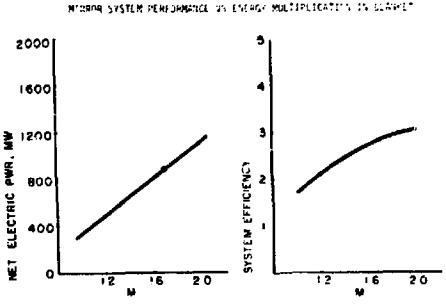
Another feature of the mirror reactor that has a dominate effect on the blanket is the geometry of its plasma and coils as shown in Figure 1-4. Implications of geometry will be discussed in detail in the following section.

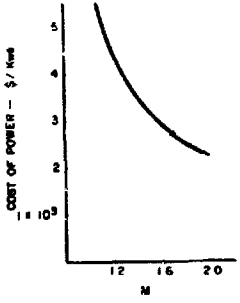
POWER FLOW DIAGRAM

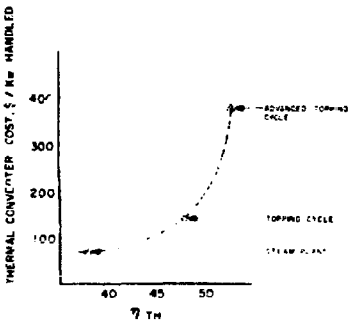


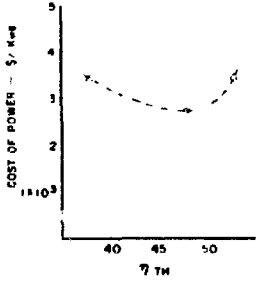
POWER IN MEGAWATTS

Fig. 1-1


NET ELECTRIC PWR, MW
2000
1600
1200
800
400
0
12
16
20
M
SYSTEM EFFICIENCY
5
4
3
2
1
12
16
20
M


COST OF POWER — $/ Kwe
5
4
3
2
1 x 10³
12
16
20
M


THERMAL CONVERTER COST, $ / Kw HANDLED
400
300
200
100
ADVANCED TOPPING CYCLE
TOPPING CYCLE
STEAM PLANT
40
45
50
η TH


COST OF POWER — $/ Kwe
5
4
3
2
1 x 10³
40
45
50
η TH

YIN YANG GEOMETRY

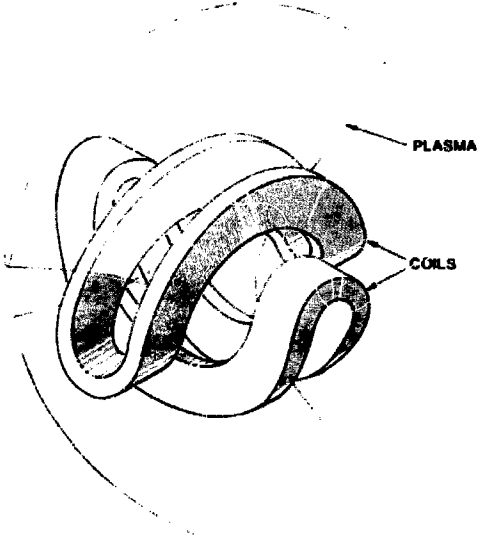


Fig 1-1

(CELLO)

FACILITY LAYOUT AND BLANKET MAINTENANCE J. H. Doggett

Facility

The blanket system in the Mirror Reference Reactor is designed to take advantage of the rough, spherical interior space and the natural parting line in the coil set which makes separation of the reactor halves practical. Fig 1-1

The facility layout is suited to accommodating the separation of the reactor in a convenient fashion. This scheme makes use of dry dock technology. One half of the reactor is floated on one or more pontoons. Changes in elevation are accomplished by raising or lowering the water level in the floatation space.

Two masses to be moved is about 50,000 tonne which is large but well within the scope of present technology.

There are several possible arrangements of components that will satisfactorily provide the separation movement required. At present we are proposing a "catamaran" system employing two large floats along side the reactor (Fig 1-2, 1-3) The upper half of the reactor is lifted and then moved aside to expose the blanket

Blanket and Blanket Replacement

The blanket and its coolant piping constitute a spherical shell inside the coil set with appendages near the midplane for coolant to enter and exit. The shell is divided into an arbitrary number of segments for convenience in handling with the blanket divided as one slices a melon, the segments may be removed by a simple movement parallel to the axis. Fig 1-4.

The neutron and gamma flux are attenuated by the blanket and its associated

shield to a level that eliminates radiation damage effects to materials behind the shield and allows contact maintenance of the outside of the reactor. The piping and wiring connections along with the main vacuum seal are outside the irradiated area.

During a blanket replacement or repair all joints can be broken and made in a contact mode minimizing the need for remote manipulators.

Tritium contamination of the reactor room is inevitable. Maintenance personnel will need to wear protective clothing suitable to the contamination level at the time they are in the reactor room and will carry emergency breathing apparatus at all times.

A blanket replacement would go as follows.

1. Shut down the reactor.
2. Allow cooldown time and prepare for separation.
3. Disconnect the coolant ducting (direct contact).
4. Break the vacuum joints (direct contact).
5. Evacuate the reactor room.
6. Separate the reactor halves (remote).
7. Remove and replace the blanket segments (remote).
8. Close the reactor (remote).
9. Remake the joints (direct contact).
10. Startup.

Our present assumption is that this duty cycle of the reactor will be 80% and that we will replace the entire blanket every year. This requires that a blanket set be built every year and from a practical standpoint 25% spares should be on hand. The total inventory of blanket materials is then 2.25 times the installed inventory. A multiplicity of identical [illegible]

[illegible]

Blanket Coverage

[illegible]

neutral particles and particle leakage from the magnet [illegible]

There are four [illegible] each [illegible] m. Their size [illegible] determined by [illegible] density limit of the [illegible] and the divergence of the beam. In this design the injectors are all in the upper hemisphere of the reactor.

The [illegible] leakage [illegible] the form of a fan at each end of the [illegible] [illegible]

The fractional blanket coverage has a strong effect on the power multiplication and tritium breeding ratio.

The blanket thickness is essentially constant independent of reactor size. This coupled with the shape of the plasma limiting field line tends to reduce fractional blanket coverage as the reactor gets smaller. (Fig. 2-6) The full thickness blanket coverage goes to [illegible] to a mirror to mirror length of about 7 m.

These considerations set the lower size limit for reactor design.

Blanket Safety

For reasons of economy the blanket segments will be fueled with reprocessed materials (Be and/or U). This implies an at least moderately radioactive unit delivered from the segment manufacturing plant for insertion in the reactor.

When the segment is removed from the reactor, it will be a strong gamma emitter, be contaminated with tritium and in the case of the hybrid, with fission products.

The tritium and fission products will be well held up in the cold blanket segment.

During the time the reactor is open there will be a large gamma flux in the containment making manned entry unacceptable (except perhaps in a shielded vehicle). During this time the tritium level will rise due to the outgassing of the large contaminated interior walls of the reactor and direct converter.

The used blanket segments will be moved to a hot shop for preparation for shipment to the reprocessing plant. The size and shape of the segments are such that they will require partial disassembly before shipment.

The new blankets will also require final assembly and inspection after delivery to the reactor site.

In summary, in the maintenance mode the reactor will pose the same type of safety problems that are found in a fission reactor (with the exception of criticallity) but on a much larger scale. If it is a pure fusion machine, the problems associated with heavy element fission products and after heat will be eliminated.

YIN YANG GEOMETRY

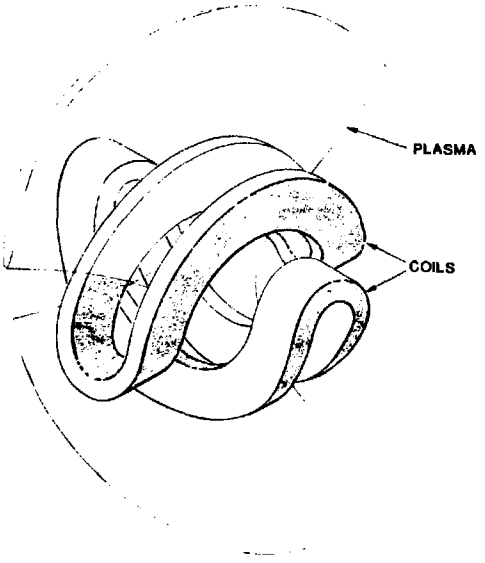


Fig. 1

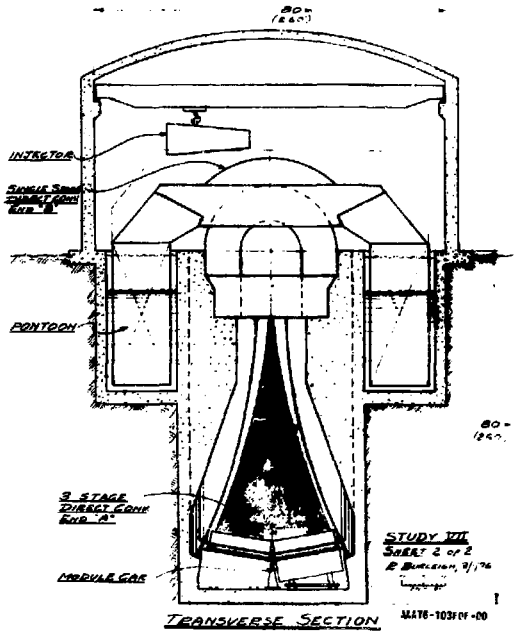

80'
(24.0)
INTAKE DUCT
PONTOON
80'
(24.0)
3 STAGE
DIRECT ZONE
END "A"
MODULE GAS
STUDY VII
SHEET 2 of 2
TRANSVERSE SECTION
AAAT6-103F01-00

Fig. 2

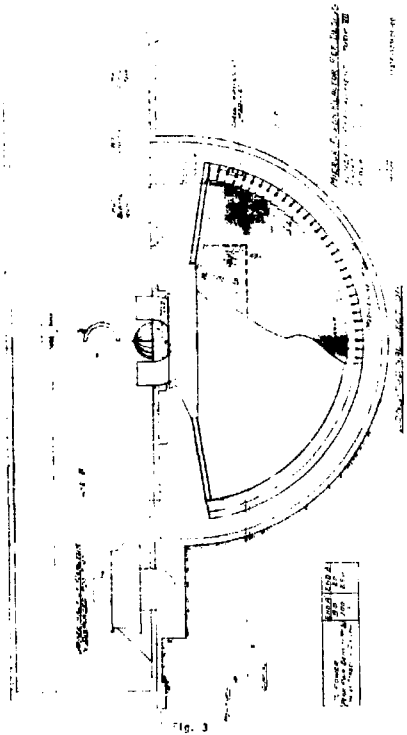
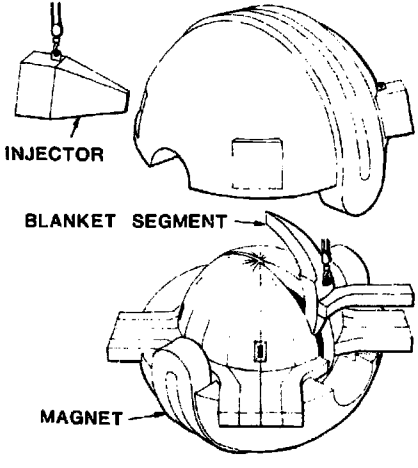


Fig. 3



MIRROR REACTOR

Fig. 4

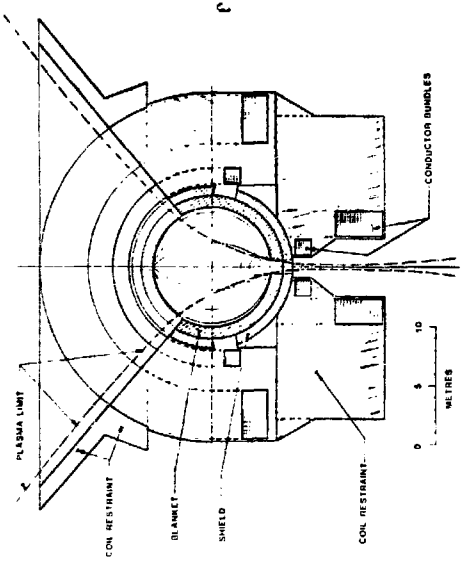


Fig. 5

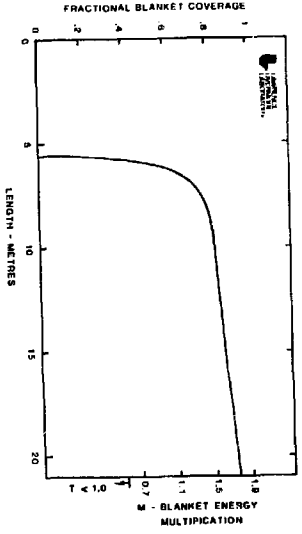


Fig. 6

TITLE: 3

HEAT TRANSFER AND THERMAL CONVERSION SYSTEM, D. [illegible]

The [illegible] of current mirror plasmas results in a power plant with a high recirculating power fraction. In a system with this type of power balance, substantial increases in net power output can be realized with modest increases in the efficiency of the thermal conversion system. Therefore, there is a strong incentive to examine the feasibility of coupling an advanced thermal converter to the nuclear island. Of the many candidate systems presently being examined by the energy R & D community, a concept that appears particularly well suited to accommodating the characteristics of a mirror power reactor is the potassium-topped steam cycle. Our design effort on the thermal conversion system is concentrated on defining the characteristics of the thermal cycle that will meet the constraints placed on it by the fusion power source.

The mirror power reactor has three major sources of thermal energy. They are

- waste heat in the injector system,
- the blanket thermal energy, and
- thermal energy in the direct converter that is a result of the inability to convert the ion kinetic energy completely to electrical energy.

A schematic of the thermal cycle is shown in Fig. 3-1. The basic concept is to boil potassium in the blanket. The potassium vapor is used to generate power in a wet potassium turbine, and the reject heat from the potassium is used to raise steam for the bottoming cycle. These features are common to previous conceptual potassium topping cycle designs.

However, we are examining some variations on the basic cycle which have the potential for meeting design constraints in all three reactor subsystems which develop significant thermal energy. It was realized at the outset of the thermal conversion system study that the choice of an advanced cycle, including some *unique modifications*, would result in a design in which several major engineering uncertainties would remain unresolved. However, it is hoped that the results of the study will be sufficiently encouraging to stimulate the investigation of these uncertainties.

Our blanket cooling concept takes advantage of the fact that at about 10% quality (≈ 85% void fraction), a potassium two-phase mixture has essentially infinite resistivity. Such a flow will not experience an MHD interaction with the magnetic field. Therefore, we use direct converter thermal energy to heat the potassium from the liquid state to a 10% quality two phase mixture, at which point it is passed into the blanket. In the blanket, the potassium experiences further boiling, where it is raised to a high-quality mixture, but below burnout. The flow, after exiting from the blanket, is separated into liquid and vapor components; the vapor is passed into the turbine and the liquid is recirculated back into the blanket flow. At the turbine exit, the potassium is condensed, rejecting its latent heat to the steam cycle. It is then preheated to the saturated liquid state with turbine bleed before entering the direct converter heat exchanger.

Heat removal from the direct converter is accomplished with helium, as the converter operates at high voltage and therefore an electrically nonconducting coolant must be employed. The balance of the direct converter heat that is not used in the potassium loop is used in the steam cycle, for high temperature feed-heating and possibly reheat. Also, the rejection (which require relatively low temperature coolant because of the proximity of

cryogenic pumping surfaces) are cooled with a water loop, this heat being used to do low-temperature feedwater heating.

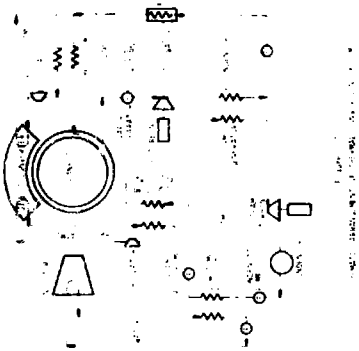
A set of preliminary thermodynamic conditions and power flows around the cycle are shown in Fig. 3-2. The potassium temperature in the blanket has been set at 850°F. Strength and chemical compatibility properties of the blanket structure would permit a higher operating temperature and therefore higher efficiency. However, the design requirements on T_2 leakage from the tungsten cans in the blanket into the potassium coolant dictated the upper limit on the temperature of 850°C. The power flow in Fig. 3-2 results in a thermal efficiency of 51%.

The basic technology of the potassium Rankine thermodynamic cycle has been established as part of NASA's space power program. However, several problems associated with our adaptation of the cycle to the mirror reactor require further investigation, and probably represent substantial R & D efforts.

- We are proposing to transport a low quality, two-phase flow over considerable distance and through complex ducting. It is not certain, on the basis of existing data, that the flow can be maintained in a sufficiently homogeneous state to assure a high electrical resistivity and stable flow regime.
- The boiler concept in the blanket employs an inlet quality of 10% and exit quality of 80%. Burnout at this high exit quality is avoided by the use of helical inserts in the boiler tubes, a technique established in the space power program. However, it is possible that the magnetic field may significantly alter the hydrodynamics of the flow. The

invalidating our present analysis which is based on data obtained in the absence of a magnetic field.

- Our present design is based on rejecting heat from the blanket heat transfer loop directly to the steam loop. In this configuration, the possibility exists for the release of activated potassium in the event of a tube failure in the heat exchanger and subsequent potassium-water reaction. Therefore, it may be necessary to include an intermediate heat transfer system between the blanket and steam loops. The incorporation of this additional heat transfer loop would lower the efficiency of the thermal conversion system. Alternately, to maintain the present high efficiency, the potassium boiling temperature could be raised, but this variation in the design would severely aggravate the problem of T_2 containment in the blanket.



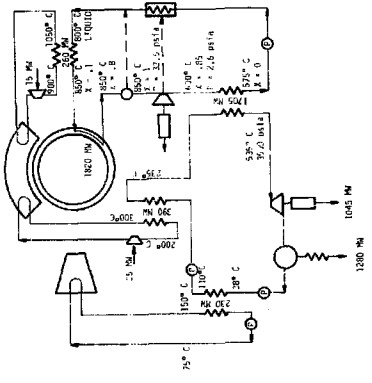


Figure 3-2. THERMODYNAMIC CONDITIONS FOR THERMAL CONVERSION SYSTEM

SECTION 4

MATERIALS, W. L. Barmore

First Wall:

The refractory alloy Nb-1Zr was selected for the first wall because of its predicted high-strength, ductility, and low swelling properties under fusion reactor conditions and because it is a commercial alloy which is fabricable and weldable. It has good corrosion resistance to flowing two-phase potassium at temperatures to approximately 1100°C under nonirradiation conditions. But the oxygen content of the alloy should be less than 500 ppm and of the potassium less than 20 ppm to insure low corrosion rates, since high oxygen content can lead to enhanced corrosion. Research should be done on the dynamic corrosion of niobium-base alloys by flowing two-phase K containing various concentrations of oxygen under irradiation at temperatures from 700°C to 1000°C.

Based on the predicted swelling of the Nb-1Zr first wall under mirror reactor conditions of approximately 3.0 MW/m^2, the first wall should have a lifetime of approximately 2 years before accumulating 10% swelling (Fig.).

For the Nb-1Zr alloy at 850°C, preliminary calculations show that creep rate of the highly stressed first wall could be enhanced by irradiation as much as 100%.

Improvements in swelling characteristics and creep resistance of the first wall might be secured by using a ternary niobium-base alloy having a solid solution strengthener like tungsten; e.g., Nb-9W-1Zr. An alloy of this type would still have good welding and fabricating properties. Future research should include development of niobium-base alloys with minor alloy additions which will maximize both swelling resistance and creep strength while retaining corrosion resistance to K under irradiation in the temperature region 700°C to 1000°C. The enhancement of creep, and the effects of helium and hydrogen on the ductility of these alloys, should be studied under irradiation conditions at temperature near 850°C.

Blanket Cooling Tubes and Tritium Permeation Barrier:

Pure tungsten was selected for this use because of its very low permeation rate to tritium. Other low-permeation materials considered --

e.g. Be and BeO -- were discarded because of high swelling rates under neutron irradiation. In spite of its poor fabricability and weldability, tungsten (or a dilute alloy) was the only reasonable choice that would limit the tritium leakage to the potassium coolant to acceptable levels at the blanket cooling temperature of 850°C (this is covered in detail under tritium containment). The predicted swelling life under mirror reactor irradiation conditions is approximately 1.0 year before reaching 10% swelling. By using the W-5 Re alloy in this application, fabricability can be somewhat improved, and this alloy should have a lower rate of swelling. From high temperature data, on corrosion of W by other alkali metals, it is predicted that tungsten and W-5 Re would have good corrosion resistance to K coolant at 850°C provided that the oxygen content of the metal and the K coolant are kept to low levels. Creep of the coolant tubes is very low because of the high creep resistance of W and low stresses in this application. Research should be directed to study the dynamic corrosion of tungsten-base alloys by flowing two-phase potassium having various concentrations of oxygen under irradiation from 700°C to 1000°C. Alloys should be developed that would minimize swelling and minimize tritium permeation under irradiation at temperatures near 850°C. Methods for fabricating and joining these alloys should be developed concurrently with the above studies.

Blanket Tritium Breeding Material and Moderator:

The tritium breeding material is a porous, but intimate, mixture of $Li_2Be_2O_3$ dispersed as micron-size particles in a Be matrix. There is small free-energy change favoring the reduction of $Li_2Be_2O_3$ by Be forming liquid Li and BeO. This tendency increases gradually as temperature is increased. By superimposing a lithium vapor on the He used as a tritium scavenging gas, the reaction tendency should be reduced by lowering the oxygen activity in the system. Tritium oxide released from the $Li_2Be_2O_3$ should be reduced by the Be metal, giving T_2 gas to be removed from the blanket.

The beryllium metal and tungsten coolant tubes should be compatible to approximately 1000°C before reaction occurs.

Compatibility of Be and $Li_2Be_2O_3$ should be investigated at temperatures in the vicinity of 850°C; if reactions are severe, then H and D should be done on coating the particles with thin layers of materials

which prevent, or greatly reduce, the reaction. Compatibility of Be with W at temperatures from 900°C to 1000°C should also be studied, and if necessary, thin unreactive cladding layers should be developed.

Fabrication methods should be developed for making Be-$Li_2Be_2O_3$ porous composites with intricate flow-through passages for the helium scavenging gas.

MATERIALS FOR CONSIDERATION IN LLL MIRROR FUSION REACTOR ON BASIS OF 1 MW/M²/YR

MATERIAL	Nb-1Zr	Nb-5W-1Zr	Pure W	W-5 Re
USE:	FIRST WALL STRUCTURE	FIRST WALL STRUCTURE	TRITIUM PERMEATION BARRIER	TRITIUM PERMEATION BARRIER
MAXIMUM FLUENCE N/CM²/YR	1.5×10^{22}	1.5×10^{22}	1.5×10^{22}	1.5×10^{22}
OPERATING TEMPERATURE °C	850	850	850	850
PEAK SWELLING TEMPERATURE °C	~800	--	~770	--
HELIUM APPM	25	~30	70	~70
ESTIMATED PERCENT SWELLING	1.5	*	4	**
ESTIMATED LIFE YEARS (10% SWELLING)	6.7	--	2.5	--
†ESTIMATED LIFE YEARS (1% CREEP AT 6 KSI)	1	~2	>5	>5

*PROBABLY LESS THAN Nb-1Zr.
**PROBABLY LESS THAN PURE TUNGSTEN.
†DOES NOT INCLUDE ENHANCED CREEP FROM RADIATION.

Fig. 1

SECTION 5

TRITIUM CONTAINMENT AND REMOVAL

T. R. Galloway

SCOPE:

Tritium containment and removal problems associated with the blanket and power-systems for a mirror fusion reactor are identified and conceptual process designs are devised to reduce emissions to the environment below 1 Ci/day. The blanket concept development proceeds by starting with this emission goal of 1 Ci/day and working inward to the blanket. At each decision point, worker safety, operational labor costs, and capital cost tradeoffs are contrasted.

The conceptual design uses air for the reactor hall with a continuous catalytic oxidizer-molecular sieve adsorber cleanup system to maintain a 40 $\mu Ci/m^3$ tritium level (5 $\mu Ci/m^3$ HTO) against 160 Ci/day leakage from reactor components, energy recovery systems, and process piping.

This blanket contains submodules with $Li_6Eu_2O_3$-Be for tritium breeding and submodules with Be for mostly energy production. Tritium production in both are handled by separately containing this breeding material and scavenging this container with lithium vapor-doped helium gas stream. The container consists of tungsten-5% rhenium tubes and tube sheets with the breeding material packed and sintered in the shell surrounding the tubes. Potassium vapor (also lithium doped) coolant passes through these tubes to recover the heat at 850°C. Tritium leakage from the scavenging helium into the potassium vapor within the blanket submodule can be reduced to 160 Ci/day for the tungsten alloy case. This small leakage would result in loss into the steam via a 100 ft tungsten clad boiler (potassium boiling) of 0.4 Ci/day. A moving getter bed is used to recover the tritium from the LiT and Li_2T scavengers in both the helium blanket scavenging flow and the potassium vapor coolant. The extent of the increased leakage resulting from replacement of tungsten alloy with TZM or niobium is examined.

A. Impact of Reactor Hall Containment on Blanket Design

The long-term goal among the fusion reactor engineering community[1] is to reduce the total tritium loss to the environment to below 1 Ci/day, the accumulated

tritium emissions from all such fusion reactor power plants to be built in the U.S. by the year 2000 would not significantly add to the present tritium background (or a reduced background from a total nuclear device test ban). And by the year 2021, the emission must be below 0.1 Ci/day to stay safely below the natural tritium production of 10,000 to 20,000 Ci/day.[2]

For a reactor hall of the approximate volume of 350,000 m^3, losses could be kept to such a low level only if the tritium concentration within the gas cover (inside the reactor hall) were kept low (i.e., 40 $\mu Ci/m^3$) and the entire wall surface be hermetically sealed and could contain a rare metal getter (scavenger). The getter approach was rejected when there was examined the cost of fabrication and resource depletion that would result from a 1 mil thick foil of cerium, titanium, or other rare metal placed on the hall (i.e., 10 tons). Also, the operating utility is faced with the problem of disposing of or regenerating this getter material and the problem of excluding air.

The other alternative involves providing a permeation barrier as the hermetic seal. A stainless steel shell (alloys 304, 316, or 321) and 2 mm (1/16") thick at ambient temperatures will provide such protection.[3] Today the technology exists to weld metal plates in place within a large concrete building structure. This shell will meet a criteria for low permeation (below 1 Ci/day) even for the case of the most incredible accident where 2.3 kg of T_2 is ignited and is dumped, at once releasing HTO and/or T_2O into the reactor hall, and quickly increasing the concentration up to 24.5 ppm (65 Ci/m^3). Release of T_2 gas inventory into the reactor hall is also serious because of high permeation losses to the environment of T_2 or HT.

In addition to maintaining a low permeation loss, the hermetic reactor hall interior design must also protect personnel just outside the building from experiencing HTO or T_2O concentration in excess of the maximum permissible concentration (MPC) during both routine operations and extreme conditions. The most conservative approach is to assume that the non-occupational, general public could be present (for four hours per week) outside the concrete reactor hall following an ignition and explosion of 2.3 kg T_2; thus, T_2O concentration must remain below 2 $\mu Ci/m^3$ (~ 0.001 ppb) for a controlled access area.[4]

The design to provide the required protection could consist of a space between the concrete building shell and the hermetic seal liner. In other words, the hermetic liner is welded together on standoffs or studs cast into the concrete shell. If we state that the facility will have the T_2, HTO, or T_2O containment and removal capability to reduce this high accident level down to the nominal value of 40 $\mu Ci/m^3$ within the building after three days of reactor hall air processing, then we can compute the length of the standoff. If the 24.5 ppm spill leaks for three days into this space at 0.03 Ci/day and the level within this space must remain below 2 $\mu Ci/m^3$, the standoff distance must be 79 cm. This concept therefore would be economic, very safe, and would qualify as secondary containment of the reactor hall.

The safety of the worker must be the next topic in the reactor hall containment, involving normal ambient operating levels. We selected the nominal level of 40 $\mu Ci/m^3$, as in the previous discussion, for very particular health and safety as well as state-of-the-art hardware reasons. An occupational worker can be exposed at 100% MPC for 40 hours per week without any clothing protection at these contamination levels of 40 $\mu Ci/m^3$ if the HTO fraction remains below 12% by volume. Such environments (with once-through air) are not atypical of production facility work environments in the U.S. Mound[5] is demonstrating 40 $\mu Ci/m^3$ and Los Alamos[6] on a smaller scale demonstrating 0.5 $\mu Ci/m^3$ for 10^6 isolation between processed glove box and room, both in recirculating systems. However, for routine work, the design basis for exposure should be well below the 100% MPC. ERDA recommends[4] that for design purposes, levels at 20% MPC should be . This would be quite feasible at work assignments of eight hours per , or one could provide analytical evidence for HTO fractions belo , the occupational worker can be protected by light-weight suits an masks that would not interfere with his needs for agility and dexterity. Such light-weight suits (two piece with overalls) with air supplied masks typically[7] can provide protection factors of 100 or more if changed hourly or factors of 10 or more if not changed for eight hours. Thus, it can be seen that this nominal level of 40 $\mu Ci/m^3$ offers the largest number of operational options while still remaining close to the economic and safety optimum.

Now there must be some assurance that in this operational design, the level of 2% or less HTO can be feasibly maintained. It is this question that is critical to the practicability of our proposed design. The HTO can be produced from pure T_2 or HT by several chemical routes: isotopic exchange with H_2O in the water-laden air within the reactor hall, autoradiolysis of T_2 in air (oxygen), catalytic conversion of T_2 or HT via active metal surfaces exposed, and radiolysis of T_2 or HT in air via radiation field escaping the mirror machine. It is for these reactions that, in the past, the use of air has been rejected for fusion reactor halls. However, careful review of contemporary research and demonstrated technology shows that the use of air must be reconsidered. If care is taken to avoid hot, precious metal (catalytic) surfaces, excess water vapor, and a buildup of T_2, then the formation of HTO can be kept at manageable levels. Estimates of the conversion rates have been made by fitting available data [8,9] with a rate equation. The results are as follows:

$$HTO = T_2(0)[1 - \exp(-kT_2(0)\theta)]$$

where θ is time for conversion in hours, $T_2(0)$ is the initial T_2 concentration in $\mu Ci/m^3$, k is the rate constant (i.e., $k = 29 \times 10^{-10}$ $m^3/\mu Ci$-HR at 100% humidity or $k = 6.24 \times 10^{-10}$ $m^3/\mu Ci$-HR for dry air) and HTO is the concentration of HTO found in $\mu Ci/m^3$. At levels of $T_2(0)$ of 40 $\mu Ci/m^3$, the rate of formation of HTO would always be less than 1.0×10^{-6} $\mu Ci/m^3$ per hour, small indeed. The conversion rate does not get significant until levels of 65 Ci/m^3 are approached, over 10^6 larger than planned for in design. For this incredible accident case, 11 Ci/m^3 of HTO would be found in the first hour. The important point from this analysis is to keep T_2 at low concentrations with maximum dedication.

The above model predictions have been made for T_2 conversion by isotopic exchange and autoradiolysis to HTO in humid air. The effects of the availability of substantial areas of active metals surfaces on this conversion rate have been examined.[9] Using this accelerated rate constant for steel surfaces, a comparison of oxidation rates is made in Table I.

Table I.

Oxidation of T_2 to HTO in Reactor Hall

(Initial Concentration = 40 $\mu Ci/m^3$)

(HTO Conc. in $\mu Ci/m^3$)

Time	Dry Air	Humid Air	Steel Catalyzed*	Rad. Field on Steel‡
1 hr	9.96×10^{-7}	4.01×10^{-6}	9.75×10^{-5}	0.6×10^{-3}
2.5	2.49×10^{-6}	1.00×10^{-5}	2.44×10^{-4}	1.58×10^{-3}
50	4.98×10^{-5}	2.01×10^{-4}	4.88×10^{-3}	3.1×10^{-2}
2,000	1.99×10^{-4}	8.00×10^{-3}	1.92×10^{-1}	~5.0
10,000	9.98×10^{-3}	4.00×10^{-2}	0.96	~25.0

* With humid air

‡ Radiation Field taken to be 4,000 R/hr. on steel catalyzed surfaces in humid air.

Also shown are the enhancement effects of a superimposed radiation field[10] on top of the catalytic surface and the humidity effects. It can be seen that overall the enhancement effect is some 500 times over that formed in dry air. However, it is clear that when the concentrations are kept low (i.e., 40 $\mu Ci/m^3$) and the air within the reactor hall is continuously processed and not allowed to sit stagnant for one year (10,000 hours), the percentage of HTO present will remain well below the 2% figure desired for the design case (2% MPC HTO) unless T_2 is left isolated within the reactor hall for three months (2,000 hours).

B. Blanket and Process Piping Leakage

Now that it has been established that a tritium level of 40 Ci/m^3 is a safe working environment within the reactor hall, let's examine the constraints

that this level places on the other components in the nuclear island. Experience in Building 331 at LLL and at other tritium handling facilities in the U.S. suggests that either in a continuous air recirculation or a once-through air exhaust system, the ambient room levels are dependent on the cleanliness of the worker. That is, how much outgassing of tritium occurs from leaking equipment and de-adsorbing from room and hood surfaces. The ambient level for a given room volume depends on the replacement rate of the air within this volume and the tritium outgassing rate. Operating practice in once-through air exhaust systems suggests twenty changes of air per hour are needed to keep room levels about 15 $\mu Ci/m^3$. When laboratory operations are not as careful, the level increases.

The relationships are fundamental with the mass balance. In Table II are indicated results of the mass balance analysis relating the maximum outgassing rate of tritium that can be tolerated to meet the 40 $\mu Ci/m^3$ level for a given air flow rate in the reactor hall (350,000 m^3).

TABLE II.

Maximum Tolerable Outgassing Within Reactor Hall to Maintain 40 $\mu Ci/m^3$

Air Flow m^3/sec	Air Flow (CFM)	Volume Changes Per Hour	Outgassing Tolerated (Ci/day)
0.16	607	1/24	1.0
5.82	10,000	0.05	18.0
58.8	100,000	0.5	180.0
2352	4,000,000	20.5	7200.0

These rates can be compared favorably with available estimates[11] for tritium leakage loss (including permeation) from state-of-the-art energy recovery system (process piping, pumps, heat exchangers, turbines, etc.) For example, a conventional steel energy recovery system for a 470 MWe plant would release about 70 Ci/day of tritium gas into the reactor hall.[11] Thus, typical for a 1 GWe

reference reactor design, air flows of around 60 m^3/sec (100,000 Cfm) are required. This rate is typical of moderate size office buildings with conventional off-the-shelf air conditioning hardware. Approximate cost would be 30M$ for complete process from Englehard[12] including 3M$ for precious metal catalyst.

It is clear that there will be an optimum trade-off between the size of the reactor hall waste gas handling system and the design of the thermal energy recovery system relative to tritium leakage and permeation. Doubly or triply contained process piping and heat exchangers represent one extreme and thin steel at high temperature, the other. A compromise may well be the use of a single tube unit with triple layers: stainless steel, low permeability layer, and stainless steel. A triple sandwich using copper as the middle layer substantially reduces the high temperature tritium permeation. The feasibility of such a copper containing design has been explored[13] and indeed looks favorable. Further reductions may be possible with a ceramic inner layer. These questions will be explored later in this report.

Also indicated in Table II is the air flow of 607 cfm which would release 1.0 Ci/day at 40 $\mu Ci/m^3$. This fact tells us that the operation of conventional "air flush" air locks through the reactor hall wall to the outside would be a problem. There are two air locks: one large lock for moving blanket "orange peel" sections and other large equipment out of the reactor hall, and several smaller ones for personnel. For special conservatism, one could operate both upon entering and exiting the reactor hall. If one expects to change 10 blanket sections per year, this would represent a flow of only 2 m^3/hr (1 cfm) if the lock were the pump-down variety and for 100 personnel accesses per day. This release would amount to 8 m^3/hr (4 cfm). This is excellent. But if a conventional lock is used where air velocities past a moving person are required to be 0.6 m/sec (120 ft/min), it rapidly becomes excessive. Thus a double lock variety where one isolated chamber is pumped down must be used to avoid excessive tritium release to the environment. A pumped double air lock design could also dump its exhaust air to the reactor hall waste gas handling system and no substantial increase in system load would result. This operation would also allow

for fresh air makeup. The total amount of air makeup, however, will still be limited to under 0.36 m^3/sec (607 cfm) depending on the actual tritium leakage out of the reactor hall. The low level achieved by the molecular sieve bed just before the waste air is stacked. More makeup air can be admitted if molecular sieve bed performance exceeds the 40 $\mu Ci/m^3$ limit. With an "air curtain," it might be possible to reduce back-mixing to such a low point that the pump down feature could be deleted.

Such a reactor hall containment processing system has been depicted in Figure 1 where one of the concepts for the mirror reactor reference design is shown with such a processing concept added. Fresh air is contained within both personnel locks Nos. 1 and 2. In operation, both doors to No. 1 are closed and No. 2 is opened. Personnel enter lock No. 2 and its reactor hall door is closed. The center door is opened and personnel move from No. 2 to 1. Lock No. 2 is pumped down with its waste gas discharged in the reactor hall. Fresh air is exchanged through No. 1 into No. 2 until No. 2 air pressure is matched to atmospheric. Personnel move from No. 1 to outside. The doors on lock No. 1 are closed and its contents are pumped down and exhausted into the reactor hall and lock No. 1 is filled with fresh air. Personnel ingress could be done without any pump-down or it could be analogous in a reverse fashion.

The hermetic seal secondary containment space inside the reactor hall structure is flushed with fresh air (about 0.36 m^3/sec or 600 cfm) when the air locks are not in operation, maintaining levels in this space of HTO below 2 $\mu Ci/m^3$, into the waste gas processing system.

This processing system as shown in Figure 1A would catalytically oxidize the T_2 or HT to HTO. Operation at high temperature (350°C) would *permit oxidation of tritiated hydrocarbons. The higher boiling refractory* organics would be trapped by the activated carbon columns. Helium impurities will also be removed by a bleed air purge system, but these problems of non-tritium impurities will be discussed in Section II. Waste air would be recycled for the most part into the reactor hall itself, with a small bleed stream exhausted to the stack, hopefully below 40 $\mu Ci/m^3$. The processed air would enter the reactor hall near floor level where occupational workers would be concentrated. The bulk flow of this reprocessed air would be expected to be 58.8 m^3/sec (100,000 cfm) or better, depending on how leaky or permeable

the blanket tritium processing system turns out to be. The air velocity near the worker might be in excess of 3.3 m/min (10 fpm) however, the net velocity *upward through the interior of the reactor hall would be only 0.33 m/min (1 fpm)* or so. Thus, no credit of the velocity could be attributed to safety (~120 fpm).

The occupational worker safety with respect to tritium within the reactor hall should be excellent, since he could work at the ambient levels of 40 $\mu Ci/m^3$ (less than 2% HTO) for a 40 hours work week without additional suits or breathing masks. We have also provided in the design a full capability for workers survival from the "incredible accident," where 2.3 Kg of T_2 explodes to form 25 ppm (65 Ci/m^3) of T_2O within the reactor hall. Even though calculations shows that such an explosion would not stet the integrity of the machine containment, the "incredible failure" postulates 100% escape of T_2O into the reactor hall.

We propose that the workers normally wear coveralls with a plastic suit underneath and outside as well, and carry a breathing mask with three minutes of air in a small belt tank. In audition, we propose to supply a large number of fresh air "breathing booths," where fresh air is flushed and exhausted through a large duct and manifold. This design would permit the worker to enter the "breathing booth" where not only protection from the high tritium levels is provided, but a safe egress port is available to him for escape to the outside world. These booths with escape ducts would be located conveniently at all building elevations to permit worker entry from all locations in less than three minutes. The suit and breathing mask would permit the worker complete safety and egress during the "incredible failure." Normally upon hearing the alarm, within seconds the worker would install his breathing mask, run *to a booth, avoid the elevated concentrations which would take many minutes* to accumulate within the reactor hall. At 65 Ci/m^3, a 12 sec. exposure would result in 25 rem, a dose equal to that allowed to the general public under accident conditions[16] assuming a quality factor of unity for the radiation.[14] There is, of course, the unlikely possibility, that the worker might be sprayed directly with 100% T_2O before he could mount his face mask. But we feel this design concept and operational procedure would provide excellent worker safety, well within even the most restrictive ERDA guidelines that could

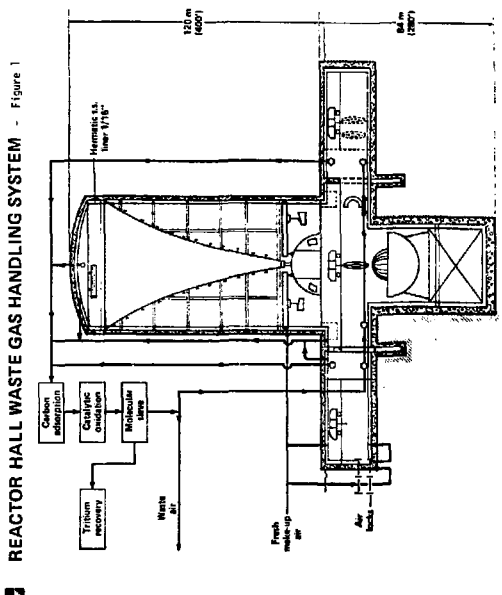
be anticipated far into the future.

Now that we have provided worker safety for the "incredible failure," the consequences for reactor hall and the waste gas processing system must be discussed. Worker re-entry times must be estimated as well. This problem is a difficult one because the transient behavior is difficult to predict. The T_2O concentration spike will drive T_2 and HT into many materials and create adsorbed layers of T_2O on all exposed surfaces.

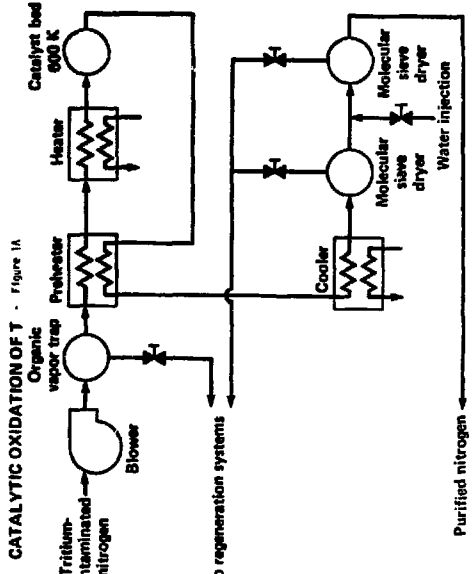
First, the complexing of the gas (65 Ci/m^3 tritium) from the "incredible accident" with the metal parts will be estimated. If the metal available is 300,000 tons of stainless steel and at room temperature, the tritium dissolved will reach about 100 Ci and the amount adsorbed onto the surface in the form of a monolayer will be around 5,000 Ci. Thus, some 5,100 Ci will become tied up with the metal parts as a result of this 65 Ci/m^3 spike. Little is known about the deadsorption of tritium from such surfaces, but the knowledge derived from practical experience tells us that the outgassing rate is very slow. It is very difficult to decontaminate tritium-fouled equipment.

The consequences of this lack of knowledge about tritium outgassing from equipment are substantial. The waste-gas processing time needed to clean up this spike and restore the normal 40 $\mu Ci/m^3$ level is difficult to predict. We can predict the processing time to reduce the 65 Ci/m^3 to 40 $\mu Ci/m^3$ in the absence of outgassing, but cannot estimate the additional time needed when outgassing occurs. There are procedures for surface cleaning designed to enhance tritium outgassing, such as steam cleaning and smooth polished surface, electrocleaning, etc. This is the one area that must be studied more carefully and on an urgent basis.

In the above discussion, the problem of outgassing into the reactor hall air will be considered, but there is also the problem of outgassing into any free water volumes (such as the water maintained to float the two hemispherical reactor halves apart). If the blanket portion of the reactor (1278 tons) were allowed to outgas into the water float volume (100,000 M^3), the concentrations would reach 0.1 $\mu Ci/m\ell$. This is 1 MPC for controlled area and many times MPC for an uncontrolled area (3×10^{-3} $\mu Ci/m\ell$).[4] The tritium contamination would remain a safety hazard to the worker and would have to be corrected. The water volume must be hermetically contained.



REACTOR HALL WASTE GAS HANDLING SYSTEM - Figure 1



CATALYTIC OXIDATION OF T - Figure 1A

Discharge into a deep well or other environmental sink would be unacceptable at such concentrations.[14] Processing by cryogenic distillation of such a huge volume would be incredibly expensive. New innovations are needed (i.e., new selective catalysts, laser isotope separation in water, selective molecular excitation, etc.) in order to carry out this decontamination.

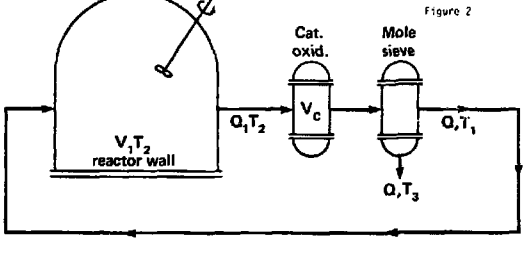
Neglecting these adsorption controlled outgassing problems discussed above for metal surfaces in reactor hall air and for the water volumes present, we can make an estimate of the processing time required to reduce the 65 Ci/m^3 from the "incredible failure," to the 40 μCi/m^3 operating level. Using the study at Mound Laboratory[16] for the determination of the catalytic oxidation kinetic rate data, a plug flow reactor can be designed.

A schematic for such a recirculating tritium removal system has been shown in Figure 1. The tritium concentration (gms T_2/m^3 of total gas) is given as T_i subscripted according to location. It is assumed that the reactor hall is well mixed and that the volumetric removal of gas (mostly T_2), q_1, is negligible compared to the recirculating flow, Q.

A mass balance on the reactor hall can be made by expressing the difference between inlet and outlet concentration as the decrease in its prevailing level:

$$T_2(\text{in}) - T_1(\text{out}) = -\frac{V}{Q}\frac{dT_2(\theta)}{d\theta} \qquad (2)$$

Figure 2



where θ is the process time V the reactor h ll volume. A similar balance can be made around the removal process unit:

$$Q[T_2 - T_1] = qT_3 \tag{3}$$

or

$$\frac{T_1}{T_2} = 1 - \frac{qT_3}{QT_2} \tag{4}$$

The plug flow reactor design equation can be written[17] as:

$$\frac{T_1(\theta)}{T_2(\theta)} = \exp\left[- k \frac{Vc}{Q} \right] \tag{5}$$

where Vc is the catalyst volume and k is the first order kinetic rate constant:

$$k = 2.27 \times 10^5 \exp[- 7100/RT] \tag{6}$$

as determined at Mound.[16]

Eliminating $T_1(\theta)$ between equations 2 and 5 results in:

$$T_2(\theta)\ [1 - \exp(- k \tfrac{Vc}{Q})] = - \frac{V}{Q} \frac{dT_2(\theta)}{d\theta} \tag{7}$$

Integrating:

$$\int_{T_2,C}^{T_2,\theta} \frac{dT_2}{T_2} = - [1 - \exp(- k \tfrac{Vc}{Q})] \frac{Q}{V} \int_0^{\theta} d\theta$$

results in:

$$\frac{T_{2,\theta}}{T_{2,0}} = \exp\left\{ -[1 - \exp(- k \tfrac{Vc}{Q})] \frac{Q}{V} \theta \right\} \tag{8}$$

In Table III are shown the results of this model above relating safe "re-entry times" as a function of required catalyst volumes. Clearly, catalysts beds larger than 200 m^3 are not inexpensive or practical; thus, re-entry times range from around one day to six months for reasonable bed designs. An interesting note can be made here regarding the 30MS quotation from Engelhard. Their bed volume was over 100 m^3 since their conservative approach (as catalyst supplier) is to meet the efficiency requirements in a single pass.

Re-entry times of the order of days to a week or so would be acceptable. Catalyst volumes from 2 to around 30 m^3 are called for with complete plant cost from 10 to 200MS. It is clear that there are many areas where costs can be cut if an optimization of the design is made to suit the particular specialized requirements for the fusion reactor hall processing. Costs for such a recirculating system might be forced to a low of 3MS, a tenfold reduction in cost over the previously quoted system.

Before we leave this topic, a comment about the ability of the above design configured for the "incredible accident" to handle the routine processing task. As indicated in Table II, a processing system scaled at around 60 m^3/sec (100,000 cfm) could handle tritium outgassing from the blanket, process piping, contaminated parts, etc., at levels of 180 Ci/day. As indicated earlier in this report, this processing capability is safely above the estimated[11] leakages around 70 Ci/day. Thus, the system presented is nearly optimally configured for both the routine leakage task and the "incredible failure." Now we have to insure that our blanket containment scheme can meet this goal of below 180 Ci/day.

Table III.

Incredible Accident: Tritium Reduction

$Q = 58.8\ m^3/sec$ (100,000 cfm)

$T_2O = 65\ Ci/m^3$

$V = 350{,}000\ m^3$

Temp = 300^0K

Reactor Hall Tritium T_2O ($\mu Ci/m^3$)

Re-Entry Times (Hours)	Vc, Catalyst Volume (m^3)			
	0.17	1.7	16.7	167
30 (1 day)	6.05×10^7	3.21×10^7	1.71×10^5	0.0
600 (1 month)	1.54×10^7	48.03	0.0	0.0
3,000 (4 months)	6.56×10^6	0.0	0.0	0.0
10,000 (1 year)	2.12×10^{-3}	0.0	0.0	0.0
4 years	0.0	0.0	0.0	0.0
Approximate Catalyst Cost (M$) ($35c/cm)	0.1	1.0	10.0	100.0

c. Blanket Design:

The problems of tritium containment in three blanket configurations will be examined which utilize potassium vapor cooled Beryllium augmented lithium beryllate ($Li_2Be_2O_3$) packaged within the blanket structure canned in (1) tungsten - 5% rhenium, (2) TZM, and (3) niobium - 1% zirconium.

A careful study of the feasibility of the various candidate solid lithium compounds for tritium recovery from fusion blankets has been very recently completed at Brookhaven National Laboratory.[16] A detailed review and analysis of these results suggests that the beryllium metal-lithium beryllate ($Li_2Be_2O_3$) mixture is the leading contender owing to its high melting point ([illegible] ^0C), low tritium solubility, sustained breeding ratio (BR $\geq$ 1.2), low residual radioactivity, low reactivity with air or other common chemicals (i.e., Be), and indications of absence of sintering problems. $LiALO_2$ would qualify except for its strong reactivity with Be or BeO.[19] The lithium beryllate compounds are ionic crystals with tightly bound oxygen such that they possess a very low (in fact, zero) solubility of the hydrogen isotopes.

The phenomenology of tritium production within the Be-$Li_2Be_2O_3$ particle mix can be visualized as follows. Neutron multiplication in the Be and the resulting bombardment results in a transmutation reaction of Li atoms within the crystalline structure to ^3H or T which diffuses through grain boundaries, dislocation, pores, etc., to the surface. Somewhere in this process, the oxide (T_2O) is formed, and this T_2O reaction product is chemically bound (complexed) on these

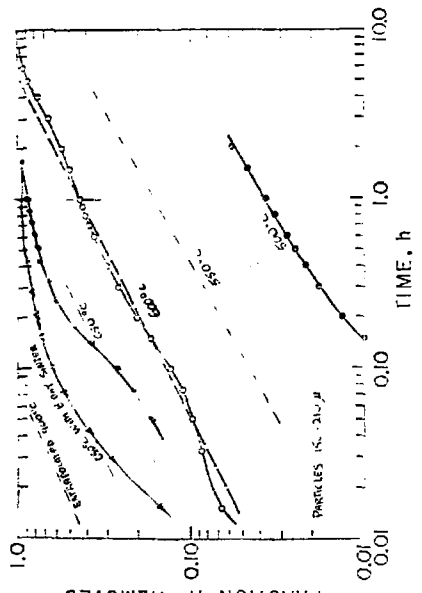


FIGURE 5: TRITIUM REMOVAL FROM $LiAlO_2$ PARTICLES VIA HOT HELIUM PURGE STREAM

internal and external surfaces. Chemical reduction of this T_2O-$Li_2Be_2O_3$ surface complex on the surrounding Be can occur. Thus releasing T_2 gas into the tritium scavenging stream. This decomposition from the surface is very slow [19] at the lowest concentrations. This fact is commonly experienced when one is attempting to pump down to very good vacuum a glass vessel that was previously wet with water. Thus, it is difficult to remove this small residual of water complexed with the $Li_2Be_2O_3$ particle and this step will most surely be rate controlling. Experiments at BNL [18] verify this phenomenon for $LiAlO_2$, which is expected to be similar in performance. This is the conclusion, although they assume the process is one of diffusion of T_2 within the particle. Their results are shown in Figure 5.

In any case, as a result of the complexity of this process, such experiments are needed to properly evaluate the rate of release of T_2 from the $Li_2Be_2O_3$ particle into the [illegible]. These experiments done at BNL [18] require special interpretation, as there are rather large excursions from the square root of time dependence. It can be seen also that preconditioning by high temperature air sintering at 1000°C has a profound positive effect on the removal rate of tritium. Much of these observed phenomena are not well understood, although it has been attributed to a 900°C phase transition. [16] Making cautious use of these experiments, the tritium containment problems associated with $LiAlO_2$ particles encapsulated within the breeding material of the blanket can be examined and extended to $Li_2Be_2O_3$. First, an equation relating blanket inventory, I_T, to reactor neutronic parameters, and solid lithium particle tritium holdup time, τ_T, can be written: [12]

$$I_T = 3 \times 10^4 \, \tau_T \frac{P_o \cdot BR}{Q \cdot N_o}$$

where Q is the total energy per DT fusion reaction (joules), P_o reactor power in watts, BR the breeding ratio in atoms T found in blanket per atom T burnt in plasma, and N_o Avagadro's number.

For the reference design of a Mirror Machine power reactor producing 1000 MW(e), Table 4 indicates the relationship of tritium inventory and $Li_2Be_2O_3$-tritium buildup. The data are available for 650°C and can be extrapolated to our operation condition of 850°C. Estimated removals have been tabulated in Table 4 for 850°C and these results indicate that holdup times around 5 minutes, 62% of the blanket contained tritium can be removed to achieve inventories of 12,500 curies or 1.2 grams.

Now let us examine this holdup time in more detail to obtain its physical meaning. Unsteady state diffusion from a sphere can be represented by the following equation:

$$\frac{C_{o,\theta} - C_{r,\theta}}{C_{o,\theta} - C_{r,\theta}} = \phi\left(\frac{D\theta}{R^2}, \frac{r}{R}\right)$$

where $C_{r,\theta}$ represents the concentration of tritium within the blanket $Li_2Be_2O_3$ particle at radius, r, at time, θ, D the diffusing of tritium within the particle of outside radius R. As a crude picture, when 87% (or $6/\sqrt{15}$ r) of tritium is lost from the particle, the time of exposure to the helium scavenging stream will be equal to the holdup time, $t_h = R^2/6D$. Thus, for 850°C and 200 u particles, this result would correspond to a D = 4.5×10^{-7} cm^2/sec.

TABLE 4

Holdup Times (min)	Tritium Blanket Inventory kCi	Fraction Tritium Removed from Blanket, 650°C	Estimated Removal Fraction 850°C
1	2.5	0.08	0.62
5	12.0	0.16	0.85
10	25.0	0.40	0.98
20	50.0	0.70	1.00
40	100.0	0.82	1.00
60	150.0	0.88	1.00

1. Tungsten - 5% Rhenium Blanket Containment

The first priority blanket concept has been shown in Fig. 6 for the Be-$Li_2Be_2O_3$ canned with a W-5% Re alloy. Our first task is to estimate the permeation of tritium to the potassium vapor coolant stream in order to properly size the wall thicknesses and obtain pressure drop estimates for the helium scavenging flow.

A helium scavenging flow rate is selected at 2 Kg/s which will provide good scavenging characteristics, low pressure drop through the blanket, and small piping costs. The porous Be-$Li_2Be_2O_3$ bed shown in Fig. 6 must be designed to provide a low pressure drop for the helium scavenging stream in order to avoid stressing the W 5% Re internal structures. A pressure drop around 21 kPa (3 psi) is desired. If a small fraction (1%) of the blanket area can be utilized for flow passages through the Be-$Li_2Be_2O_3$ bed, the flow passage diameter and number density can be computed. A 1% area would be about 2.5 m^2 (70 $ft.^2$) producing linear velocity of 2.6 m/s (25 ft/s). The diameter required to meet the pressure drop criteria will therefore be 0.7 mm. Thus, there would be about 25 holes/cm^2. The flow will be laminar at a Reynolds number of 5.

It should also be mentioned here that the total pressure of the helium scavenging stream is maintained around 1 atm. The potassium vapor coolant stream total pressure will be around 3 atms. The purpose of this difference is to insure that any failure or breeching of the W-5% Re blanket container will result in a guaranteed flow of potassium into the helium scavenging stream. This is expected to prevent massive tritium leaks into the potassium via mechanical fractures or pores.

At this helium scavenging flow of 2 Kg/s, the tritium production within the blanket of 2.6×10^{-3} g/s will result in a concentration of 1.8×10^{-6} atms or 5 Ci/m^3 in the helium flow. This concentration would constitute the driving force out of the blanket across the W-5% Re tubes into the K vapor coolant flow or any other sink. However, a scavenging agent such as lithium vapor can be added

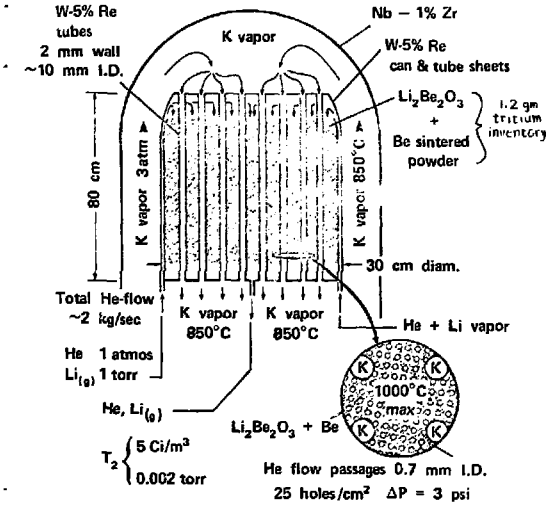


Figure 6 Blanket Concept

to the helium flow to complex the tritium into Li and Li_2T, which themselves will not permeate the W-5 Re barrier.

Data for the equilibrium distribution of gas phase constituents in the Li-D systems as a function of temperature are available[20] and are shown in Fig. 7. Here the mole fraction ratio of the total moles of D complexed with Li to the free D_2 is displayed with temperature. At 850°C, about 480 moles of D_2 are complexed with lithium for any single mole of D_2 freely available in the gas phase for permeation. In this way, the lithium vapor acts as a scavenging agent to reduce the effective D_2 concentration by a factor of 480. It is assumed that the Li-T system will behave identically.

It is obvious that this concept becomes feasible only at high temperature where the equilibrium is favorably shifted toward the complex above 500°C. There is one constraint, however. The amount of lithium vapor cannot exceed its vapor pressure or a liquid phase will condense out. In this design, a concentration of 1.0 torr (0.1%) was used, which is safely below its 4 torr vapor pressure.[21]

Available permeation data[22] for tungsten and other metals have been displayed in Fig. 8. It is assumed that the W-5% Re solid suspension system will behave as pure W. If this is true and the Li scavenging agent performs its task in the helium scavenging flow, the 2 mm W-5% Re tubes with an aggregated area[23] of about 2×10^7 cm² (about 5% of the heat transfer area) will permeate tritium at 160 Ci/day.

If this loss all appeared in the reactor hall atmosphere, the room processor could handle this load, since its maximum rating would be 180 Ci/day at 58.8 m3/s. However, most of this loss is expected to appear in the potassium vapor coolant loop with a flow around 5 kg/s,[23] resulting in a tritium concentration of around 1.0 μ Ci/m³. This level of contamination is 50 times MPC for a controlled

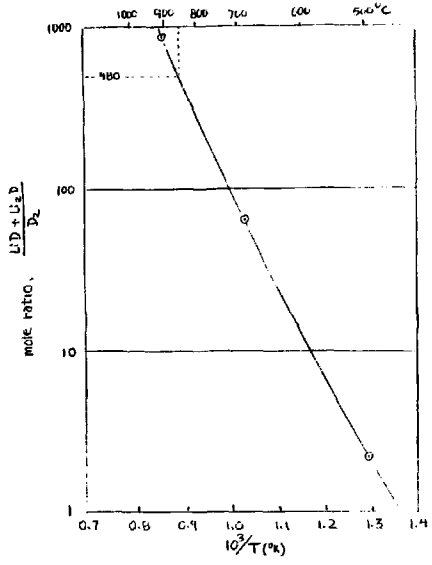


Figure 7 Distribution of Li-D Species

TRITIUM PERMEATION

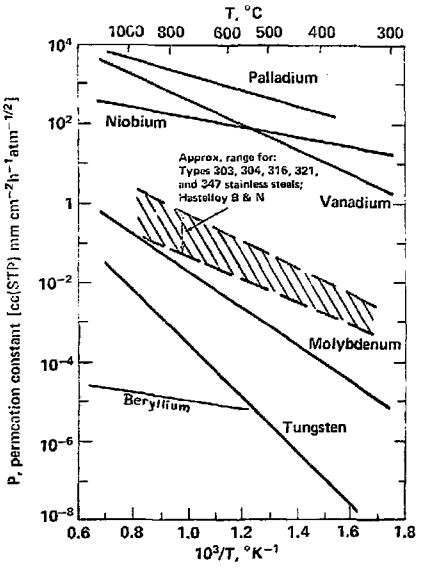


Figure 8: Available Permeation Data.

access area,[4] assuming that it was all converted to T_2O by the time it came in contact with operating personnel. This tritiumm contamination would for a line rupture emergency (authorized 25 rem dose) allow the emergency rescue worker exposures over a 24 hour period.

There are other sources of tritium, however. Tritium is produced at low levels within the beryllium in portions of the blanket that would not require a Li breeding material or an associated helium scavenging flow (added W degrading to the neutronics). For this reactor concept, this production would be 31,100 Ci/day This tritium production would be intolerable if it were totally released to the potassium vapor flow. Thus, these purely energy-producing blanket sections must be contained also with W-Re takes as before, thus reducing this leakage to 2 Ci/day.

Tritium would also enter the potassium vapor stream through the first wall (Niobium-1% zirconium) through energetic particle implantation deep into the niobium wall. Although the peak tritium concentration level will occur only a few microns[24] into the niobium surface from the plasma, the diffusion coefficient is so very large that one must assume that the wall acts as a source of tritium at a depth of 1.5 μm and in proportion to the relative distances 1.5 μm/2 mm, could leak into the potassium vapor stream. This leakage is expected to be 98.8 Ci/day, giving a concentration of 710 $\mu Ci/m^3$. These problems can be handled through the addition of a lithium vapor scavenging agent to the potassium vapor. As before lithium will complex the tritium containment via LiT and Li_2T and reduce the gaseous T_2 level some 480 times to 1.5 $\mu Ci/m^3$.

A moving bed getter[25] can be used to selectively remove tritium from the LiT and Li_2T from the helium scavenging flow and from a slip stream taken off the main potassium flow. Such getter systems[23] typically have capacities of 100 cc T_2 /gm of getter at S.T.P., thus this moving getter bed would process

the tritium leakage into the helium leakage into the helium scavenging flow with getter material moving at 5.5 Kg/min.

To handle the potassium vapor flow the moving getter bed concept can be utilized by processing a slip stream of 1.7 $\mu Ci/m^3$ T_2 through the moving getter bed. This small unit would only have to process 0.35 gm of getter per minute.

The leakage of tritium from the potassium coolant loop with 1.7 $\mu Ci/m^3$ tritium into the reactor hall can be extended assuming 100 meter. of 2.5 mm wall 15 cm I.D. SS pipe at 7.3 Ci/day. Most of this low level leakage can be purged by air ducting arranged along the pipeway channels with the flow into the reactor hall processor.

The leakage of tritium from the potassium through the steam boiler heat exchanger into the steam can be estimated for a boiler at 600°C with an area of 5 x 10^7 cm^2 with 3.8 mm thick stainless steel walls to be 19.2 Ci/day. A thin tungsten clad coating of 100 μ will reduce this leakage to 0.4 Ci/day.

A summary of these various process stream tritium concentrations and their associated leakages is given in Table 5. Clearly the process streams, where accidental exposure to occupational workers is possible, are held below MPC and the tritium leakage to the environment is well below our goal of 1 Ci/day, ranging from below 0.03 under accident conditions to 0.4 Ci/day if all the leak . from the boiler appeared in the steam cycle and escaped.

Table 5

Estimated Tritium Leakage Contributions

Source Unit	Stream Conc. ($\mu Ci/m^3$)	Leakage (Ci/day)	Recipient
Be-$Li_2Be_2O_3$ Blanket	----	3,110,000	He Scav. Flow
He Scavenging Flow	5,000,000	160	K vapor coolant
K Vapor Coolant	1.0	1.1	Reactor Hall
Be Blanket Sections	-----	31,100	He Scav. Flow
He Scavenging Flow	50,000	2.0	K vapor
Plasma	-----	98.6	K vapor coolant
K vapor coolant	0.7	0.9	Reactor hall
K Vapor Coolant (total T_2)	1.7	7.3	Reactor hall
K vapor coolant (total T_2)			
316 S.S.	1.7	19.2	Boiler Water
Haynes 25	1.7	8.7	Boiler Water
316 SS + 100μ W/ clad	1.7	0.4	Boiler Water
Reactor Hall	40	$\leq$0.03	Earth Atmosphere

Radiation Effects on Tungsten Alloys

In addition to serving as a tritium permeation barrier, the tungsten alloy tubes (W-5% Re) must exhibit acceptable mechanical properties under this high neutron flux and hydrogen environment. A brief summary of a survey[27] concerning this problem follows.

Irradiations by Moteff, et al.[28] in EBR-II, Row 7 to 10^{22} n/cm^2 (E >1 MeV) have been made over a wide temperature range in tungsten and a tungsten-25 rhenium alloy. In pure tungsten, they found 1.2% void swelling at 850°C. For a first wall fusion spectra, the damage per source neutron will be about 2-1/2 times that in EBR-II, Row 7 when the EBR fluence is given as E>1 MeV. We would thus expect 3% swelling at 10^{22} source neutrons/cm^2. The harder spectra and increased helium production might be expected to increase this. Thus pure tungsten would appear to be/marginal candidate for cooling tubes.

Tungsten-25 rhenium on the other hand showed only 0.1% swelling at 850°C and an absence of observable voids at 10^{22} n/cm^2. From experience with the effect of alloying additions on void swelling in other metals, we would expect that a few percent rhenium would be equally effective in suppressing void swelling. This would appear to be a viabl choice for the cooling tubes at 5% Re.

The impact of this damage on the overall permeation to tritium is not known at this time, but as long as the mechanical integrity of the W-5% Re tubes are not breeched, gross pathways to tritium should not appear. Much research on the tritium barrier, hydrogen resistance, neutron resistance, and fabricability of such blanket materials is desperately needed.

2. TZM Blanket Containment

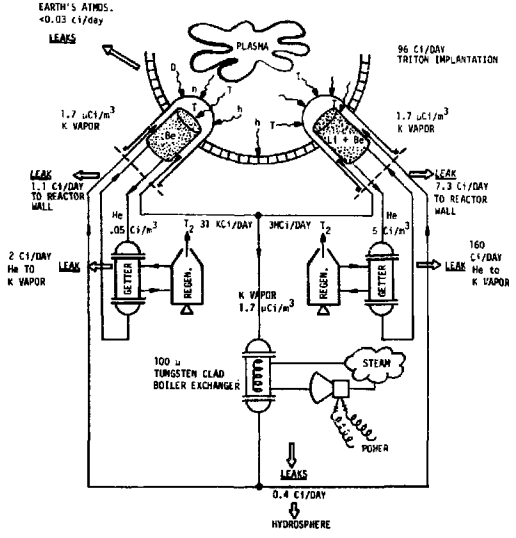
The molybdenum alloy TZM (0.5% Ti, 0.8% Zr, and 98.7% Mo) could be used for the blanket can; however, as can be seen from Figure 8, the permeation rate for molybdenum is 40 times that for tungsten thus the tritium leakage would increase to 6400 Ci/day. If this leaks into the reactor hall, this would represent some 36 times the capacity of the room air processor. If the processor capacity were increased to ~ 352 m^3/s (4,000,000 cfm), the catalyst cost alone would total

$120 million[12].

If this tritium leakage were all to appear in the potassium loop at 40 $\mu Ci/m^3$, this would far exceed by 20 the MPC as T_2O for occupational workers. In order to keep the tritium leakage to the stream below 1 Ci/day, a tungsten-rhenium alloy exchanger would be required -- a fantastic cost indeed! Or a massive helium processor could be used on the scavenging loop, processing 220 Kg/min of getter material and reducing the T_2 level to 31 $\mu Ci/m^3$ -- a very tough and expensive task. A approach might be to avoid the lithium vapor and utilize a catalytic oxidation-molecular sieve trap concept. The system could use containment for 3×10^6 Ci/day with minimum extensive piping and exchanger barriers.

3. Niobium -1% Zirconium Blanket Containment

The niobium alloy used as a blanket container would be 50,000 times more permeable than our W-5% Re first priority case. The processing requirements necessary with the present geometry and containment to meet 1 Ci/day leakage goals are far beyond our present technical ability and economic constraints. The only alternate that could be considered would be to utilize high temperature permeators that would be comparable in size to the main energy recovery heat exchangers to reduce the tritium level as much as possible and let the remaining fraction leak into the reactor hall. This extreme philosophy, of course, requires that all operations and maintenance be done remotely, that the reactor hall be hermetically sealed with very thick stainless steel lines, and that the atmosphere be continually processed to handle a load of 3 million Ci/day. Again, the processor cost would be around $1.2 billion. Equipment once committed to this reactor hall would have to remain impounded for some 70 years until the decay could allow its removal to a remote "hot lab." for decontamination and disposal.



LEAKAGE OF TRITIUM TO THE ENVIRONMENT

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SECTION 6

NUCLEAR PERFORMANCE - J. D. Lee

The physics of mirror plasma containment results in a low energy gain (Q) fusion system. The performance of low Q fusion systems is quite sensitive to the amount of energy generated in the blanket per fusion reaction (see Fig. 1-2). We call the ratio of thermal energy produced in the blanket per fusion neutron to the kinetic energy of fusion neutron, blanket multiplication (M). For DT fusion

$$M = E_{Blk} \text{ per fusion} \div 14.06 \text{ MeV}.$$

The objective of the neutronic design of blankets for mirror reactors is to get the highest M as practical while also getting a tritium breeding ratio slightly greater than unity (~ 1.1). The nuclear design of the blanket must also be compatible with requirements imposed on the blanket by geometry, structural, heat transfer, thermal conversion, materials, and tritium handling considerations.

Two major classes of blankets for mirror reactors are under investigation--one in which the DT neutrons are used to fission a heavy element such as ^{238}U (known as the hybrid)--the other in which the DT neutrons generate additional neutrons through n,2n type processes which, in turn, are captured in a material with an attractive total Q for neutron capture.

The major emphasis here is on the second class of blankets since the uranium blanket has previously been reported (ref. 6-1 & 6-2).

Blanket Description

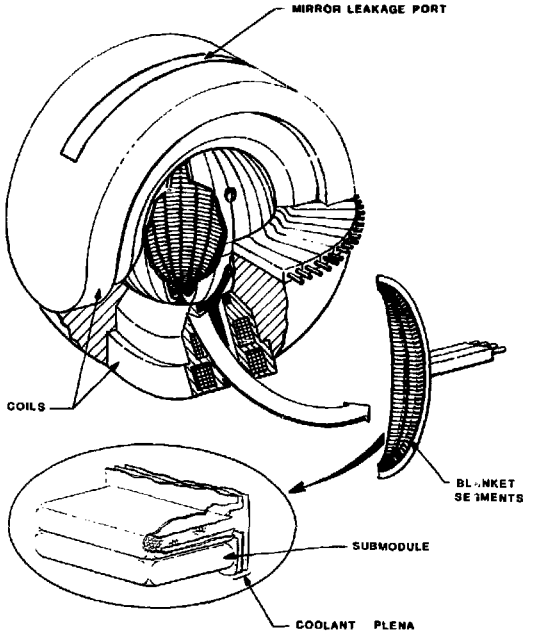
General - The overall geometry of the blanket is a spherical annulus

made up of segments as shown in Figure 6-1. The segments, in turn, are made of individual pressure vessels called submodules. Coolant is supplied to the submodules by plena that make up the outer portion of the blanket segment. The submodules contain materials to absorb and convert the 14 MeV fusion neutron's kinetic and potential energy to thermal energy.

In a hybrid the submodules contain natural or depleted uranium for energy and neutron multiplication and plutonium breeding, and lithium for tritium breeding. For the nonhybrid blanket the submodules contain beryllium (Be) for neutron multiplication, lithium for tritium breeding, and a high Q neutron absorber for energy multiplication. The overall geometry is the same for both hybrid and non-hybrid blankets.

Beryllium Blanket

The use of beryllium (Be) is dictated by the objective to get the highest energy multiplication (M) possible without resorting to heavy element fission. The Be (n,2n) 2He - 1.67 MeV reaction is used for neutron multiplication. Some neutrons are then captured by ^{6}Li to breed tritium (T) and some energy ($^{6}Li(n,T)\alpha$ Q = 4.8 MeV). Most of the remaining neutrons are then free to be captured by an element with a high positive Q such as vanadium ($^{51}V(n,\gamma)^{52}V$ Q = 7.3 MeV + [illegible] MeV from $^{52}V(\beta^-)^{52}Cr$ decay). Beryllium is by far the best neutron multiplier we have found. Its (n,2n) reaction has a low negative Q (-1.67 MeV), it has a reasonable (n,2n) cross section (0.5 barns at 14 MeV), and fraction of the total cross section (~ 0.3 at 14 MeV), and its thermal capture cross section is low (0.01 barns). The choice of a high positive Q (n,γ) material is more open. Elements such as V, Mo, Ti and Mn are examples, all have total Q's in the 8 to 10 MeV



MIRROR FUSION REACTOR

Fig. 6-1

range and thermal (n, γ) cross sections ranging between 4.5 barns for V to 13 barns for Mn.

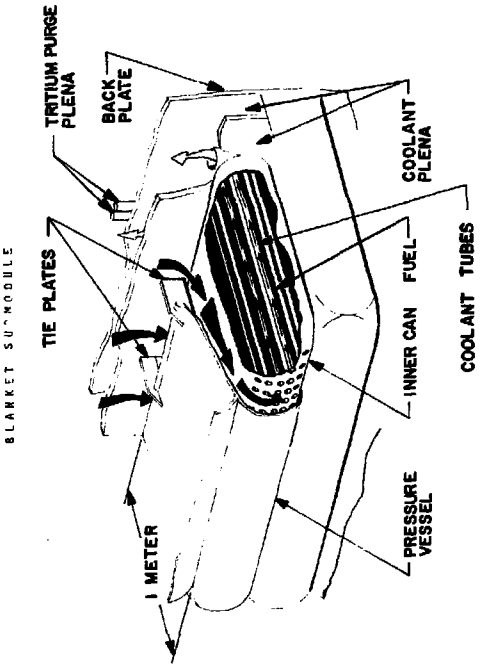
From a purely neutronics point of view the blanket should consist of regions containing homogenous mixtures of Be plus few percent of a high (n, γ) capture material and Be plus a few percent ^6Li. The amount of other materials, such as structure, should be kept to a minimum since its presents will reduce blanket performance by competing with the Be for fast neutron interactions.

Structure must be present of course to contain the fuel and coolant. Examples of potential candidates are Nb-1Zr, cobalt alloys such as "Haynes 25" and more common stainless such as 316. Our present choice for the structural material is Nb-1Zr and for the coolant potassium. Our tritium breeding material is Li_2BeF_4.

Submodule Design

The submodule consists of a rectangular vessel closed by caps on 3 ends and the coolant plena on the 4th. (Fig. 6-2). The submodules share common sides and are fed and discharge coolant in parallel. Coolant enters each submodule from the lower plenum, flows down the periphery cooling the pressure vessel, then reverses direction flowing back up through the interior removing nuclear heat from the "fuel", then is exhausted into the upper plenum.

Two types of blanket segments are invisioned for this blanket, tritium breeding and non-tritium breeding segments. The submodules of the breeding segments contain Be "fuel" seeded with a compound of lithium. In the non-breeding segments the fertile (Li) material is replaced by a high Q (n,γ) material such as vanadium or manganese. The principle



BLANKET SUBMODULE

reason for segmentinn the blanket this way is to concentrate most of the tritium in a portion of the blanket loop, which hopefully will simplify tritium handling. Unfortunately some tritium will be in the non-breeding portion of the blanket as well because a little tritium is produced from Be. Approximately 2% of the total tritium produced is from Be, therefore, ~1% would be produced in the non-breeding portion of the blanket. Of course, the breeding and "non-breeding" fuel can be combined in individual blanket segments or submodules if desirable.

The Be fuel is contained in cans and consists of a homogenous mixture of Be (~95%) plus 90% enriched lithium as $Li_2Be_2O_3$ (~5%). In non-breeding fuel the high Q (n,.) material vanadium is used in place of the lithium compound. We believe the fuel can be made by straightforward power metallurgy techniques resulting in sintered material with ~80% of theoretical density that has interconnecting voids for tritium and helium release.

Most of the tritium is removed from the fuel can by a purge gas system. As discussed in the Tritium Section ( ), the amount of tritium that will permeate into the K coolant is dependent on the choice of fuel cladding material and thickness. As you will see, this choice also strongly influences the blanket's nuclear performance. Three combinations are being considered--thin Nb-1Zr clad, thick (2mm) W clad, and thick (2mm) Mo clad.

Calculational Models & Methods

Blanket nuclear performance was calculated using a nested set of spherical annuli for a blanket model and two monte carlo transport codes, TART and TARTNP. (Ref. 1-3). TART is a neutron code and TARTNP is a

coupled neutron-gamma code. Both codes use 176 group cross section sets generated from the Livermore EMDL nuclear data library (Ref. 6-4).

The basic blanket spherical calculational model consists of a 3 meter radius volume source of isotropic 14 MeV neutrons, a 4.85 first wall radius, a 0.25 cm Nb first wall thickness, a 1 meter multi-zone fuel region and a 10 cm Nb back plate. The Be fuel zones consist of (by volume fraction) .02 Nb structure, .05 K coolant, .01 Nb clad, or .03 W clad or .03 Mo clad, with the remainder occupied with fuel at 80% theoretical density.

The calculated blanket energy multiplication (M) and tritium breeding ratio (T) for the representative cases are listed in Table 6-1. Six "local cases" are listed. "Local cases" refer to blankets with 100% coverage that contain one type of fuel and/or clad. The Be + Li case is for Be + $Li_2Be_2O_3$ fuel canned in thin Nb. The Be + Li + W and the Be + Li + Mo are the cases with Be + Li fuel clad in 2 mm W or Mo, respectively. The Be + V is for the Be + V fuel clad in thin Nb case. The Li + Mo case is included to show how natural lithium clad in 2 mm Mo compares to the Be fueled case. The U - Mo case (Ref. 6-2) is included to show how a representative uranium blanket compares.

The "Effective Values" listed in Table 6-1 are the values of M that results after tritium breeding requirements and fractional blanket coverage effects are accounted for. For example, the Be + Li blanket segments produce 2.3 tritons per DT neutron. Therefore, to get 1.1 tritons per fusion 1.1 ÷ 2.3 = 48% of the blanket spherical area must be covered by the Be + Li blanket. Since only 90% of the spherical area contains blanket (10% for injection and plasma leakage ports)

TABLE 6-1

NUCLEAR PERFORMANCE OF BLANKETS

Local Cases	M	T	
Be + Li	1.8	2.3	
Be + Li + W	1.5	1.5	
Be + Li + Mo_o	1.7	1.9	
Be + V	2.3	.02	
Li + Mo	1.2	1.4	
U-Mo	11	1.2	(1.7 Pu/n)

Effective Values (for 90% blanket coverage)

Be + Li & Be + V	1.8	1.1	
Be + Li + W & Be + V	1.5	1.1	
Be + Li + Mo & Be + V	1.7	1.1	
Li + Mo	1.1	1.3	
U - Mo	1.2	1.1	(1.5 Pu/n)

90% - 48% = 42% is available for the high M Be + V blanket segments
The resulting M is:

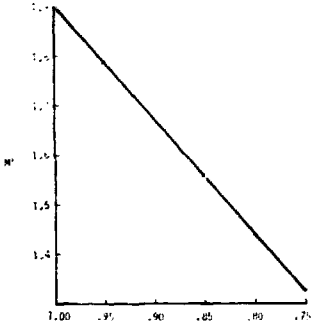
$$\therefore \; M_{\substack{(Be + Li) \\ (Be + V)}}^{90\%} = .48 \times 1.8 + .42 \times 2.3 = 1.8$$

The results listed in Table 6-1 point out 3 important trends: (1) providing an effective tritium barrier in the blanket will cause a significant drop in blanket performance, (2) the Be blankets perform 40-60% better than the standard lithium blanket, and (3) the best of the Be blankets produce 5 times less energy than the uranium blanket.

Another important factor effecting blanket performance is blanket coverage. This effect is displayed in Figure 6-3 for a representative case.

Thermal power density profiles, total flux profiles, and flux spectra for both a Be and all blanket are displayed in Figures 6-4 through 6-9. The Be profiles are normalized to wall loading of 1.0 MW/M^2.

Obviously, we have just scratched the surface of the nuclear design questions. More blanket iterations are needed to develop an optimized nuclear design that is consistent with the other blanket requirements imposed by the reactor system. We also must address the shielding and activation questions. Effective shielding design is especially critical in the leakage fan area where space is at a premium. (Fig. 6 -10 and 2-5). After we arrive at a consistent and viable design, the question of uncertainties in nuclear data and calculational methods must be dealt with.



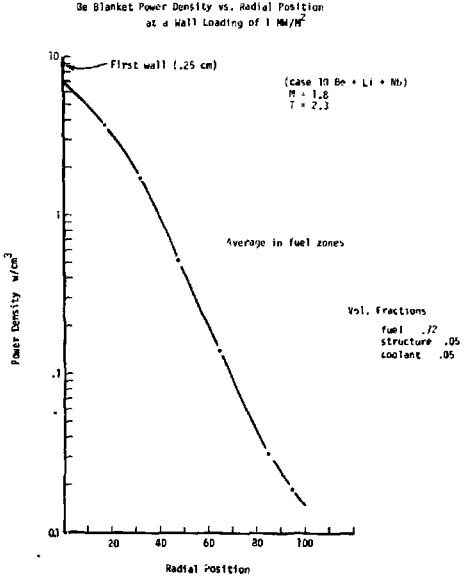

Be Blanket Power Density vs. Radial Position
at a Wall Loading of 1 MW/M²
First wall (.25 cm)
(case 10 Be + Li + Nb)
M = 1.8
T = 2.3
Average in fuel zones
Vol. Fractions
fuel .72
structure .05
coolant .05
Power Density w/cm³
10
1
.1
.01
20
40
60
80
100
Radial Position

Fig 6 4

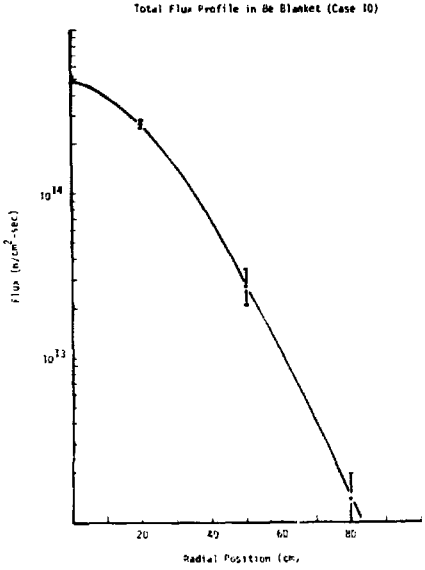

Total Flux Profile in Be Blanket (Case 10)
Flux (n/cm²-sec)
10¹⁴
10¹³
20
40
60
80
Radial Position (cm)

Fig. 6-5

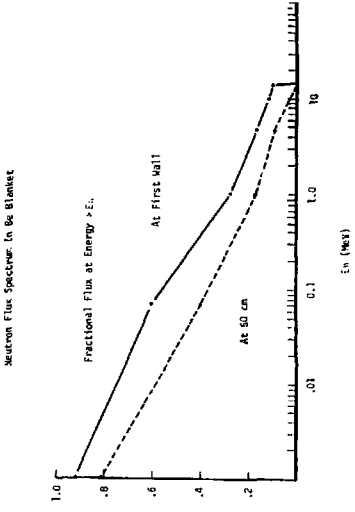

Neutron Flux Spectrum in Be Blanket
Fractional Flux at Energy >E
At First Wall
At 50 cm
1.0
.8
.6
.4
.2
10
1.0
.1
.01
En (MeV)

Fig. 6.6

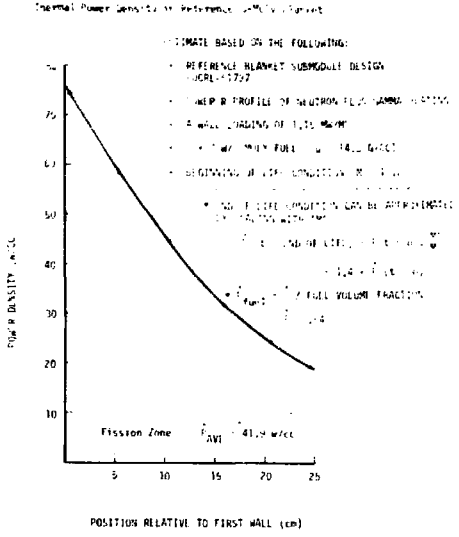

Thermal Power Density in Reference
ESTIMATE BASED ON THE FOLLOWING:
REFERENCE BLANKET SUBMODULE DESIGN
Fission Zone
POWER DENSITY
POSITION RELATIVE TO FIRST WALL (cm)

Fig. 6.7

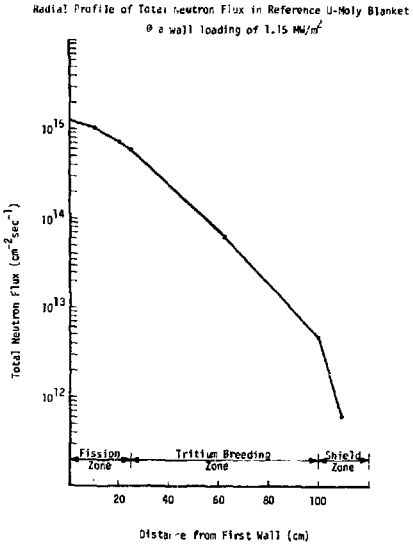
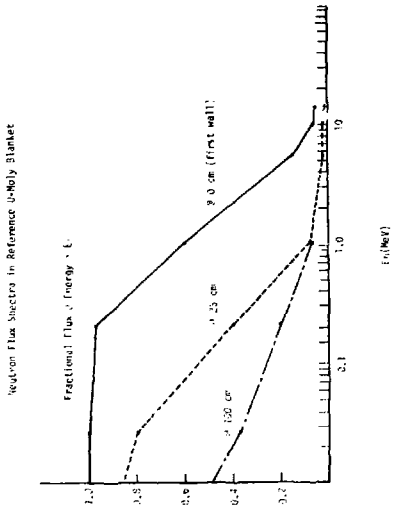
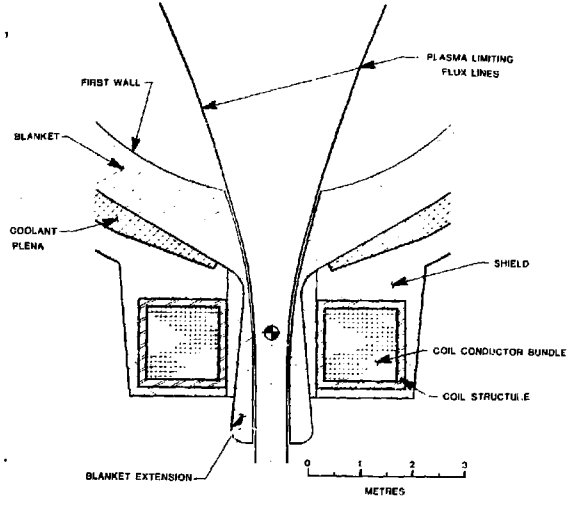

Radial Profile of Total Neutron Flux in Reference U-Moly Blanket
@ a wall loading of 1.15 MW/m²
Total Neutron Flux (cm⁻²sec⁻¹)
10¹⁵
10¹⁴
10¹³
10¹²
Fission Zone
Tritium Breeding Zone
Shield Zone
20
40
60
80
100
Distance from First Wall (cm)

Fig. 6.5


Neutron Flux Spectra in Reference U-Moly Blanket
Fractional Flux per Energy Unit
0 cm (First Wall)
100 cm
10
1.0
0.1
E(MeV)



LEAKAGE FAN DETAIL

Fig. 6.10

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6-4 R. J. Howerton, R. J. Doyas, T. C. Michels, and S. T. Perkins, An Integrated System for Production of Neutronics and Photonics Calculational Constants, Lawrence Livermore Laboratory Rpt. UCRL-50400, Vol. [illegible] (1971).

APPENDIX 1 - SUMMARY TABLE FOR MIRROR REACTOR BLANKETS

FACILITY	BLANKET #1	BLANKET #2
1. Facility Type(s)	Commerical Hybrid	Commerical Power
2. Life (years)	~ 30	~ 30
3. Powers (MW)		
a. Fusion	580	2090
b. Injection (neutral atom)	900	2000
c. Electric (from blanket loop)	1550 (η_{th} * .33)	1320 (η_{th} * .48)
d. Net	1035	890
4. First Wall Loading (MW/M^2) (ave/peak)		
a. Neutron	1.1/1.1	3.1/3.1
b. Brems. + Syn.	~ .02/.02	0.1/0.1
c. Particle	~ .06/.2	NA/.02
5. Operating Cycle		
a. Length of cycle	Steady State Plasma	
b. Burn fraction		
c. Plant factor	80% (target)	
6. Impurity Control -	Inherent to mirror plasmas	

BLANKET

7. First Wall Structure			
a. Type -	Separate where beam loads may be locally high and or if vacuum baffle is advantageous.		
b. Construction (1 and 2)	Design of a separate wall not yet developed but are considering thin radiantly cooled members. Principle first wall is integral part of blanket pressure vessel which is cooled by primary blanket coolant.		
3 Materials	SS316		Nb-1Zr
4 Joints	Welded sheet		

		Blanket #1	Blanket #2
c. Thermal and Mechanical parameters			
1	Temp °C	350	850
2	Thickness (cm)	0.5	0.5
3	Stress (design)(psi)	10^4	6×10^3
d. Effects of operation			
1	Sputtering	NA	
2	Max fluence (n/cm^2)	$\sim 1 \times 10^{23}$	
3	Ductility @ max fluence	0.5% uniform	
f. Max thermal flux during abnormal conditions		NA	
8. Balance of Blanket			
a. Blanket structure			
1	Metal & (v/o)	SS316 (9)	Nb-1Zr (5)
2	Max/Min Temp (°C)	600/250	900/850
3	Max stress (design)	8×10^3	4×10^3
b. Coolant			
1	Type	He	K
	T_{in} (°C)	250	850
	T_{out} (°C)	450	850
	v/o	20	20
2	Pressure (psia)	300	35
	ΔP (psi)	20	3
3	Max Velocity m/sec	60	50
4	Variations during cycle	-	-
5	Impurity levels	NA	NA
6	Pumping power/Thermal power (%)	~ 5	<0.1

		BLK #1	BLK #2
c. Moderator			
1	Type(s)	Uranium alloy or compound Graphite and or $LiAlO_2$	Be + $Be_2Li_2O_2$ Carbon Compounds (SiC)
	Vol. fraction %	54	60-70
2	T Max (°C)	600 (in U alloy)	950
		800 (in UC)	
		615 (in $LiAlO_2$)	
3	Canned ?	Yes	Yes
4	Max fluence	$\sim 1 \times 10^{23}$	NA
5	Dimensional change %	~ 5	NA
d. T Breeding material			
1	Type(s)	$LiAlO_2$ +C	$Be_2Li_2C_2$ + Be
	Vol Fraction %	54	60-70
2	T max (°C)	650	950
3	Breeding ratio		
	Local	~ 1.1 (at t = 0)	2.5
	Ave (in space & time)	1.1	1.1
4	Tritium content (appm)	~ 7	NA
e. Overall blanket thickness (m)		1	1
f. Blanket coverage %		87	93
g. Energy Multiplication (E/14 MeV)			
Local maximum @ time 0		10	2.3
Ave (in space & time)		10	1.7
h. Fissile fuel production (atoms/fusion)			
Local at time 0		1.8	--
Net ave (in space & time)		1.5	--

	BLK #1	BLK #2
Shield Parameters	NA (assuming ~ 4 orders of magnitude attenuation from ~ 80cm of shield)	NA
Tritium Recovery System		
13. Methods of recovery	Vandium diffuser @ 600°C Molecular sieve bed at -20°C	Diffuser and or getter
14. T inventory (g)		
in blanket	~ 40	NA
in coolant	<0.1	NA
Balance of plant	4000	NA
15. T leak	NA	NA
Power Conversion		
16. Types of systems	Rankine (H_2O)	Binary Rankine (K & H_2O)
17. Peak temp of working fluid (°C)	400	850
18. Materials		Haynes 25
19. Overall gross efficiency (%)	33	48
Net plant efficiency (%)	22	28
Blanket & First Wall Replacement		
20. Blanket & first wall life (y)	2.7	1 +
Fraction replaced/shutdown	.25	1.0
21. Scheduled shutdowns/year	1.5	1 or less
Length of shutdown (month)	.75	1
22. Method of blanket replacement	Separation of the 2 magnets	
23. Number of segments replaced/shutdown	4	16
Weight of largest segment (tonne)	~ 125	75

24. After heat (% of operating power)		
at shutdown/after 10 days	6/.5	NA
Environment		
25. Residual activity	NA	NA
26. Tritium leak - Design goal of 1 curie/day		

Printed in the United States of America
Available from
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U.S. Department of Commerce
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Springfield, Virginia 22151
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