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FUSION POWER

AN ASSESSMENT OF ULTIMATE POTENTIAL

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DIVISION OF CONTROLLED
THERMONUCLEAR RESEARCH

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FUSION POWER: AN ASSESSMENT OF ULTIMATE POTENTIAL

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Division of Controlled Thermonuclear Research
U.S. Atomic Energy Commission

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PREFACE

The objective of controlled thermonuclear research (CTR) is the development of practical fusion reactors for production of electrical and thermal energy in large central station installations. At present the principal program task is to demonstrate that a plasma of light nuclei can be confined at sufficient temperature and density for a long enough period of time to release more energy by means of a controlled thermonuclear fusion process than was required to create the plasma.

Considerable progress in understanding the complex character of the plasma state has occurred in the last few years. This understanding was sufficient to permit experiments which exhibited particle confinement times close to the "classical" upper limit -- the theoretical maximum possible in a completely quiescent plasma at a particular density and temperature. This achievement was obtained in several different experiments, and it provided a basis for renewed optimism.

Although it is exceedingly difficult to predict when fusion power will become available, it is clear that there are many technical and socio-economic variables which could speed or slow its development. Present estimates indicate that an orderly aggressive program might provide commercial fusion power about the year 2000, so that fusion could then have a significant impact on electrical power production by the year 2020.

Fusion power has been recognized as having the potential of minimum environmental insult. This expectation is very general and deserves detailed backup. Because some second generation fusion reactor system designs have recently been developed, it is now possible to analyze the ultimate potential of fusion power to a meaningful extent and that is the subject of this report. The approach taken was to evaluate the projected characteristics of fusion power plants in an absolute sense and to compare fusion systems with current or other projected energy sources.

The material contained herein was developed by a special committee organized by the Division of Controlled Thermonuclear Research of the U.S. Atomic Energy Commission. This committee was very ably chaired by Dr. Arthur Fraas of the Oak Ridge National Laboratory and consisted of the following members:

Dr. S. Burnett, Los Alamos Scientific Laboratory,
Dr. T. Coultas, Argonne National Laboratory,
Dr. D. Dudziak, Los Alamos Scientific Laboratory,
Dr. G. Hopkins, Gulf General Atomic,
Dr. G. Kulcinski, University of Wisconsin,
Dr. R. Mills, Princeton Plasma Physics Laboratory,
Dr. B. Myers, Lawrence Livermore Laboratory,
Dr. F. Ribe, Los Alamos Scientific Laboratory, and
Dr. F. Tenney, Princeton Plasma Physics Laboratory.

A second special committee assembled as part of an Energy R&D Goals Study chaired by the President's Office of Science and Technology, reviewed the material contained herein and developed additional background. This group was composed of the following:

Dr. Robert L. Hirsch, Chairman, U.S. AEC,
Dr. T. K. Fowler, Lawrence Livermore Laboratory,
Dr. A. P. Fraas, Oak Ridge National Laboratory,
Dr. M. B. Gottlieb, Princeton Plasma Physics Laboratory,
Dr. H. T. Motz, Los Alamos Scientific Laboratory,
Dr. H. Postma, Oak Ridge National Laboratory, and
Dr. A. W. Trivelpiece, University of Maryland.

Since fusion reactor technology is in an early stage of development, the conclusions in this report must be viewed accordingly. While no confinement configuration has yet emerged from plasma research as the clear choice for the future and no preferred reactor design for any configuration is clearly more desirable than another, the general conclusions reached herein are nevertheless believed to be generally representative of what might ultimately be expected from fusion power reactors.

Robert L. Hirsch, Director
Division of Controlled
Thermonuclear Research

I. INTRODUCTION

Controlled fusion research is a scientific discipline which developed worldwide over the past 20 years. In the early 1950's fusion was a classified field of research, and little was known about its root science -- the physics of high temperature plasmas. In 1958, the fusion program was declassified, and by the early 1960's a number of relevant scientific problems were identified and a systematic study of them begun.

The difficulties that arose became the central problem of fusion research -- the isolation of a reacting fusion plasma from its surroundings. The principal approach to this problem, then as now, was to confine a fusion plasma through the use of specially shaped magnetic fields, which were to control the motions of its individual ions and electrons. However, in attempting to apply this technique, it was soon discovered that spontaneously arising turbulence and unstable plasma oscillations significantly weakened the confining effect of the magnetic fields. As a result of several years of intensive theoretical and experimental research, the plasma instability problem was brought under reasonable control by the late 1960's. In fact, the understanding of instabilities and means for their control was sufficient to permit experiments which exhibited confinement close to the "classical" upper limit -- the theoretical maximum possible in a completely quiescent plasma at a particular density and temperature. This achievement was obtained in several different experiments, and it provided a basis for renewed optimism with respect to ultimate success.

The scientific, technical and size limitations of present-day fusion experiments preclude any of them from simultaneously achieving all three of the plasma parameters (temperature, density and confinement time) required for a fusion reactor. This achievement, which would demonstrate scientific feasibility, will require larger, more complex facilities than presently available. In addition, it will be necessary to continue the development of relevant technologies at an expanded level in parallel with plasma experimentation.

There clearly remain significant scientific questions about plasma behavior in reactor regimes of temperature and density, but if the presently favorable trends toward better fusion plasma confinement continue, these questions should be sufficiently resolved so that preparations for scientific feasibility experiments can begin in the mid-1970's. These might be ready to begin operation near the end of the decade, and then they should be ready to test fusion scientific feasibility in the 1980-1982 period.

The laser-fusion process was recognized in the early 1960's as being of potential military interest and as a possible method of achieving practical fusion power. In the late 1960's it was possible to envision the high power lasers capable of permitting the detailed study of the important physics questions relevant to laser-fusion, and this provided part of the basis for expanding the laser-fusion program. Large lasers are now being developed in an effort to perform the key basic studies. If these studies prove favorable, still larger new laser facilities could permit the demonstration of the scientific feasibility of the laser-fusion process in the latter 1970's.

The quest for fusion power has resulted in the development of a new field of research -- high temperature plasma physics. Plasma physicists believe that in the coming decade the scientific proof that fusion reactors can indeed be built can be obtained. Given that proof, the fusion program can shift into a development phase in which the practical problems of producing economically competitive electrical power can be fully addressed.

The primary application of fusion power will be for the production of electrical and thermal energy. It is difficult to predict when this achievement will be attained because of the myriad of physical, technological, and socio-economic variables which can either accelerate or slow its future development. An analysis of what might be accomplished in an orderly aggressive program indicates that central station fusion power might become commercial about the year 2000. Assuming any of various models for its introduction into the utility market thereafter, fusion power could then have a significant impact on electrical power production in the year 2020.

II. INHERENT CHARACTERISTICS OF FUSION POWER SYSTEMS

Even though fusion power is not yet a reality, it is possible to assess approximately its operating, safety, economic, and environmental characteristics. Fusion systems potentially offer a number of attractive advantages and a variety of choices to the utilities and to the public. To convey the flexibility of fusion as well as to provide specific information, the approach herein is to first describe some of the inherent features of the basic system and then to present the characteristics of preliminary fusion reactor designs based on the deuterium-tritium (DT) fuel cycle.

Fusion reactors will be inherently safe against nuclear runaway. All full-scale fusion reactor concepts involve small quantities of fuel (of the order of a gram) in the core region. The working fluid of fusion is a gaseous plasma, which because of its high pressure tends to expand whenever it is not suitably confined. When a plasma expands, its fusion rate, which is proportional to the square of its density, decreases drastically. Plasma confinement is a delicate process requiring a high degree of control. These basic characteristics indicate that whenever confined fusion plasmas are perturbed in any manner other than by very special kinds of compressive forces, they will tend to expand, thereby decreasing or quenching the reaction rate.

There are a number of possible fusion fuel cycles, the most promising of which are shown in Table I.

Table I

Reaction Equation	Approximate Threshold Plasma Temperature	Approximate Average Energy Gain per Fusion
$D+T \rightarrow {}^4He$ (3.5 MeV) + n (14.1 MeV)	10 keV	1800
$D+D \rightarrow {}^3He$ (0.82 MeV) + n (2.45 MeV) $D+D \rightarrow T$ (1.01 MeV) + p (3.02 MeV)	50 keV	70
$D+{}^3He \rightarrow {}^4He$ (3.6 MeV) + p (14.7 MeV)	100 keV	180

All of these cycles require an energy investment to initiate fusion, and all utilize deuterium which occurs abundantly in nature and which is available at low cost. The first requires tritium which does not occur naturally and which therefore must be bred. The third reaction utilizes ^3He which could be obtained from DD reactions. All three cycles involve a copious emission of neutrons from either the primary or secondary reactions, e.g., DD reactions in the D^3He cycle.

Because of its high energy gain and its low threshold temperature, the DT reaction is considered most attractive for first generation fusion reactors. The inherent features of the reaction will determine many of the basic characteristics of DT fusion reactors:

1. Because about 80% of the energy output is carried by the neutrons, a special blanket will be required to convert neutron kinetic energy to thermal energy, as well as to provide a biological shield.

2. Efficient neutron thermalization will require a blanket of low atomic number materials.

3. Because neutron moderation gives rise to thermal energy, DT reactors will work primarily on a thermal conversion cycle.

4. The blanket region of a DT reactor will become radioactive because nearly all materials become activated to some degree by energetic neutron bombardment. This activity can be minimized by appropriate materials choices.

5. Tritium will have to be bred, with neutron absorption in natural lithium appearing attractive. Breeding ratios to 1.5 appear possible, giving doubling times of about a month. (A ratio of 1.3 appears typical.)

6. The elemental reaction product is helium, which is inert.

7. Because the energy gain is high, there is flexibility to deal with system losses and inefficiencies.

8. Because of its high reaction rate, the DT cycle has the potential of being self-sustaining since the energetic charged fusion products (helium) can feed energy directly into the plasma.

While the other cycles have lower energy gains, they have a number of very attractive features:

1. DD cycle. DD reactions utilize naturally occurring deuterium and do not require external tritium breeding, which removes an important constraint from the blanket requirements. The DD reaction products (T and ^3He) are themselves fuel and will partially react with the deuterium before escape from the plasma. Unburned T and ^3He can be reinjected to improve the fractional burnup.

2. D^3He cycle. By increasing the operating temperature and reinjecting only the ^3He, the DD cycle can operate such that D^3He reactions contribute most of the output power, with as little as 10% of the output being from DD neutrons (and its tritium byproduct). With efficient direct conversion of the energy from the charged D^3He reaction products, increased overall system efficiencies appear possible.

In the following the DT cycle will be considered almost exclusively because it is the most likely choice for the first generation of fusion reactors. However, depending upon the ultimate performance of certain of the confinement concepts, it may prove highly desirable to actively develop systems employing alternate fuel cycles.

III. METHOD OF ASSESSMENT

For the purposes of this study the ultimate potential of fusion power has been appraised by considering a set of reference designs for full scale fusion reactors. These reference designs were developed by personnel associated with the four major fusion concepts considered to be approaching feasibility tests. They are the tokamak, the theta pinch, the magnetic mirror, and the laser-fusion system. (See the Appendix for descriptions of the various reference designs). The reactor versions of these concepts have some common and some unique features. Three involve magnetic confinement and two of these would employ superconducting magnets to contain the hot plasma in a vacuum chamber. Laser-fusion reactions are envisioned to occur so rapidly that inertial forces provide adequate confinement. In their projected first generation reactor versions, all would utilize the DT fuel cycle and all would supply power in the range of 500 MW(e), or more, corresponding to the current size-range of central power stations. In addition, the laser and mirror systems may be suited to specialized applications requiring power outputs of as little as 50 MW(e).

All conceptual fusion reactor designs are based on the best available plasma physics information. Because the fusion reactor design activity is still in an embryonic stage, these designs differ substantially in the extent to which efforts have been made to resolve the engineering problems of both core and facility design. This is particularly true of the laser-fusion system where some thought has been given to containment vessel engineering, but little can yet be said regarding the laser. In only one case has a design gone through some iterations to factor in considerations of reactor safety. Therefore this design -- the ORNL tokamak reactor -- was chosen as the Reference Controlled Thermonuclear Reactor or Reference CTR for the purposes of this study. In many ways the conclusions drawn from the analysis of this design are believed to be representative of what might be expected for the other concepts. The choice by no means implies any favoritism towards one concept over the others.

IV. BRIEF DESCRIPTION OF THE REFERENCE CTR

The principal features of the conceptual design[1] of a full scale tokamak chosen as the Reference CTR are shown in Figure 1. The torus structure is divided into six sectors to facilitate construction and maintenance. Four of these are shown assembled and positioned around the poloidal magnet core. In the left foreground a fifth is assembled and ready to be moved into position. In the right foreground partially assembled magnet coils for the sixth are illustrated. Note the massive steel reinforcing rings that contain the superconducting coils in their inner flanges. Figure 2 is a schematic of the approximately one meter thick blanket region which surrounds the toroidal plasma. It consists of a set of 60 segments, each of which consists of a 2.5 mm thick niobium shell. Two such segments are illustrated in the lower right foreground of Figure 1. These segments contain a long, slender, central "island" of graphite surrounded by a lithium-filled duct. Lithium coolant would be circulated at about 30 cm/sec around this closed loop by an electromagnetic pump at one end. Tritium is bred by neutron absorption in the lithium. A typical breeding ratio is 1.3, giving a doubling time of about a month. (Addition of neutron absorbers can easily reduce this ratio when excess tritium is no longer needed). A set of tubes installed in the lithium blanket utilizes the heat generated in the blanket to boil potassium. One set of the ring-shaped manifolds would carry the liquid potassium feed to the blanket from pipes in a duct beneath the reactor floor, and the other set carries potassium vapor to vapor pipes that extend around under the reactor and out to a potassium vapor turbine in the adjacent turbine hall (see Figures 3 and 4).

A magnet shield about 1 m thick attenuates radiation leaking from the blanket region into the liquid helium-cooled superconducting magnets so that the radiation energy deposited in them would be about 1 kW(t), and hence the power required for the liquid helium refrigeration system can be held to about 2 MW(e).

Six neutral beam injectors for plasma heating and refueling are mounted near the top of each sextant so that fuel injection takes place through the parting planes between sextants.

Figure 4 shows the reactor installation in a 60 m diameter evacuated shielded cell. The vacuum pumps, helium refrigeration system, and tritium recovery and handling system are located in rooms beneath the reactor.

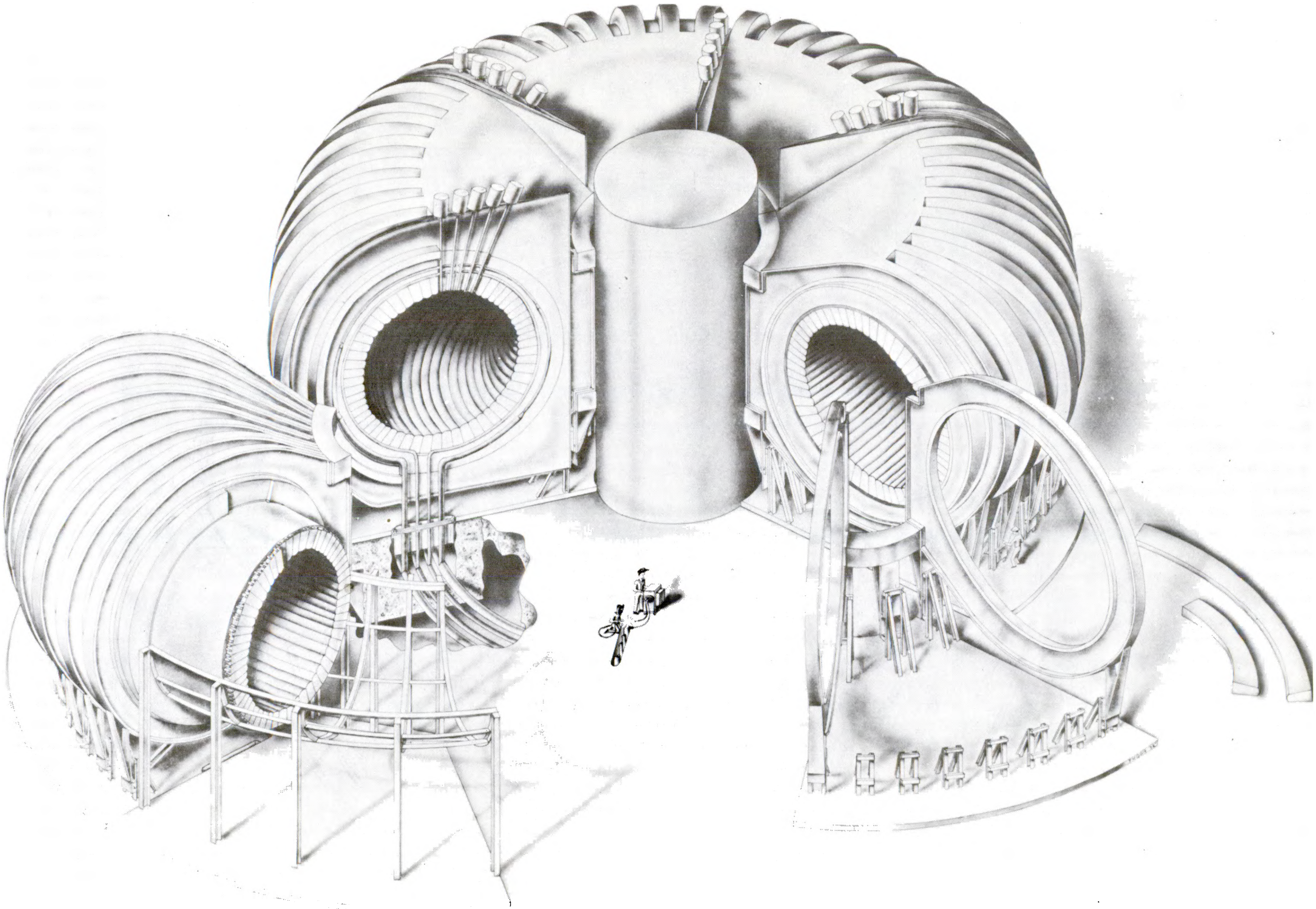

Fig. 1 Isometric view of an ORNL full-scale Tokamak reactor in the latter stages of assembly. Two partially assembled sectors are shown in the foreground.

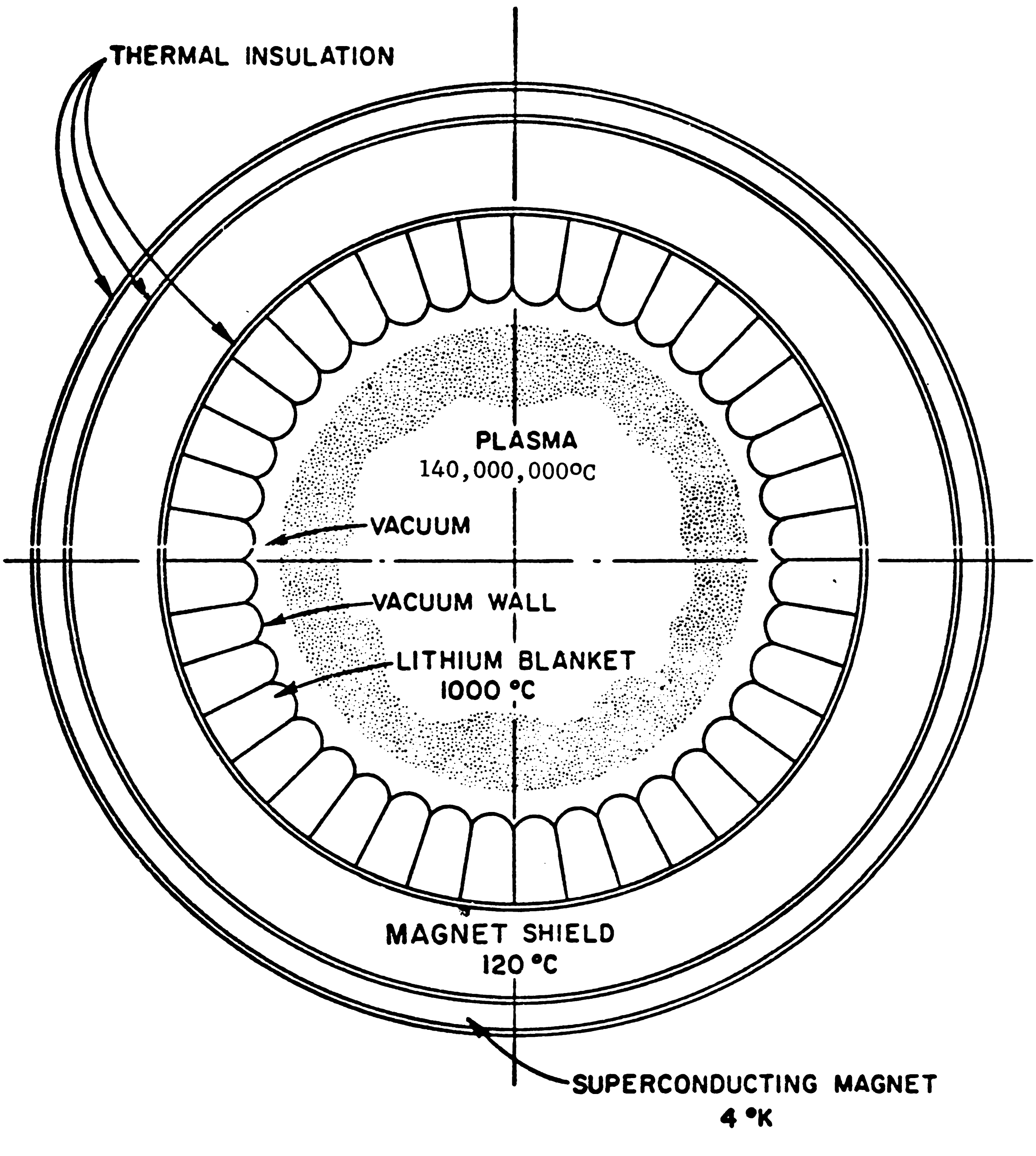

Figure 2. Cross section of the toroidal core of the Reference CTR.

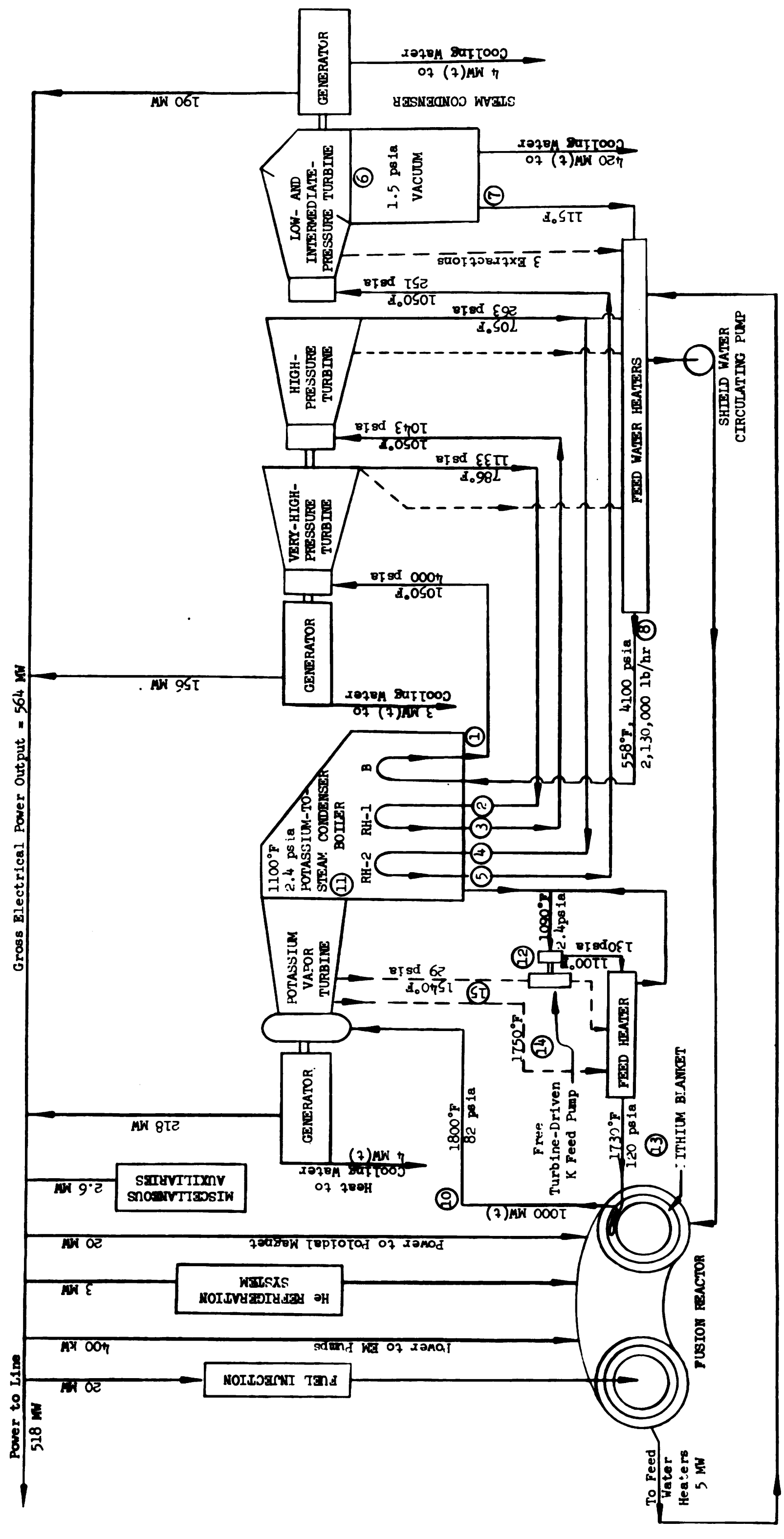

Fig. 3 Flow diagram for the potassium-steam binary-vapor-cycle power conversion system.

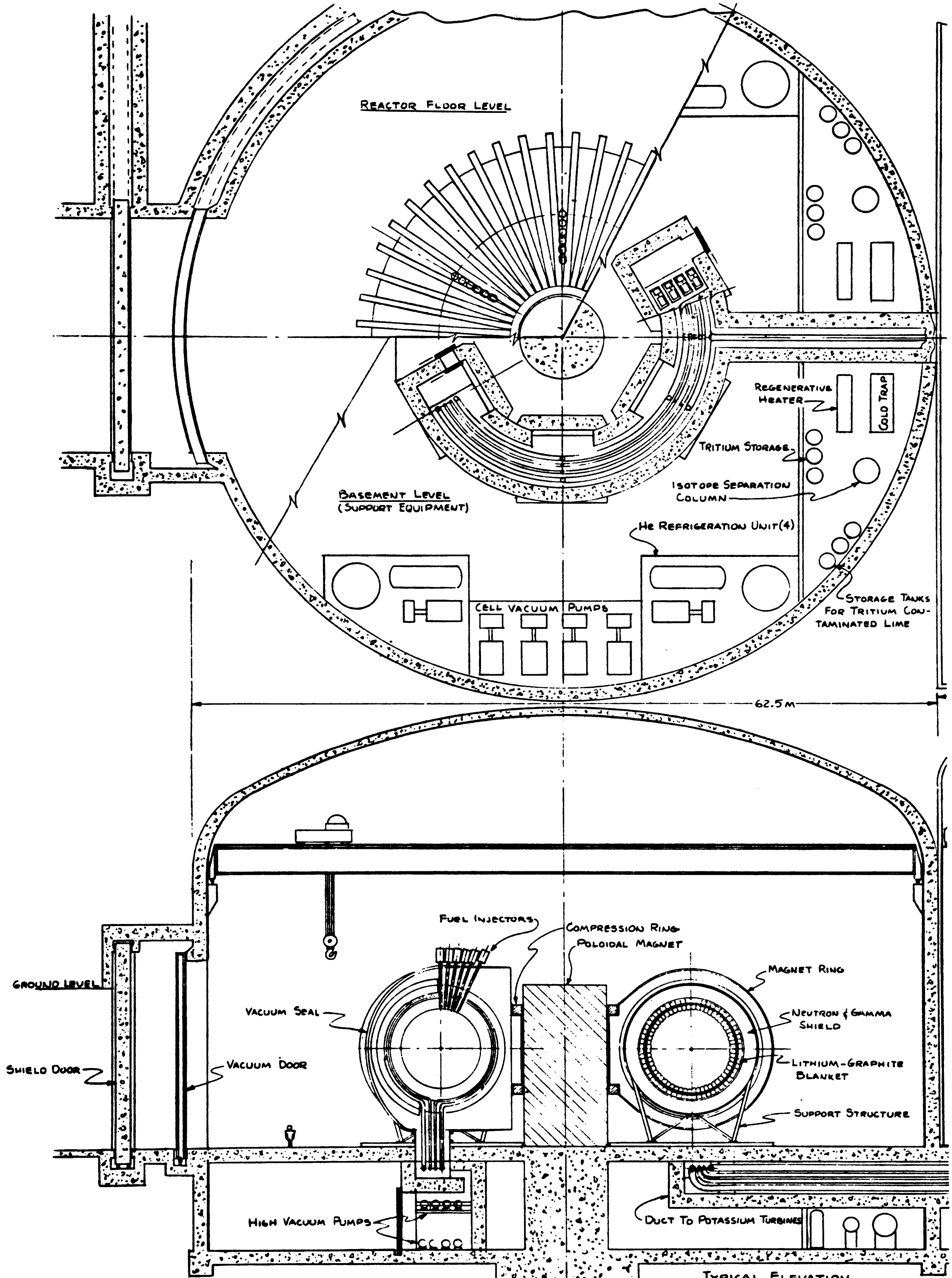

Figure 4. Section through the power plant building.

V. PROBABLE ENVIRONMENTAL CHARACTERISTICS DURING NORMAL OPERATION

A. Radioactive Effluents

The only possibility of radioactivity release during routine plant operation is tritium leakage. On the basis of preliminary design considerations, it appears that tritium leakage can be maintained at very low values. In assessing the fusion reactor tritium leakage rate, a number of key points should be noted. The first is that the thermally hot niobium core section, which contains the tritium, would be surrounded by a cold wall with the intervening space evacuated. Tritium drawn from this space would be recycled. As a consequence, the problem becomes one of tritium leakage through the heat exchanger, because the diffusion rate of tritium through the cold walls would give a trivial loss rate to the atmosphere. The second point is that any tritium diffusing through the walls of the potassium boiler into the potassium system, and thence through the potassium condenser-steam boiler, would react with the water to form HTO. Its recovery would then be difficult because an isotope separation process would be required. Use of a very tight system for the steam power plant would keep HTO leakage to the biosphere at a negligible level, or the rate of tritium diffusion into the steam could be held to a very low level by using tungsten or oxide diffusion barriers. The latter choice appears more attractive and was chosen as the basis for the Reference CTR. It was also evident that both the lithium and potassium systems must be made highly leak-tight to avoid the loss of tritium dissolved in the fluid that might leak from these systems. This latter choice does not appear to present a problem because liquid metal systems are commonly designed to be sufficiently leak-tight that normal liquid metal leakage losses are essentially zero. Stainless steel systems for potassium vapor cycles have been operated at ORNL, for example, for periods of

10,000 hr with no sign of leakage.[2] (The limit of detection was about 0.001% of the system volume in 1000 hr. This is about 10^{-6}%/hr or 2.4 x 10^{-5}%/day.) Examination of the Reference CTR design and present experience with operating liquid metal systems indicates that the leakage can be kept to 0.0001%/day.

If ventilating air discharged from the reactor building is directed up through a 200 ft. stack, the maximum tritium concentration downwind at ground level would produce a dose rate of about 1 mrem/yr. This is less than 1% of the average dose to the population from natural radioactivity of 110 mrem/yr.

B. Long-Lived Radioactive Wastes

Fusion reactors will produce nonvolatile, long-lived radioactive wastes in modest quantities. The primary source of radioactive waste from the Reference CTR will be the activated structural material of the blanket, which will have a finite useful lifetime within the reactor owing to radiation damage. Table II shows the principal long-lived activities of the Reference CTR blanket structure (niobium or vanadium). This table gives the annual rate at which the activity is generated, normalized to one megawatt of reactor thermal power, the accumulated activity resulting from 1000 years of continuous generation,* and the biological hazard potential associated with this amount of accumulated activity. Note that in Table II the Maximum Permissible Concentration (MPC) in water is used, which seems more appropriate than the MPC in air in the context of underground disposal. For niobium as the structural material the biological hazard potential associated with the accumulated Reference CTR radioactive waste is significant and would have to be treated accordingly.

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* In 1000 years the accumulated hazard potential will approach its steady-state value.

Table II. Long-Lived Activities in the Blanket Structure of the Reference CTR

Nuclide	Mean Life (yrs.)	Activity Generation Rate (curies/MW(t)-yr)	Accumulated Activity at 1000 yrs. (curies/MW(t))	Maximum Permissible Concentration* in Water (μcuries/cm^3)	Biological Hazard Potential Activity at 1000 yrs ÷ MPC (km^3 of water/MW(t))
			Reference CTR with Niobium		
^{93m}Nb	19.6	8,800	173,000	4×10^{-4}	0.4
^{94}Nb	2.9×10^{4}	2.9	2,900	3×10^{-6}	1.0
			Reference CTR with Vanadium		
Long-Lived Activities Due to Activation of Niobium Impurity in Vanadium					~ 0.00014 - 0.0014

* Abbreviated MPC in the text.

The use of vanadium as the blanket structural material dramatically reduces the problems associated with radioactive waste disposal. Vanadium exhibits no known long-lived activity as a result of activation; therefore the long-lived activities result only from the activation of impurities and alloying additions within the vanadium. Niobium is typical of such an impurity and might be present in vanadium at an atomic concentration somewhere between 100 to 1000 ppm (parts per million). Assuming this concentration range, the biological hazard potential associated with the activated vanadium structure would be three to four orders of magnitude lower than that associated with the niobium structure (see Table II). The same arguments would also be valid for several promising vanadium alloys, i.e., those containing titanium and chromium.

The activated structure of a fusion reactor could be reused after reprocessing if necessitated by a scarcity of niobium resources. In view of the rapidly growing use of automation in industry, the remote handling and recycling of radioactive material may prove practicable and economical, thus virtually eliminating the need for long-term radioactive waste management in a fusion power economy, e.g., recycle of the blanket structure after allowing time for radioactive decay.

C. Waste Heat Rejection

The DT fuel cycle requires use of a thermal power conversion system. The efficiencies of such systems are determined in large part by the maximum temperature of the heat transfer fluid, which is determined by the maximum temperature of the core structure. The Reference CTR utilizes a niobium structure which appears capable of operation at 1000°C. This may allow use of a potassium topping cycle in addition to the main steam generators, the combination of which appears to provide overall plant efficiencies greater than 50%.

The use of cooling water versus wet or dry cooling towers has not been considered in detail for fusion reactors because the choice of heat rejection mode is such a sensitive function of plant site considerations. Obviously the high operating temperatures of the Reference CTR would allow increased flexibility in system optimization using cooling towers over systems operating at lower temperatures. Because of the potential of urban siting and the high peak cycle temperature, heat can be rejected from fusion power plants at 100 - 200°C without seriously reducing plant thermal efficiency. This heat energy may then be used for building heating and cooling and/or industrial processes, and it would thereby not represent a waste.

D. Land Despoilment

There are three aspects to fusion power related to land despoilment. The first is the direct land use by the power plant itself, which includes buildings, switchyards, transformer yards, transmission lines, cooling equipment, etc. To a significant extent fusion reactors would be similar to fission reactors in this regard, and fusion fuel storage space requirements will be negligible.

A second aspect of land despoilment is associated with the procurement of the fuel and construction materials. DT fusion power plants would consume deuterium and lithium as fuels. Deuterium is obtained from water which is available to all countries. Its extraction results in no despoilment but rather provides useful quantities of commercial grade hydrogen and oxygen and modest quantities of purified water.

Lithium is obtainable from surface and underground brines (the least expensive extraction process) and from the oceans (a more expensive process but still relatively insignificant in cost). The land despoilment associated with the extraction of lithium and the metals incorporated in the structure of the Reference CTR are shown in Table III, which shows that the residues of lead and copper are of greatest concern.

Table III - Yield of Required Metals from Their Ores

	Requirement for 10^7 MWE - metric megatons	Approximate average yield of metal from crude ore - percent	Ore Requirement for 10^7 MWe - metric megatons
Nb	7	2	350
Be	.6	2	30
Cr	11	5	220
Ni	5	$\sim$ 1	500
Li	5	$\sim$ 5	100
Cu	40	.9	4400
Pb	107	1.5	7100
Al	10	10	100
V	4	5	80
Mo	6	2	300
Sn	.8	10	8
Fe	170	45	380
Zr	.07	$\sim$ 5	--
		Total	13,600

The third aspect of land despoilment is associated with the projected flexibility of fusion reactor siting. If urban siting is indeed acceptable, then the large land areas usually required for power transmission from rural to urban areas would be significantly reduced.

E. Transportation

To start up a fusion power plant an initial fuel charge of deuterium and tritium will be needed. Thereafter a continuous supply of deuterium and lithium will be required at the rate of about a kilogram per day. Tritium shipment will be necessary only to supply the initial charges to start new power plants, i.e., possibly about 10 kg quantities from each operating plant every few years on the average, depending upon the rate of growth of the fusion power industry.

The blanket structure of a fusion plant will become radioactive and will have a lifetime of the order of 10-20 years. When the blanket structure is replaced, the used activated unit will have to be shipped from the power plant to a site wherein it would be either stored or reprocessed. The structure itself will be nonvolatile and consequently its hazard potential should be relatively low. It will not require a large amount of shielding during shipment nor would it present a difficult cooling problem.

VI. EFFECTS ON NON-RENEWABLE RESOURCES

A preliminary survey has been made of U.S. and world resources of the various materials needed for fusion reactor construction. The results are shown in Table IV, where the approximate quantities of materials needed to fabricate a single 1000 MWe reactor are tabulated. These figures are for current reactor designs. The development of other materials for the various components, i.e., blanket structure and superconducting magnet, is clearly possible and would alter these requirements accordingly.

To emphasize maximum resource requirements, the largest quantity of a given material required by any of the several reactor designs available (see the Appendix) has been used. For instance, a pulsed theta-pinch reactor would use more copper and less superconducting material than would a tokamak reactor. The larger needs for both materials are included in the Table. Clearly no one reactor design would use all of the material listed and this approach thereby overestimates the quantities of material needed.

In the extreme of a fully developed world fusion power economy, ten terawatts (10^7 MWe) of electric power might be generated by fusion reactors. Therefore the third column of the Table shows the mass of materials in metric megatons needed to construct and operate ten thousand 1,000 MWe fusion reactors. Plant replacement at about 5% per year would be required at a later time but is not considered here.

Also presented in the Table are estimates of the total production of the various materials projected to be required in the year 2000, along with quantities of known reserves at present prices and estimates of resources available at increased prices.

A great many evaluations of U.S. and world raw materials resources have been made but these are usually a matter of expert opinion. Consequently, values such as "known resources at current costs" vary widely from one source to another. Often the estimated quantities of a raw material available at increased costs are based on industrial projections. But when adequate reserves of a given ore are available to supply the demand for 20-30 years, exploration for additional reserves is usually curtailed with the result that total projected reserves can be underestimated to a significant degree. Most of the values quoted are from the 1970 edition of "Mineral Facts and Problems". In addition to estimating materials needs, some comments concerning environmental problems associated with a particular raw material are included in the Table.

It is apparent that the production of 10^7 MWe of fusion power would give rise to some resource use conflicts which will have to be resolved. For example the requirements for niobium could just be met by known reserves. However, additional reserves may be found or other superconducting materials developed. In addition to niobium, other possible resource conflicts exist in the projected usage of beryllium, titanium, helium, lead, vanadium, and molybdenum, and some of these problems will also be common to other power generating concepts.

Table IV CTR Resource Utilization

Material	Approx. Mass In Metric Tons Per 1000 MWe Reactor	For Reactor[†]	Mass In Metric Megatons For 10^7 MWe	Total Estimated Production in Year 2000 In Metric Megatons		Known Resources Present Prices Metric Megatons		Resources At Increased Prices In Metric Megatons		Comments
				U.S.	WORLD	U.S.	WORLD	U.S.	WORLD	
Nb	~ 400 structural, ~ 130+180 in NbTi and Nb_3Sn	4,1,2	7	.009	.020	.07	6	.14	NA	Present mining operations are relatively nonpolluting; greatly increased demand might necessitate strip mining to obtain low grade deposits
Li	~ 900	1	9	.01	.016	5	6-8	9	250,000	100 metric megatons probable land resources; extraction from sea water possible, 1.5 lbs. of Li/100,000 gal. of sea water
Be	~ 60	2	.6	.002	.003	.026	.38	.072	1	Little information on world Be resources available, Be presents health hazards in mining and handling
Cr	~ 1100 in SS	5	11	1	4.3	0	700	1.6	NA	Resources almost entirely outside of U.S.
Ni	~ 500 in SS	5	5	.5	1.3	.2	68	5.0	NA	World estimates are based on fragmentary information and are possibly low
Ti	~ 400 structural, ~ 80 in NbTi	1,1	5	2.3	6.9	.15	6.4	.4	30	Significant quantities of mud and slimes result from dredging Ti minerals from sand deposits
He	~ 350	3	4	.012	.015	1.2*	1.2*	5*	29,000+	*In the ground +Extracted from atmosphere at up to 30 times current prices
Cu	~ 2900 coil, ~ 1100 in NbTi	3,1	40	6-12	35	77	280	180	1,100	Considerable secondary recovery possible; significant land-use conflict will result from an expanded copper industry
Graphite	~ 2200	1	22	.1	1.4	.5	> 100	NA	NA	Very rough estimates of world reserves available
Pb	~ 10,700	1	107	3	7.3	32	86	45	95	Considerable secondary recovery possible
Al	~ 570 structural, ~ 390 in Nb_3Sn	3,2	10	30	75	12	2200	275	NA	Large land areas and great amounts of energy needed to mine and process Al
V	~ 400	4	4	.03	.06	.1	9	3	NA	
Mo	~ 400 structural, ~ 200 in SS	4,5	6	.08	.24	2.9	5	NA	> 10	Substantial resources of sub-marginal-grade ore throughout the U.S. and world
K	~ 20	1	.2	11	56	120	> 10,000	770	Virtually unlimited	
Sn	~ 80 in Nb_3Sn	2	.8	.12	.41	.006	4	.042	7	Some secondary recovery possible
F	~ 500 in flibe	2	5	2.2	7.5	4.9	35	NA	NA	Increased price would stimulate expanded exploration for fluorspar
Fe	~ 12,600 steel,	1,5	170	180	800	2000	90,000	20,000	> 300,000	Potential reserves are vast

[†]Reactor code:
1) ORNL Tokamak, 2) PPPL Tokamak, 3) LASL Theta-Pinch, 4)LLL DT Mirror, 5) LLL D^3He Mirror

VII. FUSION POWER ECONOMICS

At the present stage of fusion development many physical and technical uncertainties clearly exist. Fusion power costs are therefore impossible to accurately predict. Nevertheless, cost estimates are of value because they indicate a general order of magnitude, and they help to identify particularly sensitive components for which further cost-reducing development could have a major impact. In this section the costs for the plant and the fuel will be considered.

The safety and environmental characteristics of fusion reactors will very likely make them acceptable for urban siting. The power costs of urban fusion power plants would be significantly reduced by savings in transmission costs as well as possible savings associated with the sale of waste heat for building heating and/or industrial processing.

To estimate fusion power capital costs, the reactor designs developed for the various concepts were analyzed to determine the approximate amounts of the various materials used in their construction. Current prices for the required quantities of these materials in finished form were then used to estimate component costs. The prices for the superconductor material correspond to present large order levels. The unit winding costs and structure costs have been scaled somewhat less than the square of the magnetic field.

Amongst the auxiliaries for magnetic confinement systems, the greatest uncertainties are associated with injection systems and the theta-pinch reactor energy switchgear. Injector development has not yet progressed to fabrication of reactor-sized units and factors of two or so cost uncertainties are felt to exist. The switchgear estimate is based on an estimate of $2 to 7 million for similar equipment designed for two synchrotrons requiring 1-2 second switching of 100 to 1000 megajoules. Theta-pinch reactors would require 10 msec switching of 200 gigajoules. Development and fabrication costs of $100-200 million are considered probable.

A comprehensive projection for a fully developed superconductor industry has shown that it appears possible to obtain cost reductions for finished magnets of factors of four to five over present levels. In such a well developed situation the cost of the conductor moves from being the largest single cost to being secondary, and structure costs become dominant. Winding costs are expected to decrease from the present level of $33 per kg of conductor to near $10 per kg.

The results show that prototype reactor costs might be about $500/kwe for the nuclear "island." Ultimate magnet costs would reduce mirror and tokamak reactor costs substantially. Superconductor in the theta-pinch reactor serves as an energy storage element separated from the plasma vessel, and it operates at low fields. It represents a small fraction of the system total cost and is little affected by the ultimate magnet cost patterns. Maturing of the fusion reactor industry should bring reductions associated with production quantity manufacturing and the removal of design uncertainties, further reducing costs. Final projected fusion reactor capital costs then correspond roughly to the level projected for other types of plants in the year 2000. Because of the uncertainties, it is believed that these exercises in cost estimation serve only to suggest that fusion power capital costs could be competitive with other energy sources. To conclude any capital cost advantage at this stage of development would clearly be premature.

Fusion fuel cycle costs are determined by the costs of deuterium and lithium which are shown in Table V. Fuel transportation costs will be negligible because of the small quantities of materials involved and because handling techniques for gases and liquid metals are already well developed and inexpensive.

Table V

Fusion Fuel Cycle Cost Based Upon Current Prices

Element	Cost Per Gram	Cost Per Kilowatt-Hour
Deuterium	\$0.20	6×10^{-3} mills
Lithium	0.02	10^{-3} mills
Total Cycle Cost		7×10^{-3} mills

VIII. ACCIDENT HAZARDS

Any reasonable appraisal of accident hazards requires a detailed examination of a specific design because many potential problems are in large measure dependent upon specifics of the system. As mentioned previously, only one fusion reactor design has been iterated through a number of steps in an attempt to maximize safety and minimize accident potential. That Reference CTR served as a basis for the following analysis.

A first step in appraising the possibilities and consequences of a fusion reactor accident is to determine the maximum energy stored in the system in nuclear and chemical forms and in the form of high pressure steam or gas. Table VI lists the principal hazards sources in the Reference CTR and shows that the largest potential source of accidental energy release is associated with the lithium in the blanket. In the design considered here, no lithium is situated near any water, and it would require the rupture of three successive envelopes for the lithium to react with air. Further, the lithium inventory in the Reference CTR is divided into many separate segments, thus significantly limiting the energy release from a single leak. In addition,the lithium region is well protected. If, for example, an airplane were to crash into the containment shell and rupture it, the lithium region would still be well protected not only by the magnet shield but also by the massive structure of the steel reinforcing rings carrying the superconducting magnet coils.

A second concern is the possibility of the abrupt release of a substantial amount of energy via nuclear reactions. In this case the only fuel that could possibly react would be that actually inside the plasma region, i.e., about a gram. If all of the fusion energy obtainable from this charge were to be dumped into the blanket in a few seconds, the average temperature of the lithium would rise about 30°C -- a minor perturbation. A dump of such magnitude appears impossible from what is known today about plasma behavior. Further, the kinetic energy of the unburned plasma is a factor of about 1000 lower than the total available fusion energy so that a full plasma loss to the walls would have a much smaller effect. This low total available energy in the fuel charge and the low probability of liberating more than a small fraction of it in a fault situation are major factors in the inherent safety of fusion reactors.

Table VI. Energy Release Potential of Components of a Reference CTR Producing 1000 MW(e)

	Energy in Megajoules	Equivalent Gallons of Fuel Oil
Plasma, complete fusion	6.9×10^4	~ 430
Magnet	2.4×10^5	~ 1500
Lithium + water + air	6.4×10^7	$\sim 4 \times 10^5$
Potassium + water + air	6.4×10^5	~ 4000
Primary vacuum vessel	640	~ 4
Secondary vacuum vessel	1.6×10^4	~ 100

From experience to date, a localized plasma dump onto the adjacent wall appears very unlikely. Clearly the probability of such an instability occurring must be made extremely low in a practical system. This question can be specifically studied in the larger, more energetic plasma systems to be fabricated later this decade. In any event, local wall burnout due to an inadvertent concentration of plasma would at worst cause a lithium leak into the plasma but would not cause an accident affecting the public.

A third possible failure mode is associated with a magnet failure. There are two faults of concern. In a high current density coil a transition from super to normal conduction could progress over the total conductor volume in a period short compared to the overheat time in the conductor. The rate of stored energy dissipation could be handled with the insertion of an external load resistor. If, somehow, the resistor was not cut-in, the coil assembly temperature would rise to near that of the room while liquid helium evaporated and was vented with no damage to the coil. There is satisfactory experience with this type of quench fault. At low current densities in a coil the quench might not spread rapidly, and a load resistor could be inserted automatically in a time interval of the order of a minute to drive the current down, thus preventing local heating which could damage the conductor. A third possible fault mode is the breaking of a conductor in the coil. This would lead to further damage of the coil by arcing and probably a quench. In the latter two cases proper design can insure that the damage would be limited to the coil itself.

Study of the afterheat problem in connection with the Reference CTR indicates that it is possible to evolve a design that is virtually unaffected by a loss of coolant accident. A basic reason for this is given by Table VII which shows the average afterheat power density at shutdown in watts per cubic centimeter for the Reference CTR and the rate of temperature rise after a cooling system failure. An analysis of the consequences of a complete loss of coolant in both the blanket and the shield region of the Reference CTR indicates that all of the afterheat could be removed by thermal radiation and conduction with a temperature rise of no more than about 100°C in the high temperature zone during the first week after the outage, assuming that no action whatsoever were taken by the plant operating personnel. This refers to a blanket structure built of niobium. If stainless steel were employed, the afterheat would be reduced by a factor of about two relative to that of niobium, or, if vanadium were employed, the afterheat immediately following shutdown would be reduced by a factor of about four. Further, in the vanadium case the afterheat would fall off much more rapidly than with the niobium.

Table VII. Afterheat Power Density Associated with the Niobium Structure of the Reference CTR

	Reference CTR
Average* afterheat power density at shutdown	0.15 watts/cm^3
Rate of temperature rise if uncooled immediately after shutdown	0.06°C/sec

The probability of a lithium leak will be low because the lithium blanket can be designed so that the lithium pressure will differ from that in the plasma region by only about 1/10 of an atmosphere, and hence both the pressure stresses and the driving force for a leak will be small. Further, the blanket has been designed to keep all of the thermal stresses well within the elastic range both during normal operation

*Average over the first wall

and in the course of any of the transients that have been envisioned, and this would minimize the probability of a crack induced by thermal cycling strain.

An obvious cause for concern is a leak of lithium into the plasma region. If this occurs, even a small amount of lithium will quench the plasma because of the increased loss via bremsstrahlung radiation from the lithium atoms and/or conduction cooling.

If a lithium leak occurs in the region between the blanket and the shield, the multilayer stainless steel foil reflective insulation should prevent the lithium from reaching the titanium shield tank. If it does reach the tank, the lithium will simply solidify because the tank temperature would be below the freezing point of the lithium.

The initial design of the Reference CTR envisioned the use of a magnet shield that consisted primarily of water and lead. While the probability of lithium coming in contact with this shield water seemed exceedingly low, it was decided that with relatively little increased cost the water could be replaced with graphite or a metal oxide such as alumina or magnesia. The presence of water in the shield can be avoided completely by employing helium rather than water as the shield coolant. The energy deposition in the magnet shield as a consequence of nuclear and thermal radiation and thermal conduction will represent less than 1% of the total reactor output. Analysis indicates that the shield can be cooled easily with helium at about ten atmospheres, and thus the designer can eliminate the possibility of a substantial energy release from a lithium-water reaction.

The consequences of a lithium leak are greatly reduced by the fact that the lithium blanket is segmented into many independent elements. Any lithium leak will be quickly detected as a consequence of its effects on the plasma or the vacuum system.

The above discussion has been concerned with single point failures. It should be noted that the design is such that even a double failure would not lead to any serious difficulties. If, for example, there were a lithium leak into the region between the blanket and the shield, and a leak from this region out into the reactor cell, there would still be no serious reaction because a vacuum is maintained in the cell. Again, a leak could be readily detected. Note, too, that there is no apparent way in which a leak from the lithium system could induce a secondary leak through the walls of the high vacuum region into the reactor cell.

If a leak develops in the potassium condenser-steam generator of the Reference CTR, the steam jetting into the potassium condenser would react with the potassium to form potassium oxide and hydrogen. Inasmuch as the potassium condenser will have a large vapor volume space available, there would be adequate space to accommodate the hydrogen gas, and no explosion or even large increase in pressure would occur. (This situation differs from that in a liquid metal-heated boiler in which there is little or no free volume on the liquid metal side into which the hydrogen from the reaction can expand). As the hydrogen builds up in the condenser, it would block the flow of potassium vapor into the condenser and produce a back pressure which would provide an obvious signal to an operator or which could be used to trigger a warning signal. If a large steam leak were to develop as a consequence of a burst type of failure, the inherent nature of the inlet orificing of the reentry tube boiler is such that vapor rather than water would be injected into the potassium region, and as a consequence the rate of injection would be relatively low -- a few lb/sec per ruptured tube. In the Reference CTR this would lead to an increase in the condenser at a rate of about 1 psi/sec. Thus, if the potassium condenser were designed to take an internal pressure of 60 psi, and if the flow of either steam or potassium into the condenser could be stopped within a minute after the first evidence of the rupture, the damage would be limited to the broken tube. For the extreme case of an abrupt, complete rupture of a steam generator tube, the potassium condenser pressure would rise faster. Again this should be easily and reliably detectable and could be the basis for closing valves in the feed water supply line. If this were done in an additional 10 seconds, the inventory of superheated water in the boiler design proposed would be exhausted in another 15 seconds, and the peak pressure in the potassium condenser would be held to about 15 psig (30 psia). To protect against the contingency that no action might be taken, a rupture disc could be provided to blow off at perhaps 40 psia.

A leak from the potassium boiler into a segment of the lithium blanket will cause the liquid level in the header tank for that segment to rise and in extreme cases overflow into a dump tank. This will lead to a forced shut-down, but no other ill effects have been envisioned.

The inventory of volatile radioactive material is probably the most important factor to be considered in appraising the requirements for engineered safety features for any type of nuclear power plant. For a fusion reactor this means that the tritium inventory, particularly the active inventory in the liquid metal system, is the most vital consideration because it will be the only volatile activity in a fusion reactor. One of the systems proposed appears to be capable of holding the tritium concentration in the lithium to roughly 1-10 ppm irrespective of the type of fusion reactor, the total lithium inventory or the tritium generation rate. Thus, the tritium inventory in the lithium system would be primarily a function of the total lithium inventory.

Practically all of the tritium outside of the liquid metal systems will be contained in components in the tritium equipment room. These components will be at or close to room temperature, and the atmosphere in the room would be carefully controlled and monitored so that, if any tritium leakage occurs in that room, it would be well contained.

The only substantial inventory of radioactive material other than tritium will be that in the blanket structure, and the bulk of this activity will be in the region close to the first wall. Table VIII shows the estimated quantities of the principal radioactive inventories (in curies per kilowatt of reactor thermal power) for the Reference CTR, using niobium and vanadium as alternate structural materials. The biological hazard potential is provided for each item; it is defined here as the activity divided by the maximum permissible airborne concentration (MPC) as specified in the radiation protection standards for continuous exposure to individuals living in the vicinity of controlled areas.

Table VIII. Principal Radioactive Inventories of the Reference CTR

Inventory	Activity (curies per kW of thermal power)	Maximum Permissible Airborne Concentration (μ curies/cm^3)	Biological Hazard Potential Activity $\div$ MPC (km^3 of air/kW thermal)
	Reference CTR 10-year Operation		
^3H (combined in H_2O)	12[a]	2×10^{-7}	0.06
	Niobium as the Blanket Structure		
^{95}Nb	155	3×10^{-9}	52
Total Niobium Structure	714	c	240
	Vanadium as the Blanket Structure		
^{48}Sc	4.20	5×10^{-9}	0.84
Total Vanadium Structure	55.1	c	0.86[b]

a The specific activity of tritium is approximately 10^4 curies per gram.

b Impurities within the vanadium might increase this number by a factor of two.

c MPC's for each individual isotope were estimated to get the composite Biological Hazard Potential.

IX. RELIABILITY AND VULNERABILITY

As with any infant technology,when fusion reactors first become commercially available, their reliability will not be as high as that of the more mature power plant types. Areas where there will be relatively little experience include large refractory metal structures, superconducting magnets at very high fields, potassium-steam turbines, generators, and boilers, and to a lesser extent large high temperature vacuum systems. Because of this unfamiliarity, a certain amount of redundancy will be required which can be eliminated as the technologies develop. The reliability of the steam system and other standard elements of a fusion power plant should of course match the reliability of similar equipment used in other plants.

Fusion power plants, like other systems, will be vulnerable to both internal and external hazards. Clearly care in design can eliminate many potential problems. Inherently a number of potential problems will represent minimal hazards. Failure of a magnet would cause the plasma to strike the wall, extinguishing the reaction with relatively minor effects on the wall. Failure of an injector would reduce the fuel supply causing the plasma to slowly diffuse away. Failure of the on-site reprocessing system would result in an impure fuel replenishment which would markedly reduce the plasma temperature and thereby the reaction rate. At a later date when improved fusion reactor designs are developed, the matter of internal vulnerability can be considered in greater detail and the results of such analyses factored into plant design.

In terms of external influences, such catastrophies as earthquakes, tornadoes, hurricanes, lightning, aircraft crashes, etc. must be considered. Rather than attempting to consider these possibilities in any detail at this time, reference is made to the previous discussion wherein the potential sources of energy release and the radioactivity inventories were estimated. From these considerations and fission reactor experience to date, the hazards of a fusion power plant appear to be readily manageable.

X. MATTERS RELATED TO NATIONAL SECURITY

A variety of matters associated with fusion power plants could have an impact on the national security. First, there is the question of fuel availability. The U.S. has ample access to water for a deuterium supply and also a lithium supply,if this more expensive source is needed. More important, the U.S. has known lithium reserves sufficient for hundreds of years of fusion power before even the ocean source need be considered. This independence of foreign fuel supply is clearly one advantage of fusion power.

Next, there is the question of dependence on foreign supplies of structural materials. This matter was discussed above in Section F "Effects on Non-Renewable Resources." The U.S. is dependent upon foreign sources for a variety of basic metals at the present time, and this dependence will probably increase in the future. Again the flexibility of fusion reactor design permits a range of materials choices, and, if the foreign dependency question becomes a more significant matter in the future, it could be factored into fusion reactor design.

Another matter of concern is associated with the possible diversion of weapons type materials from power plants. Fissionable materials are not present in pure fusion power plants. Conceivably attempts could be made to breed fissionable materials using the neutron fluxes in fusion reactors, but designed-in limited access to the reactor core and normal plant security should prevent such actions. The tritium utilized in a fusion power plant would be generated, circulated, and burned within the plant. The only shipment of tritium would be for the initial startup of new plants so its availability external to a power plant would be minimal.

The development of laser-fusion for civilian power could have implications for national security because this achievement would carry with it the following additional laser-fusion applications: neutron and x-ray photography for weapons design studies, weapons simulation, and possibly the direct production of pure-fusion weapons.

Fusion power could be of direct use in military applications, such as for ship or space propulsion, which would thereby affect the national security. The development of the industrial base associated with a fusion economy could be of value in terms of an improved national technological base and in terms of an improved balance of payments resulting from foreign sales of fusion reactors.

XI. SUMMARY

For the purposes of this study the ultimate potential of fusion power has been appraised by considering a set of reference designs for full scale fusion reactors based upon the deuterium-tritium (DT) fuel cycle. One design -- referred to as the Reference Controlled Thermonuclear Reactor or Reference CTR -- was analyzed specifically.

Deuterium for the Reference CTR is obtained directly from sea water at low cost. Tritium is bred in a blanket surrounding the plasma region by neutron absorption in lithium. Typical breeding ratios are about 1.3, giving a doubling time of about a month. With neutron absorbers this ratio can be easily reduced when excess tritium is no longer needed.

During routine power plant operation, tritium is anticipated to be the only radioactive effluent, and it appears to be readily controllable. A tritium leakage rate to the atmosphere from the Reference CTR of 0.0001%/day (based on a system inventory of 6 kG of tritium) appears reasonable from a design standpoint. Assuming that this leakage is to be discharged from the reactor building through a 200 foot stack, the maximum concentration at ground level would be reduced to the point where it would give a maximum dose rate downwind of 1 mrem/yr, i.e., less than 1% of the average dose to the population from natural radioactivity.

The primary source of radioactive waste from a fusion reactor will be the activated structural material of the blanket, which will have a finite useful lifetime within the reactor owing to radiation damage. Approximately 9000 Ci /MW yr. of long-lived radioactivity would be produced in the niobium structure of the Reference CTR. If vanadium were substituted for niobium, this activity would be reduced by a factor of 1000-10,000, depending upon the type and concentration of alloying material.

The DT fuel cycle requires use of a thermal power conversion system. The Reference CTR utilizes a niobium structure which appears capable of operation at 1000°C, which is sufficiently high to provide cycle efficiencies greater than 50%. Using stainless steel for the structure, temperatures are limited to about 500°C, which would give cycle efficiencies near 40%.

Urban siting of fusion power plants would allow rejected heat to be used for heating and cooling and industrial processing. The land despoilment associated with fusion plants appears to be similar to that for fission plants with the exception that urban siting would decrease the land requirements for power transmission.

To start up a fusion power plant, an initial fuel charge of deuterium and tritium will be needed. Thereafter, a continuous supply of deuterium and lithium will be required at the rate of about a kilogram per day. Further tritium shipment will be necessary only to supply the initial charges to start up new power plants. The blanket structure of a fusion plant will become radioactive and will have a finite lifetime of the order of 10-20 years. It will then have to be shipped for reprocessing or storage.

A projected worldwide production of 10^7 MWe from fusion and/or many other types of power will give rise to some resource use conflicts which will have to be resolved. Fusion requirements for niobium for magnets and structure could just be met by known reserves. However, additional reserves may be found or other superconducting magnet materials developed.

To estimate fusion power capital costs, reactor designs developed for the various concepts were analyzed to determine the approximate amounts of the various materials used in their construction. Current prices for the required quantities of these materials in finished form were then used to estimate component costs. These estimates yielded capital costs for the nuclear "island" of roughly the same order as projected for other types of plants in the year 2000. Because of major uncertainties, it is believed that these projections serve only to suggest that fusion power capital costs could be competitive with other energy sources.

Fusion power fuel costs are determined by the costs of deuterium and lithium, and they are essentially negligible--of the order of 0.007 mils/KWh. The safety and environmental characteristics of fusion reactors should make them potentially acceptable for urban siting, which would further reduce total fusion power costs by savings in transmission costs as well as possible savings associated with the sale of waste heat for building heating and cooling and/or industrial processing.

Fusion reactors appear very attractive when considered from the point of view of accident potential. A runaway reaction will not be possible in a fusion reactor both because of the inherent nature of plasmas and because of the low fuel inventory--about one gram--that would be resident in the core during operation.

Studies of the afterheat produced in the Reference CTR indicate that it is possible to evolve a design that is virtually unaffected by a loss-of-coolant accident. An analysis of the consequences of a complete loss of coolant in both the niobium blanket and the shield region of the Reference CTR indicates that all of the afterheat could be removed by thermal radiation and conduction with a temperature rise of no more than about 100°C in the high temperature zone during the first week after the outage, assuming no action whatsoever by automatic controls or the plant operating personnel. If stainless steel were employed for the blanket structure, the afterheat would be reduced by a factor of about two relative to that of niobium, or, if vanadium were employed, the afterheat immediately following shutdown would be reduced by a factor of about four.

The inventory of volatile radioactive material is probably the most important factor to be considered in appraising the requirements for engineered safeguards to protect against accident hazard. For a fusion reactor this means that the tritium inventory, particularly the active inventory in the liquid metal system, is the most vital consideration because it will be the only volatile activity present.

By holding the tritium concentration in the lithium to 1-10 ppm and isolating the lithium and tritium handling equipment in a single, well sealed and monitored compartment, this potential accident hazard can be kept very low.

The national security aspects of fusion power would be many-fold. The U.S. has plentiful deuterium and lithium resources and would therefore be independent of foreign sources. Fusion reactors do not utilize fissionable materials which may be subject to diversion for clandestine purposes. A mature fusion reactor industry would strengthen the country's technological base and foreign sales of fusion reactors would have a favorable effect on the balance of payments. Some reliance on foreign sources of materials such as nickel and chromium will be inherent to fusion as well as many other power sources.

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3. Nucleonics Week. Vol. 15, No. 44, pp. 26-31, November 3, 1969.

Appendix

Summaries of the Reference Fusion Reactor Designs

Brief descriptions of five reference fusion reactor designs are provided along with core schematics. Key characteristics of a number of these designs are then summarized in Tables A-1 and A-2.

No attempt has been made to define specialized terms. The reader who is unfamiliar with fusion terminology is therefore referred to other texts for further description and terminology.

PPPL Tokamak Fusion Reactor

The guiding principles on which this design was based were as follows:

1. The maximum magnetic field at the superconductor of the toroidal field coils were to be limited to 160 kilogauss. This field strength is somewhat higher than the present state-of-the-art level.
2. A divertor was to be included since the reactor was expected to operate essentially on a steady state basis.
3. Inexpensive, readily available materials and common techniques were to be utilized as much as possible.
4. The "safety factor", q, was chosen to be 2.0, a reasonable expected improvement over present experimental accomplishments.
5. The aspect ratio, A, was expected to exceed 3.0; the plasma ion density to approximate $10^{14}cm^{-3}$; the plasma temperature to be about 15 kev. The plasma composition was assumed to be equal parts of D and T. The reactor's electrical output was expected to be about 2000 MW(e) and a thermal cycle efficiency of 40% was assumed.

The resulting design (Figure A-1) in part reflects the difficulty in placing a divertor on a tokamak reactor. The divertor windings were placed outside the neutron shield in order for them to be either superconducting or cryogenically cooled. The divertor windings also provide the vertical magnetic field that is necessary for plasma equilibrium. Furthermore, the size scale had to be sufficient to permit adequate neutron shielding between the reacting plasma and the superconducting toroidal field coils thereby limiting the heat deposition in the coils by the neutrons to acceptable levels.

In keeping with Item 3 above, stainless steel is the chief construction material. The vacuum wall is constructed of stainless steel plates welded on a steel framework. Liquid lithium is not used as a coolant to avoid associated MHD problems, but lithium in the form of flibe is used for tritium breeding. The blanket is cooled by helium gas which in turn is used to drive helium gas turbines.

The use of stainless steel limits the blanket operating temperatures to about 550°C. Thence the design foregoes the advantages of higher thermal cycle efficiencies that can be achieved with higher operating temperatures. However, the use of higher temperatures would require the use of a refractory metal, such as niobium, which is not in common use today.

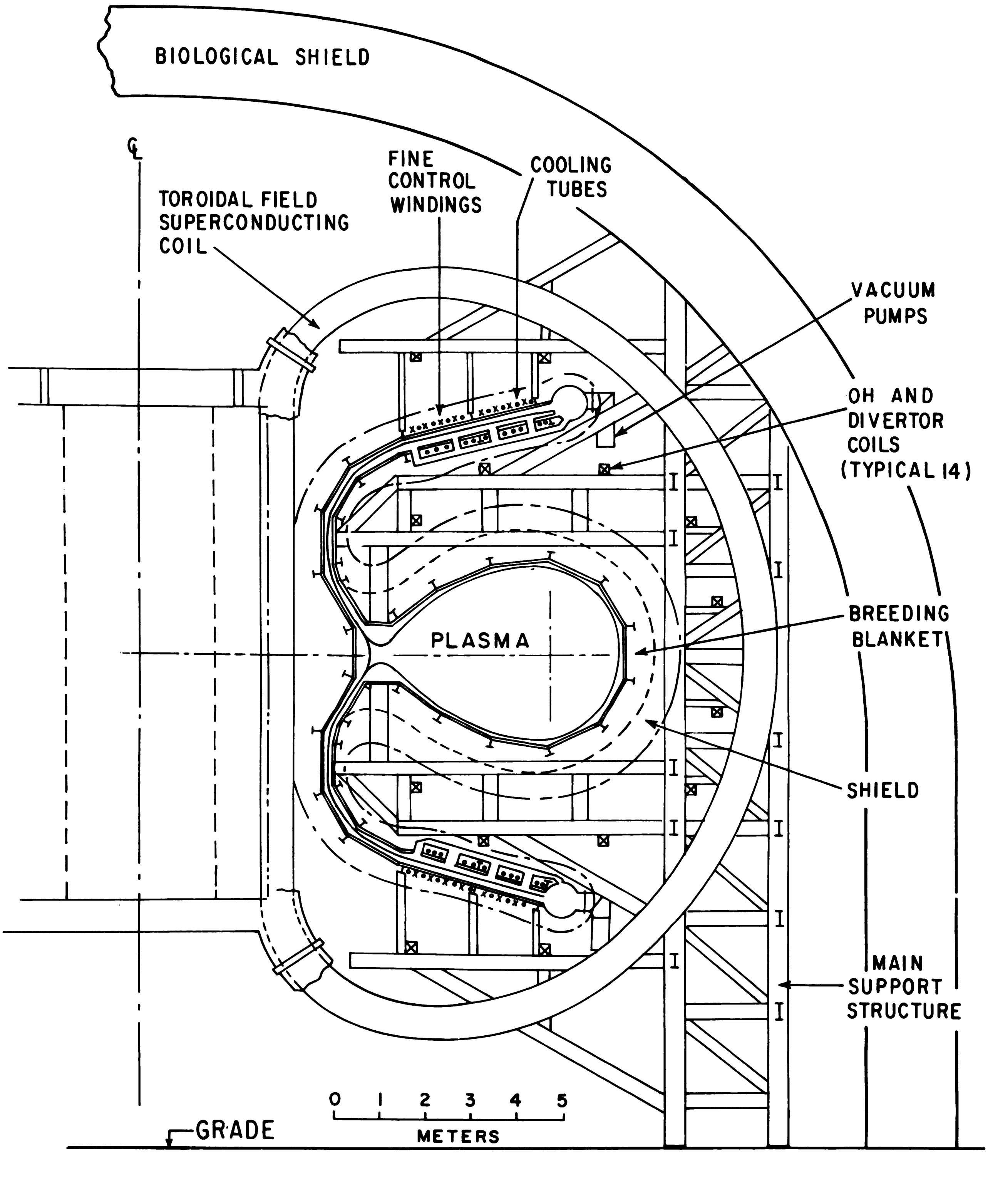

Figure A-1. Cross Section of the Princeton Tokamak Reactor Design

LASL Theta-Pinch Fusion Reactor*

A theta-pinch fusion reactor would utilize a shock-heating phase and an adiabatic compression phase. The shock-heating phase would have a risetime of a few hundred nsec and a magnitude of a few tens of kG to drive an implosion of a fully ionized plasma whose density is of the order of $10^{15}cm^{-3}$. After the ion energy associated with the radially directed motion of the plasma implosion has been thermalized, the plasma would assume a temperature characteristic of equilibration of ions and electrons. After a few msec the adiabatic compression field (risetime $\sim$ 10 msec and final value B $\approx$ 100 to 200 kG) would be applied by energizing a compression coil.

A schematic diagram of a theta pinch reactor system is shown in Figure III-2. The inner shock-heating coil with (for example) 8 radial transmission-line feeds is surrounded by a Li-Be-C blanket which has three functions: (a) it absorbs all but a few percent of the 14-MeV neutron energy from the plasma, which its flowing lithium carried out to heat exchangers in the electrical generating plant. (b) It breeds tritium by means of the Li^7 $(n,n'\alpha)$ T and Li^6 (n,α) T reactions. (c) The high Reynolds-number flow of liquid lithium cools the first wall (shock-heating coil).

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*S. C. Burnett, W. R. Ellis, T. A. Oliphant, Jr., and F. L. Ribe. Parameter Study of a Pulsed High-Beta Fusion Reactor. LA-DC-72-234. (1972).

Outside the inner blanket region is the multiturn compression coil which is energized by the slowly rising current ($\sim$ 10 kA per cm of its length) from the secondary of the superconducting magnetic energy store. The compression coil consists of the coiled up parallel-sheet transmission lines which bring in the high voltage to the feed slots of the shock-heating coil. Each side of the horizontal feed of the secondary coil also serves as a ground plane for the high-voltage shock-heating field. Each transmission line delivers of the order of 100 kV to one slot of the shock-heating coil.

Outside the compression coil and its titanium coil backing is the remainder of the neutron blanket for "mopping-up" the last few percent of neutron energy and breeding the last few percent of tritium. Unlike the inner blanket, which would run at $\sim$ 800°C to provide high thermal efficiency of the generating plant, this portion of blanket could run much cooler. Surrounding the outer blanket is a neutron shield, and beyond the shield the radially emerging transmission lines are brought around to make contact with the secondary coil current feeds and the high-voltage shock-heating circuits. To the right is shown the cryogenic energy storage coil in its dewar. At the bottom of the storage coil is the variable-inductance transfer element which reversibly transfers energy from the storage coil to the compression coil and back again.

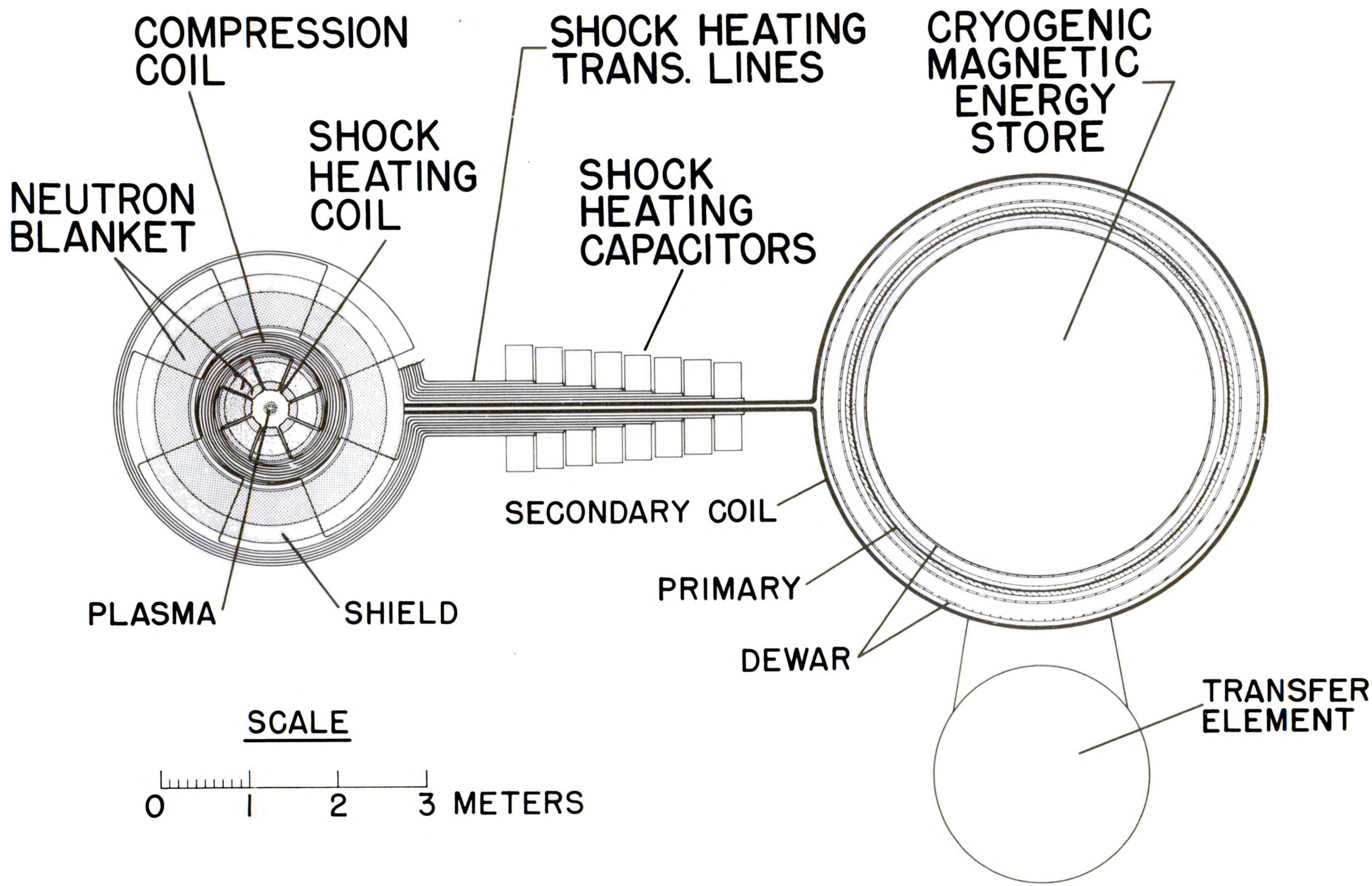

Figure III-2. Schematic of a Theta Pinch Fusion Reactor
(Cross Section of a Torus)

LLL Mirror Fusion Reactor

Designed to produce 500 MW(e), the LLL DT mirror reactor design may be considered as having three main parts: a magnetically contained plasma volume in which the fusion reactions take place, an ion injection and plasma heating system requiring electrical power input, and a combination thermal and direct energy converter system. The thermal portion of the converter system converts the neutron kinetic energy to thermal energy in a blanket surrounding the plasma confinement zone. The blanket breeds tritium for fuel replenishment. The second element of the energy converter system is the direct converter which accepts energetic charged particles which escape from the plasma confinement zone and it converts their energy to high voltage dc power. A fraction of this direct converter power is then fed back to the ion injection system to sustain the reaction and maintain the plasma. The reactor may be generally classified as a relatively low gain energy amplifier. This concept of combining thermal and direct conversion should be applicable to any fusion containment system; however, it is especially attractive for mirror systems because it furnishes a means to minimize the adverse effects of end losses. The direct conversion subsystem operates in a sequence of four steps: (1) expansion, (2) charge separation, (3) deceleration and collection, (4) conversion to a common potential. The first three steps of this process are as follows. The reaction products escape from the mirrors at a low ion density ($10^8 cm^{-3}$) which is further decreased to $10^6 cm^{-3}$ by expansion into a large, flat, fan shaped chamber. Expansion is accomplished by coupling an external radial magnetic field to the mirror field and allowing the field to decrease from its high level at the mirrors (approximately 150 kilogauss) to levels of about 500 gauss. The expansion also converts particle rotational energy to translational energy in inverse proportion to the field change. At the end of this expander field, electrons are separated from the ions by abruptly diverting the field lines. The electrons behave adiabatically and remain on the field lines while the ions cross the field lines and enter the collector region.

The ions emerge from the expander with a considerable spread in energy. To recover this energy at high efficiency the ions are passed through a series of electrostatically focusing collectors within which they are progressively decelerated. The ions are decelerated to a low residual energy and then diverted into a collector. Experiments at LLL have demonstrated overall collection efficiencies in excess of 80% and further improvements are expected.

The final step of direct conversion is the transformation of the electrical energy to a common potential. This is accomplished by an inverter-rectifier system using commercially available equipment.

The approximate plasma conditions are as follows: average ion energy = 400 keV, average electron energy = 40 keV, total power output = 1330 Mw, plasma beta = 0.9, plasma density = $10^{14} cm^{-3}$, and plasma radius = 4.3 meters. A schematic of the system is shown in Figure III-3.

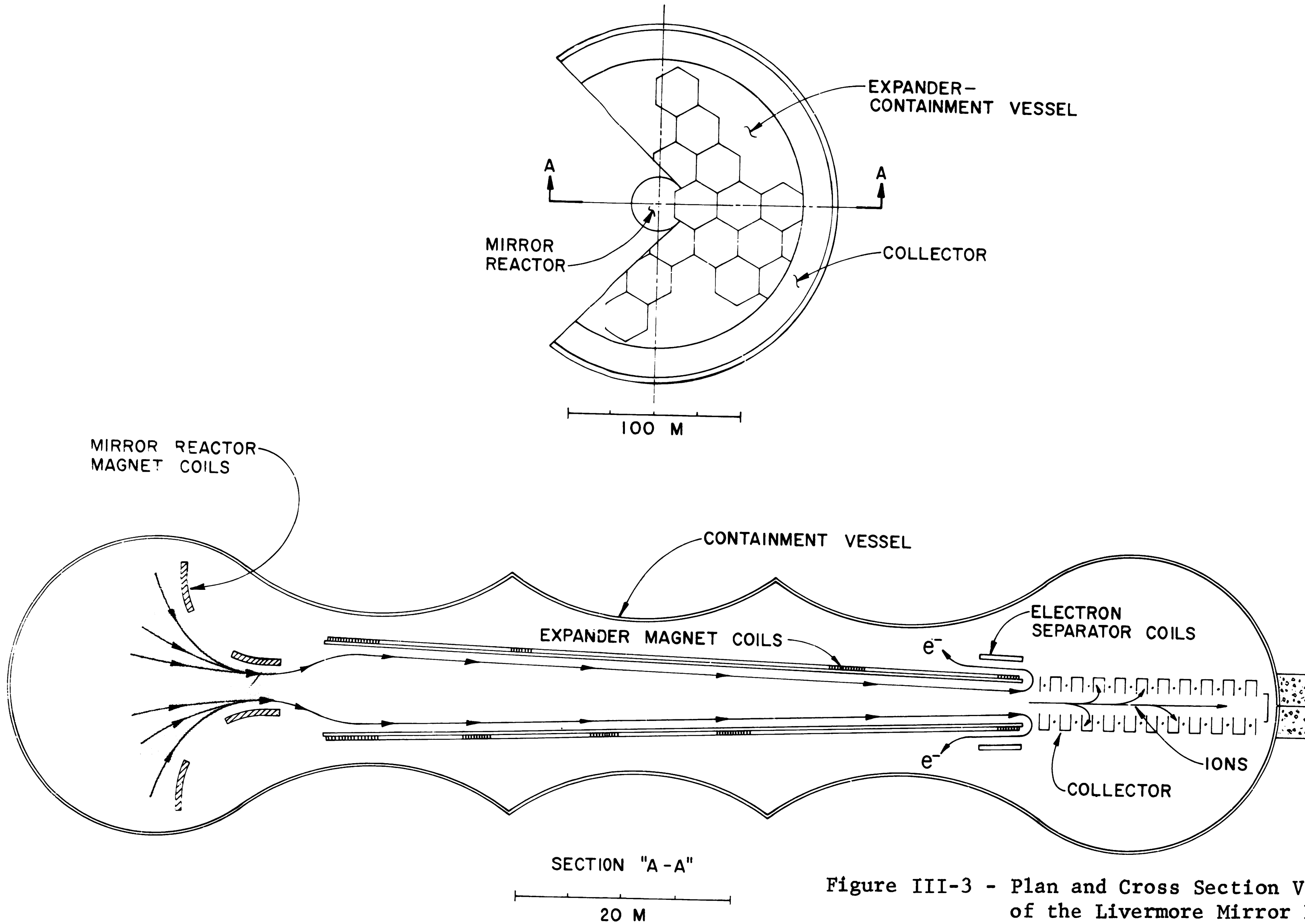

Figure III-3 - Plan and Cross Section Views of the Livermore Mirror Reactor with Direct Converter

ORNL Laser-Fusion Concept (BLASCON)*

If lasers can be economically utilized to ignite DT pellets to give small thermonuclear explosions, it may be possible to build reactors for central stations, ships, and spacecraft propulsion. Analyses and model tests indicate that, by igniting the pellets in the cavity of a vortex formed in a pool of liquid lithium, the explosion can be contained in conventional pressure vessels at a vessel capital cost of only about $10/kw(e). The neutron economy would be excellent -- the breeding ratio could be 1.3 to 1.5. If applied to reactors for central stations or ships, the concept would permit the construction of economic, thermonuclear reactors in sizes possibly as small as 100 MW(t). There would be no need for large cryogenic magnets, and no problem with fast neutron damage or neutron activation of structure. If applied to spacecraft propulsion the laser-exploded pellets might give a system whose propellant requirement for a typical Earth-Mars-Earth mission would be only about 10% those of a Rover-type nuclear rocket.

Frozen DT particles could be ignited at intervals of 10 to 20 sec and the energy of the explosions absorbed in a rapidly swirling pool of molten lithium contained in a massive pressure vessel perhaps 10 or 15 ft. in diameter having a configuration similar to that of Figure III-4. With a sufficiently high swirl velocity, a free vortex would form at the center of the swirling pool to provide a cavity

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*A. P. Fraas, "The Blascon - An Exploding Pellet Fusion Reactor". ORNL-TM-3231, July, 1971.

into which a deuterium-tritium pellet could be fired. When the pellet approached the bottom of the cavity in the vortex, a laser beam could be triggered to ignite the pellet, and the energy released in the subsequent fusion reaction could be absorbed in the molten lithium. Drawing off the lithium from the bottom of the pressure vessel would help stabilize the vortex. The lithium would be circulated to heat exchangers that could serve either to boil the working fluid for a Rankine cycle or heat the gas of a Brayton cycle. Other thermodynamic cycles could of course be employed, but the Rankine and Brayton cycles appear to be the most attractive. The lithium would be returned through pumps to tangential nozzles in the perimeter of the pressure vessel to maintain the desired vortex so that particles would be injected to a point close to the center of mass of the lithium. The operating temperature of the lithium would depend in part on the choice of containment system material, e.g., about 900°F if a chrome-moly steel were used and perhaps 1800°F if niobium were employed.

ORNL DWG. 71-6077

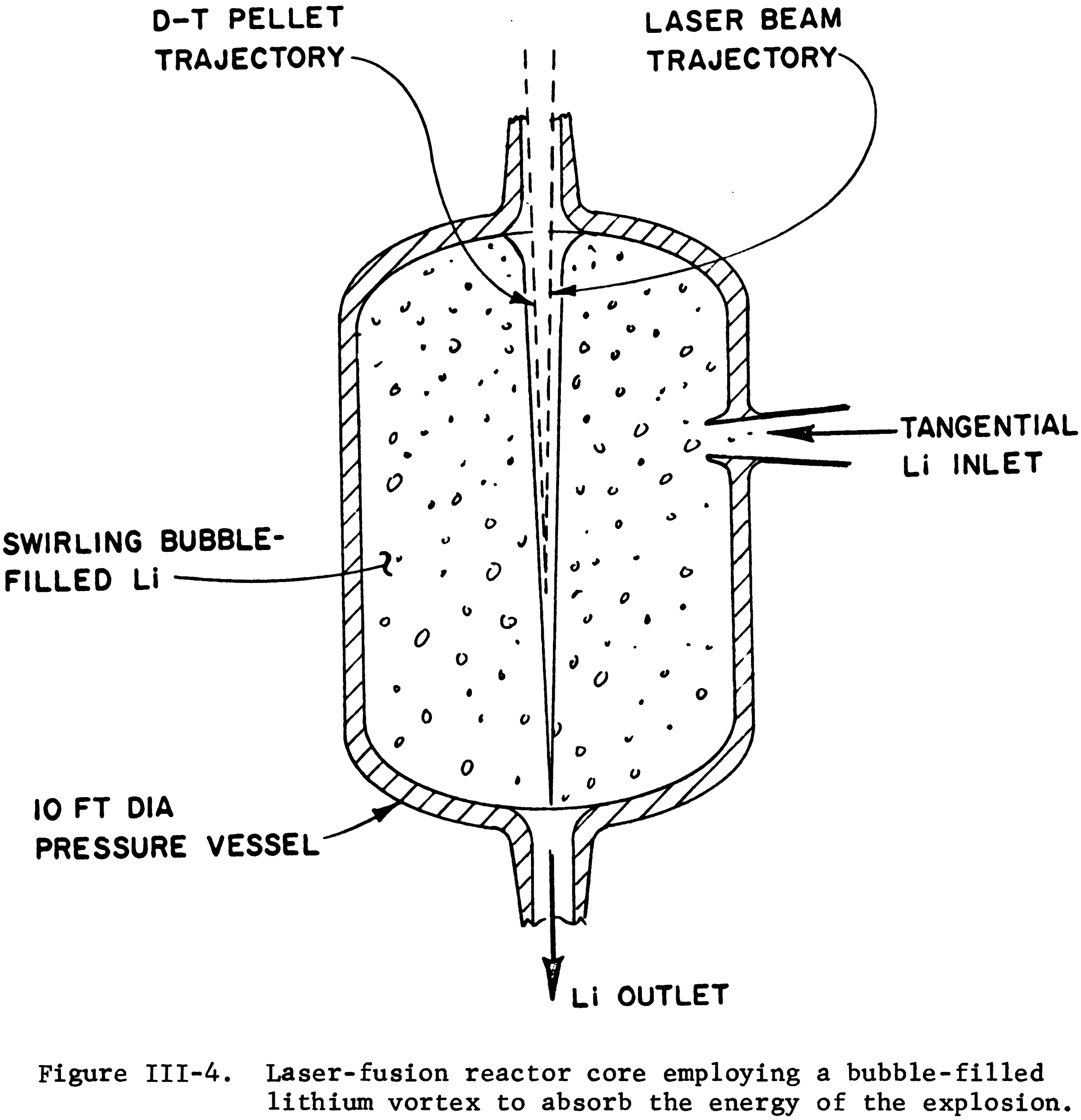

Figure III-4. Laser-fusion reactor core employing a bubble-filled lithium vortex to absorb the energy of the explosion.

LASL Laser-Fusion Reactor*

A schematic of a wetted-wall Inertial Confinement Thermonuclear Reactor (ICTR) is shown in Figure III-5. A DT pellet is injected through a port, which penetrates the blanket, and is initiated at the center of the cavity by a laser pulse; the cavity is defined by the wetted-wall located at a radius of 1.0 m from the center. The subsequent (D+T) burn releases 200 MJ of energy. Within fractions of a microsecond, 50 MJ is deposited within the pellet and 152.5 MJ is generated within the blanket lithium and structural materials.

Within $\sim$ 0.5 ms the pressure pulses generated by the interaction of the pellet with the lithium at the wetted-wall will subside. Within the next few milliseconds, the cavity conditions are equilibrated, $\sim$ 1.6 kg of lithium are vaporized from the protective layer at the wall, and sonic flow conditions of the cavity gases are established at the outlet port.

The flow of hot gases through the cavity outlet port is expanded in a diffuser to supersonic conditions, and the gases are then condensed in a downstream length of duct where a finely atomized spray of liquid lithium is injected. (The spray of atomized droplets is recirculated from the liquid pool at the bottom of the condenser). Downstream of the condenser duct, the mixture of gas and liquid droplets, still at supersonic velocity, is decelerated by turbulent mixing created by a spray of large lithium droplets.

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*L. A. Booth, et al. Central Station Power Generation by Laser-Driven Fusion. LA-4959-MS, Vol. 1. February, 1972.

(The coarse-droplet spray is provided from a side-stream of the 400°C return flow from the heat exchanger.) The kinetic energy of this mixture is finally absorbed by impacting with a pool of liquid lithium at the bottom of the condenser system.

After $\sim$ 0.2 s, the pressure within the cavity decreases to less than atmospheric, and the blow-down continues during the remaining 0.8 x of the pulse cycle, reducing the cavity pressure to less than 133 N/m^2 (1.0 mm Hg). The cycle is then repeated with the initiation of another pellet.

The energy deposited within the blanket is removed by circulating the lithium through an external heat exchanger. Lithium, flowing at 400°C from the heat exchanger, is returned to a plenum between the 1.0 cm-thick wetted-wall and the 5.0 cm-thick inner structural wall, which serves to restrain the movement of the inner blanket boundary caused by the pressure waves generated within the blanket and the cavity pressure. Located a few centimeters behind the wetted-wall, the inner structural wall also serves as a flow baffle for distributing the radial outflow. The wetted-wall moves along with the structural wall through hydrodynamic coupling, and, if needed, through mechanical attachments.

The minimum power level is based on a thermal output of $\sim$ 200 MW, from one ICTR. Higher power levels may be obtained by combining several ICTRs in a reactor system, thereby increasing both the versatility and the overall ratio of actual operating power to full design power. The nominal thermal power level for a conceptual plant was arbitrarily chosen to be $\sim$ 2000 MW, requiring ten modular ICTRs.

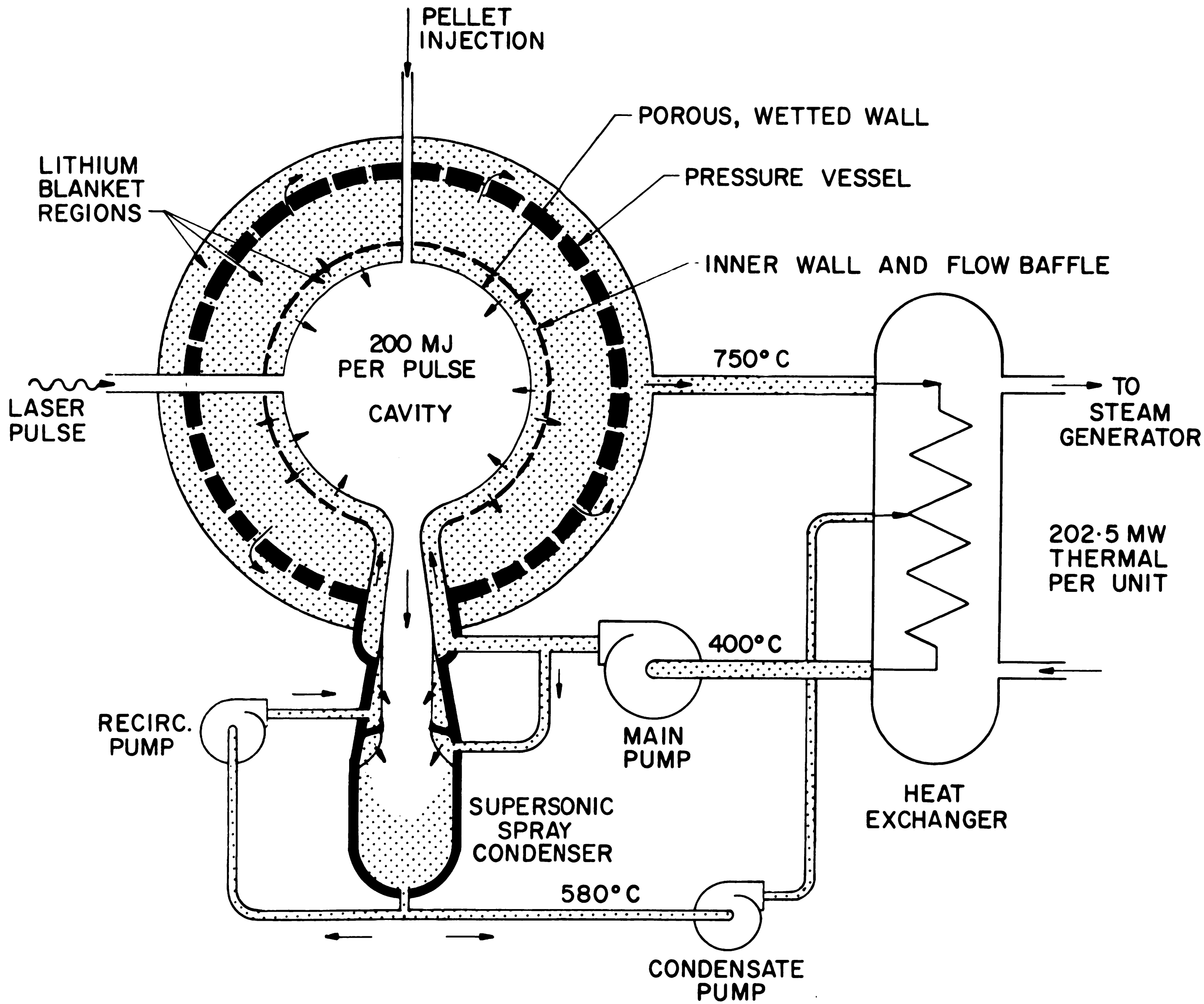

Figure III-5. Schematic of a Laser-Fusion Reactor Design

TABLE A1: THERMONUCLEAR ISLAND PROTOTYPE COST SUMMARY

BASIC PLASMA REACTOR SYSTEMS		ORMAK R=10.5m, r=3.5m		θ PINCH R=56m, r=30cm		PRINCETON TOKAMAK R=9m, r=2m		LLL DT MIRROR (bb) R_c=90m (f), r_p=3m (g)		ORNL LASER		LASL & LLL LASERS	
MW(e) & TOTAL FIRST WALL FLUX IN MW/m²		520 & 0.6		446 TO 4400 (c) & 0.67 TO 6.7 (c)		1840 & 4.5		500 & 1.3		56 & 13		1000 & ~50	
—BASIC MAT'L —	$/Kg.	METRIC TONS	$ M	METRIC TONS	$ M	METRIC TONS	$ M	METRIC TONS	$ M	METRIC TONS	$ M	METRIC TONS	$ M
Nb-1Zr	60	143	8.6	132	8			200	12	110	6.6	CONCEPTUAL DESIGN ONLY	
Li	9	460	4.1	1780 (b)	32 (b)			100	.9	20	.2		
Be	100			33	7								
STAINLESS STEEL	15	200	30	2500	38	8800 (x)	132	1000	15	25	.4		
Ti	12	200	2.4	560	7					10	.1		
He	10 (a)	1		600	6	81	.8						
Cu	10	450	4.5	5100	51								
NbTi (dd)	VARIES	1300 (n)	82	200 (o)	12	800 (p)	47 (ee)	500 (r)	59				
Nb_3Sn (dd)	VARIES					1200 (q)	237 (ee)						
GRAPHITE	2.50	1100	2.8	530	1			400	1				
Pb	.60	5350	3.2	390	2	350	.2						
Al	7			1000	7			210	1.4	100	.06		
FLiBe	8					1200	96						
Na	1.50												
K	2.20	10	—										
STRUCTURAL STEEL	2	6300 (k)	12.6	12K	24	5000	10						
INSULATORS	LUMP SUM				5		5 (y)						
CONCRETE	.12	20 K	2.4	5200	.6	2300	.3	25K	3				
-SUB TOTAL FOR BASIC MAT'L -			125.6		198.8		441.9		92.3		7.4		
$/KW(e) FOR BASICS		242		45 TO 450		240		185		132			
-REACTOR AUXILIARIES - (f)													
HELIUM REFRIG. SYSTEM			4		30		8		26				
INJECTION SYSTEM			20		—		20		13.0				
TRITIUM RECOVERY SYSTEM			6		8		10		6				
POWER SUPPLIES			—		—		SEE INVERTERS & RECTIFIERS		—				
CONTROLS & INSTRUMENTATION			2		5		4		2				
PUMPS & COMPRESSORS			—		—		SEE OTHER		—				
HEAT EXCHANGERS			2		2 TO 20		SEE OTHER		1				
LASER & OPTICAL SYSTEM			—		—		—		—	~50 TO 2K		~500 TO 20K	
DT PELLET SYSTEM			—		—		—		—	~.5 TO 2		~5 TO 20	
INVERTERS & RECTIFIERS			.1		—		15		2.0				
VACUUM SYSTEMS			1.3		1		7		.3				
TRITIUM, $1M PER Kg (v)			1		1		7.3		1				
REMOTE HANDLING EQPT.			10		10		10		10				
CONTAINMENT & SHIELD VESSEL			10 (m)		20 (u)		20		30				
OTHER			—		~150 (e)		50 (aa)		—				
SUBTOTAL FOR REACTOR AUXILIARIES			56.4		227 TO 245		151.3		67.9				
-TOTAL -			182		425.8 TO 443.8		593.2		160.2				
$/KW(e) TOTAL			350 (h)		100 TO 1000 (h)(w)		320 (h)		320 (h)				

NOTES

(a) He EXTRACTED FROM THE ATMOSPHERE- $300 PER Kg.
(b) 80 METRIC TONS Li^6 (93 % enr) AS OUTER COOLANT FROM 1135 METRIC TONS Li, 645 METRIC TONS ELSEWHERE. ENRICHMENT COST ESTIMATED TO BE $200 PER Kg ON BASIS OF 8 KWh/gm.
(c) FOR .25 & .025 PULSES PER SEC. RESPECTIVELY LINEAR WITH REP RATE
(d) TIME AVERAGE
(e) ENERGY SWITCHGEAR, VERY PRELIMINARY COILS ARE INCLUDED ELSEWHERE.
(f) OUTSIDE RADIUS OF EXPANDER-COLLECTOR
(g) RADIUS OF PLASMA
(h) DOES NOT INCLUDE ENGINEERING @ 20% AND OVERHEAD @ 30%
(k) 300 METRIC TONS IN CONTAINMENT LINER, 1500 IN REACTOR STRUCTURE 4500 IN POLOIDAL MAGNET CORE.
(m) STEEL & CONCRETE INCL IN BASIC MAT'L LIST
(n) 53 KILOGAUSS AT CONDUCTOR
(o) 45 KILOGAUSS AT CONDUCTOR IN ENERGY STORAGE COIL
(p) 400 METRIC TONS EACH AT 80 & 60 KILOGAUSS AT COND
(q) 400 METRIC TONS EACH AT 160, 120 & 100 KILOGAUSS; AT CONDUCTOR
(r) 90 KILOGAUSS AT CONDUCTOR
(s) 90 " " "
(t) 150 " " "
(u) ANNULAR SHIELD
(v) BASED ON 239/3 TIMES PLUTONIUM COST
(w) VARIATION OF COST IS DUE TO DUTY FACTOR CHOSEN TO GIVE A SPECIFIC WALL LOADING.
(x) INCLUDES DEWARS & COIL HOLDERS BUT NOT THERMAL INSULATION
(y) THERMAL & ELECTRICAL INSULATION
(z) COSTS INCLUDE FIELD INSTALLATION.
(aa) HELIUM CIRCULATION & STEAM GENERATION SYSTEM.
(bb) Q = 1.0
(cc) Q = 0.4
(dd) STABILIZED CONDUCTOR, INCLUDES Cu OR Al AS REQUIRED.
(ee) COST OF COIL HOLDER INCLUDED IN STAINLESS STEEL ITEM.

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TABLE A2: - Data for CTR Reference Designs

Reactor	ORNL Tokamak	PPPL Tokamak	LASL θ Pinch	LLL Mirror		ORNL Laser	LASL Laser	LLL Laser
Reference Report	ORNL-TN-3096	-	LA-DC-72 234-A	UCRL-72883		ORNL-TM-3231	LA4858MS Vol.1	Freeman et al APS Mtg. 11/71
				DT	D^3He			
Design Power, MW(t)	1000	5700	3500	1250	1220	150	2025	~ 200
Design Power, MW(e)	520	1840	1750	500	500	56	825	~ 100
Net Thermal Efficiency, %	52	40	50	40	41	37	41	~ 50
Reactor Size	Torus R=10.5m r-3.5m	R=9m r=2m	R=56m r-.3m	r_p=3m[a] r_e=90m[b]	r_p=4m[a] r_e=150m[b]	4m sphere[c]	2m sphere[c]	1m sphere[c]
Magnetic Field in Plasma, kG	24.5	64	150[v]	42	70	0	0	150
Vacuum Wall								
Heat loading, MW/m^2	0.69	4.50	3.5[h]	1.3	0.5	13	10^4[e] - 10^5[f]	~ 50
Surface area, m^2	1450	800	670	430[w]	200[w]	11.4	π	π
Thickness, cm	0.25		.3[s]	0.1	0.1	25	1.0	thin
Enclosed volume, m^3	2120	720	101	170[w]	52[w]	5.2	4.2	0.5
Blanket Material								
Structure	Nb	SS	Nb	Nb	SS	Nb	SS	
Moderator	Graphite	Flibe	Graphite	C+Li	H_2O	Li	Li	Li-LiD
Coolant	Li	Helium	Li	Li	H_2O	Li	Li	Li-LiD
Activity after 1 year[m], curies	7×10^7			10^9		10^4	3×10^9[d]	
Tritium Inventory, g				1000	k			
Blanket, g	375	300	35			20		
Blanket coolant, g	0.2	.5				0.1		
Recovery system, g	300	7000	35			100		
Major Material Inventories, Metric Tons							t	t
Nb - 1 Ar	143	0	132	200		110		
Li	460	0	725[p]	100		20		
Be	0	0	33			0		
SS	200	8800	2500	1000	3000	25		
Ti	200	0	560			10		
He	1		600			0		
Cu	450	0	5100			0		
Superconductor, Nb Ti in Cu	1300	800	1000	500	490	0		
, Nb_3Sn in Cu	——	1200	0		240	0		
Graphite	1100	0	530	400		0		
Pb	5350	350	390			100		
Al	0	0	1000	210	750	0		
Flibe		1200						
Concrete	20K	2300	5200	25K	47K			
Structural Steel	6300	5000	12K	24	14			
K	10							

Notes

a. r_p=plasma radius

b. r_e=expander collector radius

c. pressure vessel diameter

d. from ten modules

e. average during 1 ms pulse, 1 ms pulse per sec

f. peak during pulse, 1 pulse per sec

h. time average

k. low but not determined. No breeding required.

m. One year at full power

n. See also Scientific American June, 1971.

p. Coolant is 50% enriched lithium. Shield is normal lithium

s. Plus .1 cm Al_2O_3

t. No detail design

v. External to plasma field is zero in plasma

w. Primary plasma vessel only.