Implementation Considerations of Coupling Dedicated Fissile and Fusile Production Fusion and Fission Reactors

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FUSION TECHNOLOGY INSTITUTE

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IMPLEMENTATION CONSIDERATIONS OF COUPLING DEDICATED FISSILE AND FUSILE PRODUCTION FUSION AND FISSION REACTORS

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ABSTRACT

The steady state operational mode of coupled systems of dedicated fusion reactors producing fissile fuel for fission reactors which, in turn, produce the fusile fuel for the fusion reactors, is discussed. It is found that if the breeding capacity of either system is increased, the breeding capacity in the other system can be relaxed, and that the ratio of total fission to fusion power depends on these breeding capacity choices. The coupling is shown to be potentially more economical than a pure-fusion system. The different alternatives for producing the tritium feed for fusion reactors are compared. Existing Light Water Reactors would require major design modifications and do not look attractive. Savannah-River-type reactors are judged inadequate because of their inability to produce power. Graphite-moderated Hanford-type reactors seem to be the most appropriate existing systems. However, since they are of outdated design, enriched-uranium CANDU-type reactors or graphite moderated, water or gas-cooled reactors are judged as better choices for a practical fusion-fission coupling implementation, provided their development cost is less than that of the fusion reactors.

1. INTRODUCTION

The possibility of sharing fissile and fusile fuel production among coupled systems of fission and fusion reactors has been considered by several authors [1-5]. The sharing is based on the premise that fission reactors are "power rich (~200 MeV/fission) but neutron poor (2 MeV energy)" and would be best used in such a coupling to produce power and breed some fusile fuel for the fusion reactors in a predominantly thermal spectrum. On the other hand, fusion reactors are considered "power poor (~17.6 MeV/fusion) but neutron rich (~14.1 MeV energy)," so that they can be better used for neutron multiplication and fissile fuel breeding in a fast spectrum. This bred fuel can then be shipped for burning in the fission reactors. In particular, systems which breed ^{233}U in the fusion reactors to be shipped and burned in the thermal fission reactors with tritium, in turn, produced in the fission reactors for shipment to the fusion reactors are here considered. Practical aspects of such coupling including constraints on the systems and the ensuing economic parameters are analyzed. Information available in the declassified literature about the production of tritium in dedicated facilities is summarized, and the conversion of existing reactors to tritium production is discussed.

2. CONSTRAINTS ON STEADY STATE OPERATIONAL MODE

Consider the case of D-T fusion reactors dedicated to the production of fissile fuel coupled to fission reactors, in turn, dedicated to the production of power and tritium fuel. The fusile and fissile fuel are exchanged among both types of reactors. Such a system is described by the two equations,

$$\frac{dN_T}{dt} = B_T(1 + \alpha)R_U - R_T \tag{1}$$

$$\frac{dN_U}{dt} = B_U R_T - (1 + \alpha)R_U \tag{2}$$

where:

T,U denote tritium and ^{233}U, respectively,

N are the number of tritium or ^{233}U atoms,

R is the rate of tritium consuming or ^{233}U fission reactions,

B is the rate of tritium or ^{233}U produced per U or tritium atom consumed,

$(1 + \alpha)$ is the number of consumed ^{233}U nuclei per fissioned ^{233}U nucleus, $(1 + \alpha) \approx 1.1$ for ^{233}U. (α is the capture to fission ratio.)

The thermal powers produced in the fusion and fission reactors are:

$$\begin{aligned} P_T &= R_T W_T \\ P_U &= R_U W_U \end{aligned} \tag{3}$$

where:

$W_T \approx 20$ MeV/fusion reaction

$W_U \approx 200$ MeV/fission reaction

Substituting from Eq. (3) into Eqs. (1) and (2), we can write:

$$\underline{\dot{N}} = \underline{\underline{M}}\ \underline{P} \tag{4}$$

where:

$$\underline{\dot{N}} = \begin{pmatrix} \frac{dN_T}{dt} \\ \\ \frac{dN_U}{dt} \end{pmatrix} \quad \underline{\underline{M}} = \begin{pmatrix} -\frac{1}{W_T} & \frac{B_T(1+\alpha)}{W_U} \\ \\ \frac{B_U}{W_T} & -\frac{(1+\alpha)}{W_U} \end{pmatrix}, \quad \underline{P} = \begin{pmatrix} P_T \\ \\ P_U \end{pmatrix}.$$

Different operational modes [4] for such a system can be considered. Blinkin and Novikov [3] considered a case where:

$$\frac{1}{N_T}\frac{dN_T}{dt} = \frac{1}{N_U}\frac{dN_U}{dt} = \frac{1}{\tau}.$$

This implies a constant ratio of fuel inventory, $\frac{N_U}{N_T}$, with both the fissile and fusile fuel inventories increasing with a time constant, τ. For an expanding fusion inventory, one can use:

$$\frac{dN_T}{dt} > 0, \qquad \frac{dN_U}{dt} \geq 0,$$

while an expanding fission inventory economy implies:

$$\frac{dN_T}{dt} \geq 0, \qquad \frac{dN_U}{dt} > 0.$$

For a steady-state operational mode where the collection of fusion and fission reactors are self-sustaining and no accumulation of fissile or fusile fuel is occurring, we choose to consider for our analysis:

$$\underline{\dot{N}} = \underline{\underline{M}}\,\underline{P} = 0. \tag{4}'$$

For a nontrivial solution of Eq. (4), the determinant of the coupling matrix, $\underline{\underline{M}}$, should be zero:

$$|M| = 0, \tag{5}$$

from which

$$B_T = \frac{1}{B_U}. \tag{6}$$

Substituting from Eq. (6) in Eq. (4'), we get for the ratio of power production in the fission reactors to the power production in the fusion reactors(support ratio):

$$\frac{P_U}{P_T} = \frac{W_U}{W_T} \cdot \frac{1}{(1 + \alpha)B_T} \cdot \tag{7}$$

Youssef, Conn and Vogelsang (Ref. 1, Eq. (18)) considered a special case in which $B_T = 1$ and concluded that the thermal power of the tritium production reactor per unit of fusion power does not depend on the breeding capacity, B_U, of the fusion reactor. This is not true in our case, since B_T and B_U are related by Eq. (6), and B_T appears in Eq. (7). Equations (6) and (7) govern such a coupled system, and Table I displays some ensuing discrete values of the parameters. Since the product $B_T B_U = 1$, as one obtains more fissile breeding in the fusion reactor, then the tritium breeding requirements of the fission reactors can be relaxed. The ratio of fission reactors' power to fusion reactors' power has no minimum and is lowest for higher tritium breeding in the fission reactors, and lower fissile breeding in the fusion reactors. The sum, $B_U + B_T$, has a minimum at $B_U = B_T = 1.0$, corresponding to a system consisting of a single fusion reactor supplying fissile fuel to about nine fission reactors of the same power, which in turn supply it with its need in tritium.

The case where $B_T = 0.6$ corresponds to existing LWRs with a conversion ratio of ~0.6 which conceivably would be converted to breeding tritium rather than producing Pu. The ratio of LWRs to fusion reactors of the same power would be about 15. The fusion reactors

Table I. Some Values of System Parameters

$B_U + B_T$	B_U	B_T	P_U/P_T	$C_T/C_U = 2$		$C_T/C_U = 10$	
				C_1	C_2	C_1	C_2
2.50	0.50	2.00	4.55	1.18	0.59	2.62	0.26
2.00	1.00	1.00	9.09	1.10	0.55	1.89	0.19
2.01	1.11	0.90	10.10	1.09	0.55	1.81	0.18
2.27	1.67	0.60	15.18	1.06	0.53	1.56	0.16
2.50	2.00	0.50	18.18	1.05	0.53	1.47	0.15
3.33	3.00	0.33	27.27	1.04	0.52	1.32	0.13
4.25	4.00	0.25	36.36	1.03	0.51	1.24	0.12
5.20	5.00	0.20	45.45	1.02	0.51	1.19	0.12
6.17	6.00	0.17	54.55	1.02	0.51	1.16	0.12

would have a fissile breeding ratio of about 1.67, which is well within the range of existing DT-fusion conceptual designs, provided tritium production is substituted by fissile production. The case for $B_T = 0.9$ would correspond to advanced converters and the relaxation of fissile breeding requirements from $B_U = 1.07$ to 1.11 can be noticed.

We next consider the economic advantages or disadvantages of such a coupling. If the costs per unit installed capacity for the fusion and fission islands are, respectively, c_T and c_U, then the capital cost of the total system will be : $c_U P_U + c_T P_T$. It is reasonable to expect that the cost of electricity to be produced will be proportional to the last sum. At the early stages of introduction of fusion energy, one can obviously expect that $c_T > c_U$. One can define a cost ratio, C_1, of the coupled system to that of a system where all the combined system power would be generated solely from fission reactors:

$$C_1 = \frac{c_U P_U + c_T P_T}{c_U (P_U + P_T)} = \frac{\frac{P_U}{P_T} + \frac{c_T}{c_U}}{\frac{P_U}{P_T} + 1} \tag{8}$$

and another cost ratio C_2 of the coupled system to that of a system where all the combined system powers would be generated solely from fusion reactors:

$$C_2 = \frac{c_U P_U + c_T P_T}{c_T (P_U + P_T)} = \frac{\frac{P_U}{P_T} + \frac{c_T}{c_U}}{\frac{c_T}{c_U}\left(\frac{P_U}{P_T} + 1\right)} = C_1 \frac{c_u}{c_T} \tag{9}$$

For the case, $c_T/c_U = 2$, 10 values from Eqs. (8) and (9) are displayed in Table I. If the cost of the fusion island is twice that of the fission island ($c_T/c_U = 2$), one can notice that the coupling of fusion and

fission reactors (for $B_U = 1.67$ and $B_T = 0.6$) leads to a cost saving of approximately 47% compared to a case where fusion reactors alone were used, and the savings are more pronounced (84%) if $c_T/c_U = 10$. For $c_T/c_U = 2$, the coupling is only 6% more costly than if all the power would have been produced by fission reactors. If $c_T/c_U = 1$, then no economical advantage is obtained from the coupling.

The previous simplified analysis shows that substantial savings would be possible by using coupled systems of fusion and fission reactors compared to an unmixed fusion system. The cost penalty compared to a pure fission system is not real, since the fission reactor economy would be running out of fissile fuel in the near future, unless breeder reactors or fusion reactors are introduced as a source of fissile fuel.

Other than the power production cost advantage discussed here, releasing the complicated fusion design from tritium breeding in the blanket with its diffusion and leakage problems, and concentrating it in the relatively simple and known fission reactors may have technological advantages in terms of reliability, maintainability and accessibility of the fusion reactor components. The fusion reactor blanket can also be designed to be relatively clean by discouraging the fissioning of the produced fissile fuel in it.

We proceed to analyze the possibilities of producing tritium in dedicated fission reactors. Taking into account practical considerations, such as maximum limits on fuel burnup, is important, since they do not allow us to exactly reach the theoretical estimates of the last section.

3. POSSIBLE SOURCES OF TRITIUM PRODUCTION

Cosmic radiation produces 1/2 - 1 kg/year of tritium, resulting in a natural inventory of 7-14 kg in air and water, which is unrecoverable due to its low concentration [6]. Artificial sources are needed to produce tritium fuel for future D-T fusion reactors. Table II shows that the yearly tritium consumption is 110-170 kgs per 1000 MWe capacity [7-11] in fusion reactors.

In general, possible sources of tritium as a feed to fusion reactors would be:

—The U.S. stockpile

—Light-Water Reactors (LWRs)

—Modified LWRs

—Fuel Reprocessing Plants

—Liquid Metal Fast Breeder Reactors

—Production Reactors

—Heavy Water Reactors (HWR)

United States' tritium stockpiles may not be available for commercial uses, since they are kept for national security purposes [12].

Tritium is generated in LWRs by ternary fission. An amount of 500-1000 Ci/1000 MW(e)-yr is generated by neutron capture reactions, primarily in the shim control boron in the primary coolant [12]. Tritium generation by the year 2000 would amount to a meager 51.3 g/yr for a 683.9 GWe installed capacity, or 32.6 g/yr for a 434 GWe installed capacity. This amounts to only 7.50×10^{-8} kg/(MWe-yr) of tritium (Table III).

Table II. Tritium Consumption for Some Fusion Power Plant Designs

Design	Consumption kg/(MWe•year)
UWMAK-I[7]	1.7×10^{-1}
UWMAK-II[8]	1.3×10^{-1}
UWMAK-III[9]	1.1×10^{-1}
NUWMAK[10]	1.5×10^{-1}
SOLASE[11]	1.7×10^{-1}
RANGE	0.11 - 0.17

Table III. Possible Tritium Producing Sources

Source	Production kg/(MWe•year)	Remarks
Existing Light Water Reactors (LWRs)	7.50×10^{-8}	From shim control boron
Slightly Modified LWRs	1.80×10^{-4}	90% ^{6}Li enrichment in burnable poison rods and instrument wells
Largely Modified LWRs	6.51×10^{-3}	Conversion Ratio = 0.6, 30,000 MWD/TU burnup
Heavy Water Reactors	1.90×10^{-4}	Activation of D_2O
Fuel Reprocessing	1.18×10^{-3}	Ternary fission tritium
Liquid Metal Fast Breeder	3.33×10^{-3}	Li used instead of Na
Savannah River Reactors	$\sim 1.66 \times 10^{-2}$	Maximum, No power production
Hanford N-Reactor	4.34×10^{-3}	Minimum + Pu + power
	1.14×10^{-2}	Maximum, No Pu, + power
Modified Heavy Water Reactors	1.95×10^{-2}	Conversion Ratio = 0.9, 2.0% enrichment, 10,000 MWD/TU burnup

Fuel Consumption in Fusion Reactors: 0.11 - 0.17 kg/(MWe•year)

If existing LWRs are slightly modified to produce further tritium by neutron capture in Li enriched to more than 90% in ^{6}Li, and eight instrument wells of a 1000 MWe Pressurized Water Reactor (PWR) are filled with solid lithium pellets, 100g/yr of tritium can be produced. If solid lithium compounds are substituted for the burnable poison rods used to provide a negative temperature reactivity coefficient when using fresh fuel, another 80 g/yr can be obtained [12]. This is again a meager total of 1.80×10^{-4} kg/(MWe/yr) of tritium.

Major modifications to LWRs can conceivably use the neutrons captured in ^{238}U to produce ^{239}Pu to breed T from ^{6}Li in suitably modified cores. Assuming that the production of tritium in such a system can be approximated by its atomic conversion ratio (ACR):

$$ACR = \frac{N_p}{N_c} = \frac{\text{Average rate of fissile/fusile atom production}}{\text{Average rate of fissile atom consumption}}$$

and considering an average ACR of 0.6, and burnup of 30,000 MW-days/MTU for 3% enriched fuel, the maximum tritium production from a 1000 MWe (3582 MWth) reactor at a 65% duty cycle can be easily calculated.

For this purpose, let us define the Weight Conversion Ratio (WCR) as:

$$WCR = \frac{m_p}{m_c} = \frac{\text{Average weight of fissile/fusile atoms produced}}{\text{Average weight of fissile/fusile atoms consumed}},$$

and knowing that $N = m \frac{A_v}{M}$, where A_v is Avogadro's number, and M is the molecular weight, we can deduce the relationship:

$$ACR = \left(m_p \frac{A_v}{M_p} \right) \Big/ \left(m_c \frac{A_v}{M_c} \right) = \frac{m_p}{m_c} \cdot \frac{M_c}{M_p} = WCR \cdot \frac{M_c}{M_p}$$

or

$$WCR = \frac{M_p}{M_c} \cdot ACR. \qquad (10)$$

The WCR for an LWR producing ^{239}Pu from ^{235}U will be:

$$WCR = \frac{239}{235} ACR = 1.017\ ACR,$$

however, if tritium is produced from ^{235}U, then:

$$WCR = \frac{3}{235} ACR = 1.277 \times 10^{-2}\ ACR.$$

For our assumed modified LWR, we have:

$$\text{Tritium weight}^{*} = 1.277 \times 10^{-2} \times 0.6 \frac{\text{kg T}}{\text{kg}^{235}\text{U}} \times \frac{1}{30000} \frac{\text{MTU}}{\text{MWth•day}}$$
$$\times 10^{3} \frac{\text{kg U}}{\text{MTU}} \times 0.03 \frac{\text{kg}^{235}\text{U}}{\text{kg U}} \times 3582\ \text{MWth} \times 365 \frac{\text{days}}{\text{year}}$$
$$\times 0.65 = 6.51\ \text{kg T/year.}$$

This gives a ratio of $\frac{P_U}{P_T} = 16.9 - 26.11$ for a practical system, taking into consideration fuel burnup limits and duty factors. In Table I, the theoretical value of $\frac{P_U}{P_T}$ for ACR = B_T = 0.6 is 15.18 in approximate agreement to the lower limit of the practical calculated value of $\frac{P_U}{P_T}$.

Such a conversion may not be possible, since the replacement of ^{238}U by ^{6}Li with its high neutron absorption cross-section (~675b) could not allow criticality if distributed over the core. Seed-blanket reactor concepts may be used where highly enriched (>90%) uranium oxide

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*In Ref. 12, page 20, the following calculation is given:

$$0.6 \times 0.65 \times \frac{\text{MTU}}{30000\ \text{MWth•day}} \times 365 \frac{\text{days}}{\text{year}} \times 3582\ \text{MWth}$$
$$\times 10^{3} \frac{\text{kg}}{\text{MTU}} \stackrel{?}{=} 6.85\ \text{g/yr.}$$

Consideration of the per atom basis of the definition of ACR and that the burnup is defined in terms of heavy metal leads to the different calculations.

mixed with ZrO_2 as inert material, is used as a seed and is surrounded by blanket elements containing lithium. This would be similar to Light Water Breeder Reactor (LWBR) systems.

As a result of neutron activation of D_2O, 190 g/yr of tritium are produced in a 1000 MWe CANDU-type reactor. Modified CANDU reactors, using enriched fuel, would be necessary for breeding tritium because of neutron economy problems with the currently used natural uranium. Assuming that an enrichment to 2% would allow a burnup of 10,000 MWth•day/MTU, at a conversion ratio of 0.9, a calculation similar to the one carried out for the LWR case yields:

$$\text{Tritium weight} = 1.277 \times 10^{-2} \times 0.9 \times \frac{1}{10000} \times 10^3 \times 0.02 \times 3582 \times 365 \times 0.65$$

$$= 19.5 \text{ kg T/year},$$

which amounts to 1.95×10^{-2} kg/(MWe•year). This yields a $\frac{P_U}{P_T}$ ratio of 5.64 - 8.72. For comparison, the theoretical $\frac{P_U}{P_T}$ ratio for a conversion ratio of 0.9 from Table I is 10.1.

Liquid Metal Fast Breeder Reactors (LMFBRs), are conceivable competitors to fusion reactors in the long run; it does not seem useful to have them supply tritium to fusion reactors. Nevertheless, using Li as a coolant in an LMFBR with a breeding ratio of 1.15, it was estimated [13] that $\sim 3.33 \times 10^{-3}$ kg/MWe•year of tritium can be produced at an overall 30% efficiency. They do not look as good as modified LWRs, however.

If reprocessing is allowed, ternary fission tritium will be released in the effluent of fuel-reprocessing plants. Assuming an 850 MWe nuclear generating capacity by the year 2000, it has been estimated that a production of 1 kg/yr of tritium is possible [12]. This amounts

to 1.18×10^{-3} kg/MWe•yr. This can be useful for the start-up of the cycle and supplement tritium producer fission reactors.

It is known that tritium is produced for thermonuclear weapons and industrial uses (e.g., luminous dials) in the Department of Energy (DOE) production reactors by the irradiation of Li-Al alloys. Production reactors exist at two sites: The Savannah River Plant, South Carolina, and at the Hanford Site, Richland, Washington. Co-production of both Pu and tritium is possible in these reactors [12], so we consider them in more detail, since they are already existing facilities for tritium production and discuss their possible role in fusion-fission coupling.

4. SAVANNAH RIVER TYPE REACTORS

The Savannah River reactors are thermal reactors cooled and moderated with heavy water. They are descendants of the early NPD Reactor. As shown in Figs. 1 and 2, a reactor consists of a tank 17 ft in diameter and 15 ft high. The core of the reactor contains 612 fuel and/or target assemblies and 61 control assemblies, all arranged in a hexagonal pattern as shown in Figs. 3 and 4. The tritium-producing elements are placed around the reflector while the Pu ones are dispersed in the core. The D_2O exits the reactor tank through six symmetrically located nozzles, and then flows through six external loops. Each loop has one pump and two heat exchangers in parallel. Returning from the heat exchangers, most of the D_2O flows to the plenum via six nozzles located along its rim. The remainder of the D_2O flow goes to a header below the reactor tank and supplies cooling to the control assemblies. The D_2O flows at high velocity from the plenum down the fuel and target assemblies and enters the moderator space at the bottom

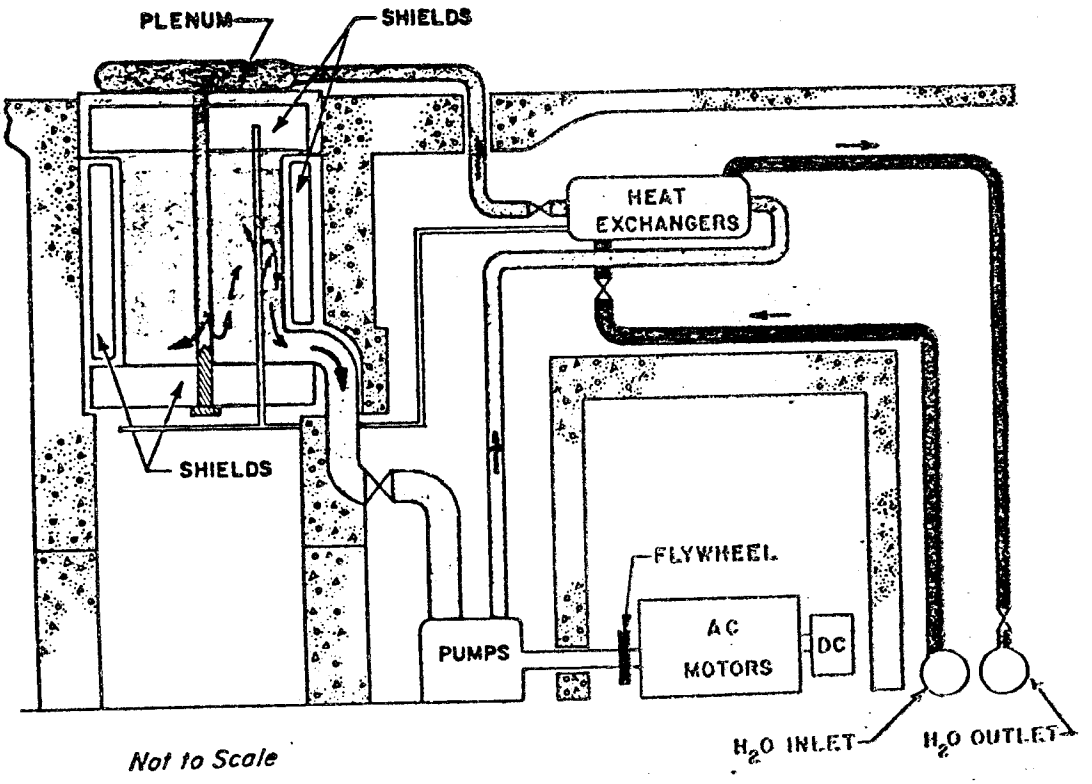


Fig. 1. Schematic Diagram of the Savannah River Reactor.

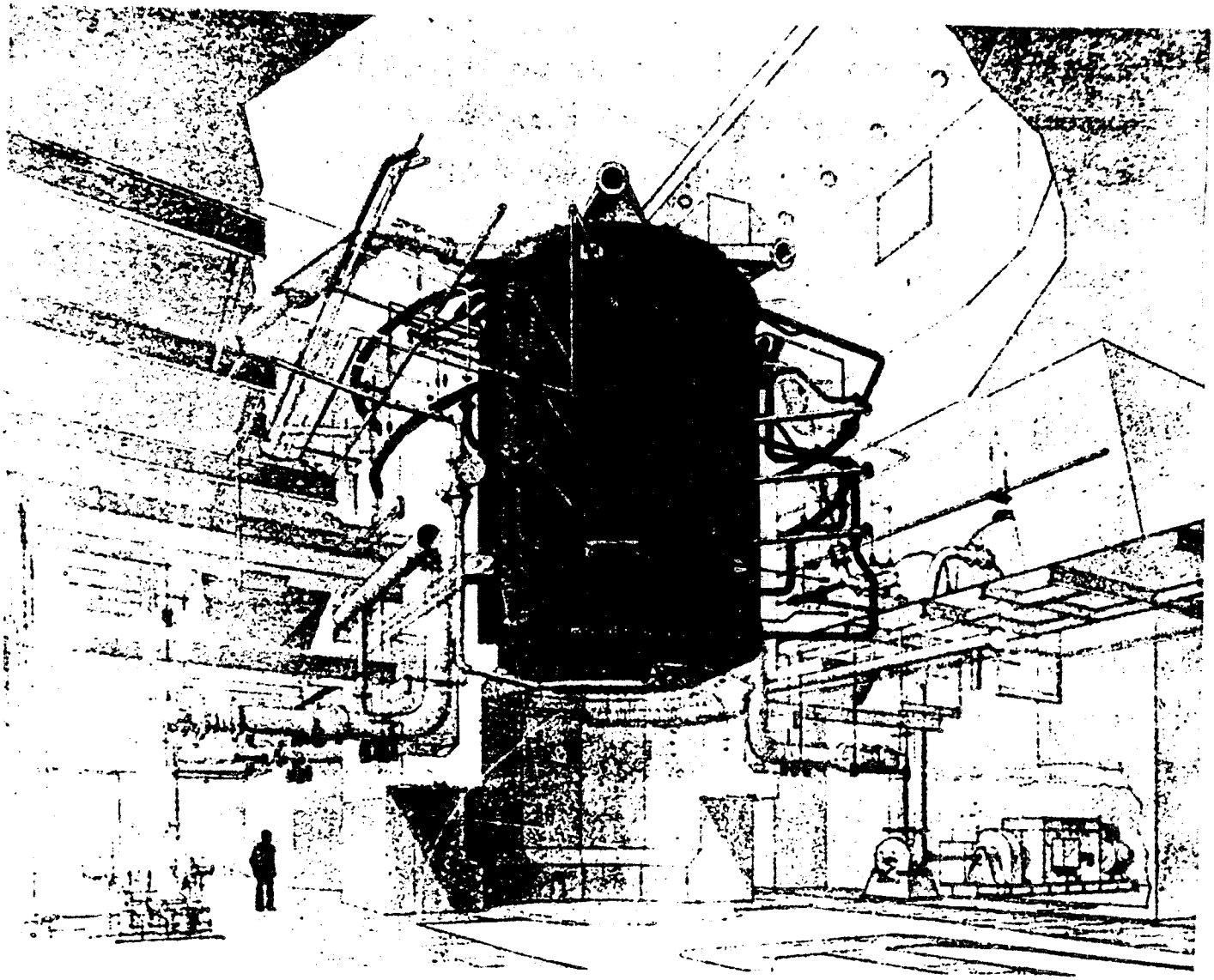


Fig. 2. **SAVANNAH RIVER REACTOR**

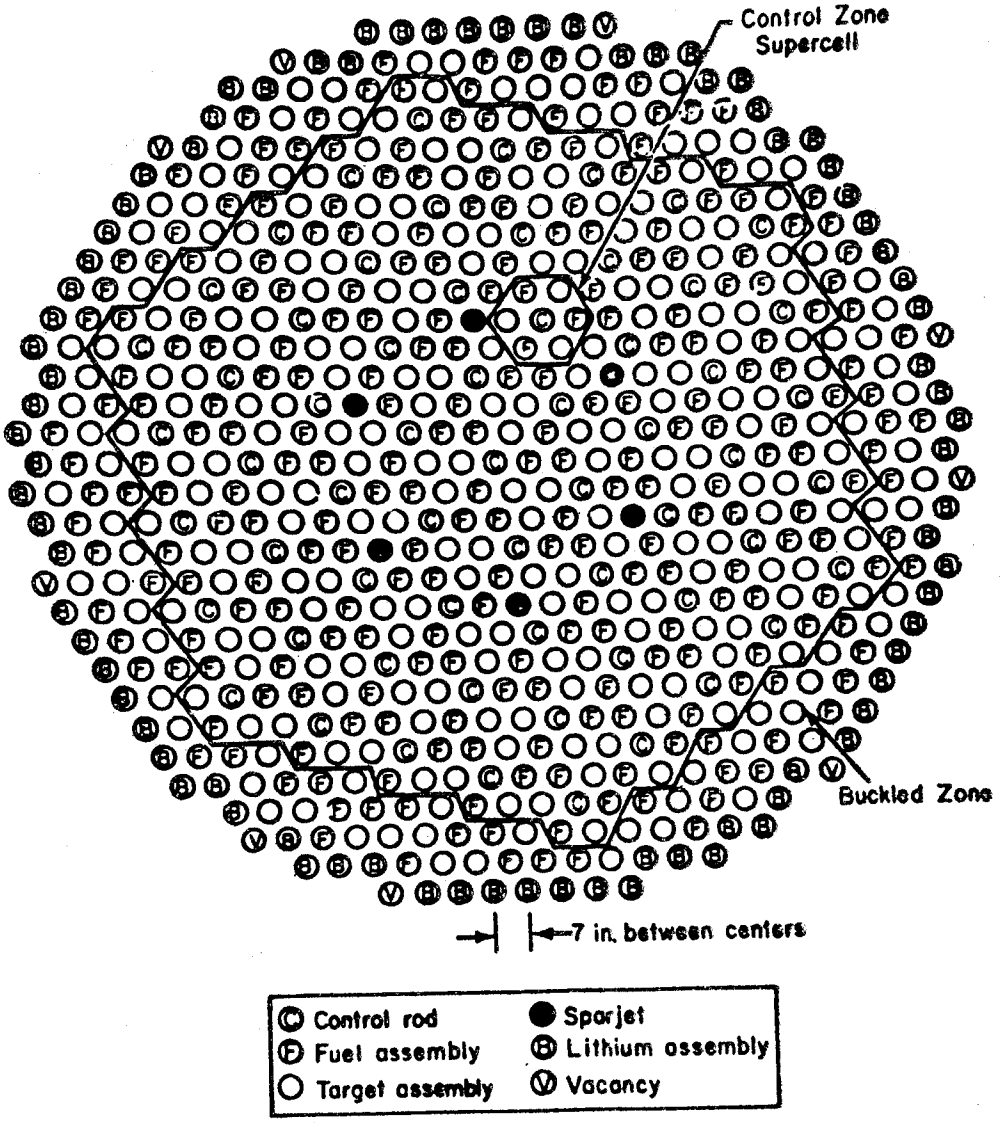
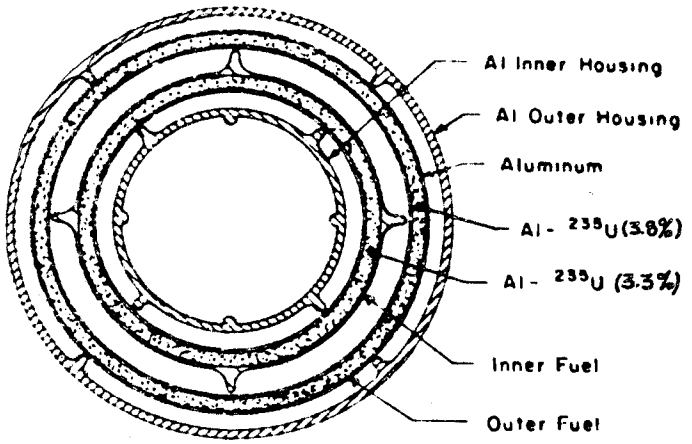
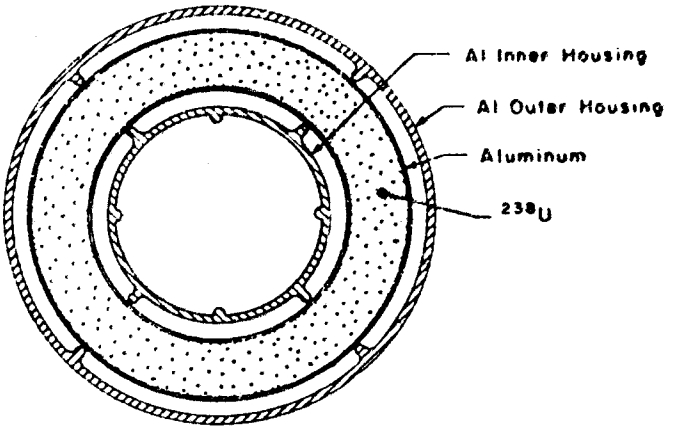


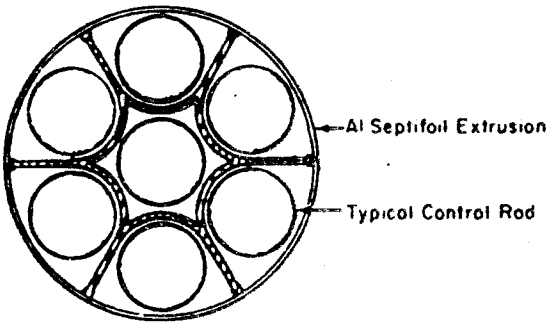
Fig. 3. Face Map of a Typical Mixed Charge Lattice for a Savannah River Reactor



a. Fuel Assembly



b. Target Assembly



c. Control Assembly

Fig. 4. Typical Savannah River Reactor Assemblies

of the tank. The D_2O, now acting as a moderator, flows upward, outward, and then downward in a three-dimensional path across the bank of assemblies to the exit nozzles. The reactor operates with single-phase flow throughout the reactor system. A vent is located above the reactor core but below the plenum, and maintains a low over-pressure on the reactor system. The vent system permits large resistance flow of D_2O from the reactor tank to an overflow system or to the process room above the reactor [14]. Five production reactors exist at the Savannah River plant, one or more of them can be assumed capable of tritium production or coproduction of plutonium and tritium. Three of these reactors are in operation and two are inactive. As shown in Fig. 1, no power generation is possible from these reactors. The pumps are driven by AC motors, and the heat generated is rejected from the heat exchangers to an ordinary water loop, probably river water. The reactors operate on an intermittent way at high power (11 days) and then are shut down for removal of the target elements. The temperature rise of the coolant is low and the reactor is operated at relatively low temperature. They are designed as special purpose systems and it is doubtful that they can be modified for continuous operation and power production. Not only so, but they need external power for reactor operation; not so for the Hanford reactors as explained later.

At Savannah River, ~93% enriched U-Al is used under high power operation to breed either Pu or tritium in target elements containing either ^{238}U or Li enriched in the ^{6}Li isotope. The target elements for tritium production are made of the compound LiAl (melting point 718°C). It exhibits a wide solubility range and the eutectic temperature with Al is approximately 600°C. Alternately, a solid solution

of lithium in Al might also be used. The low limit temperatures involved preclude the conversion of a Savannah River type reactor to high-temperature pressurized water operation. A maximum tritium production capability of $\sim 1.66 \times 10^{-2}$ kg/MWe yr has been estimated assuming a thermal efficiency of 30% [13].

The Savannah River Reactors may not be operational late in this century [6].

5. HANFORD GRAPHITE MODERATED TYPE REACTORS

We specifically consider the graphite-moderated, light water cooled reactor that was originally scheduled for initial operation solely for Pu production. However, it was designed in such a way that it could be converted with modifications to produce electrical power. The conversion of the otherwise wasted heat substantially reduces the cost of the product. Steam is supplied to an adjacent utility-owned electric generating plant. The conversion capability was considered at its design stage in the event that future international disarmament agreements dictated that the production of reactor products for military uses be curtailed or stopped [15]. Although there were rumors of placing the reactor on standby in 1977, it is still functioning in a dual-purpose capacity.

Table IV shows the economic parameters for the operation of the plant from a study on its conversion to production of Pu with high ^{240}Pu content [16]. The important observation is that the steam credits reduce the product cost by about $\frac{20.00 - 14.93}{20.00} \times 100 = 25.35\%$, a substantial savings. A Savannah River plant is at a disadvantage from this point of view, since it does not produce power and even needs to buy external power for its reactor's operation.

Table IV. Economic Parameters of the N-Reactor

(1972 Projected Figures)

Reactor Production:	
Power Level, MW(th)	4800
Burnup MWD/T	3000
Pu generated, kgs	1028
Uranium Throughput, Tons/yr	498
Production Costs (in $ Million)	
Off-site Costs	
Uranium	1.992
On-site Costs	
Fuel Fabrication	3.595
Irradiations	6.130
Separations	1.404
Burnout Costs	7.438
Total Product Cost	20.559
STEAM CREDITS	(5.208)
Net Product Costs Without Np Credits	15.351
Unit Costs Before Steam Credits, $/gm	20.000
Unit Costs After Steam Credits, $/gm	14.930
% ^{240}Pu Content of Product	15.400

A recent study has considered its conversion to coproduction of Pu and tritium, and tritium has actually been produced in it for a short period around 1967 [17,18]. Table V shows its production data [19]. For an operation at 4800 MWth, 815 kg of Pu can be produced per year together with 6.25 kg of tritium. At 30% thermal efficiency, this amounts to 4.34×10^{-3} kg/MWe•yr.

Assuming that all the produced Pu could have been replaced by tritium, this amounts to (Av is Avogadro's Number):

$$815 \text{ kg Pu} \times 10^{3} \frac{\text{g Pu}}{\text{kg Pu}} \times \frac{\text{Av}}{239} \frac{\text{atom Pu}}{\text{g Pu}} \times 1 \frac{\text{atom P}}{\text{atom Pu}}$$

$$\times \frac{3}{\text{Av}} \frac{\text{g T}}{\text{atom T}} \times 10^{-3} \frac{\text{kg T}}{\text{g T}} = 10.23 \text{ kg T/yr},$$

which is equivalent to 7.10×10^{-3} kg T/MWe yr. In this last case, higher enrichments may have to be used with the fissile fuel supplied by the fusion reactor, but we may obtain a total of $7.10 \times 10^{-3} + 4.34 \times 10^{-3} = 1.14 \times 10^{-2}$ kg T/MWe•year, which is much better than converted LWRs. This is not astonishing in view of the large conversion ratio of such a system in spite of its low fuel burnup of 1675 MWD/T.

One can calculate a value of $P_U/P_T = 25.35 - 39.17$ in case of production of tritium and Pu. However, in the case of tritium production alone, this is: $P_U/P_T = 9.65 - 10.26$. According to our theoretical results of Table I, such a system would have a value B_T slightly less than one (~0.9), which is the case and would require a fusion reactor with a value of B_U slightly larger than 1. If $c_T/c_U = 2$, a cost saving of 45% is achieved by the fusion-fission coupling compared to a pure fusion case. The excess cost over a pure fission system is about 10% which can well be compensated by simpler fusion reactor designs.

Table V. Data for Combined Tritium-Plutonium Production in the N-Reactor

	4800 MW	4000 MW[a]
PRODUCTION DATA		
Fuel Enrichment - % U-235		
In	2.10	2.10
Out	1.91	1.91
Fuel Exposure - MWD/T	1675	1664
Tons Uranium per Year	878	737
No. Fuel Assemblies per year (average length - 24")	60,140	50,480
Conversion Ratio - g Pu equiv/MWD		
Plutonium	0.554	0.548
Tritium	0.340	0.340
Production		
Plutonium - kg/yr	815	672
Tritium - kg/yr expressed as Pu equiv	500	417
Total Production - kg Pu equiv/yr	1315	1089
Tritium - kg/yr	6.25	5.2

(a) Fuel design capable of operation at reactor power level of 4800 MW.

Since such a system appears as the most promising compared to other cases, we give a description of such a reactor collected from several sources, since it cannot be found in a single one.

The reactor is basically of a simple design and consists of interlocking graphite bars acting as moderator and reflector, forming a stack of about 33-1/2 ft high, 33 ft wide and 39-1/2 ft long. The active core is about 25 ft high, 23 ft wide and 35 ft long. Figures 5 and 6 show face and side views of the reactor. The stacking pattern, as shown in Fig. 7, includes venting provisions to permit escape of the coolant which would be released in case a process tube ruptures. The vent channels have sufficient area to limit damage to the particular graphite bar in which the tube rupture would have occurred. There is a total of 1180 lattice units, 38 in the horizontal direction and 34 in the vertical direction, with 28 missing from each corner.

Gas plenums are placed between the moderator and the reflector graphite at the inlet and outlet of the stack as well as at both sides. A helium atmosphere is maintained normally, circulating through the active core from front to rear. The inert atmosphere prevents graphite oxidation, dries it when necessary, detects water and air leaks, prevents in-leakage of air by maintaining a positive pressure and provides a heat transfer medium for removal of moderator heat.

The stack, moderator and reflector are penetrated by 1003 process tube channels, extending from front to rear on an eight-inch horizontal by a nine-inch vertical lattice. A number of 86 horizontal control rod channels penetrate the graphite stack from side to side on a 32-inch horizontal by 36-inch vertical lattice. To achieve structural graphite temperature meeting contraction criteria, 640 moderator cooling channels enter the stack from side to side on a 16-inch horizontal by

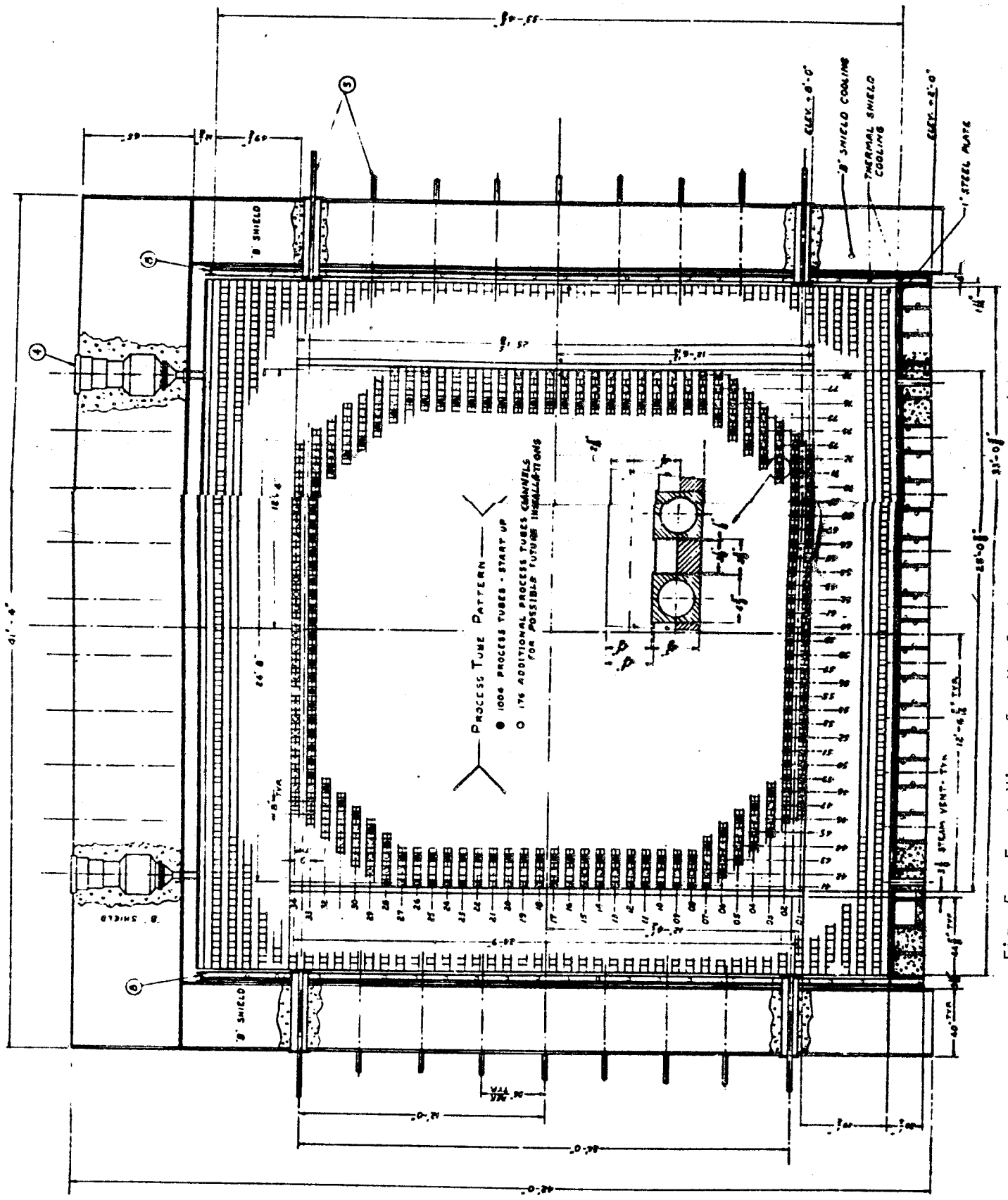


Fig. 5. Front View of a Hanford-Graphite Reactor

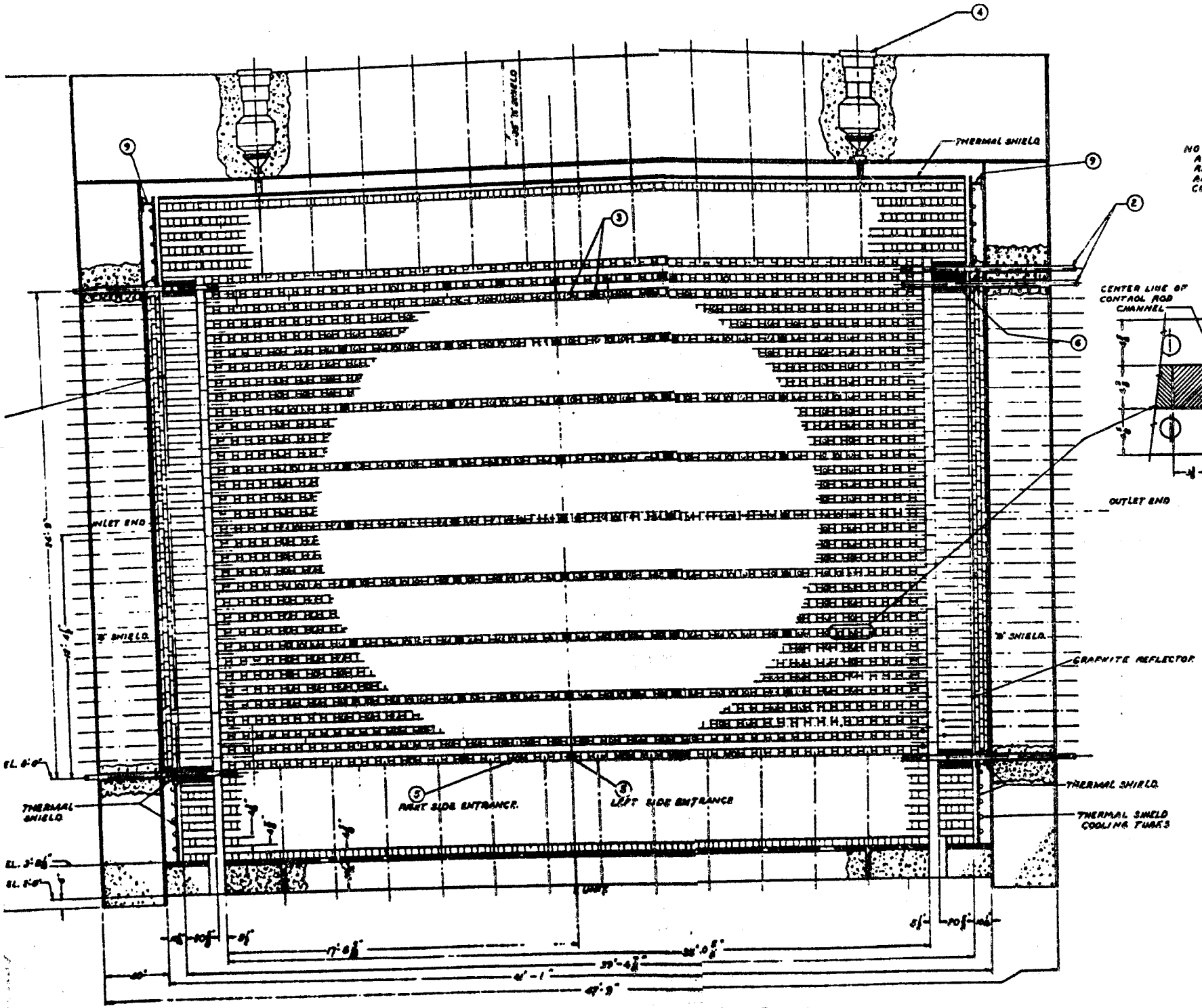


Fig. 6. Side View of a Hanford-Graphite Reactor

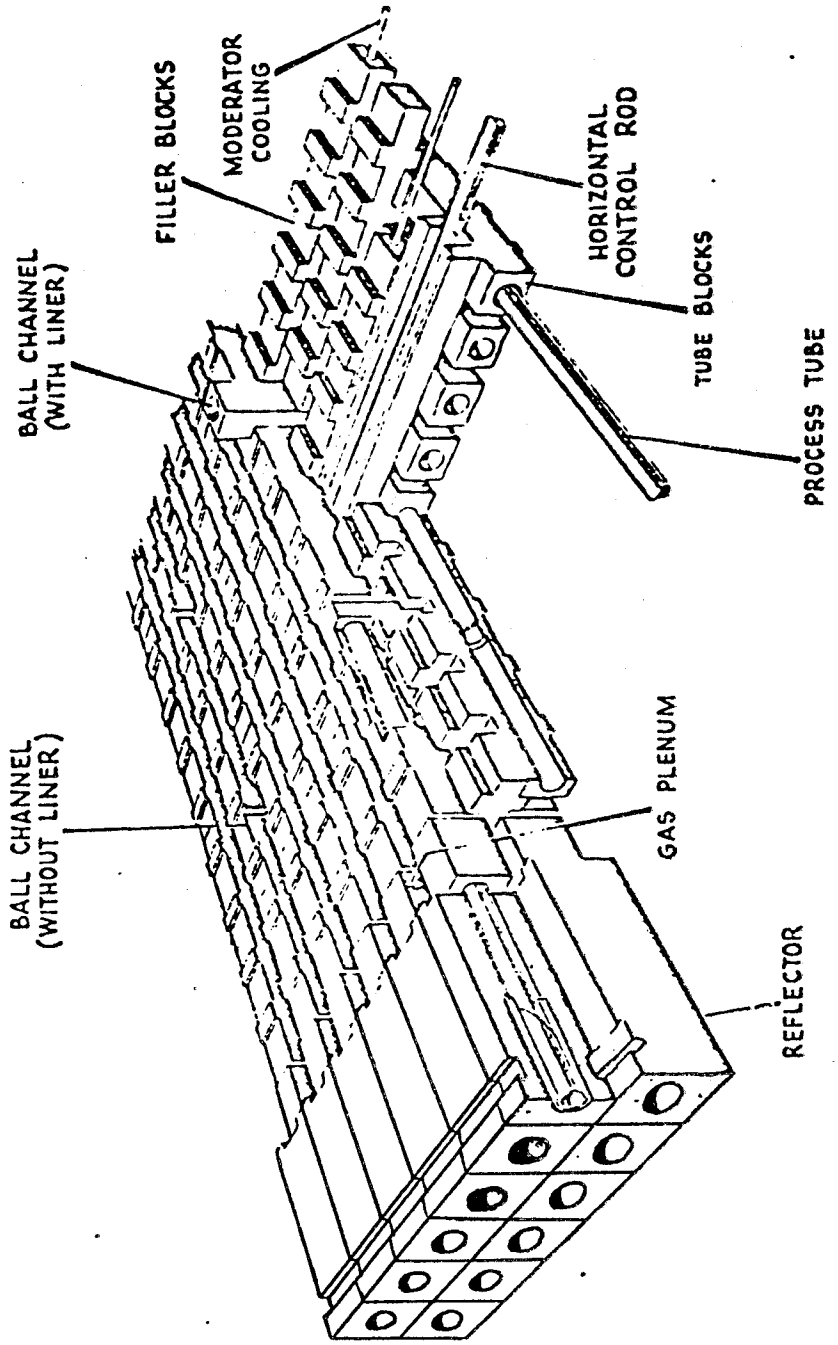
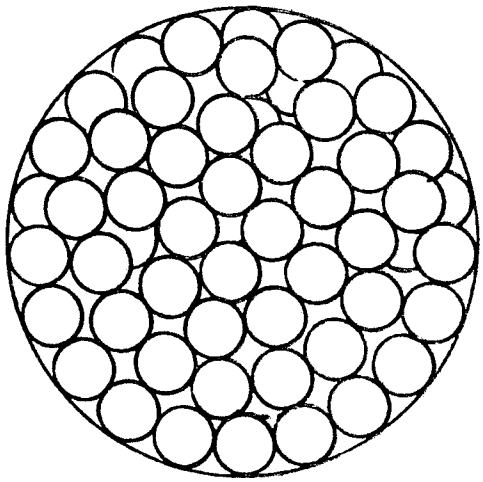


Fig. 7. Graphite Moderator Isometric

a 9-inch vertical lattice. Safety ball channels numbered 107, extending from top to bottom on a 32-inch by 32-inch spacing. These would be filled by 3/8-inch stainless steel boron containing balls (Fig. 8).

One-inch thick boron steel is used as the thermal shield material except at the inlet and outlet ends of the reactor where eight-inch thick cast iron blocks are used. Cooling tubes are welded to the boron steel plate. High density concrete is used as the primary shield on all sides of the reactor except at the bottom where ordinary concrete is used with thicknesses ranging from 40 to 72 inches. Personnel access is allowed only after shutdown. A biological shield surrounds the whole reactor structure and maintains radiation levels below 0.1 mr/hr in continuously occupied areas of the building.

Table VI shows the lattice physics parameters for coproduct tritium and Pu fuel and those of the standard fuel loading. Eta (η), the ratio of neutrons born to thermal neutrons captured in fuel, is higher in the coproduct fuel because of higher fuel enrichment (2.1%), although the difference is somewhat diminished by the buildup of Pu, which is a more pronounced effect in the standard fuel (9.947% enrichment). The fast fission factor (ϵ) is lower in the coproduct fuel because of the reduction in uranium content. The resonance escape probability (p) is higher in the coproduct fuel for the reduced U content. The thermal utilization (f) of the coproduct fuel is lower because absorptions in the target are considered parasitic for reactivity calculations—were these absorptions to be considered as fuel absorptions, the coproduct, f, would increase to about 0.92, or three percent higher than the standard fuel. The Fermi age (τ) increases in the coproduct case primarily because the area of the coolant passages is reduced by 15 percent, reducing the total slowing down power of



3.000" O.D. Cylinder

3/8" stainless steel balls @ 1½ % boron by weight or equivalent.

Fig. 8. Balls Control System

Table VI. Lattice Physics Parameters for Standard and Coproduct Operational Modes of the N-Reactor

Parameter	Lattice Physics Parameters			
	Cold		Hot, Operating	
	Standard	Coproduct	Standard	Coproduct
η	1.400	1.651	1.352	1.572
ϵ	1.039	1.029	1.040	1.030
p	0.836	0.866	0.820	0.848
f	0.866	0.736	0.893	0.748
τ (cm^2)	395	433	41[illegible]	448
L^2 (cm^2)	139	130	161	152
k_∞	1.053	1.082	1.029	1.028
k_{eff}	1.033	1.060	1.008	1.005
ρ, mk	32.2	56.3	7.9	5.2

the lattice. The thermal diffusion area (L^2) is decreased because of the increased blackness to thermal neutrons of the coproduct fuel. This discussion implies that for a system dedicated to tritium production higher enrichments will be needed, substantially changing the design parameters to the extent of needing a new design of the reactor.

Table VII shows the operating parameters and data of the system. The riser's configuration is shown in Figure 9. Demineralized ordinary water is used as a coolant. For a maximum credible accident where all water is lost to the reactor, the graphite system removes fission product decay heat and has a backup source of raw water. The coolant system is operated at low temperatures and pressures to permit the use of aluminum together with Zircaloy-2 tubing. A maximum outlet water temperature of 535°F is attained with a maximum fuel temperature of 492°C.

Driver and fuel elements are shown in Figure 10 and consist of a seven-rod cluster element. Two types of clusters are used. Cluster A, for driver fuel, is composed of slightly enriched uranium with Zircaloy-2 cladding. Cluster B, for target fuel, is composed of natural Uranium with Al cladding, for Pu production. Normal operation enrichment of the driver elements is 9.947%, but for the coproduction case, it is 2.1%. The discharge fuel enrichment is 0.68 - 0.74 percent. The fuel pile loading is 350-516 tons and the maximum fuel element length is 35 inches. The pressure tubes are made of Zircaloy-2 with 2.7 inch inside diameter and 0.25 inch minimum wall thickness.

There are two types of control rods, as shown in Figures 11 and 12. The first type is a boron control and safety rod composed of Boron Carbide and Aluminum. The second is a Lithium-Aluminum control rod. Both are enclosed in a steel sheath. Tritium will be produced in the

Table VII. Operating Data and Physical Parameters of the N-Reactor

	4800 MW	4000 MW[a]
OPERATING DATA		
Inlet Process Water Temperature - F	390	386
Maximum Outlet Process Water Temperature - F	567	535
Target Temperature - approximate	process water temperature	
Maximum Fuel Temperature - °C	548	492
Average Fuel Temperature - °C	434	396
Average Pressure Drop Across Fuel Column - psi	177	177
Rear Riser Pressure - psia	1500	1500
Process Water Flow per Channel - #/hr	94,400	94,400
Total Process Water Flow - Millions #/hr	94.7	94.7
Core Length - ft	31	31
Maximum Tube Power - MW (full load)	5650	4700
Maximum Tube Power - MW (transition)	5940	4950
Heat Flux (Driver) - Btu/hr/sq ft		
Inner Surface	785,000	659,000
Outer Surface	683,000	571,000
Heat Flux (Target)	Negligible	
Maximum Heat Flux/Burnout Heat Flux	<0.33	<0.23
Tube Maximum Power Density - kw/ft	240	208
Reactor Average Power Density - kw/ft	151	131
Gas Volume Ratio in Target	3.5	3.4
Steam Pressure - spia	150	150
PHYSICAL PARAMETERS		
Shutdown Margin (Rods Only) - mk	17	20
Control Rod Strength - mk	77	77
Ball System Strength - mk	62	62
Maximum Cold Xenon-Free Reactivity - mk	60	57

(a) Fuel design capable of operation at reactor power level of 4800 MW.

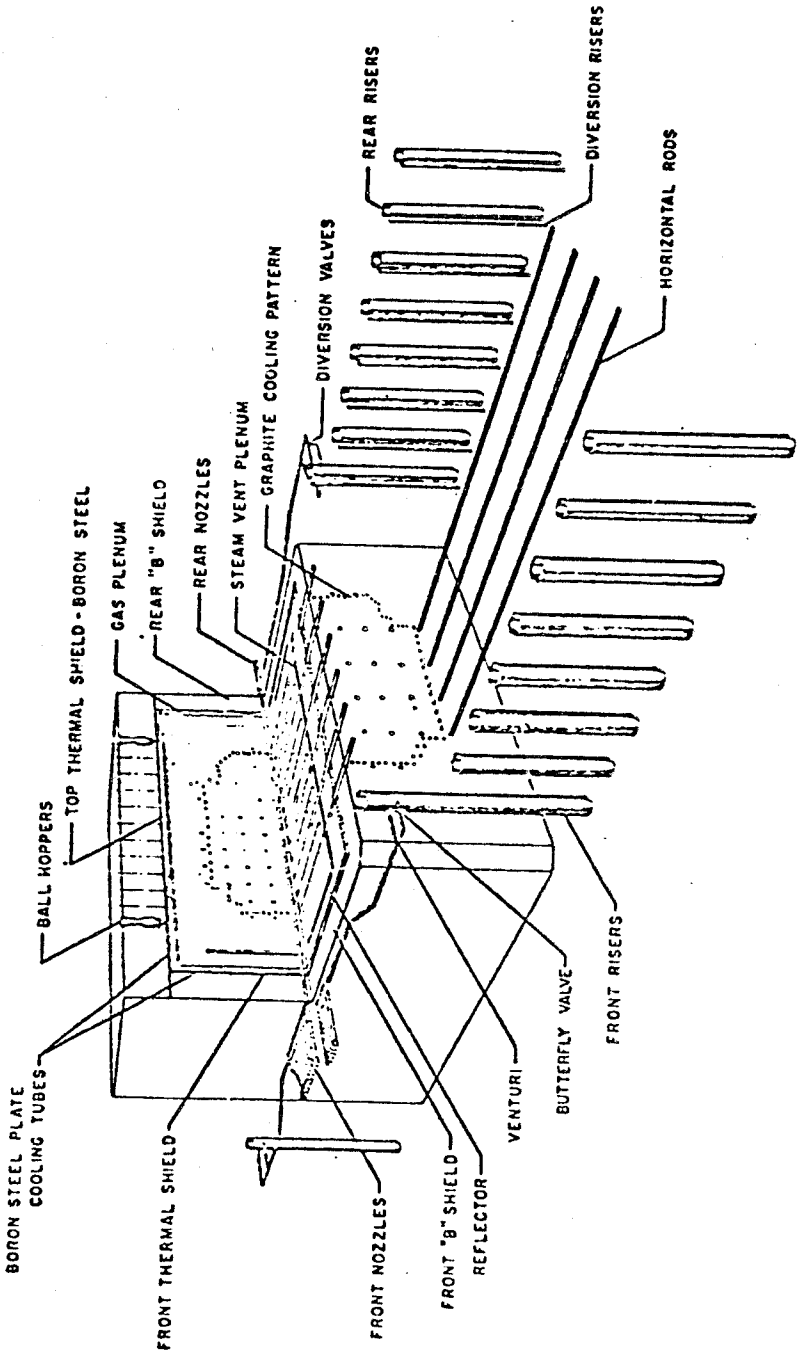
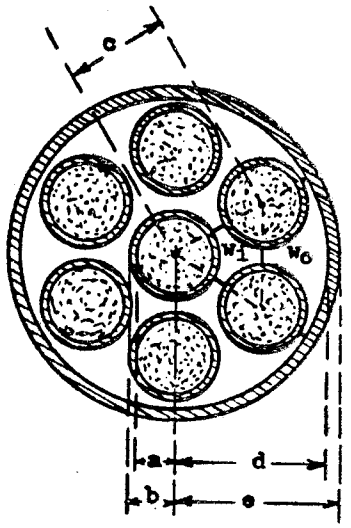


Fig. 9. Configuration of the 105-N Reactor



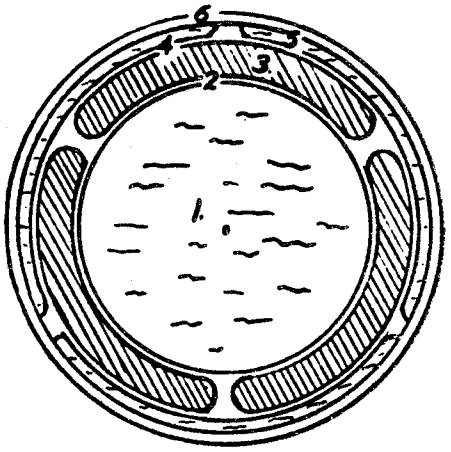
Cluster Dimensions

Cluster	a (in)	b (in)	c (in)	d (in)	e (in)
A	.250	.298	.648	.997	1.0605
B(771)	.322	.352	.851	1.35	1.60

Cluster A: Natural uranium, 28 aluminum tubes. 7 1/2 inch lattice

Cluster B: 0.95% enriched uranium, zircalloy-2 tubes. 8 3/8 inch lattice assumed in calculation.

Fig. 10. Driver and Target Cluster Configurations



BORON CONTROL & SAFETY ROD

Region	Material	
1	Water	Radius = 0.8835"
2	Aluminum	Thickness = 0.065"
3	Boron Carbide	0.218"
4	Aluminum	0.065"
5	Water	0.078"
6	Steel	0.065"

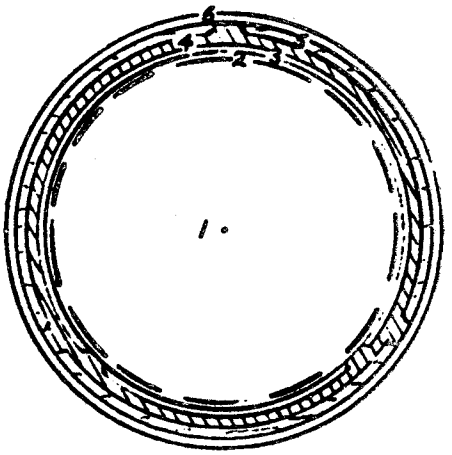
HCR Channel Dimensions

At tube block $4\frac{5}{8}" \times 4\frac{3}{8}"$

At filler block $4\frac{5}{8}" \times 6\frac{11}{16}"$

Average dimensions $4\frac{5}{8}" \times 5\frac{3}{8}"$

Fig. 11. Boron Control and Safety Rod



LITHIUM ALUMINUM CONTROL ROD

Region	Material	
1	Lithium Aluminum	Radius = 1.025"
2	Aluminum	Thickness = 0.040"
3	Water	0.102"*
4	Aluminum	0.065"
5	Water	0.078"
6	Steel	0.065"

* After irradiation, which results in swelling of lithium aluminum slug, water thickness will be reduced to 0.078".

Fig. 12. Li-Al Alloy Control Rod

Li-Al rods. In the studies conducted on coproduction of tritium and plutonium, Li-Al alloys with 3 wt% Li in Al and 41% enriched in ^{6}Li, as well as Li-Mg alloys were considered. Li silicates, aluminates and aluminosilicates were also studied. Double Zr and Al canning is used where the Al acts as an efficient barrier to the tritium produced from the $^{6}Li + {}_{o}n^{1} \rightarrow T + \alpha$ reaction.

Prototype lithium aluminate target elements shown in Figures 13 and 14 were irradiated in the N-Reactor. These elements consisted of 24-inch long Zircaloy-2 cans, 1.316-inch OD, having either 0.03- or 0.095-inch walls, each containing two aluminum-canned ceramic cores. The Al cans were approximately 1.25 inch OD, approximately 12 inches long and had a 0.050-inch wall. The $^{6}Li/Li$ weight percent at 78% bulk density was: 2.84, 1.59, 6.43 for the base, spike and poison (natural lithium) compositions, respectively.

Steam pressures of 75-500 psia were achieved. Steam is fed to turbine generators of 200-300 MWe each for a generating total capacity of 300-900 MWe net.

6. SUMMARY AND CONCLUSIONS

In a steady state operational mode of a system of coupled dedicated fusion reactor fissile producers, and fission reactor fusile fuel producers, the breeding capacities of fission and fusion reactors are related by: $B_U B_T = 1$. For a choice of B_U or B_T, a fission-to-fusion power support ratio can be deduced, as given in Eq. (7). In all cases of the coupling, a substantial saving in the power production costs can be expected compared to an overall pure fusion power producing system.

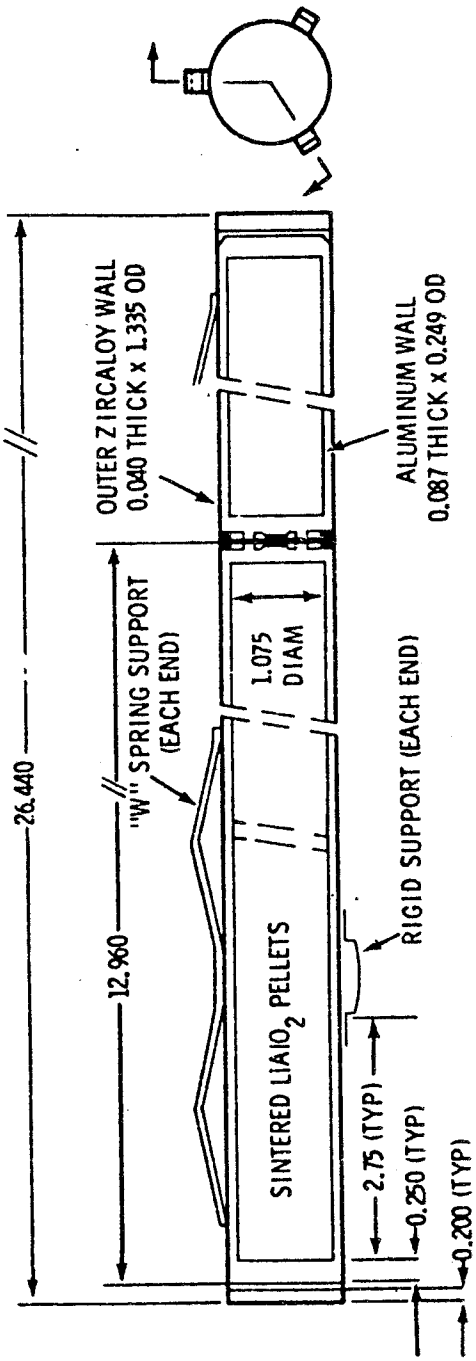


Fig. 13. Cross Section of Coproduct Target Element

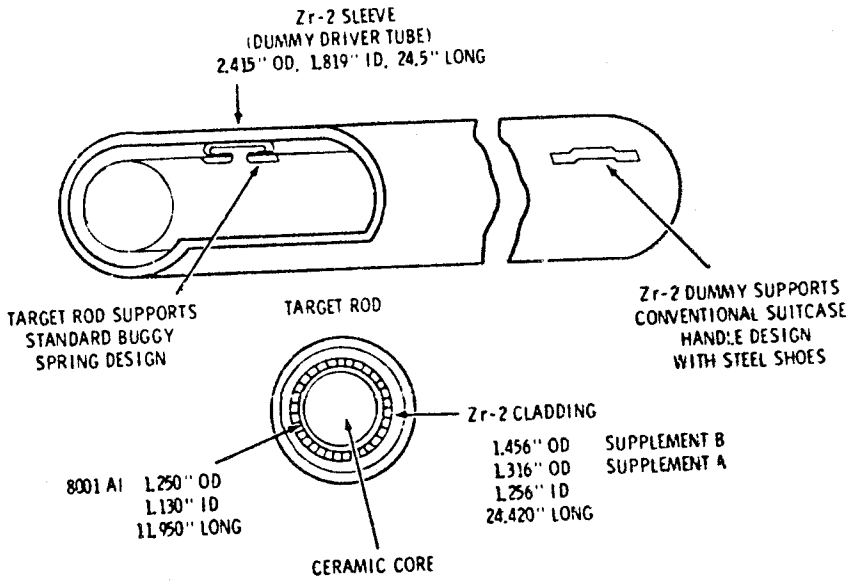


Fig. 14. Coproducer Target Test Element

The sources of tritium as a fuel to the fusion reactors are surveyed. Light water reactors are thought to need major modifications if tritium is to be produced in them, and do not seem to be likely candidates. Savannah-River type production reactors are not adequate, since they are not power producers. Graphite-moderated Hanford-type reactors produce power, and tritium has been produced in them from Li aluminates and seem to be the best option due to their high conversion ratio. However, these are of outdated designs, and a system that would easily replace them would be enriched-fuel CANDU-type reactors. The latter need to be modified for power and tritium production. Otherwise, advanced converters, such as graphite-moderated water or gas-cooled systems need to be developed for the implementation of such a concept, provided that their development implementation and operation costs are lower than that of the fusion reactors in the system. If this cannot be achieved, then it makes more sense to produce both the fissile and fusile fuel in the fusion reactors, and supply existing fission reactors with the produced fissile fuel.

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