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ORNL-EPR STUDY — RESULTS AND IMPLICATIONS*

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ABSTRACT

A two-year preliminary design study of a tokamak experimental power reactor has been completed by the Oak Ridge National Laboratory. The major engineering features, plasma physics characteristics, and technological requirements of the device are discussed. Plasma confinement is provided in a toroidal chamber of major radius, R_0 = 6.75 m and minor radius, a = 2.25 m. The toroidal magnetic field strength is 4.8 T. A unique poloidal magnetic field system creates the fields required for plasma equilibrium and stability. The power extraction system is centered around the blanket, which absorbs ∿ 90 percent of the energy produced in the plasma. The operating characteristics and nuclear performance of the system are given.

The results of the study indicate a low benefit-to-cost ratio for this design. Recent developments have suggested that some of the design constraints were too restrictive. The advisability of a large scale test of the ideas linked to these developments has become apparent. To this end, ORNL has started the design of a high-β tokamak. The basis for the high power density device is discussed.

INTRODUCTION

In the early 1980's the Tokamak Fusion Test Reactor (TFTR) will be operated at the Princeton Plasma Physics Laboratory. It will be the first magnetic confinement device to burn deuterium and tritium fuel and many of the scientific and technological principles required for the development of magnetically confined fusion power will be investigated for the first time in that facility. However, as the first of what will be a series of reactors, the TFTR will demonstrate some of the most basic feasibility features of a fusion reactor system. Many of the other feastures such as high duty cycle, plant availability, thermonuclear ignition, and energy conversion will be demonstrated in devices which follow the TFTR. One facility of this type will be the Experimental Power Reactor (EPR).

The objectives of the EPR are broad. From the scientific point of view it must generate fusion power by deuterium-tritium plasma reactions in a tokamak configuration. Technologically the EPR systems must be capable of being extrapolated to operate in a commercial reactor plant. With respect to engineering the EPR must demonstrate the ability to achieve safe, reliable, and cost-effective plant operation. A conceptual study aimed at developing a reference design consistent with these goals was completed by the Oak Ridge National Laboratory (ORNL) in June, 1976. Similar studies were performed by the Argonne National Laboratory (ANL) and the General Atomic Company (GAC). The studies have resulted in reference designs, have served as forcing functions both for the areas of science and technology, and have focused attention on the requirements for the device to follow TFTR.

The key features of the ORNL-EPR Reference Design (1), conclusions drawn from the study, and characteristics of the device now under design, which is itself an outgrowth of the study, are summarized in this paper.

THE OAK RIDGE EPR REFERENCE DESIGN

The EPR is based on the tokamak confinement concept. The device will contain an ignited deuterium-tritium plasma which produces a thermal power of about 400 MW during nominal operation. Superconducting

Fig. 1. Oak Ridge EPR Design

toroidal field coils encircle the plasma. The poloidal field system, which drives the plasma current and provides the additional magnetic fields necessary to maintain plasma equilibrium and stability during operation, includes both normal (copper) and superconducting coils. Those located inside the toroidal field coils are copper. The plasma current of 7.2 MA is induced by an air-core transformer. The burn cycle has a duration of 100 seconds and the duty cycle is 87 percent.

Figure 1 shows the EPR elevation. The major radius of the device is 6.75 meters and the plasma radius is 2.25 meters. A blanket surrounds the plasma and is capable of transferring 800 MW(th) from the burning plasma to a steam power cycle. This thermal output is equivalent to a source neutron wall loading of 1 MW/m^2 and a thermal power density in the plasma of 1.2 MW/m^3. (Increasing the power density is considered necessary and forms the basis for the current design work.) A shield located outside of the blanket and inside of the toroidal coils further attenuates the radiation produced. The neutron flux coming through the blanket and shield is attenuated by a factor of 2000.

An external means of heating is required to raise the plasma temperature to the level required for thermonuclear ignition. A neutral beam injection system, which deposites 50 MW of power into the plasma in the form of 200 keV deuterons, is used in the EPR design. Fast neutral atoms are injected since they pass freely through the magnetic fields, enter the plasma chamber, and are ionized and trapped by the plasma. Six beam lines, with two ion sources per line, supply the power needed during the 5-second heating phase of operation.

The primary vacuum system establishes a base pressure of 2×10^{-10} torr prior to the initiation of a sequence of pulses and reduces the pressure between pulses (for plasma purity reasons) by a factor of 1000. There are two vacuum ports located between each pair of toroidal field coils. A cryosorption system is used and 40 one-meter diameter cyropanels provide a net pumping speed of 1.1×10^6 liters/second. The roughing vacuum system consists of three 5000-cubic feet/minute Roots Blowers coupled to three 500-cubic feet/minute mechanical pumps.

The EPR will be fueled by deuterium and tritium. A tritium handling system is provided to process about 2 kg of tritium per day. It includes equipment to evacuate and maintain a vacuum in the plasma chamber and neutral injectors, to purify and recycle tritium and deuterium for the fuel cycle, to recover tritium from experimental breeding modules, and to provide tritium containment and atmospheric clean up.

Table 1 shows the major parameters for the EPR. Figure 2 shows schematically one full operating cycle. Note that the initial start-up procedure only occurs at the beginning of a sequence of pulses and that the normal operating cycle is 115 seconds in length.

TABLE 1 EPR PARAMETERS

A. General Characteristics	
1. Requirements	
Device lifetime	10 years
Duty cycle	87%
Availability — normal	80%
Thermonuclear power (nominal)	410 MW
Thermonuclear power (design)	800 MW

TABLE 1 (cont'd)

2. Machine Parameters	
Plasma radius	2.25 m
Major radius	6.75 m
Aspect ratio	3.0
Plasma edge to winding distance	1.40 m
Plasma volume	675 m^3
Overall height	15 m
Overall diameter	23 m
B. Plasma Parameters	
Burn time	100 sec
Current	7.2 MA
Ion temperature	12.2 keV
Electron temperature	13 keV
Electron density	$7.4 \times 10^{13} cm^{-3}$
Fuel ion density	$7.0 \times 10^{13} cm^{-3}$
Impurity level (Z_{eff})	1.34
Injection power into plasma	50 MW
Injected deuteron energy	200 keV
Particle confinement time	9.6 sec
Energy confinement time	3.7 sec
C. Electromagnetic Parameters	
Maximum toroidal field	11 T
Toroidal field on axis	4.8 T
Number of toroidal coils	20
Magnetic field ripple at plasma edge	2.2%
Average poloidal field at plasma edge	0.64 T
Poloidal coil volt-seconds	165 Wb
D. Thermal Hydraulics	
Maximum power rating of blanket	800 MW
Blanket coolant	helium
Outlet temperature of blanket coolant	370°C
Blanket coolant pressure	70 atm
Shield coolant	borated water
Coil coolant (superconducting)	helium
E. Nuclear Parameters	
Neutron wall loading (nominal)	0.55 MW/m^2
Neutron wall loading (design)	1.07 MW/m^2
Tritium breeding ratio (experimental module)	1.15

THE EPR MAGNET SYSTEMS

The toroidal field (TF) coil system provides the axial field required for plasma stability and confinement. It is anticipated that superconducting coils will be necessary for this application in a full-scale reactor power plant to maintain a favorable energy balance; thus, superconducting TF coils are used in the EPR design. Although the dependence of plasma confinement time on magnetic field strength is at present unknown, there is strong reason to believe that confinement time will increase with an increase in field strength. An additional incentive for high magnetic fields exists because increased field strength may permit the operation of higher density

plasmas, and therefore higher power densities. Considering existing superconductors and those under development, a maximum field at the coil winding of 11 T was used in the design. The corresponding field strength on the plasma axis is 4.8 T. The plasma size and the blanket and shield thicknesses and tolerances necessitate a TF magnet bore > 7 meters.

Fig. 2. EPR Operating Chart.

The TF coil conductor uses multifilamentary Nb_3Sn to achieve the maximum design field of 11 T. NbTi is used in the lower field regions of the conductor. The conductor is in a cable configuration and is cooled by forced-flow supercritical helium that flows in strand interstices. Each conductor is housed in a stainless steel hollow conduit which serves as a helium-tight jacket and as the support structure. The technique for cryostabilizing a bundled cable housed in a hollow conduit using forced-flow supercritical helium is a new concept developed at the Massachusetts Institute of Technology (2). Figure 3 summarizes the features of the TF coil configuration. A more general discussion of superconducting magnet development for fusion research is given in Ref. (3).

The poloidal field (PF) coil system establishes and maintains the plasma current during a pulse and provides the fields necessary for plasma equilibrium and stability. The plasma current is driven by transformer action where the plasma represents a 1-turn secondary. The primary includes the ohmic heating (OH) coils which produce and control the magnetic flux swing. A decoupling coil prevents electromagnetic "short circuiting" of the plasma by the shielding-vertical field (S-VF) coils which are located near the plasma boundary. The trim-vertical field (T-VF) coils guide the flux from the air core outside of the TF coils and adjust the equilibrium field pattern to that precisely required. Figure 4 shows the relative locations of these four coil systems. The conductor cable for the PF coils (except the S-VF coils, which are normal) is NbTi in Cu and a mixed-matrix of Cu-Ni to reduce ac coupling losses since these are pulsed systems.

The electromagnetic shielding feature of the PF system is unique. This system minimizes the volt-seconds required to set up the plasma current, permits an increase in pulse length and duty cycle, and reduces the pulsed field at the superconducting TF coils by a factor of about 6. Since the S-VF coils are closely coupled to the plasma, they will respond rapidly to plasma motion. Use of the magnetic shielding scheme reduces power supply requirements by 41 percent and stored magnetic energy by 34 percent compared to the requirements of a similar system without the shielding feature.

Fig. 3. Winding and conductor configuration.

Fig. 4. Poloidal field system coil orientations.

THE EPR HEAT SOURCE

The heat source in the EPR system is the ignited deuterium-tritium plasma. The burning plasma produces a flux of 14 MeV neutrons which escapes into the blanket and 3.5 MeV alpha particles which are confined in the

plasma chamber, thermalize due to collisions with the background plasma, and provide the energy input needed to balance energy losses.

The plasma system is modeled as a multi-fluid mixture. Physics models for the various loss processes which occur in the plasma are not well understood. At present theoretical estimates of transport coefficients and radiation effects must be used as the basis for reactor plasma simulations. The detailed time-dependent model used in this study is discussed in ref. (4).

Figure 5 shows the ion temperature of the reference system vs time for various assumptions relative to impurity control. The introduction of impurities into the plasma has several detrimental effects on the plasma energy balance; impurities enhance radiation losses, provide a source of atomic radiation, and decrease the fuel ion density relative to the electron density. In the reference case (No. 1) shown in Fig. 5, a stainless steel wall is assumed, the impurities reside in the plasma for one deuteron confinement time, and the neutral density at the plasma edge is $10^8/cm^3$. Under these conditions a 90 percent-efficient divertor (i.e., an attenuation of 90 percent of the charged particles which diffuse toward the wall and a 90 percent attenuation of returning impurities) permits a 100-second burn in the EPR at nominal conditions (cf. Table 1). Variations from this base case lead to limited burn times (typically on the order of 40 seconds, except in case 5). It is the use of the low-Z wall coating or liner concept which permits long burn times to be assumed in the EPR without incorporating a divertor system.

For this discussion the plasma is considered a source of energy deposited on the blanket first wall mostly by conduction and radiation and throughout the blanket and shield by the neutron and subsequent gamma ray fluxes produced.

Fig. 5. Time-dependent ion temperature for several conditions of impurity control.

NUCLEAR CRITERIA AND DESIGN

The essential functions of the nuclear systems of the EPR are to absorb greater than 90 percent of the available plasma energy and convert this energy to heat at high temperature, to attenuate the plasma neutrons and provide radiation shielding to protect other system components, and to provide a system where tritium breeding can be demonstrated in at least a few experimental modules.

A minimum requirement imposed on the design is that the first structural wall handles safely the cyclic heat loads associated with power levels of several hundred megawatts, while holding impurity influx to the plasma to a rate that permits fusion pulses of the required length. A design objective is to handle the anticipated heat loads safely and reliably, using configurations and materials that can be adapted to higher power reactors.

The blanket will quickly become highly radioactive due to neutron reactions; thus, reliable, tested, completely remote means of assembly and disassembly of the blanket and shield must be provided. For the size devices under consideration, the blanket will be massive and modular construction will be essential.

The reference material for the EPR blanket and shield structure is type 316 stainless steel. The design data base for this material is more extensive than for any other candidate material, and some data are available on its high-temperature properties after irradiation to high levels of both displacement damage and helium contents. The major concern at this time is fatigue properties. No fatigue tests have been conducted on materials irradiated to high dpa levels and high helium contents. The most important parameter in determining low-cycle fatigue behavior is ductility, and tensile and creep-rupture tests show that the ductility of 316 stainless steel is reduced at nearly all temperatures. It is thus expected that fatigue life will also be adversely affected.

It would be highly desirable to maximize the ductility and minimize swelling of the structural material. Previous work has shown that in austenitic stainless steels the swelling rate is extremely sensitive to composition, especially the collective influence of minor elements such as carbon, nitrogen, sulphur, manganese, and silicon (5). Preliminary evidence (6) indicates that small additions of titanium can improve the ductility of a material with both a high dpa level and a high helium content. These results suggest that an optimum alloy similar to type 316 stainless steel can be developed that will perform satisfactorily for the life of the EPR, provided stress levels can be reduced sufficiently in the design.

The blanket design consists of 60 blanket segments forming an annulus around the torus. Each of the 60 blanket segments is subdivided into 12 blanket modules in the poloidal direction. Each of the 12 modules has its own autonomous cooling circuit (using helium to cool the bulk of the blanket). The 12 coolant circuits in each blanket segment are in parallel, and each is connected to a helium inlet and outlet header. These 12 modules are bolted to a skeletal framework which consolidates the modules into a self-contained blanket segment. The skeletal framework is designed to support the blanket modules so that they project toward the plasma with support entirely from behind, thereby minimizing the exposure of the framework to neutrons. An exploded view of the blanket configuration is shown in Fig. 6. The vacuum boundary is located on the inboard side of the blanket. The vacuum wall is designed to be cooled separately from the blanket.

one-dimensional discrete ordinates code ANISN, a P_3 scattering expansion, and an S_{12} angular quadrature. The transport cross sections were taken from the coupled 100 n - 21 γ cross section library (7) prepared for EPR calculations and compiled from the latest available END/B-IV data.

Fig. 6. Schematic of blanket construction.

A side view of a typical module is shown in Fig. 7. There are basically three regions in each module: 1) a helium-cooled liquid metal (potassium or lithium) region as the principal neutron absorber (breeder), 2) a graphite reflector, and 3) a gamma shield region made from concentric cylinders of stainless steel. The annuli between these cylinders are filled with liquid metal. As indicated, each blanket module is cooled autonomously. The coolant is helium at 70 atm, which flows in numerous ¼ in. stainless steel coolant tubes which traverse the inside front surface of the module and are then directed as indicated in Fig. 7 until they exit in an outlet header near each corner. The three principal regions of each module and the coolant tubes are contained in the stainless steel module structure and each module is supported by an outer support frame. All of the internal structure of each blanket module is supported from the top plate. Each blanket module has the same thickness (i.e., inboard to outboard) of ∿ 51 cm and the same arc length of ∿ 1.18 m). However, the width varies depending on the poloidal location of the blanket module in the blanket segment.

Mounted on the inboard surface of each blanket segment is an independently-cooled, thin vacuum vessel. The wall of the vessel is a composite stainless steel structure; as envisioned for EPR it would be about 1 cm thick with a honeycomb interior enclosed between two stainless steel surface plates, each about 0.16 cm thick. The present plan would be to circulate water through the interstitial spaces of the honeycomb in the interior of the wall in order to remove the heat generated by the energy deposited on the surface.

The shield consists of a "can" structure made of 316 stainless steel. This structure contains stainless steel spheres cooled with borated water which is circulated through a heat exchanger to remove the heat generated in the shield. Adjacent to and bolted to the outside shield container wall is a 6.2 cm layer of lead.

The neutronics and photonics calculations for the blanket and shield were performed using the

Fig. 7. Blanket module (side view).

Table 2 presents a comparison of results for the present design with the water-cooled vacuum vessel. Tabulated are values of fractional heat (in percent) in the various regions for three absorber materials: potassium (35 volume percent) and stainless steel (65 volume percent), pure potassium, and lithium. The cases with potassium represent non-breeding blanket modules while the case with lithium represents blanket modules where tritium breeding would occur. These cases indicate the distribution of heat fraction in the major regions of the blanket and in the toroidal field coils.

Also given in Table 2 are estimates of displacements and gas production in the stainless steel first structural wall. The data are presented for a neutron wall loading of 1.0 MW/m^2. The atomic displacement rates of 10 — 11 dpa/year are consistent with values obtained for earlier EPR blanket designs investigated at ORNL and are also consistent with estimates obtained by other designers. The gas production values of ∿ 400 appm for hydrogen and ∿ 130 appm for helium are also consistent with earlier estimates.

The tritium breeding ratio for the case where lithium is the absorber is also presented. Note that the value of 1.15 does not account for the adverse effect of penetrations on the breeding; it is estimated that penetrations will amount to about 5 percent of the blanket volume.

HEAT TRANSFER AND THERMAL HYDRAULICS

Alternative coolants have been considered for the EPR application. The heat transfer and thermal hydraulic design assumes low-pressure, low-temperature water as the coolant for the vacuum vessel and helium gas at 70 atm as the coolant for the bulk of the blanket.

In the vacuum vessel system, the ΔT is ∿ 80 C, with an inlet temperature of 38 C and an outlet temperature of 121 C. The water flows at ∿ 50 psia.

TABLE 2 FRACTIONAL HEATING RATE IN THE CURRENT EPR DESIGN AS A FUNCTION OF BLANKET ABSORBER MATERIAL

	Absorber Material		
	K + SS-316	Li	K
Total Heating Rate (W/n/sec)	3.18×10^{-12}	2.73×10^{-12}	2.83×10^{-12}
MeV per neutron	19.84	17.04	17.66
Fractional Heating Rate in Reactor Components (%)			
Graphite curtain	1.57	1.56	1.61
Vacuum vessel with water coolant	5.32	5.52	5.30
Blanket			
First structural wall	11.89	8.72	10.78
Absorber compartment	65.89	49.59	30.79
Reflector[1]	6.19	15.72	24.84
Gamma-ray shield	6.02	12.27	16.56
Total	89.99	86.20	82.97
Shield	3.11	6.61	10.03
TF Coil			
Inner SS-316 structure	5.88×10^{-4}	3.37×10^{-3}	5.68×10^{-3}
TF coil windings	8.90×10^{-4}	5.31×10^{-3}	8.67×10^{-3}
Outer SS-316 structure	1.05×10^{-5}	6.45×10^{-5}	8.00×10^{-5}
Radiation Damage			
Displacements per atom/year[1]	10.9	10.7	10.1
Gas production (appm/year)			
H	424	452	421
He	129	134	129
Tritium Breeding Ratio	—	1.15	—

[1]Based on an effective displacement energy of 40 eV.

The surface heat flux is ∿ 27 W/cm^2, and the neutron and gamma energy heat load that must be handled by the water coolant system is about 7 percent of the total. This system appears capable of handling the heat loads associated with the assumed power levels (i.e., ∿ 1 MW/m^2) and in fact appears capable of handling much higher heat loads.

Numerical calculations of the temperature distributions in the blanket were performed. Transient cyclic calculational results for selected duty cycle cases are presented in Table 3. Results are given for three cases with a plasma burn time of 100 seconds and downtimes varying from 10 – 100 seconds; results for a case with the plasma burn time increased to 200 seconds and a 10-second downtime are also given. The duty cycle selected for EPR consists of a plasma burn of 100 seconds and a downtime between pulses of 15 seconds. The front plate of the vacuum vessel experiences variations of < 20 C, depending on the duty cycle. The temperature variations in the stainless steel structural members of the blanket itself are < 15 C.

The temperature distribution through the blanket during a given cycle of the 100/10 (burn/downtime) duty cycle is presented in Fig. 8. These results indicate that temperatures remain below acceptable upper limits at all locations in the blanket. The results also show the magnitude of the temperature variation to be expected during the cycle in the different regions of the blanket. As indicated by these results, temperature variations are for the most part modest in all blanket regions. However, in the separately cooled vacuum vessel significant cyclic temperature variations do occur under the design conditions evaluated. An additional design effort is warranted to identify changes that could permit operation with reduced cyclic temprature variations.

POWER CONVERSION AND HEAT BALANCE

The EPR has a nominal thermonuclear power output of 410 MW(th) at a duty cycle of 87 percent [equivalent to an average power input of 356 MW(th)], all of which was initially assumed to be transferred from the blanket to the power conversion system. Twenty percent of the power is transferred there by the process of water-cooling the first wall, and the remaining 80 percent is transferred by the process of helium-cooling the liquid metal absorber portion of the blanket. The specification of the initial power conversion system and associated heat balance was based on blanket coolant temperatures of 38 C inlet and 121 C exit for the water, and 204 C inlet and 371 C exit for the helium.

An analysis based on these parameters indicates that utilization of the thermal energy from the pressurized water loop of the first wall as input to a power conversion cycle would not be practical.

TABLE 3 THERMAL CYCLING RESULTS FOR SELECTED LOCATIONS

Times Within the Cycle		Average Output Power	Vacuum Vessel Front Surface		Blanket Structure Front Surface		Lithium Peak	
Burn (sec)	Down (sec)	[MW(th)]	Temp (°C)	Var (°C)	Temp (°C)	Var (°C)	Temp (°C)	Var (°C)
100	100	400	131	± 47	278	± 35	423	± 27
100	50	533	130	± 35	297	± 29	490	± 17
100	10	727	136	± 14	317	± 7	555	± 4
200	10	762	137	± 13	318	± 7	571	± 3

Fig. 8. Temperature distributions in blanket for a 100-second plasma burn.

Thus, this energy was set up in the design to be dissipated to the environment by evaporate cooling provided by mechanical draft towers. For the 285 MW (th) of energy removed from the blanket by the helium loop, the compressor power needed for circulating the helium was estimated to be 6.7 percent of the blanket thermal energy or 19 MW. Thus 304 MW(th) is available as heat input to the steam generator. An overall energy schematic of the thermal energy from the blanket is given in Fig. 9.

The cyclic operation of the reactor will result in variations in the temperature of the helium coolant. A thermal energy storage system is required to minimize variations in the rate of delivery of thermal energy to the power conversion cycle. An intermediate heat transfer and energy storage loop would introduce an additional temperature difference between the heat source and the fluid in the steam generator, however, and would consequently result in a decreased thermal efficiency. An alternative considered in this study, which avoids the penalty of an additional temperature difference, is a helium branch loop for energy storage. This concept would produce helium at essentially a constant temperature at the steam generator by varying the helium flow rates through the branch and parallel flow paths during the operating cycle of the reactor. The necessary 70 atm helium storage capacity of the branch loop is estimated to be 934 m^3.

Fig. 9. Overall energy schematic for thermal energy from EPR blanket.

EPR RESULTS AND THE NEXT TOKAMAK STEP

In the assessment of the EPR it is clear that the basic plasma engineering assumptions concerning minimum size and field for ignition play a dominant role in setting the overall size and difficulty of EPR. Recent advances, both experimental and conceptual, have led to a reconsideration of these basic assumptions. The advances indicate: 1) that it may be possible to achieve higher ratios of plasma to magnetic field pressure than had previously been supposed, and 2) that operation at high density could provide improved performance. In setting up the basis for the next tokamak step, use will be made of the benefits of high density plasma operation and close attention will be given to achieving it. More information on the scientific basis for the postulated (8 and 9) high-density, high-beta operation will be forthcoming from

the next generation of large experiments now under construction.

ORNL has begun the design of a machine which will follow the TFTR. The device, referred to as TNS (The Next Step), will generate a reactor core plasma using moderate extensions of the technology that will be qualified by ∿ 1980. The device will contain a burning deuterium-tritium plasma and produce about 1000 MW(th). Superconducting coils will be employed, permitting a pulse length on the order of 30 seconds. Using the principles of flux conservation the plasma pressure will be 3 – 4 times that normally envisaged for low-beta systems. The plasma shape will be non-circular, i.e., moderately elonaged by a factor of about 1.5. The increased power density and neutron wall loading will lead to a smaller device size than that projected for the EPR. Presently TNS plasma volume is expected to be one-third of that in the EPR. The features of high duty cycle, plant availability, and high temperature energy conversion will be reserved for the EPR itself. TNS will focus attention on the fusion plasma and its supporting technologies and will demonstrate that when operated in this way, the tokamak can be extrapolated to a viable reactor system.

ACKNOWLEDGMENTS

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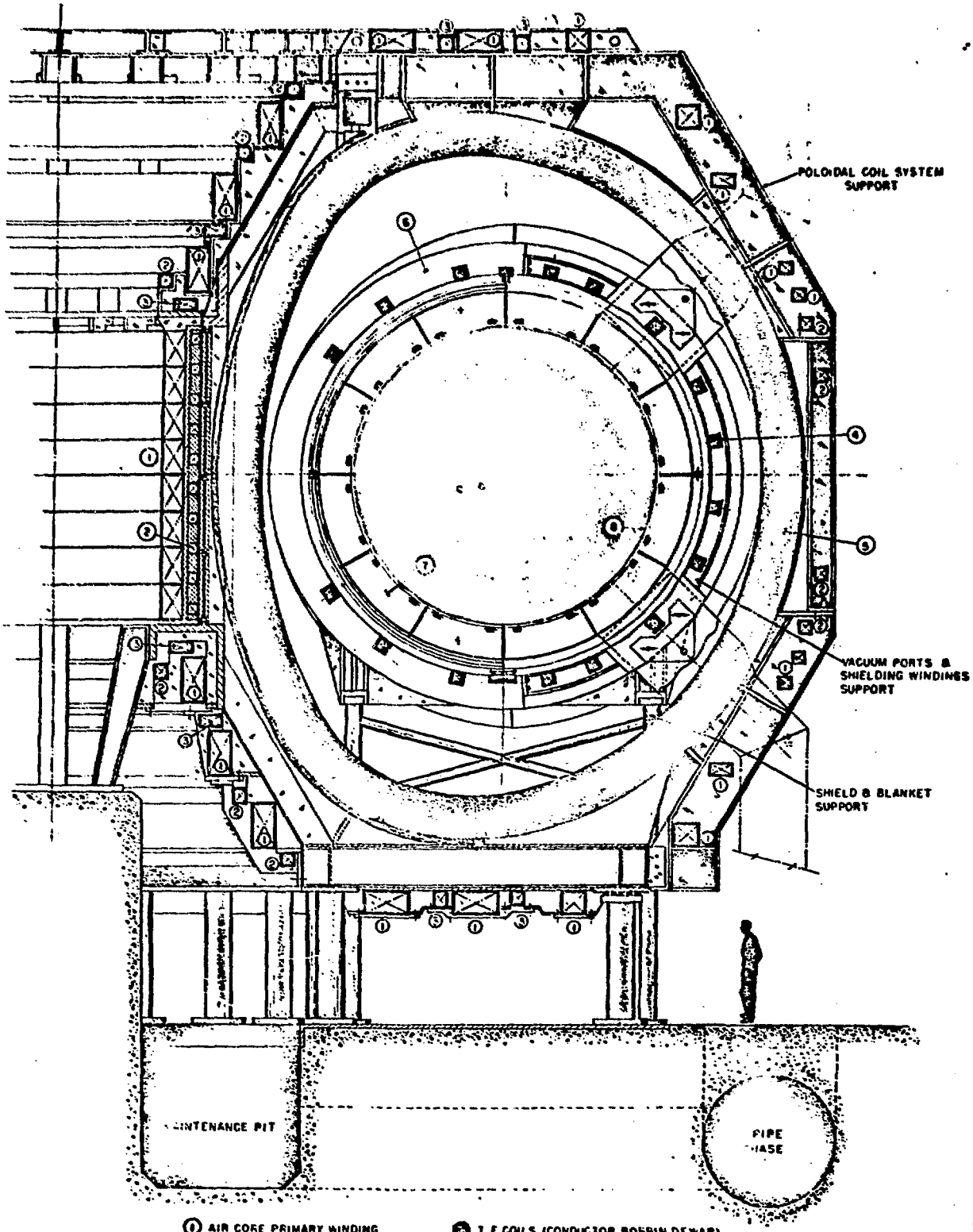
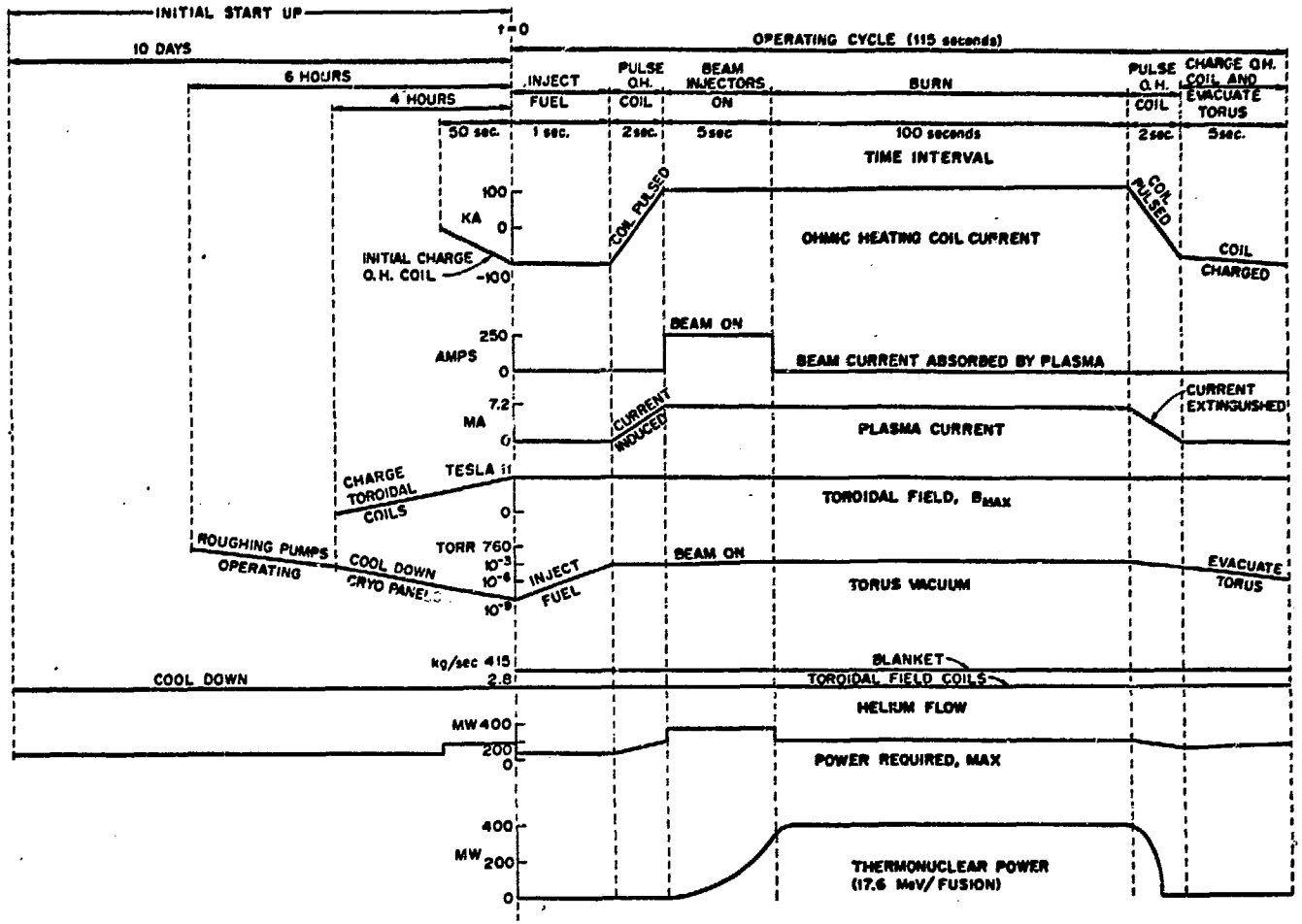
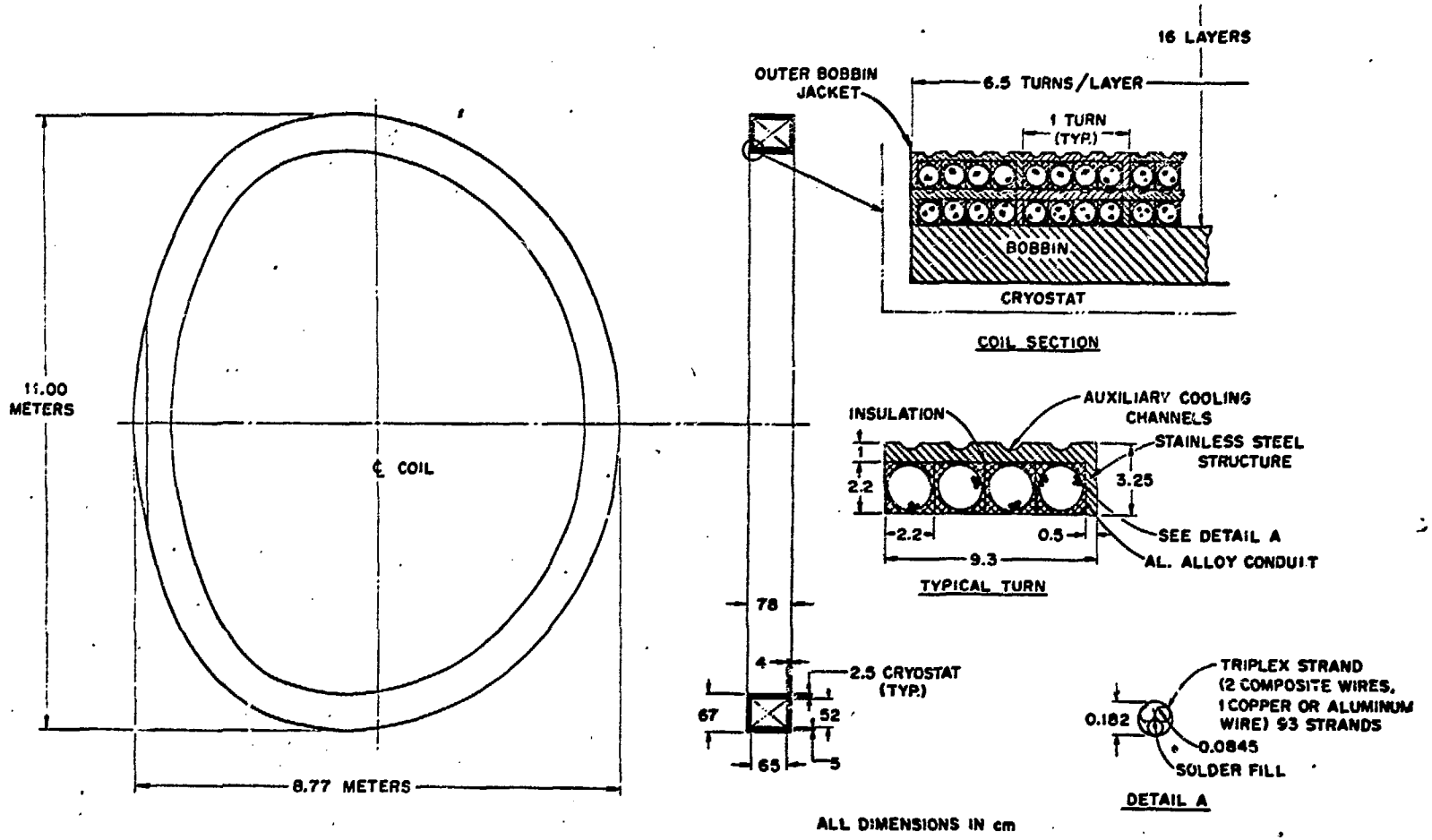

POLOIDAL COIL SYSTEM SUPPORT
VACUUM PORTS & SHIELDING WINDINGS SUPPORT
SHIELD & BLANKET SUPPORT
MAINTENANCE PIT
PIPE CHASE
1 AIR CORE PRIMARY WINDING
2 DECOUPLING WINDINGS
3 V. F. TRIM WINDINGS
4 SHIELDING/ V. F. WINDINGS
5 T. F. COILS (CONDUCTOR, BOBBIN, DEWAR)
6 SHIELD
7 BLANKET
8 VACUUM CONTAINMENT


INITIAL START UP
10 DAYS
6 HOURS
4 HOURS
t = 0
OPERATING CYCLE (115 seconds)
INJECT FUEL
PULSE O.H. COIL
BEAM INJECTORS ON
BURN
PULSE O.H. COIL
CHARGE O.H. COIL AND EVACUATE TORUS
50 sec.
1 sec.
2sec.
5sec
100 seconds
2sec.
5sec.
TIME INTERVAL
100
KA
0
-100
INITIAL CHARGE O.H. COIL
COIL PULSED
OHMIC HEATING COIL CURRENT
COIL PULSED
COIL CHARGED
BEAM ON
250
AMPS
0
BEAM CURRENT ABSORBED BY PLASMA
7.2
MA
0
CURRENT INDUCED
PLASMA CURRENT
CURRENT EXTINGUISHED
TESLA
CHARGE TOROIDAL COILS
0
TOROIDAL FIELD, B_{MAX}
TORR 760
10^{-3}
10^{-6}
10^{-9}
ROUGHING PUMPS OPERATING
COOL DOWN CRYO PANELS
INJECT FUEL
BEAM ON
TORUS VACUUM
EVACUATE TORUS
kg/sec 415
2.8
BLANKET
TOROIDAL FIELD COILS
COOL DOWN
HELIUM FLOW
MW 400
200
0
POWER REQUIRED, MAX
400
MW 200
0
THERMONUCLEAR POWER (17.6 MeV/FUSION)


11.00 METERS
℄ COIL
8.77 METERS
OUTER BOBBIN JACKET
78
4
2.5 CRYOSTAT (TYP.)
67
52
65
5
16 LAYERS
6.5 TURNS/LAYER
1 TURN (TYP.)
BOBBIN
CRYOSTAT
COIL SECTION
INSULATION
AUXILIARY COOLING CHANNELS
STAINLESS STEEL STRUCTURE
1
2.2
3.25
2.2
0.5
9.3
SEE DETAIL A
AL. ALLOY CONDUIT
TYPICAL TURN
TRIPLEX STRAND (2 COMPOSITE WIRES, 1 COPPER OR ALUMINUM WIRE) 93 STRANDS
0.182
0.0845
SOLDER FILL
DETAIL A
ALL DIMENSIONS IN cm

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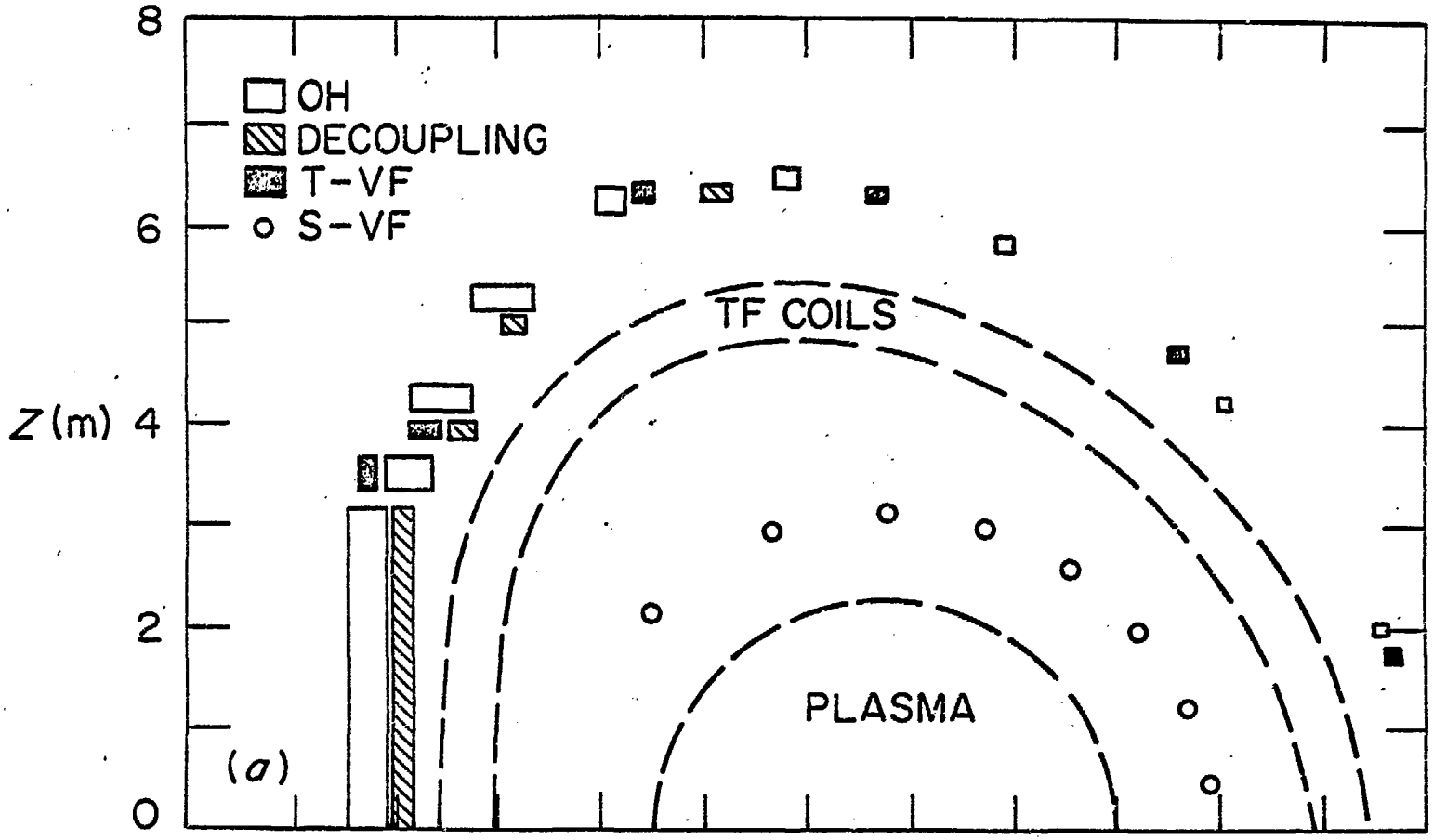

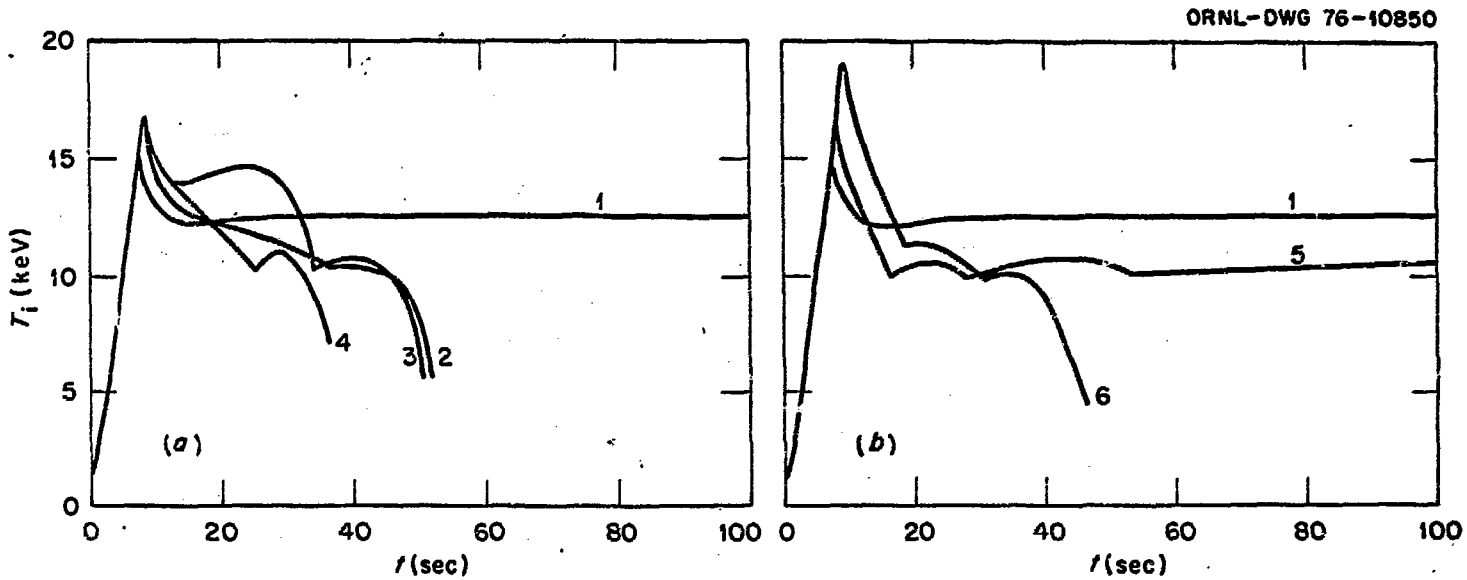

1. REFERENCE CASE: DIVERTOR EFF. = 90%, $\tau_{imp} = \tau_D$, N_0(edge) = 1 X $10^{15}/m^3$, STAINLESS STEEL WALL
2. NO DIVERTOR
3. DIVERTOR EFF. = 70%, τ_{imp} = INFINITY
4. N_0(edge) = 5 X $10^{15}/m^3$
5. NO DIVERTOR, CARBON LINER
6. NO DIVERTOR, CARBON LINER, τ_{imp} = INFINITY

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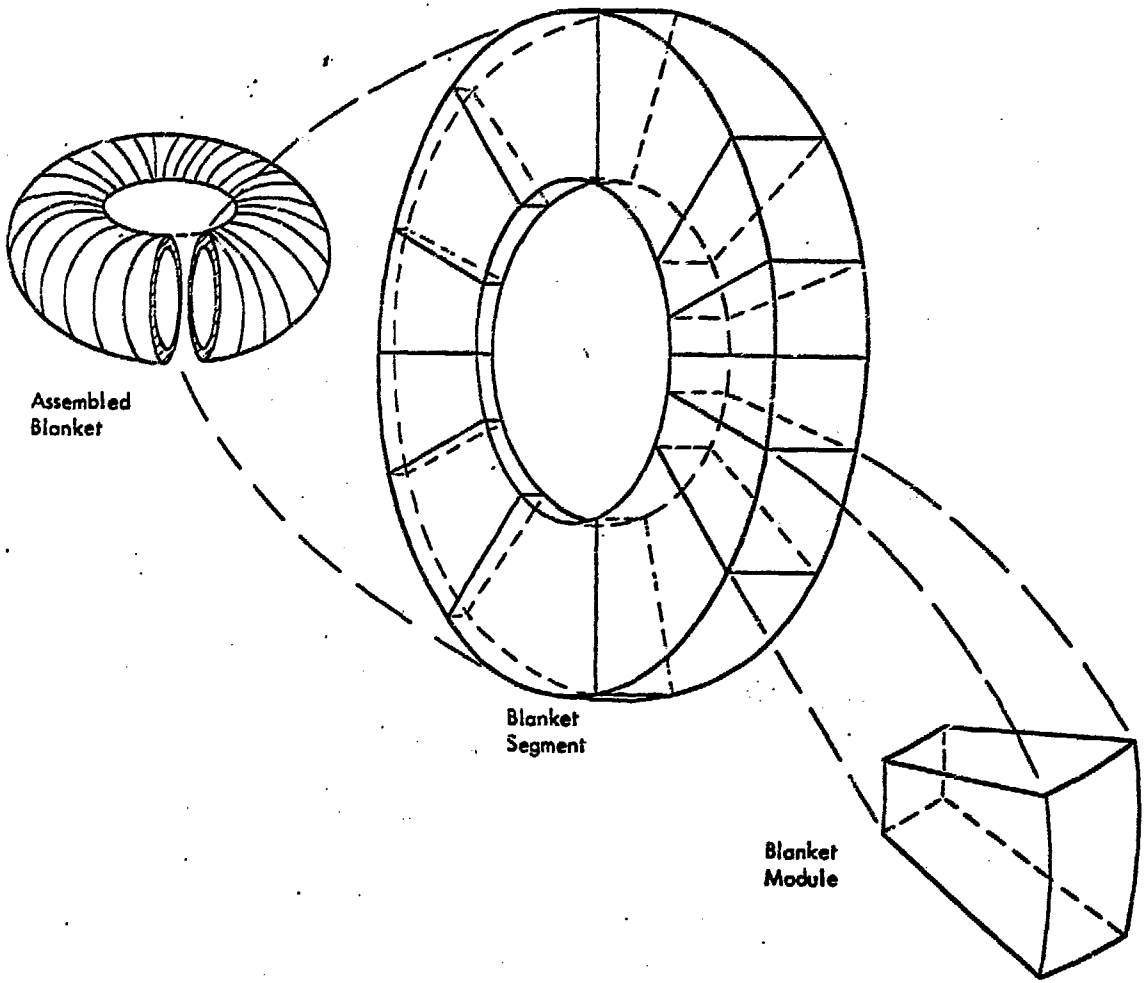

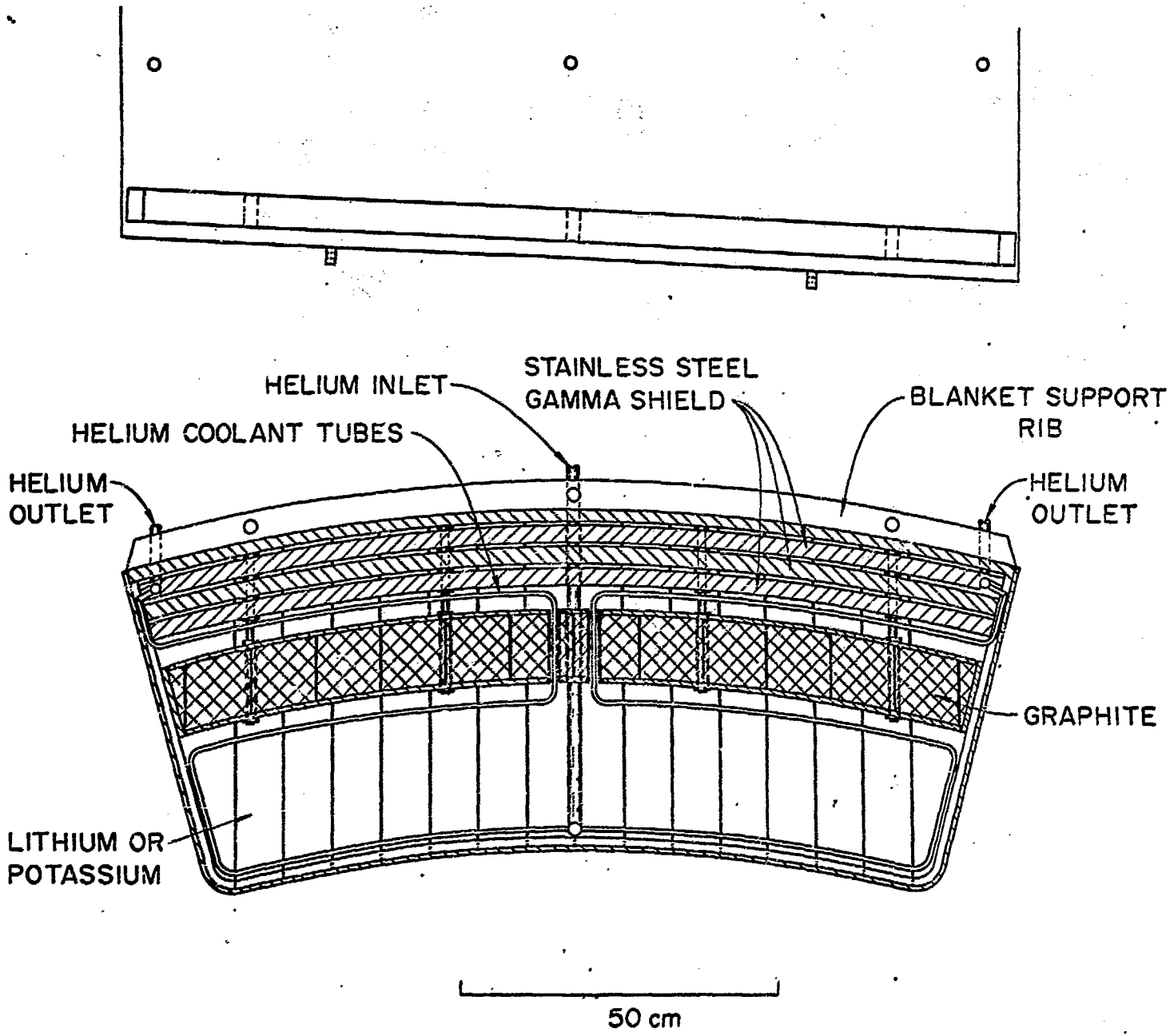
HELIUM INLET
STAINLESS STEEL
GAMMA SHIELD
BLANKET SUPPORT
RIB
HELIUM COOLANT TUBES
HELIUM
OUTLET
HELIUM
OUTLET
GRAPHITE
LITHIUM OR
POTASSIUM
50 cm

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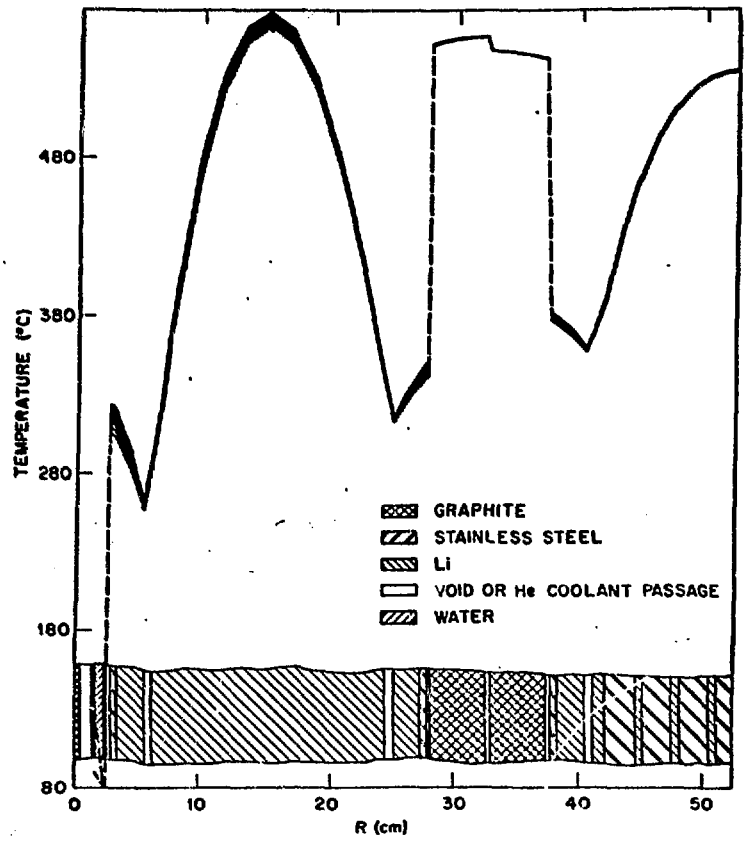
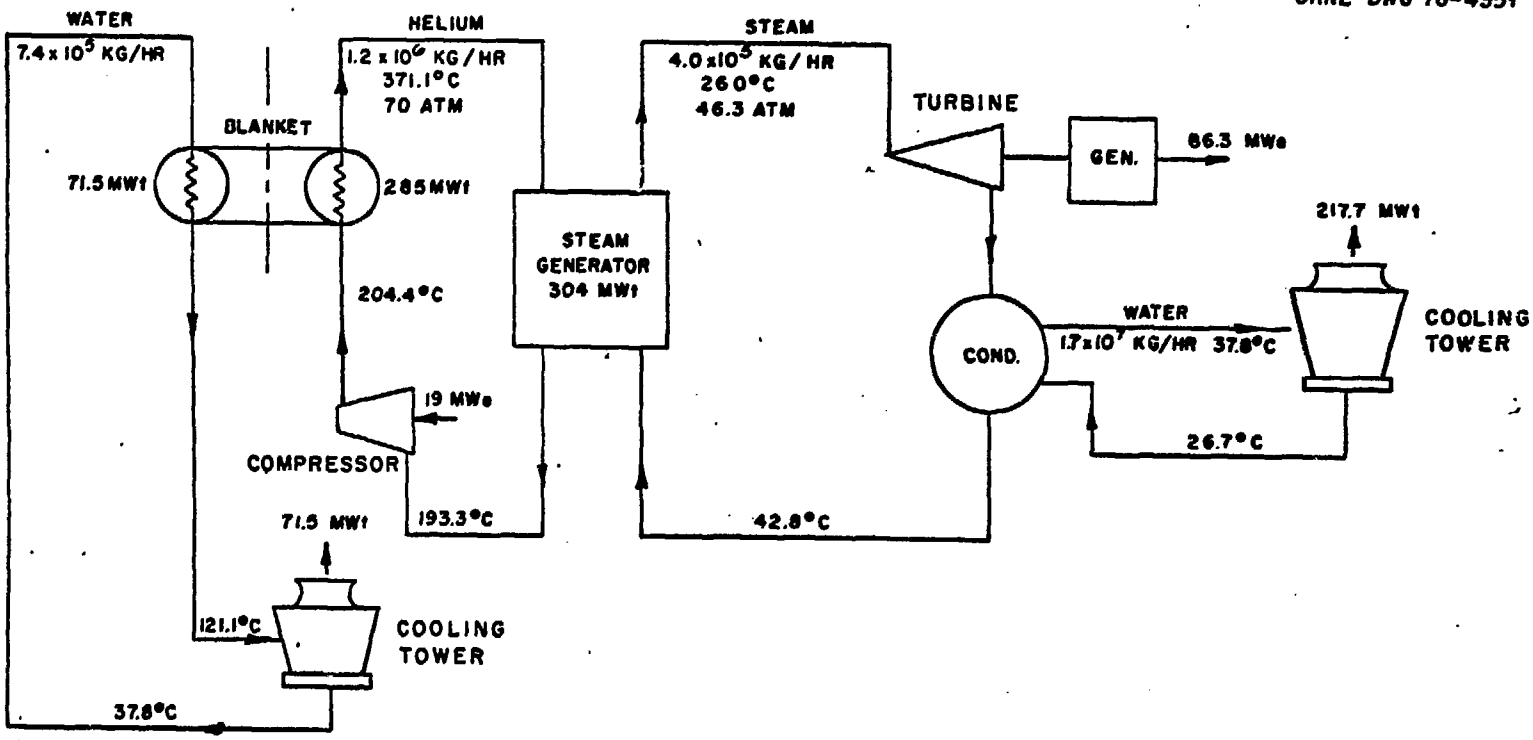
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